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ANALYSIS OF SRE POWER EXCURSION OF JULY 13, 1959

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REACTOR TECHNOLOGY
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ANALYSIS OF SRE POWER EXCURSION OF JULY 13, 1959

By
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ABSTRACT

Early in power run 14, during the time that the reactor core and coolant were contaminated with tetralin and its decomposition products, the SRE experienced the only power excursion in its operating history. The power level rose from about 4 Mw to about 14 Mw, or 70% of full power, before the reactor was scrammed. The shortest period reached was about 7.5 sec. This report presents the results of a study to determine the cause of the excursion. A description of reactor operating conditions and a chronology of events preceding the excursion are presented. The extent of fuel channel plugging and the temperature within severely plugged channels are discussed. It is concluded that the most probable cause of the excursion and relatively short reactor period was the evacuation of sodium from several fuel channels, which produced a sudden reactivity increase of about 0.3%. The conditions under which the channels could be evacuated of sodium are directly attributable to the presence of substantial quantities of organic material in about one-fourth of the fuel channels in the reactor.

I. INTRODUCTION

In June 1959 during power run 13, a quantity of an auxiliary organic coolant, tetralin, leaked into the primary cooling system of the Sodium Reactor Experiment (SRE). The tetralin decomposed in the hot sodium into a variety of organic compounds, carbon, and hydrogen. The reactor was shut down when the leak was discovered, the point of in-leakage was repaired, steps were taken to remove the organic materials from the system, and the reactor was started on power run 14.

The reactor behaved in an unusual manner during run 14; the unusual behavior included a power transient to ~ 70% of rated power level at 18:25 hours on July 13, 1959. A description of behavior during run 14 and all pertinent previous operating history of the SRE was presented in the Interim Report on SRE Fuel Damage[1] to which the reader is referred for background material. The present report is primarily concerned with investigating the cause of the power excursion although certain other anomalies are also considered.

A description of reactor behavior prior to the power excursion is first given. This is followed by a study of the effect of reactivity changes on reactor behavior. The intent of this study is to determine the magnitude and approximate time of occurrence of unexplained reactivity changes that took place before the start of the power excursion. In order to accomplish this, the reactor power level for a time interval of ~ 1 hour prior to the excursion was calculated by solving the kinetic equations. A knowledge of the magnitude of the unexplained reactivity changes needed to produce agreement with observed reactor behavior is of assistance in identifying the causes of anomalies in the behavior.

The sodium flow rate and the extent of plugging in fuel channels is next considered. This is followed by a discussion of local temperatures within severely plugged channels. The sodium exit temperature from fuel channels which are partially plugged is considered during the time interval of two minutes just preceding the scram at 18:25. Although this temperature instrumentation is quite satisfactory for normal reactor operation, the records do not provide sufficient information during the rapid increase in power that preceded the scram. Therefore, an estimate of temperatures during and just prior to the power excursion is presented. These results are of use in identifying the cause of the 7.5-sec reactor period.

It is believed that the reactivity required to produce the 7.5-sec period was introduced by the creation of void in the reactor core. Several mechanisms by means of which void could be created in partially plugged fuel channels are examined. The most likely mechanism appears to be the generation of vapor in severely plugged fuel channels due to local overheating and the consequent expulsion of sodium and perhaps other material from these channels as a result of pressure buildup.

II. DESCRIPTION OF REACTOR BEHAVIOR

A. CHRONOLOGY

The following description of reactor behavior covers the period from 1728 hours to the scram at 1825 hours. Additional background information can be obtained from reference 1. The solid curve in Figure 1 shows the relative reactor power during this time interval. Unity on this curve corresponds to about 1.6 Mwt. It should be realized that a possible systematic uncertainty of perhaps 10% is present in this value. Full power is 20 Mwt. Reactor operating conditions at approximately 1728 and 1823 are given in Table I.

TABLE I

OBSERVED REACTOR OPERATING CONDITIONS

Property			
Time of Day	1728	1823	1825
t (sec)	0	3300	3420
Power (Mw)	1.6	3.7	14
Period (sec)	∞	---	7.5
Sodium Flow Rate (gal/min)	850	850	850
Moderator Temperature, corner channel 61 (°F)	557	617	620
Inlet Sodium Temperature (°F)	470	475	475
Mixed Mean Outlet Temperature (°F)	520	585	---
Selected Fuel Channel Exit Temperature (°F)			
Channel R-54	495	540	630
Channel R-10	590	755	---
Channel R-25	605	740	---
Channel R-31	590	725	---
Channel R-35	590	710	---
Channel R-68	580	695	---
Channel R-24	575	685	---

At about 1728 the reactor was in a steady-state condition (infinite period) at a power of 1.6 Mw. Fuel channel exit temperatures and moderator coolant temperatures were behaving erratically (see reference 1, section III-G). In view of operating experience gained in December 1959 when similar reactor conditions

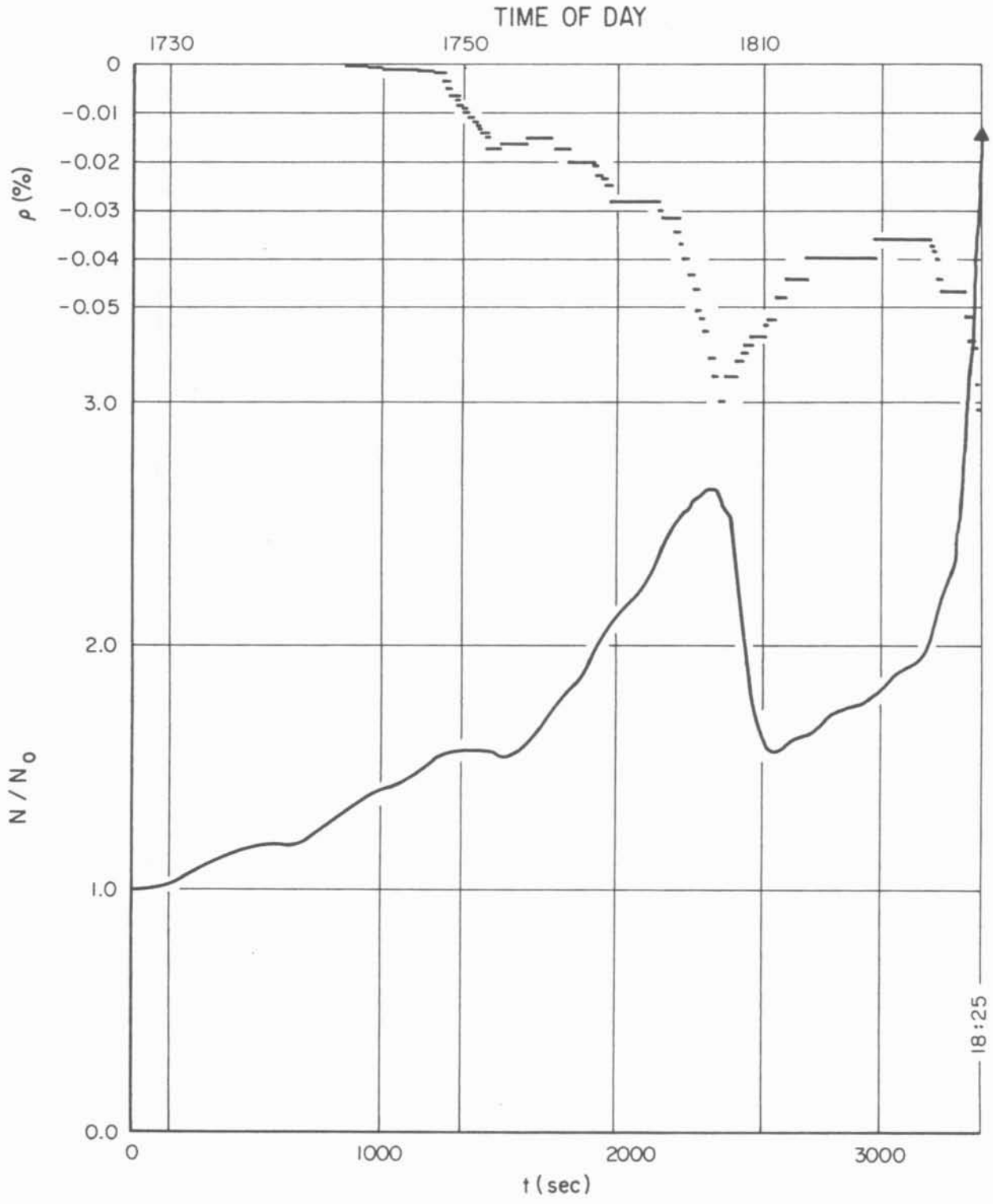

Figure 1. Relative Power and Reactivity Introduced Due to Shim Rod Motion

occurred (see reference 1, section III-A), it was believed that operating the reactor at a slightly higher temperature but with a restricted power level would improve its behavior. A planned power increase was started, and it was soon observed that the power was increasing faster than expected. Shim rods were slowly inserted in order to reduce the rate of rise. Table II and Figure 1 give the reactivity values introduced by shim rod motion. At 1752 the power rise halted briefly, and the power made a slight dip before resuming its upward climb. The rate of increase was now more rapid than before and shim rods were again inserted slowly in order to reduce it. By 1807 the power had increased to 4.2 Mw and, as this point was approached, the increase in power gradually stopped. Soon after this, the power started to fall abruptly and reached a value of 2.4 Mw about 3 min later. The shortest negative reactor period during this power decrease was recorded at the start of this drop and was about a 45-sec period. This soon changed to a negative period of $\sim$ 200 sec which continued for $\sim$ 2 min.

Shim rod withdrawal was started as soon as the power started to decrease at 1807. By 1811 the reactor was again critical and a slow rate of increase in power was maintained. By 1821 the power had risen to 3.0 Mw, and at this time started to rise at a more rapid rate. Shim rod insertion was again started in order to reduce the rate of rise. However, the power continued to rise and reached 3.7 Mw at about 1823. From this point on, the rate of increase was more rapid although shim rods were still being inserted. By about 1824 the power had increased to 4.7 megawatts and was still rising. At 1825 when the reactor was scrammed, the power had risen to about 14 Mw which corresponds to 70% full power.

Five to ten seconds before the reactor was scrammed, the operators noted that the period meter made three distinct positive swings indicating periods of about 50 sec. Following this, the period meter rose rapidly and indicated a 7.5-sec period.

B. PERTINENT STRIP CHART RECORDINGS OF REACTOR CONDITIONS

The reactor period is recorded on a strip chart which moves at a speed of 2 in./hr. Hence, it is not possible to see clearly the above three positive period swings that preceded the 7.5-sec swing. However, from a careful inspection of the instrument chart it appears that these three instrument swings are distinguishable by means of the blackness of the ink on the chart paper. It appears that the periods indicated were about 70, 60, and 45 sec.

TABLE II

REACTIVITY CHANGES INTRODUCED BY CONTROL ROD MOTION

Time (hr:min:sec)	t (sec)	$\Delta\rho$ (%)	Time (hr:min:sec)	t (sec)	$\Delta\rho$ (%)
17:42:44	864	-0.00027	18:04:32	2192	-0.00245
:43:48	946	-0.00044	:04:45	2205	-0.00300
:44:38	998	-0.00024	:05:10	2230	-0.00326
:47:05	1145	-0.00061	:05:30	2250	-0.00345
:48:07	1207	-0.00008	:05:40	2260	-0.00421
:48:48	1248	-0.00190	:05:50	2270	-0.00177
:48:53	1253	-0.0078	:06:13	2293	-0.00245
:49:00	1260	-0.00082	:06:30	2310	-0.00559
:49:04	1264	-0.00078	:06:46	2326	-0.00381
:49:10	1270	-0.00082	:07:08	2348	-0.00490
:49:38	1298	-0.00082	:07:35	2375	+0.00490
:49:41	1301	-0.00089	:08:16	2416	+0.00326
:49:58	1318	-0.00078	:08:45	2445	+0.00163
:50:14	1334	-0.00089	:08:49	2449	+0.00163
:50:26	1346	-0.00099	:09:15	2475	+0.00190
:50:46	1366	-0.00089	:10:10	2530	+0.00231
:51:00	1380	-0.00075	:10:20	2540	+0.00150
:51:05	1385	-0.00072	:10:53	2573	+0.00421
:51:10	1390	-0.00078	:11:32	2612	+0.00340
:51:34	1414	-0.00089	:12:54	2694	+0.00464
:51:36	1416	-0.00231	:17:25	2965	+0.00381
:52:46	1476	+0.00096	:21:20	3200	-0.00109
:54:14	1574	+0.00121	:21:28	3208	-0.00122
:56:07	1687	-0.00055	:21:35	3215	-0.00231
:56:10	1690	-0.00041	:21:44	3224	-0.00300
:57:10	1750	-0.00395	:21:54	3234	-0.00326
:58:43	1843	-0.00068	:21:57	3237	-0.00314
:59:00	1860	-0.00204	:23:45	3345	-0.00177
:59:24	1884	-0.00082	:23:55	3355	-0.00510
:59:36	1896	-0.00121	:24:03	3363	-0.00150
:59:48	1908	-0.00326	:24:24	3384	-0.00762
18:03:15	2115	-0.00218	:24:35	3395	-0.00503
:03:27	2127	-0.00109	:24:45	3405	-0.06220
:04:15	2175	-0.00314	:25:00	3420	scram

The neutron flux is recorded on a strip chart which moves at a speed of 2 in./hr. Careful readings taken from this chart provided the data for plotting the solid curves in Figures 1, 2, 3, and 4. The small wiggles that appear in the curves shown in Figures 1, 2, and 3 are visible on the chart and were reproduced free hand in the figures. The time scale in Figure 4 is greatly expanded for convenience in plotting the calculated results shown there. It must be recognized that the solid curve shown in Figure 4 is subject to a large uncertainty due to this expansion of the time scale.

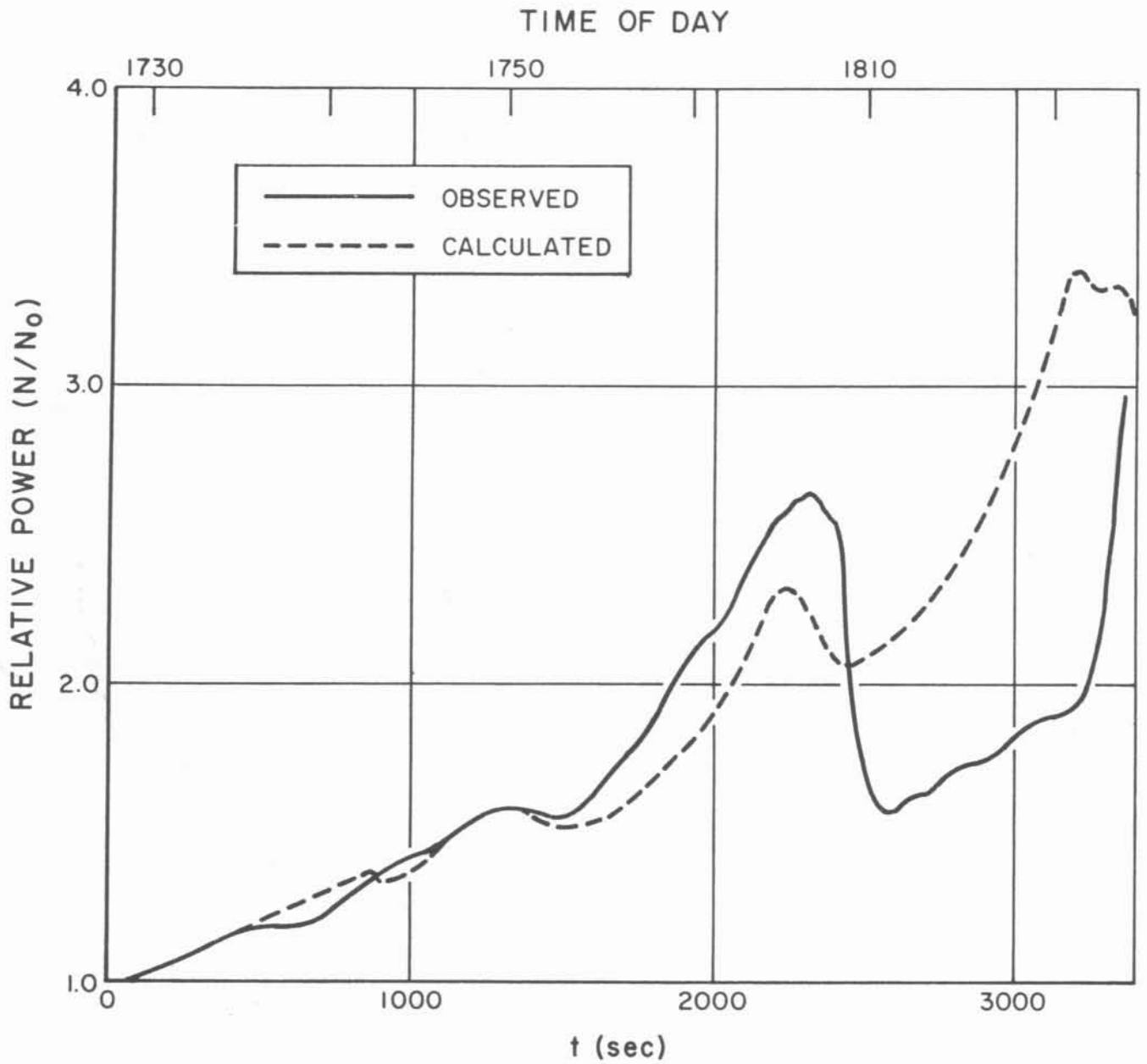

Figure 2. Relative Power vs Time (Unity corresponds to 1.6 Mw total reactor power; calculation used normal fuel and moderator temperature coefficients and observed control rod motion reactivity changes.)

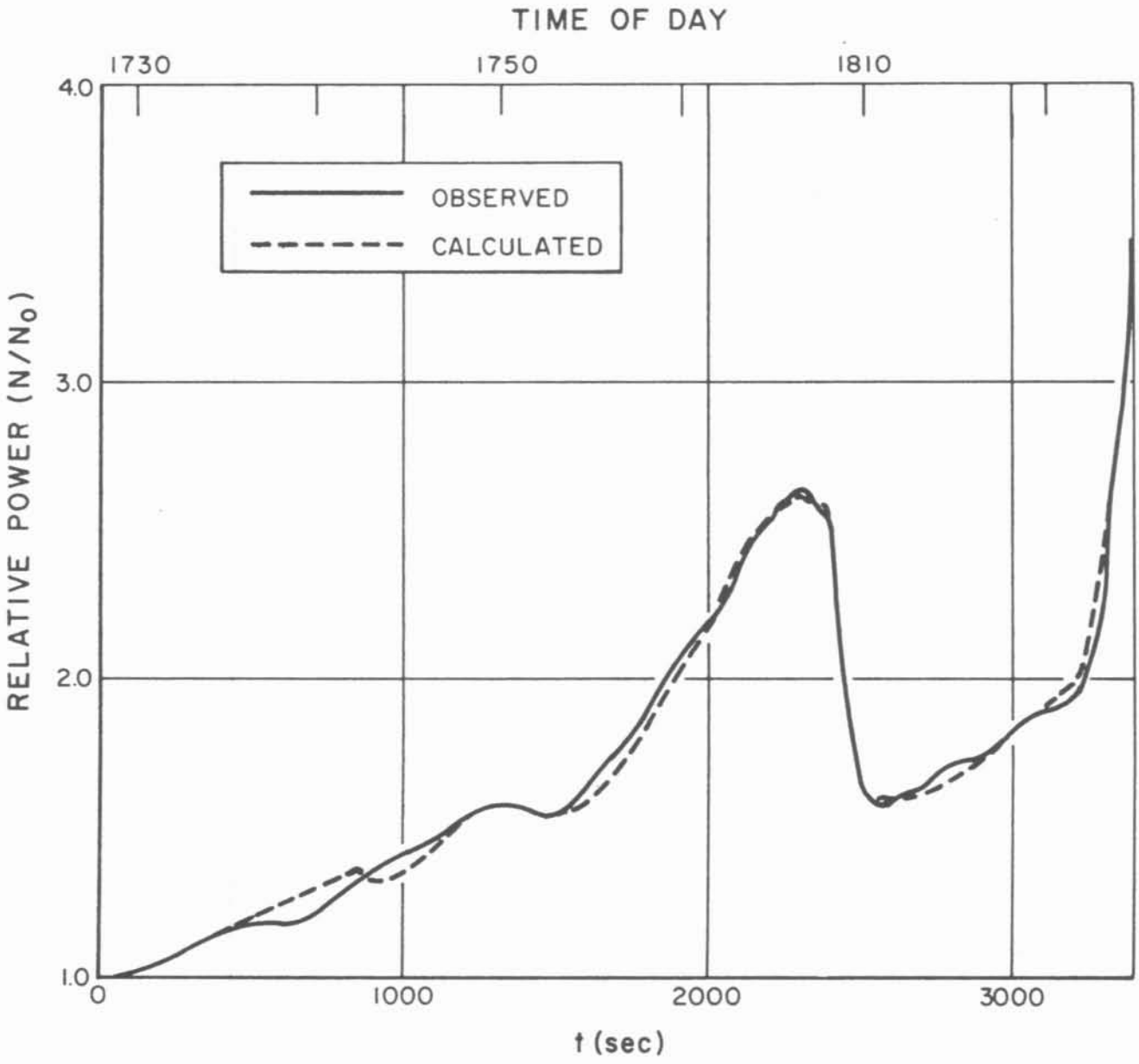

Figure 3. Modified Relative Power Calculation (The reactivities given in Table V were used in addition to those used for the calculation shown in Figure 1.)

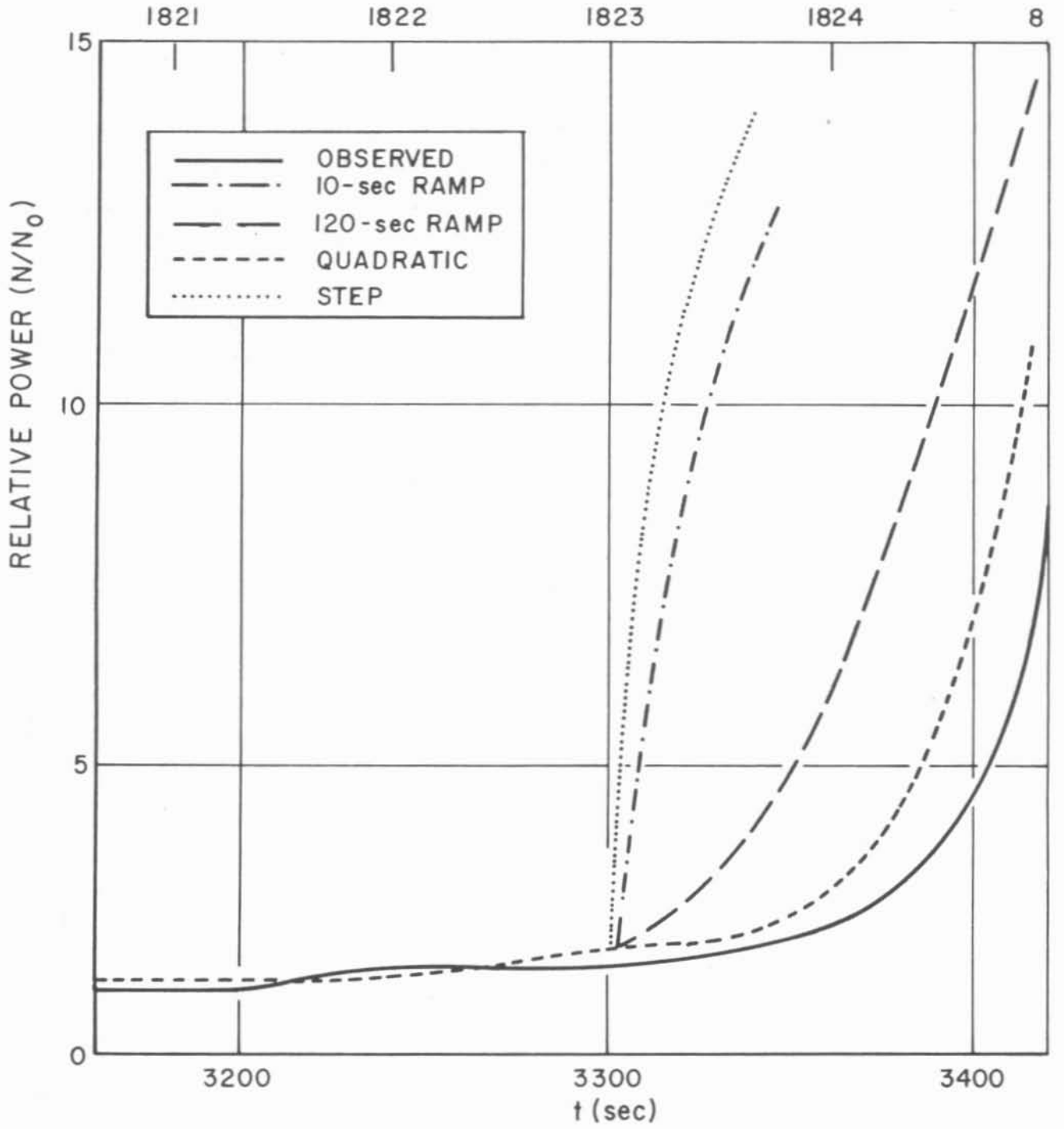

Figure 4. Calculations for the Positive Excursion

All fuel channel exit temperatures with the exception of that for core channel 54 are printed by a point recording instrument. A point is printed for a given core channel every 5 min, so detailed information as to fuel channel exit temperature during the final upsurge of the power excursion is not available. The maximum temperature printed on the chart is 775°F (core channel 10) but this point was printed about 2 min before the scram. Table I shows other fuel channel exit temperatures that were printed at about the same time. An examination of the instrument chart shows that the exit temperatures for all damaged fuel elements were behaving in an erratic manner. Some temperatures were fluctuating by as much as 100°F between successive points on the chart. The fuel channel exit temperature for all channels that contained damaged fuel elements was abnormally high. This is demonstrated by the data presented in Table III.

TABLE III
SRE FUEL CHANNEL FLOW RATES

Core Channel Number	ψ = Percent of Nominal Flow			Core Channel Number	ψ = Percent of Nominal Flow		
	July 14, 0900	July 15, 0400	July 22, 2200		July 14, 0900	July 15, 0400	July 22, 2200
4	47	46	49	45	169	133	140
9	118	85	116	46	139	133	140
10	31	35	34	47	139	133	96
11	101	114	72	53	139	133	(A)(3)
12	39	39	37	54	150	170	140
19	139	133	178	55	51	55	53
20	121	133	99	56	121	133	121
21	51	46	52	57	139	133	121
22	90	88	161	58	139	133	126
23	48	51	46	65	113	98	(A)(3)
24	45	39	48	66	139	133	121
25	38	33	46	67	139	133	102
31	50	43	48	68	38	37	40
32	152	151	140	69	52	48	62
33	139	133	140	70	139	133	136
34	139	104	140	71	82	133	42
35	48	41	49	73	91	133	113
36	83	84	110	74	139	133	120
41	163	133	148	75	129	133	92
42	139	133	140	76	113	133	54
43	42	48	47	80	102	133	79
44	(A)(1)	(A)(2)	(A)(3)				

(A) Thermocouple reading below reactor inlet temperature
(1) Flow assumed 139% of nominal
(2) Flow assumed 133% of nominal
(3) Flow assumed 140% of nominal

Totals =	44.24	43.81	42.18

C. NOTEWORTHY ASPECTS OF THE EVENTS LEADING UP TO THE SCRAM AT 1825

1. General Features

With the exception of the large drop in power that occurred at about 1807, all changes in power level until about 30 sec before the scram were quite slow. However, the power rose at a more rapid rate than was expected based on past experience and shim rods were inserted to restrain the rise. It was of interest to determine to what extent the observed power changes could be accounted for by using the reactivity changes introduced by shim rod motion and normal reactor temperature coefficients.

2. Minor Fluctuations in Power

Many minor power fluctuations are seen in the solid curve shown in Figure 1. Such behavior is not normal. It could be caused by temperature fluctuation of a few degrees in one or two fuel clusters (Doppler effect). A temperature change of 40°F in one fuel element will produce a reactivity change of about 0.001%. The behavior of the fuel channel exit temperatures suggests that this is a probable cause. The calculations of reactor power to be described in the next section make no attempt to duplicate these minor fluctuations in power.

3. The Negative Transient

The drop in power that occurred at about 1807 seems to represent a substantial loss in reactivity. This point was examined to find out how much unaccounted for negative reactivity was involved, and how rapidly it was inserted.

4. Apparent Amplification of Reactivity Changes

The tendency during run 14 for the rate of the power rise to accelerate as the power rose requires an explanation. This acceleration was slow, to be sure, but it seemed to increase as the power rose. This is illustrated by the size of the shim rod motions shown in Table II.

5. The Positive Transient

The excursion at 1825 is the main point of this study. The amount of unaccounted-for positive reactivity needed to produce it, and the mechanism which introduced this reactivity is of great interest, as is the temperature of fuel and coolant in partially plugged fuel channels during the excursion.

III. CALCULATIONS OF THE POWER CURVE AND DISCUSSION OF RESULTS

A. THE FIRST CALCULATION

1. Description

The reactor kinetics equations were solved by means of the AIREK IBM code.[2] The starting time for the calculation was taken at 1728 hours. Since exact shim rod changes during the time interval from 1728 to 1742 are not available from reactor records, the calculation was started by introducing reactivity steps which were chosen so as to approximate the power curve up to about 1750 hours. These reactivity steps are shown in Table IV.

TABLE IV

REACTIVITY STEPS USED TO START CALCULATION OF THE POWER CURVE

Time (hr/min/sec)	Time Introduced (sec)	Reactivity Introduced by the Step (%)
17:28:00	0	+0.00226
17:30:00	120	+0.00195
17:42:24	864	-0.00790

Shim rod motion following 1742 was obtained from a strip chart recorder which records the motion of rod 3. A rod movement of a few thousandths of an inch can be detected. The 4 shim rods were operated in ganged condition starting at 1742 through 1825. Changes in rod position were converted to reactivity by means of the experimentally measured rod calibration curve. The reactivity changes introduced by shim rod movement are shown in Table II.

Reactivity changes due to shim rod motion and to average fuel and moderator temperature changes were introduced into the AIREK calculation at 1742. The arbitrary reactivity steps that were used to start the calculation were adjusted so as to produce a good fit to the power curve in the neighborhood of 1750. The calculation was continued beyond 1750 using only those reactivity changes produced by shim rod motion and fuel and moderator temperature changes.

The fuel temperature coefficient was taken to be -1.1×10^{-5} per °F and the moderator temperature coefficient to be $+1.7 \times 10^{-5}$ per °F.[3] This moderator temperature coefficient is for an average moderator temperature of 600°F.

The change in fuel temperature was taken to be proportional to the change in power. The following empirical relationship was obtained by inspecting reactor records:

$$T_{fuel}\ (°F) = 23\ \frac{N - N_o}{N_o}\ ,$$

where N represents the relative reactor power and N_o corresponds to 1.6 Mw.

Moderator temperature changes as a function of time are not normally recorded and for the time interval being examined are not available from reactor records. A thermocouple located in a beryllium probe in a dummy element measures the temperature of the moderator coolant. This dummy element is located in a corner channel between moderator cans (channel 61). Sodium coolant flows through the annulus around this dummy element, so the temperature changes recorded in it are undoubtedly smaller than those occurring in the hexagonal graphite moderator blocks. Nevertheless, the temperature changes as indicated by the probe were used in this calculation for the moderator temperature changes. Hence, these calculations of reactor power use a low value of the reactivity changes caused by changes in moderator temperature. Table V shows the channel 61 temperature changes as a function of time. Linear interpolation was used for points between table entries. The results of the calculation made using the data given above are shown in Figure 2. The prompt neutron lifetime was taken[3] to be 0.5×10^{-3} sec, and the total delayed neutron fraction $\beta = 0.0075$. The solid curve shows the observed relative power while the dotted curve gives the calculated result.

2. Discussion of the Results of the First Calculation

Several interesting features shown by the experimental curve also appear in the calculated result. In particular, the drops in power that occur at 1752 and 1807 appear in the calculated result although the calculated peak occurring at 1807 is about 15% low. Use of a slightly greater moderator temperature rise in the calculation would increase this peak to the observed value. This is consistent with the belief expressed above that the moderator temperature changes used in the AIREK calculation are too small. The slow increase in power following 1811 is also obtained by the calculation, but the positive excursion at 1825 is completely missing. In fact, the calculated result shows a small decrease in power after 1821.

TABLE V

MODERATOR TEMPERATURE RECORDED IN CORNER CHANNEL R-61

Time (hr:min)	Temperature (°C)	Temperature (°F)
17:42	316.0	602
17:55	320.0	609
18:01	328.0	623
18:07	337.5	639
18:10	336.4	637
18:13	334.7	634
18:16	336.4	637
18:25	344.0	652

B. THE SECOND CALCULATION

1. Description

A modification of the above calculation was next made, the results of which are shown by the dotted curve in Figure 3. In addition to the reactivities used in making the first calculation, several steps and ramps were introduced and adjusted by trial and error so as to produce a reasonably good fit to the observed power curve. These additional reactivity changes are summarized in Table VI. They were chosen to illustrate the magnitude of the changes required to give agreement with the observations rather than to attempt to fit the small details shown in the experimental power curve.

2. Discussion of the Results of the Second Calculation

The reactivity modifications made prior to 1708 are small and correspond to temperature changes of a few degrees in the moderator or fuel. It is believed that the discrepancy between the first calculation and the observed power curve up to this time can be accounted for in terms of the uncertainty in reactor temperatures.

a. The Negative Transient

(1) Description

The reactivity step of -0.06% that was introduced into the modified calculation at 1808 hours produced a negative power response similar to the one

TABLE VI

SUMMARY OF ADDITIONAL REACTIVITIES INTRODUCED FOR THE MODIFIED CALCULATION SHOWN IN FIGURE 3

Time (hr:min:sec)	Time Started (sec)	Type	Time Ended (sec)	Total $\Delta\rho$ (%)
17:49:40	1300	ramp*	2000	+0.0077
18:00:20	2000	ramp	2200	-0.0060
18:04:00	2200	ramp	2400	+0.0200
18:08:00	2400	step	3420	-0.0600
18:10:20	2540	ramp	2910	+0.0740
18:16:40	2910	ramp	3162	-0.125
18:20:42	3162	ramp	3300	+0.0406
18:23:00	3300	†	3420	+0.30

*At the end of each ramp, the total reactivity so introduced is retained until 3420 sec.

†This reactivity was introduced in 4 different ways (see Figure 3):

a) By a step
b) By a 10-sec ramp
c) By a 120-sec ramp
d) By the quadratic relation — $\Delta\rho(t) = 2.19 \times 10^{-5}(t - 3300)^2$ %.

observed. The calculation gave a value of ~ 200 sec for the stable negative period, which is close to the observed value given in section II-A. However, it was found that the calculated power level resulting from this negative step continued to drop after 1811 and fell to values far below those observed. The positive ramp introduced into the AIREK calculation at 1810 corrected this. This ramp introduced a total reactivity of 0.074% in 6 min and 20 sec time, which shows that a substantial amount of the reactivity loss that is required to produce the observed negative period must have been recovered within a few minutes after the onset of the negative period.

Examination of records of shim rod motion and reactor temperature also shows that the two reactivity changes referred to above were of nearly the same magnitudes. From Table II, we see that at 1807 the shim rods were inserted to a greater depth than they were at 1756, but the power levels are the same at these two times. The extra reactivity being held down by the shim rods at 1807 is 0.029%. However, the moderator temperature as indicated by the beryllium probe thermocouple is about 30°F greater at 1807 than at 1756. This

temperature increase corresponds to a reactivity increase of 0.05%. The difference between these two reactivity values provides an estimate of the difference in magnitude of the two reactivity changes being discussed.

(2) Possible Explanations of the Negative Transient

(a) Doppler Effect

A change in temperature of 55°F in all the fuel is required to produce a reactivity change of 0.06 by means of the Doppler effect. The fuel temperature coefficient has a time constant of the order of 10 sec while the moderator temperature coefficient has a time constant of ~ 10 min. It thus appears that although graphite temperature changes would be much too slow to account for the observed negative reactor period, a sudden change in fuel temperature could have contributed substantially to this effect. It was mentioned earlier that the exit temperatures for several fuel channels were fluctuating by as much as 100°F. Although it is not clear that these observed temperature fluctuations were large enough to cause a reactivity loss of sufficient magnitude to produce the observed reactor period, there is a distinct possibility that such was the case. The movement of material that was causing plugging of fuel channels could cause sudden wide temperature changes in one or more fuel channels. If the temperature fluctuations in several fuel elements were to get in step, a substantial reactivity change would result.

(b) Fuel Slug Rain

A second manner in which the introduction of reactivity changes of the type described above could have been achieved is by the parting of a damaged fuel cluster. Calculations[1] have shown that if a fuel cluster parts near its center and the fuel in the bottom portion drops 10-3/4 in. (which is the maximum drop physically possible) a reactivity loss of as much as 0.08% can result. If, after such an event the fuel remaining in the top portion should rain down upon the already displaced fuel, a reactivity gain of nearly the same magnitude would result. This sequence of events would explain the observed behavior. However, no evidence of fuel dropping has been found, and it is believed that this explanation of the negative transient is unlikely. In fact, the examination of the fuel from core channel 24 indicates that extensive fuel swelling occurred before the elements parted and was of sufficient severity to wedge the fuel rods firmly in the channel.[4]

(c) Gas Bubbles in the Core

If large gas bubbles were present in the core, their movement could explain the negative transient. The movement of a large bubble from a region of high importance to one of low importance would in effect transfer sodium from a region of low importance to a region of high importance. If a new bubble were then to grow in the region of high importance, sodium would be displaced and reactivity recovered. If the presence of large gas bubbles in the core is granted, which seems reasonable with decomposing organic in the core (although difficult to prove), this phenomenon is a very reasonable explanation for the negative transient.

(d) Formation of Vapor Within Partially Plugged Channels

In section V the nature of the plugging in fuel channels will be discussed. Evidence will be presented which shows that plugging in the 13 channels containing damaged fuel was extensive and that at reactor power levels of ~ 2 Mw local temperatures within these plugs were probably high enough to vaporize sodium. The concept of vapor production in regions of severe plugging as the power level rises leads to the belief that the following sequence of events is the most likely course of the negative excursion.

The evolution of vapor as the power rose introduced void in the plugged fuel channels by the displacement of sodium. This resulted in the introduction of a small amount of positive reactivity. Fuel temperature in the immediate vicinity of the plug rose as the formation of vapor took place. Control rods were inserted in order to restrain the power increase, and this insertion in fact halted and started to reverse the rise in power. As power started to drop, the volume of vapor in fuel channels decreased, thereby introducing negative reactivity and speeding the drop in power. The collapsing bubbles would allow more coolant to enter the plugged regions, thereby rapidly cooling the overheated fuel contained therein. The reduction of fuel temperature introduced positive reactivity due to the Doppler effect, thereby reducing the rate at which power was dropping. This recovery of reactivity allowed the reactor to be made critical again with a relatively small withdrawal of control rods.

b. Power Rise Preceding the Positive Transient

(1) Description

The steady increase in the rate of power rise during the 4 min prior to the onset of the power excursion that occurred just before 1825 indicates that reactivity was being introduced into the reactor during this time. It is clear from the power curve shown in Figure 2 that this reactivity was introduced in some gradual manner. The modified power calculation was made using a positive ramp which introduced a total reactivity of about 0.04% over a time interval of 2-1/3 min. This value was chosen so as to give good agreement with the observed rise in power (next to the last entry in Table VI).

An interesting observation can be made from a study of the reactivity changes due to shim rod motion shown in Figure 1 and their correlation with reactor power. During the time interval from 1756 to 1807 reactor power increased steadily from 2.5 Mw to 4.3 Mw. A large number of shim rod insertions were made as the power rose, resulting in a net reactivity change of $\sim$ -0.05%. The power rise was halted at $\sim$ 1807 and shortly after this the negative reactor period was observed. Directing our attention next to the time interval from 1811 to 1821, reactor power increased from 2.5 Mw to 3.0 Mw. Three shim rod withdrawals were made during this time resulting in a net reactivity change of +0.014%. During the time interval from 1821 to 1824, power increased more rapidly, reaching a value of about 4.7 Mw. During this time shim rods were inserted resulting in a total reactivity change of -0.022%. It is clear from the above figures that a much smaller reactivity insertion was made during the last power increase than was made during the one ending at 1807.

Still another point to consider is the observation that during the last half of run 14 the reactivity changes showed a strong correlation with temperature (reference 1, p IV-D-15). For a given change in power, the observed reactivity changes as indicated by shim rod motion were nearly double those which past experience indicated were to be expected.

(2) Probable Cause of the Slow Power Rise

Two mechanisms which could contribute to the reactivity increase needed to produce the slow power rise will now be given. Both mechanisms were probably acting at the same time, each contributing a portion of the reactivity.

The first mechanism is the formation of sodium vapor within partially plugged channels. This was mentioned earlier in the section relating to the negative transient. At power levels above ~2 Mw, local temperatures within severely plugged channels were probably high enough to vaporize sodium. Sodium vapor would displace liquid sodium and result in the introduction of positive reactivity.

The second mechanism is an abnormal rise in moderator temperature. A moderator temperature rise of 25°F in excess of the amount registered by the thermocouple located in corner channel 61 would introduce a 0.04% reactivity increase. The temperature rise indicated in corner channel 61 is undoubtedly less than the temperature rise in the bulk of the moderator. This is because the thermocouple in channel 61 responds to changes in moderator coolant temperature. Moreover, if (as seems likely) plugging occurred between moderator cans or at the grid plate where moderator coolant enters, the cooling of graphite hexagons in the neighborhood of such plugs would be reduced. Temperature changes in the moderator would therefore be substantially greater than those indicated by the thermocouple in channel 61. The positive reactivity introduced as a result was not entirely compensated for by insertion of shim rods because an attempt was being made to increase the reactor power level. Fuel temperature fluctuations are unlikely as a cause of this power increase because of the slowness of the power rise. Reactivity changes due to fuel temperature changes would appear very rapidly.

c. The Positive Transient

(1) Description

Attention is next directed to the fast excursion. The solid curve in Figure 4 shows the observed power. Although there is a large uncertainty in this curve during the interval from 1824 to 1825 due to the chart speed of the flux recorder being only 2 in./hr, it is evident from an examination of the chart that the steep part of the rise did not start until nearly 1825. Moreover, the operators reported that three positive swings on the period instrument were observed approximately 5 or 10 sec before the reactor was scrammed. Although the power had increased by ~1 Mw during the minute just before this, the reactor was not on a short period during that time.

(2) Calculations

Four different AIREK calculations were undertaken in order to examine the nature of this excursion and to obtain an estimate for the amount of reactivity needed to cause it. The time 1823 was arbitrarily chosen as the starting point and the total additional reactivity introduced into each calculation was 0.3%. The results are plotted in Figure 4. The 10-sec ramp case reached a minimum period of 8.2 sec, 10 sec after 1823, while the 120-sec ramp reached a minimum period of 54 sec, 56 sec after 1823. The minimum period given by the quadratic was 41 sec and it occurred at 1825. The step did not show the usual asymptotic period which a step reactivity produces since other reactivity changes and shim rod motion were being continually introduced into the AIREK calculation.

These results indicate that a total of nearly 0.3% reactivity was introduced very rapidly in order to cause the observed 7.5-sec period. It may have been introduced either as several closely spaced steps or as a steep function of time, lasting no longer than ~ 5 sec. This is consistent with the observations reported by the operators.

(3) Mechanism for the Reactivity Change

The question now arises as to the cause of the large reactivity insertion of nearly 0.3% which produced the 7.5-sec period. In this connection, it is helpful to refer to Table VII which is a revision of Table IV-D-1 of reference 1. Various mechanisms for reactivity changes are tabulated together with an estimate of the magnitude and speed of the reactivity change. An examination of these mechanisms has resulted in the conclusion that the most likely mechanism for inserting 0.3% reactivity within a time interval of ~ 10 sec or less is the creation of void in ~ 10 fuel channels. The calculations which provide an estimate of the reactivity worth of a void in one fuel channel are given in the Appendix.

TABLE VII

MECHANISMS OF REACTIVITY CHANGES

Mechanism	$\Delta\rho$ (%)	Action Rate
Doppler Effect (800°F temperature rise in 1 fuel cluster)	-0.03 to -0.04	fast
Drop half of 1 fuel cluster	-0.03 to -0.08	fast
Drop 1 entire fuel cluster	-0.005 to -0.014	fast
Accumulation of hydrogenous substance		
Among rods in 1 fuel cluster	0 to -0.02	
Complete plugging of 1 fuel channel	0.007 to 0.014	
Around 1 moderator can	0 to 0.15	probably slow
Sodium flooding 1 moderator can	0 to -0.7	slow
Moderator temperature coefficient, 180°F temperature rise in		
Entire graphite stack	0.30	slow
One moderator can	0 to 0.01	slow
Small gas bubbles in circulating sodium (5% of the sodium volume)	0.3	slow
Volume increase in a gas bubble trapped between moderator cans (bubble size equal to Na volume around 1 can)	0 to 0.2	probably slow
Voiding one fuel channel	0.03	fast
Rain of fuel slugs from top part of a parted fuel element onto the bottom part	-0.08 to 0.1	fast
Accumulation of hydrogen atoms in core	0 to 0.2	slow

IV. FUEL CHANNEL TEMPERATURES AND SODIUM FLOW RATE

Before discussing ways in which a void could be introduced into a fuel channel, it will be helpful to examine first the probable sodium flow and temperature in plugged fuel channels. An estimate of the extent of plugging which probably occurred will also be helpful.

A. EVALUATION OF THE SODIUM FLOW RATE IN FUEL CHANNELS

The fuel channel exit temperatures can be used to detect abnormal sodium flow in any channel. Under normal operating conditions, the sodium flow is not the same in all channels because the orifice plate settings are not the same for all channels. The radial power distribution across the reactor is not flat and the orifice plates are adjusted so that the temperature drop across all fuel channels is nearly the same. Define the nominal flow rate in each channel to be that flow which causes all fuel channel exit temperatures to be the same. Any deviation in sodium flow through a channel from its nominal value must produce a change in the channel's exit temperature.

It is reasonable to assume that the flow rate is inversely proportional to the channel exit temperature except at extremely low flow rates. If flow in a channel is nearly zero, the channel exit thermocouple will reflect the temperature of the upper plenum because of its location which is ~ 16 in. above the top of the fuel. If the fractional flow ψ in each channel is defined as the ratio of actual flow to nominal flow, then, except for extremely low flow rates,

$$\psi_i = \frac{T_{\text{average exit}} - T_{\text{inlet}}}{T_{i\,\text{exit}} - T_{\text{inlet}}} = \frac{\Delta T_{\text{average}}}{\Delta T_i}$$

For a given reactor power and total flow rate, a plot of ψ_i vs $\Delta T_i = T_{i\,\text{exit}} - T_{\text{inlet}}$ for a typical channel is shown in Figure 5. As ψ_i approaches zero, ΔT_i must approach $\Delta T_{\text{average}}$ because the thermocouple then reflects the temperature of the upper plenum and the above equation is no longer valid. It is not known precisely where the peak of the curve occurs, but it is believed that it occurs at a value of ψ_i of ~ 0.05.

It is also reasonable to assume that the total flow through all fuel channels is independent of the extent of plugging in any one of them. This is because the

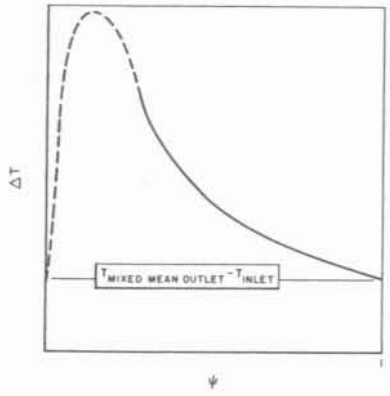

Figure 5. Typical Plot of Fuel Channel ΔT vs Fractional Flow Rate ψ

pressure drop through the core is a small fraction of the total pressure drop in the primary loop. With this assumption we can say that the sum of the fractional flows ψ_i is equal to the number of fuel channels:

$$\sum_{i=1}^{43} \psi_i = 43 \quad .$$

Therefore, if all flow rates are large enough to be correctly indicated by the fuel channel exit thermocouples, the sum of the 43 quantities $\Delta T_{average}/\Delta T_i$ should be 43. On the other hand, if several of the flow rates are nearly zero so that the fractional flow in them is on the left side of the peak of the curve shown in Figure 5, this sum will be greater than 43. This follows because the contribution of the zero flow channels will be assigned too higha value in the above sum.

The calculations described above were made for three different sets of reactor data.[5] The results are shown in Table III, and show agreement with the predicted value of 43 to within ±3%. This is quite good agreement considering the accuracy of the thermocouple data upon which the calculation was based. However, the accuracy of this result is not sufficient to prove that none of the flow rates was incorrectly represented by the channel exit thermocouple readings. In fact, as indicated in Table III, three of the thermocouples were reading below reactor inlet temperature and are obviously in error. Nevertheless, it does indicate that only a few, probably no more than about 4 or 5 of the fuel channels, could have had flow rate lower than about 10% nominal flow.

Table III shows that the channels which contained damaged fuel elements all had a severely restricted flow rate. The most severe restriction indicated was in channel 10 where the flow rate was about one-third nominal flow. However, the accuracy with this method of determining flow rates in severely plugged channels is not very good because the exit temperature for these channels was fluctuating by many degrees between points printed on the instrument chart. A point is printed once very 5 min for a given channel, and some of the fluctuations between successive points were as much as 100°F. Table VIII shows the fractional flow rates for seven fuel channels from temperatures recorded on July 13,1959

just prior to the power excursion. Included are the corresponding values taken from Table III for comparison which shows that values obtained before and after the excursion are in reasonably good agreement. An examination of this table gives an indication of the uncertainty in the calculation of individual flow rates by this method. However, it appears unlikely that many of the flow rates were much less than one-third their nominal value.

TABLE VIII

SRE FUEL CHANNEL FLOW RATES

Core Channel Number	ψ = Percent of Nominal Flow				
	July 13, 1728	July 13, 1823	July 14, 0900	July 15, 0400	July 22, 2200
10	42	39	31	35	34
24	48	52	45	39	48
25	37	41	38	33	46
31	42	44	50	43	48
35	42	47	48	41	49
54	200	200	150	170	140
68	45	50	38	37	40

B. EXTENT OF PLUGS IN FUEL CHANNELS

A calculation was made to relate the size of a plug in a fuel channel to the restriction in sodium flow through the channel. The plug length and cross sectional area was varied and the fractional flow and sodium exit temperature for steady-state conditions calculated. Figure 6 shows the results.[6] It is seen that in order to reduce the flow to one-third normal, the fraction of the flow area plugged would be from 73% to 95%, depending on the length of the plug.

Examination in the hot cell of the moderator assembly containing fuel channel R-24 showed that the plugging in this channel was indeed severe.[7] Large plugs were found at two locations within the fuel channel and there was evidence that there had been plugs at other locations as well. A region of Fe-U alloy melt-through was found approximately one-third the way up the fuel element, and above this location the fuel cladding had burst. The burst cladding is attributed to thermal cycling through the uranium α-β phase transformation temperature of 1220°F.

These observations show that abnormally high temperatures were reached within this fuel channel. Extensive plugging is required in order to produce such high temperatures.

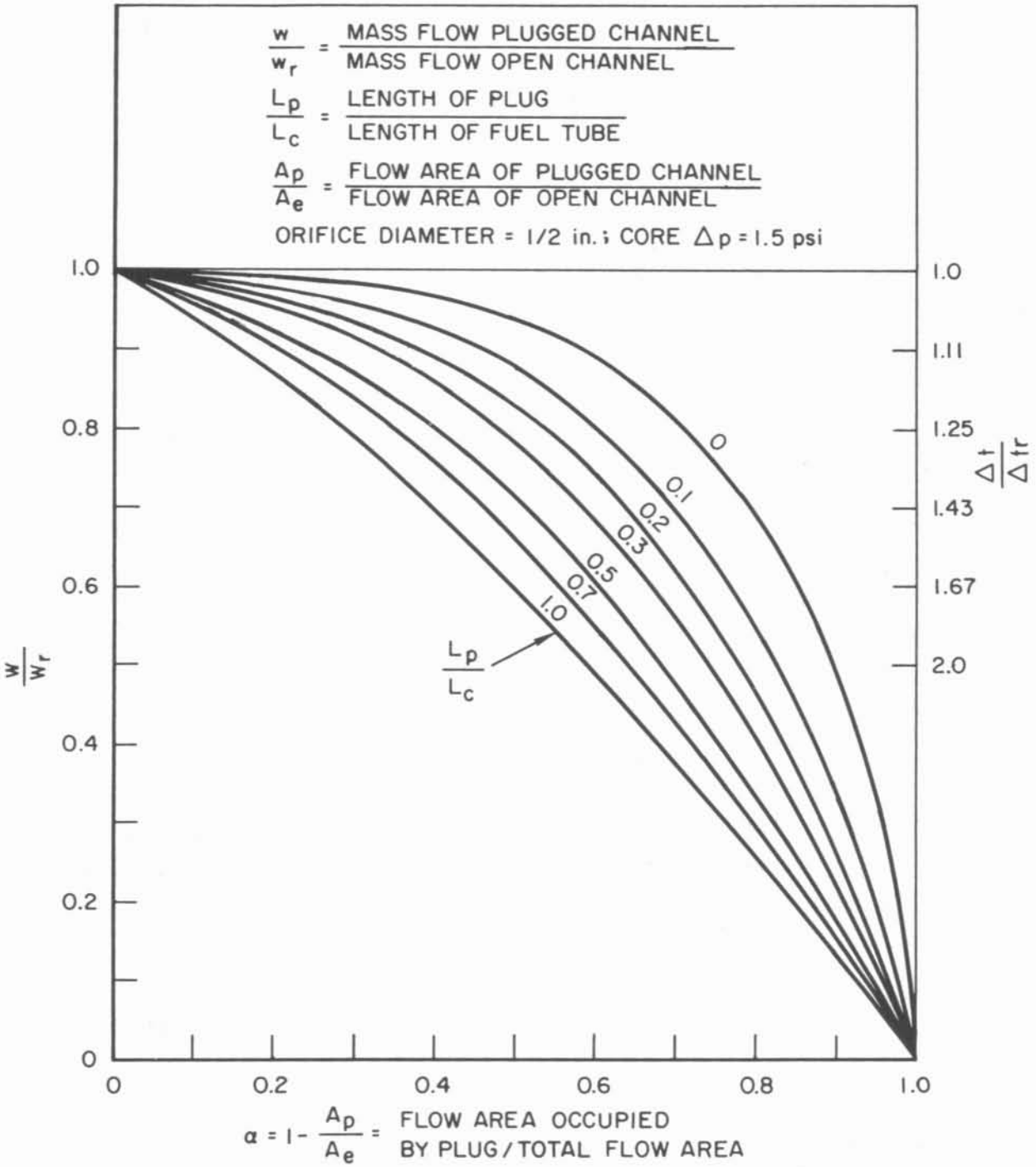

Figure 6. Mass Flow Reduction in SRE Fuel Channel Due to Formation of Plugs in Fuel-Tube Region

This plugging would easily restrict the flow of sodium to about half its nominal value, which is the flow indicated by the fuel channel exit thermocouple reading for this channel. Moreover, with this flow rate the design temperature limits for the fuel element at the power levels of run 14 (~ 2 Mw for most of the run) could not have been exceeded unless the plugs interfered locally with heat transfer. To reach the fuel temperatures observed, it was necessary for a plug to have occupied the region of failure at the time of failure.

C. ESTIMATE OF FUEL CHANNEL SODIUM EXIT TEMPERATURES

1. Method

The method used to estimate the fuel channel sodium temperature at the top of a fuel element before and during the excursion will next be described. Since data for a detailed calculation are not available, an estimate of sodium temperature rise vs power increase was made by considering the energy release during a 2-min time interval prior to the scram and the temperature changes this energy would produce in the fuel and sodium.

Assume that the power level at 1823 will maintain the fuel and sodium temperatures that were observed at that time. The excess power above this level after 1823 will produce temperature increases. The excess energy release from 1823 to time t is

$$Q = \int_{1823}^{t} \left[P(t) - P(1823) \right] dt \quad .$$

This integral was evaluated numerically from the curve of relative power shown in Figure 3. The excess energy was equated to the sum of the products of the mass, specific heat, and average temperature rise of each material whose temperature changed. The energy balance equation used was as follows:

$$Q = r\, C_{Na} \Delta t \overline{\Delta T}_{\text{Na discharge}} + (MC\overline{\Delta T})_{\text{fuel}} + (MC\overline{\Delta T})_{\text{Na in channel}} + (MC\overline{\Delta T})_{i} \quad ,$$

where

r = sodium flow rate

Δt = time interval from 1823

$\overline{\Delta T}$ = average temperature change during time t of the material contained in an entire fuel channel

M = mass of material whose temperature is changed by the amount $\overline{\Delta T}$.

C = specific heat.

The above procedure assumes that any foreign material displacing sodium in the fuel channel has the same value of MC as does sodium.

The last term (subscript i) represents the hardware, graphite, etc., whose temperature is changed along with that of the sodium.

The choice of the values to use for the various $\overline{\Delta T}$ appearing in the above expression is difficult to make. The quantity we desire to calculate is ΔT_{exit} as a function of time. If the above $\overline{\Delta T}$ can be expressed in terms of ΔT_{exit}, the calculation can be carried out. The choices made below to relate the $\overline{\Delta T}$ to ΔT_{exit} are believed to be reasonable. They are based on our knowledge of the temperature profile for an unplugged fuel channel. Average values for the temperature rise in each material were estimated using this as a basis.

$$\overline{\Delta T}_{fuel} = 3/4 \Delta T_{exit}$$

$$\overline{\Delta T}_{Na\ in\ channel} = 2/3 \Delta T_{exit}$$

$$\overline{\Delta T}_{Na\ discharge} = 1/2 \frac{\Delta T_{1\ exit} \Delta t_1 + (\Delta T_{2\ exit} + \Delta T_{1\ exit}) \Delta t_2 + \ldots}{\Delta t_1 + \Delta t_2 + \ldots}$$

$$\overline{\Delta T}_i = 2/3 \Delta T_{exit} \text{ at t-20 sec.}$$

An arbitrary 20-sec time delay was inserted into the last expression to allow for a time lag in heating up portions of the hardware associated with a fuel element. In the next to the last expression, $\Delta T_{1\ exit}$ represents the value of ΔT_{exit} after the time interval Δt_1, etc. The right hand member is an approximation for the average increase in the exit sodium temperature over the time interval $\Delta t_1 + \Delta t_2 + \ldots$. ΔT_{exit} represents the last member of the series $\Delta T_{1\ exit}$, $\Delta T_{2\ exit}, \ldots$

When these expressions for $\overline{\Delta T}$ are introduced into the energy balance equation above, the ΔT_{exit} can be calculated by a stepwise process. First a time interval Δt_1 is chosen, Q_1 is obtained for that interval and $\Delta T_{1\ exit}$ is calculated. Next an additional time interval Δt_2 is chosen, Q_2 obtained for that interval and added to the Q_1 already obtained for Δt_1. Since $\Delta T_{1\ exit}$ has already

been found, we can now calculate $\Delta T_{2\ exit}$. A third time interval Δt_3 is chosen and the process repeated. This stepwise procedure is continued until the entire time interval has been covered.

None of the terms appearing in the preceding energy balance equation are known very accurately. Consequently, the results obtained are subject to a large uncertainty. However, they indicate the magnitude of the temperature changes that may have occurred.

The following data were used in the above calculations:

r = 1070 gm/sec-channel (850 gal/min total sodium flow)

C_{Na} = 0.31 cal/°C-gm = 0.17 cal/°F-gm

C_{fuel} = 0.04 cal/°C-gm = 0.022 cal/°F-gm

M_{fuel} = 69.4 kg/cluster

M_{Na} = 2.800 kg/channel

$P(1823) = 2.3 \times (1.6 \times 10^6)/(43)$ watts/channel

$(MC)_i$ = 730 cal/°C (assumed) = 406 cal/°F

2. Temperature Rise for a Severely Plugged Channel

The resulting ΔT_{exit} as a function of time and fractional flow rate are given in Table IX. It is seen that quite large temperature rises can occur for extremely small flow rates. However, if the flow rate is as much as 30%, it seems very unlikely that the exit sodium temperature reached the boiling point during the excursion. On the other hand, if flow rate is as low as about 5%, it is possible that the boiling point of sodium may have been reached before the scram occurred.

TABLE IX

CALCULATED FUEL CHANNEL EXIT TEMPERATURE RISE ABOVE TEMPERATURE EXISTING AT 18:23

Fractional Flow Rate	Time (hr:min:sec)			
	18:24:20 (°F)	18:24:40 (°F)	18:24:52 (°F)	18:25:00 (°F)
1.2	23	56	111	163
0.2	78	168	306	440
0.05	119	246	434	628
ΔT_{54} (observed)	~20	---	---	90

3. Temperature Rise for an Unplugged Channel

The calculated ΔT_{exit} at 1825 for a typical unplugged channel is substantially higher than that observed for channel 54, which is believed to be unplugged. The fractional flow rate in a typical unplugged channel was assumed to be 1.2 instead of 1.0 because of the reduced flow through the plugged channels which constituted about one-fourth of the core. However, the fractional flow through channel 54 is nearly 2.0. Also the channel exit thermocouple, being ~16 in. above the top of the fuel, will lag behind the temperature of the sodium at the top of the fuel during the fast portion of the excursion. The time delay results from the thermal inertia of the thermocouple assembly (~5 sec time constant), the coolant transport delay between fuel and thermocouple (1/2 sec), and the cooling of the fuel channel sodium by the graphite and sodium in the top reflector region. It is believed that these facts account for the discrepancy between the calculated and observed temperature rise for channel 54 during the fast part of the excursion.

D. PROBABLE LOCAL TEMPERATURE WITHIN SEVERELY PLUGGED FUEL CHANNELS

Calculations have been made which show that severe plugging in a fuel channel can lead to quite high local temperatures in the fuel due to the thermal insulation provided by the plug.[6] It was found that at a reactor power of 2 Mw the temperature of the hot spot on the surface of a fuel rod that is insulated over 25% of its surface can be about 180°F above the local sodium temperature.

Another point which supports the belief that local temperatures within severely plugged fuel channels were considerably higher than temperatures indicated by their exit thermocouples is the following. In reference 1 it was stated that the initial instance of fuel damage probably occurred at about 1500 hours on July 12. This was the time at which high levels of radioactivity in the high bay area were first noted. Reactor power had been raised from ~2 Mw to nearly 4 Mw just prior to the observation of excessive radioactivity, and this was the highest power level achieved up to that time during run 14. Fuel channel exit temperatures were behaving in the same manner as they did just before the power excursion on July 13. If fuel damage occurred at 1500 on July 12, temperatures inside the fuel channels must have been very much higher than the values of about 750°F that were being registered by the exit temperature thermocouples on the hottest of the fuel channels. The same situation undoubtedly existed on July 13 just prior to the excursion.

Further evidence of extremely high local temperatures within severely plugged fuel channels is obtained by inspecting the record for core channel 55. (See Figures 7 and 8.) During the last part of run 14, fuel meat temperatures in this element were being recorded on an instrument located in the high bay area (1).

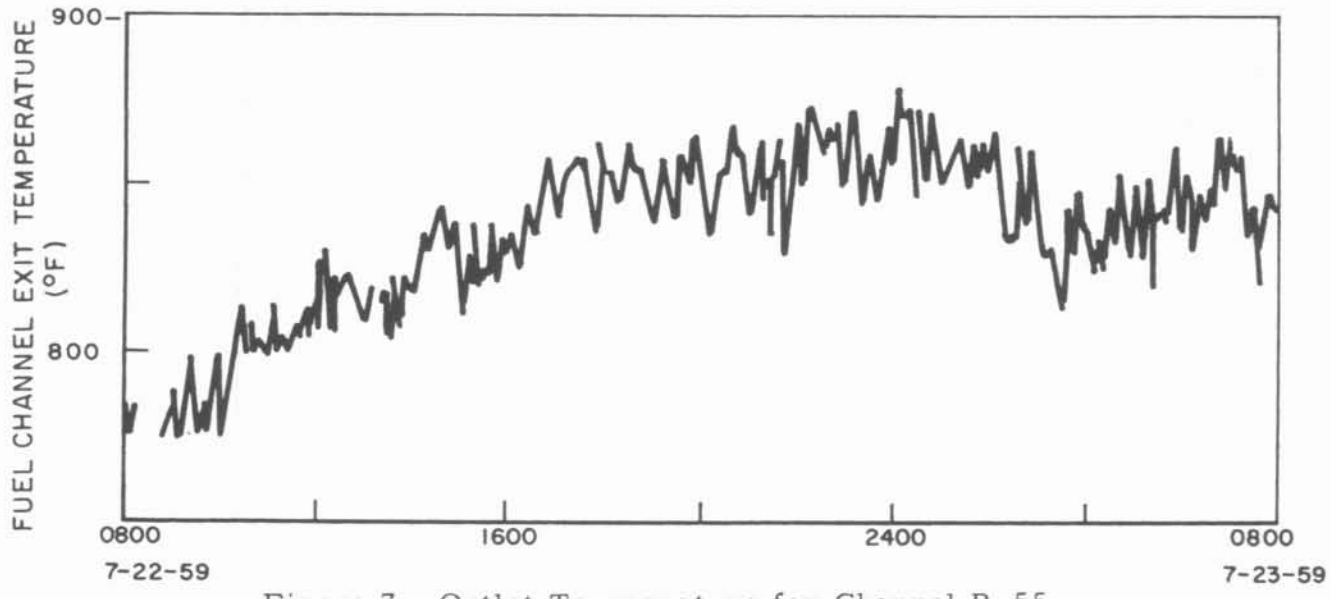

Figure 7. Outlet Temperature for Channel R-55

Channel 55 contained an experimental fuel element and this temperature recording instrument was not part of the reactor operating instrumentation, and the chart was not examined until after the reactor had been shut down. On July 22, fuel temperatures as high as 1400°F were recorded while the fuel channel exit temperature registered about 850°F. Reactor inlet temperature at that time had been increased to 700°F. The flow in channel 55 was a little over 50% of its nominal flow according to the fuel channel exit thermocouple. Reactor power was at ~4 Mw. This shows that local temperatures within channel 55 were exceeding the temperature indicated by the exit thermocouple by as much as 550°F even with 50% flow in this fuel channel.

The concept of local hot spots within severely plugged fuel channels throws some light on the nature of the plugs. In order to reach temperatures above 1400°F a substantial degree of insulation from the flowing sodium was required. Such insulation could not be obtained with a plug containing a large amount of liquid sodium. Since the plugs were probably porous, as indicated by the examination of core channel R-24, it is difficult to escape the conclusion that gas bubbles were contained in them. The gas could have been formed by decomposing

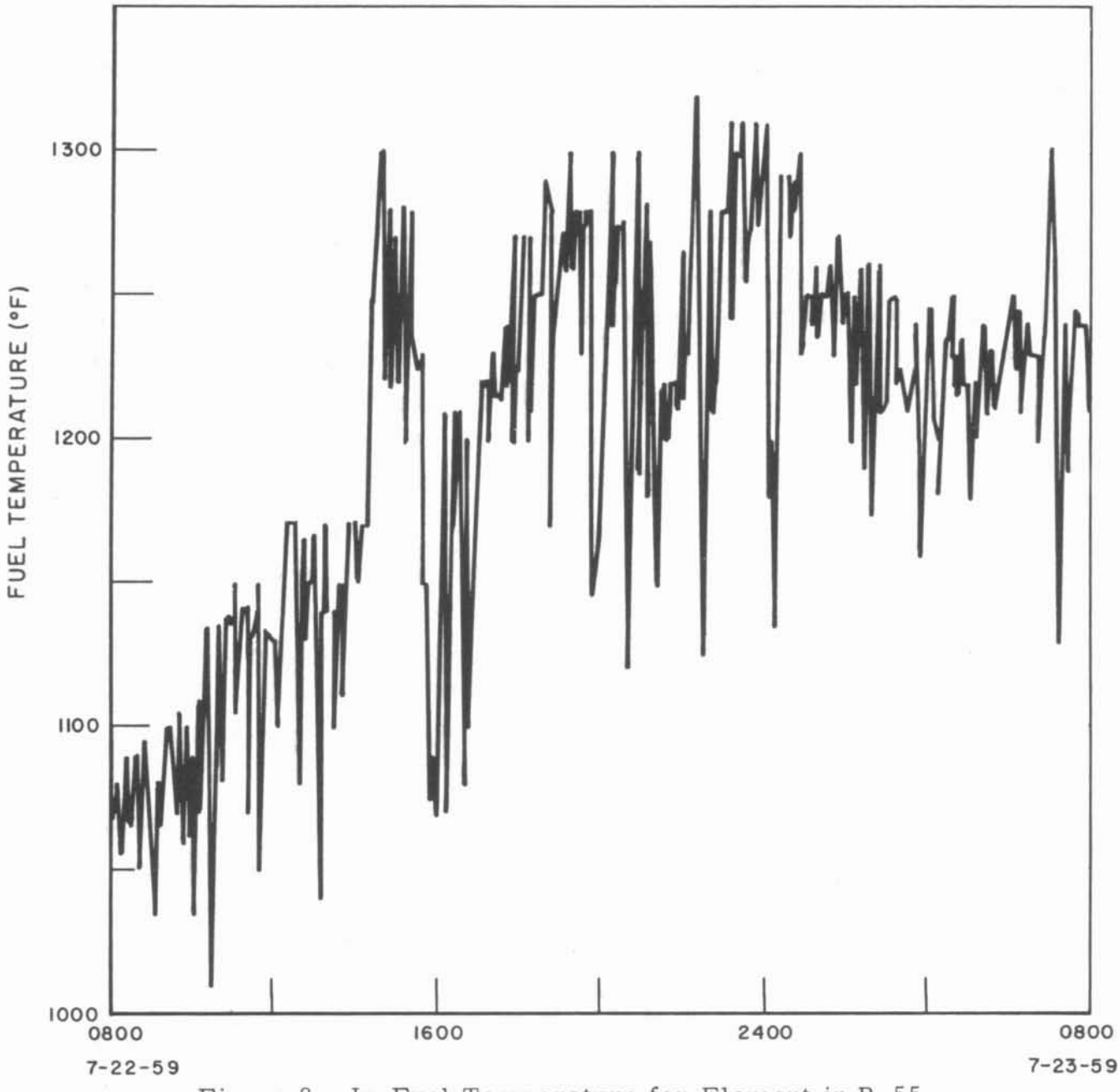

Figure 8. In-Fuel Temperature for Element in R-55

hydrocarbons, but sodium vapor seems a likely contributor. Vapor would start to form in areas of good thermal insulation as the reactor power rose. This would further increase the insulation and the formation of vapor would tend to propagate through the plug volume.

E. THERMAL CYCLING AND LOCAL BOILING OF SODIUM

The evidence showing that thermal cycling of the fuel was one of the causes of fuel element failure is given in reference 7. The behavior of fuel channel exit temperatures is consistent with this mode of failure.

A review of the fuel channel exit temperature chart from run 14 showed that all of the channels in which fuel failures occurred had shown temperature fluctuations. Fluctuations in channel exit temperatures also occurred for a few hours in some channels where fuel elements did not fail. Some indication of the relation between fuel temperatures at some regions of the element and the channel exit temperatures can be gained from the data for R-55 shown in Figures 7 and 8. As indicated by the figures, the fuel temperature was cycling over a range of about 150°F while the channel exit temperature variation was about 50°F.

Other information on the cyclical variation of fuel channel exit temperatures was obtained from a temperature record taken for a few hours during run 14 using a continuous recording strip chart instrument. This record shows a cyclical variation in the exit temperatures for core channels 10, 19, and 65. The size of the oscillations shown for channel 10 is over 100°F and the size of those for channels 19 and 65 is somewhat less than 100°F. The period of the oscillations shown on this chart is between 1 and 2 min.

The cycling of the channel exit temperature demonstrates a corresponding variation in sodium flow through the channel. With the type of plug described in previous paragraphs, the variation in flow can readily be ascribed to boiling of sodium within the plug. Similar oscillations have been observed in an experimental sodium boiling – condensing apparatus at Atomics International.

The local boiling postulate is supported by evidence that local temperatures in severely plugged channels could have reached the boiling point of sodium. The local boiling model also provides a logical explanation of reactivity changes which occurred during the positive and negative excursions. Whatever the mechanism, it is clear that thermal cycling occurred and with sufficient severity to cause cladding failure by expansion of the fuel.

V. MECHANISMS FOR THE INTRODUCTION OF VOID IN FUEL CHANNELS

A. VAPOR FORMATION DUE TO LOCAL OVERHEATING

It has been shown that local temperatures within severely plugged fuel channels reached values in excess of 1400°F. Some of these temperatures were probably high enough to cause local boiling of the sodium trapped in the plugs, thereby creating gas bubbles within the plugs which further interfered with the heat transfer from the plugged region. The discovery of thermal cycling on the fuel in some channels has produced a picture of cyclical pulsing of sodium vapor within channels.

The mechanism believed to be the most likely cause of the void introduction and consequent 7.5-sec reactor period will next be described. As the reactor power rose, the evolution of gas within plugged channels increased. This caused an increase in the size of the vapor pulses. The vapor pulses introduced reactivity changes which at power levels below ~4 or 5 Mw were not large enough to influence the cycling in other channels. However, ~10 sec before the reactor was scrammed, the power had increased to ~5 Mw and was still rising. At this higher power level the vapor volume was larger than before and it appears plausible that the effect on reactivity was of sufficient size to produce a significant additional increase in power. Finally, the effect became large enough so that there was interaction among the cycling channels leading to a cascading of the vapor pulses. The expulsion of the sodium and probably some of the plugging material from severely plugged fuel channels followed. The resulting void in a channel would introduce a positive reactivity step of ~0.03% for each channel voided. About 13 fuel channels were effected and in this manner the required 0.3% positive reactivity necessary to cause the 7.5-sec period was introduced.

B. VAPOR CHOKING OF SODIUM FLOW

Vapor choking is another mechanism that has been considered as a means of introducing void in plugged fuel channels.

1. Description of Vapor Choking

Under conditions of stable flow in a fuel channel, the mass flow rate increases with an increase in pressure drop across the channel. Under conditions of unstable flow the reverse is true. At the onset of boiling the flow is stable if

the power is below about twice full power. However, buoyancy and frictional effects become very important to the nature of the flow under boiling conditions. It has been shown[8] that unstable boiling will occur in a channel under certain circumstances. In an unplugged channel, boiling at low flow rates is stable over a wide range of power. However, in the event of severe plugging, depending on the location of the plug, boiling may be unstable because of the considerable extra flow resistance introduced by the plug. When unstable boiling occurs, the mass flow rate decreases abruptly to a very small fraction of its original value. Pressure effects in the channel cause most of the sodium to be expelled suddenly, creating a large amount of void in the channel. This phenomenon is called vapor choking, and must be distinguished from plugging which is the accumulation of foreign material in a fuel channel.

2. Propagation of Vapor Choking

When the coolant flow in a fuel channel chokes, most of the sodium is suddenly expelled from it, thereby introducing a sudden reactivity increase of ~0.03%. This causes a prompt jump in the power level of ~4% with a prompt period of ~0.07 sec. Choking in one channel will also cause an increase of about 1/43 in the flow rate in other channels which tends to suppress choking in other channels. However, if the prompt power jump is a little greater than about 1/43, the choking of one channel can initiate choking in another channel which is on the verge of choking. If the reactor is also on a positive stable period, this slower rise in power also contributes to the probability that additional channels will choke. In this manner, choking can be propagated until several of the hot channels have choked.

In order for vapor choking to be the sole cause of the 7.5-sec reactor period, it would be necessary for about 10 fuel channels to choke. In view of the sodium flow and temperature studies presented earlier, it seems very unlikely that choking could have occurred in this many channels. In fact, if the fuel channel exit temperatures give reliable indications of flow rate in all channels, it is inconceivable that choking could have occurred in any channels. However, because of the uncertainties associated with the data used in the flow and temperature studies, there is the possibility that choking may have occurred in not more than about four fuel channels.

3. Doppler Effect as a Restraining Mechanism

The objection may be raised to the above hypothesis that the fuel temperature in a voided channel will rise rapidly, thereby introducing a negative reactivity change due to the Doppler effect, and that this would tend to restrain the excursion. The increase in fuel temperature after voiding occurs is proportional to the energy release, since there is then very little cooling. An estimate of the change in the temperature of the fuel in such a channel shows that a maximum ΔT over a time interval of 10 sec after voiding is $\sim 300°F$. This neglects any cooling of the fuel. The vaporization of any residual sodium in the channel will absorb energy from the fuel and restrain its temperature rise. Also, at temperatures where voiding occurs the fuel will lose heat by radiation which will further restrain its temperature rise. A temperature rise of $\sim 400°F$ in one fuel cluster is needed to introduce a reactivity of -0.01%. On the other hand, the voiding of one channel introduces reactivity of about 0.03%. It is therefore clear that the negative reactivity introduced by the Doppler effect is too small to restrain an excursion which is initiated by voiding a fuel channel. In any case, the negative reactivity which the Doppler effect introduces would appear after voiding has occurred.

VI. OTHER SUGGESTED CAUSES OF THE 7.5-SECOND PERIOD

In addition to the creation of void in fuel channels, several other mechanisms have been considered as possible causes for the 7.5-sec period. Although these mechanisms cannot be definitely excluded as possible causes, each of them meets with some serious objection which renders it relatively improbable. It is, of course, possible that some combination of these effects and the voiding occurred simultaneously. There does not seem to be any way to evaluate this possibility.

A. GAS BUBBLES

An alternate mechanism that has been suggested to account for reactivity changes is the presence of gas other than sodium vapor. The movement and growth of gas bubbles could easily account for all three of the reactivity changes noted. Thus, the introduction of a bubble into the core either in fuel channels or between moderator cans would introduce positive reactivity. If such a bubble were then to move out of the core, it could introduce the negative reactivity required to cause the negative excursion. A re-accumulation of the gas in the core would then add reactivity so that the reactor could again be made critical with a relatively small withdrawal of control rod. The positive excursion could have been caused by the movement of large bubbles up through the reactor core. The transit time for sodium to flow through the fuel channels is on the order of 2 sec. This time is consistent with the required rate of insertion of the 0.3% reactivity needed to cause the 7.5-sec reactor period.

Evidence to support the existence of such bubbles is the pressure coefficient data taken during run 14, and it is known that gas bubbles have occasionally been inadvertently introduced into the primary sodium system in the past and their effect on reactivity observed.

It is not obvious how bubbles of gas other than sodium vapor could have produced the cyclical variation in the outlet temperature of the partially plugged fuel channels. The explanation of observed anomalies on the basis of local boiling is therefore deemed more plausible.

B. HYDROGENOUS MATERIAL

The sudden accumulation of a substantial amount of hydrogenous material in the core, or the sudden movement of such material from a region of low importance to a region of high importance was considered as a possible mechanism for

causing the 7.5-sec period. The objection to this mechanism is that a large quantity of material is required. A volume at least equal to the sodium volume around two moderator cans would have to be filled with hydrogenous material in order to introduce 0.3% reactivity. It is difficult to imagine how such a large quantity could be created or relocated in just a few seconds time.

C. FUEL SLUG RAIN

The dropping of the bottom half of several fuel elements, one at a time, followed by the falling of the fuel slugs remaining in the upper portion of several fuel clusters down onto the broken portion was also considered. A reactivity loss of approximately 0.3% would accompany the dropping of the bottom halves of about 10 fuel elements. This loss would have to occur in order to make it possible to introduce 0.3% reactivity by the "fuel slug rain" mechanism. Moreover, the "rain" of the fuel slugs in the upper half of the fuel elements would have to involve all of the remaining slugs in the parted fuel elements in order to produce a reactivity increase of 0.3%. This fuel "rain" would also have to take place in not more than ~10 sec in order to produce a 7.5-sec reactor period. Since so many fuel clusters and such a short period of time for the "rain" is involved, it seems very unlikely that this mechanism could have caused the excursion. Also, no evidence has been found that fuel slugs actually dropped.

VII. CONCLUSION

A study was made of the reactor behavior over a period of 1 hour prior to the excursion which occurred on July 13, 1959. The results and conclusions are summarized below.

Figure 2 shows a comparison between the actual relative reactor power observed and that which was computed by taking control rod motion, fuel, and moderator temperature coefficients into account. The general features of the power trace up to the time of the power excursion can be reproduced in this way. Figure 3 shows a calculation in which additional reactivities are inserted at various times. In this way, good agreement between the actual power trace and the computed power is obtained. The minor fluctuations in the observed power curve could easily be accounted for by fluctuations in fuel or moderator temperature, or by fluctuations in the volume of vapor trapped in severely plugged fuel channels.

The solution of the reactor kinetics equation shows that all but three of the changes in power level that occurred during the time interval considered are explainable in terms of reactivities introduced by control rod motion and by fuel and moderator temperature changes. The three exceptions are:

a) The negative excursion that occurred at 1807 hours

b) The gradual increase in power from 3.0 to 4.5 Mw between approximately 1821 and 1824 hours

c) The positive excursion with 7.5-sec period that occurred at about 1825.

The reactor kinetics calculations showed that the unexplained reactivity changes involved in these three exceptional instances had approximately the following characteristics:

a) Negative excursion: = -0.06% step followed by a recovery of most of this reactivity loss within the next few minutes.

b) Slow rise: = +0.04% in a 3-min ramp.

c) Positive excursion: = +0.3% in a 5 to 10-sec ramp.

These reactivity changes, in addition to those being caused by control rod motion and reactor temperature changes, produced a reasonably good approximation to the observed changes in power level and reactor period.

Reasonable explanations for these reactivity changes follow:

a. Negative Excursion

In section V a mechanism for formation of vapor within partially plugged channels was discussed. The evolution of vapor as the power rose introduced void in the core by the displacement of sodium. This resulted in the introduction of a small amount of positive reactivity. Fuel temperature in the immediate vicinity of the plug rose as the formation of vapor took place. Control rods were inserted in order to restrain the power increase, and this insertion in fact halted and started to reverse the rise in power. As power started to drop, the volume of vapor in fuel channels decreased, thereby introducing negative reactivity and speeding the drop in power. The collapsing bubbles allowed more coolant to enter the plugged regions, thereby rapidly cooling the overheated fuel contained therein. The reduction of fuel temperature introduced positive reactivity due to the Doppler effect, thereby reducing the rate at which power was dropping. This recovery of reactivity allowed the reactor to be made critical again with a relatively small withdrawal of control rods.

This model fits all phases of the observed reactor behavior during this period.

b. Slow Rise

A moderator temperature rise of about 25°F above that used in the kinetic studies would account for the reactivity change of 0.04%. Heat transfer to the moderator coolant during run 14 was undoubtedly very poor, so overheating of the moderator is a reasonable expectation. The large reactivity changes as a function of power level that were observed later in run 14 support this belief.

Another possible cause of the 0.04% reactivity increase is an increase in sodium vapor volume within severely plugged fuel channels as the power level was increased. If the picture of sodium vapor within severely plugged channels is correct, some expansion of vapor volume would accompany the 1.5-Mw increase in power level.

c. Positive Excursion

In order to produce a 7.5-sec reactor period, it is necessary to introduce about 0.3% reactivity in a time interval of not more than ~10 sec. A reasonable mechanism for accomplishing this is the creation of void in ~10 fuel channels.

The discovery of thermal cycling effects on the fuel in some channels produced a picture of cyclical pulsing of sodium vapor within channels. Several channels were involved, each with a cycle period of ~2 min. The cycling in each affected channel introduced reactivity changes which, at ~4 Mw of power, were not large enough to influence the cycling in the other channels. However, shortly before the excursion the reactor power level was ~5 Mw. It appears plausible that at this power level, the vapor volume was larger, and the effect on reactivity was of sufficient size to produce a significant increase in reactor power level, so that all cycling channels were affected. This action produced a combining of the reactivity contributions from several channels, resulting in an accelerating reactivity increase.

APPENDIX

REACTIVITY CHANGE DUE TO A VOID IN A FUEL CHANNEL

The expulsion of sodium from a fuel channel causes a positive reactivity change because of the removal of neutron absorption by the sodium. There will also be a small decrease in the effective resonance integral and an increase in the migration area, but these are small effects. The reactivity loss due to neutron streaming should not be large because the mean free path of neutrons in sodium is ~12.5 cm.

Calculations were made to determine the change in nuclear parameters caused by removing the sodium from a typical fuel channel. The following results were obtained:

$$\frac{\Delta f}{f} = +0.0122$$

$$\frac{\Delta p}{p} = +0.0057$$

$$\frac{\Delta L^2}{L^2} = -0.0120 \text{ due to neutron absorption change.}$$

The above results were converted into reactivity changes by means of the relation

$$k_{eff} = \frac{\eta \epsilon p f}{1 + M^2 B^2}$$

Since there are 43 fuel channels, this must also be divided by 43 in order to obtain the Δk_{eff} for a typical fuel channel. The final results are

$\Delta k_{eff} = +0.00028$ due to increase in f

$\Delta k_{eff} = +0.00013$ due to increase in p

$\Delta k_{eff} = -0.000017$ due to increase in L^2 from change in neutron absorption

$\Delta k_{eff} = +0.00040$ (total without neutron streaming).

Neutron streaming will reduce this value a little. It is estimated that a probable value for the Δk_{eff} due to voiding a typical fuel channel is about 0.03%.

REFERENCES

1. A. A. Jarrett, Editor, "SRE Fuel Element Damage, An Interim Report," NAA-SR-4488 (December 1959)

2. A. Schwartz, "Generalized Reactor Kinetics Code," NAA-SR-MEMO-4980 (March 1956)

3. J. G. Lundholm, Jr., C. W. Griffin, "Measurement of the SRE Power Coefficients and Reactor Parameters Utilizing Pile Oscillation Techniques," NAA-SR-3763 (1960)

4. R. L. Ashley, et al., "SRE Fuel Damage Final Report," NAA-SR-4488 Supplement (March 1961)

5. E. H. Spoehl, private communication

6. R. C. Noyes, private communication

7. J. L. Baliff, "Examination of Damaged SRE Fuel Elements," NAA-SR-4515 (June 1961)

8. J. H. Bick, "The Stability of Two-Phase Flow in Parallel Heated Channels," NAA-SR-4927 (May 1, 1960)