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by C.P. Caven 3, date 4-4-57

ARGONNE NATIONAL LABORATORY
P. O. Box 299
Lemont, Illinois

REACTOR ENGINEERING DIVISION QUARTERLY REPORT

June 1, 1953 through August 31, 1953

Compiled by

Members of the Reactor Engineering Division

September 15, 1953

Photostat Price \$ 21.30
Microfilm Price \$ 6.90
Available from the
Office of Technical Services
Department of Commerce
Washington 25, D. C.

Previous Reports:

ANL-5060 March 1, 1953 through May 31, 1953

ANL-5012 December 1, 1952 through February 28, 1953

ANL-4951 September 1, 1952 through November 30, 1952

Operated by The University of Chicago
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(The following information was obtained from the records of the FBI, New York Office, dated 10-26-68.)

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REACTOR DESIGN AND DEVELOPMENT

ARGONNE RESEARCH REACTOR (CP-5) - G. A. Anderson, O. W. Childs, J. J. Dickson, J. H. Dyer, L. C. Livesey, E. E. Hamer

The reactor shielding is in place, and the graphite on the outside of the aluminum tank has been completed. The aluminum tank is set and is ready for the hole liners. The 10-1/2 ft. diameter annular shield plug has been completed and is being shipped from the Rock Island Arsenal. The vertical experimental thimbles and plugs are about 90% completed in our shops. The shield plugs above the fuel assemblies are complete and ready for wiring.

The personnel and cargo air lock doors have been accepted by the Laboratory and are now in operating condition.

During the past quarter, approximately 95% of all control and instrumentation equipment was installed and checked, and about 75% of the wiring completed as follows:

Control room equipment and panel wiring is approximately 90% complete; control process and auxiliary conduit work 75% complete; interconnections checking, etc., 40% complete; and intercommunications system approximately 60% complete.

With the reactor tank in position and the main piping system complete and leak tested, the tie-in between the two is scheduled for the week of September 14. The secondary piping will be completed the same week, and testing of the systems will commence. The sight gauge proper is tentatively delayed by the vendor.

The reactor shield cooling system was operated and is now complete except for the tie-in to the reactor annular shield and lower plug, not yet in place.

The chilled water tank is ready for the refrigeration operation test; pre-operational tests have been completed on the cooling tower system.

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POWER BREEDER REACTOR (PBR)

A. Reactor Design and Evaluation - L. J. Koch

A number of schemes for a high-temperature, liquid metal cooled, fast reactor have been studied. The studies and completed experimental work have established certain fundamental criteria for the design of this type of reactor as follows:

1. Design Criteria

a. The objective is the economical production of power from a nuclear reactor operating on fast neutrons, but the design must be flexible enough so that it can be inexpensively converted to the production of weapons-grade plutonium as its primary function.

b. The construction and operating costs must be a principal factor of design along with function, safety, and reliability. Every effort is to be made to demonstrate competitive position relative to existing types of power plants. The design of fuel and blanket elements, in particular, must be compatible with the requirement of fast, inexpensive processing and refabricating of these elements while the metal is highly radioactive. This is the only way that fissionable material inventory can be kept to a minimum.

c. There should be no external pipes carrying radioactive sodium. The entire primary cooling system is to be contained in a double-walled tank. The space between the walls is to be partially evacuated to serve as an insulating zone and a leak detector.

d. No seals shall be used below the free liquid level of sodium in the tank.

e. All mechanisms, bearings, etc., that may require maintenance must be readily accessible outside of the shield.

f. Expensive and complicated coffins are to be avoided.

g. Shutdown cooling must not require a pump; natural convection of liquid sodium is to be utilized.

h. The heat sink (in an emergency case where it is impossible to absorb the heat in the normal production of power) must be infallible.

i. The design must be such that presently available structural materials will give satisfactory service. This particularly concerns high temperature creep, thermal stresses, warpage, and change of properties under neutron bombardment. Of course, much experimental data on certain materials need to be gathered to make this possible.

j. The entire sodium system is to be sealed off from the air during operation and while loading and unloading is being accomplished.

k. The loading and unloading scheme must be such that the processing of spent fuel assemblies can be done very soon or as long after shutdown as is required, i.e., there is to be a clean break between chemical and metallurgical processing design and reactor design. The reactor design must proceed without final information as to reprocessing of fuel and blanket elements.

l. The design of the prototype power breeder reactor must be such that the same schemes are readily usable in a full scale central power station design.

Thought has been given to many schemes and arrangements. Five of these have been the subject of brief studies. Three of the concepts show promise of fulfilling the criteria itemized above, and these are being studied further. Layouts are being made of major components, and specific problems are being analyzed. A description of the schemes studied is given below.

2. Conceptual Design Studies

a. One design consists of a large, double-walled tank filled with liquid sodium and contains the primary pump and heat exchanger, as well as a large rotating cylinder which is below the free liquid level of the sodium. The reactor core and reflector elements stand in a vertical position in one-half of the cylinder. The control rods are vertical with their mechanisms mounted on the fixed shield above the place where the core rests while in running position. These rods uncouple at the top of the core when the reactor is shut down ready for a loading and unloading operation. The cylinder is rotated 180° by turning the shaft which suspends the large assembly in the sodium. The suspension bearing and drive for this large shaft is mounted on top of the fixed shield where sodium vapor will not reach it to give trouble that might be encountered if the bearing was located where sodium could condense. When the reactor is 180° from its running position, it is in a pit under one end of a chemical and metallurgical process cell. The upper end of the pit is in the floor of the cell, and the lower end terminates in the sodium tank at the top of the rotating cylinder. A hinged seal plug at the top of the pit opens to make the reactor accessible to the remotely operated crane. This crane loads and unloads the reactor and also serves the process operations in the helium filled cell where the fission products will be removed from irradiated fuel and then refabricated into new elements to go back into the reactor. One or more fuel or reflector blanket elements can be removed from the reactor and placed in a rack at the side of the pit without leaving the sodium. Fresh elements are placed in the reactor and the cylinder is rotated 180° back into running position.

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raise it to a point where the knob at the top of the element is just below the bottom of the shield plug. In the next step in the operation an arm, which extends horizontally from a shaft installed in the fixed shield, swings around and couples to the element. The lift rod is uncoupled and the arm swings the element away from the place where it was hanging above the reactor to a position above a storage pit. There are two horizontal arms mounted on vertical shafts installed in the stationary plug over the storage pit. Each of these arms are double-ended, thus allowing two radii for storage positions below these arms. The plug has four access holes for element removal into a coffin after the residual radioactivity has subsided to a value where liquid cooling is no longer required. This could, of course, be converted to the process pit scheme and eliminate the coffin and cooling time requirement.

In this scheme the control rods would be actuated from above the top shield by means of an off-set shaft which extends down outside of the blanket to a point below the reactor where it is provided with a horizontal arm which couples to the bottom of the vertical control rods in the core. It should be mentioned that all of the schemes studied propose to control by moving fissionable material.

d. An arrangement was studied where the axis of the fuel and reflector assemblies are horizontal, a loading and unloading plug makes up one vertical face of the reactor shield, the control rods are horizontal and their mechanisms are located on the vertical shield face opposite the loading and unloading mechanism, and the primary heat exchanger and pump are located in the tank below the lattice.

e. An effort was made on a conception involving small uranium balls mixed with a bonding sodium which was pumped slowly into the reactor vessel. The cooling sodium circulated through tubes running through the vessel. The effluent uranium balls would move out only as fast as required by the reprocessing cycle.

B. Reactor Physics and Critical Assembly Work

1. Multigroup Diffusion Theory Analyses - W. E. Drummond, H. Greenspan, D. Okrent

Several spherical-geometry multigroup problems pertaining to PBR design have been solved to a sufficient degree of convergence on the Argonne IBM. A partial analysis of the results is presented.

Six groups were used in these particular solutions, and the breakdown of the neutron energy spectrum into groups was determined in such a way as to throw light in particular on blanket design and plutonium quality. Hence, only three "fast" groups were used, plus two "resonance" and a thermal group. The exact energy breakdown and the cross sections used are presented in Table I.

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Table I

IBM SPECTRUM*

Group		Energy Interval		Fraction Fission Neutrons Born Into			
1		∞ - 1.6 mev		0.5			
2		1.6 - 0.4 mev		0.42			
3		0.4 - 0.05 mev		0.08			
4		0.05 - 280 ev		0.0			
5		280 - 1 ev		0.0			
6		thermal		0.0			

Material		C	Na	Fe	U ²³⁸	U ²³⁵	Pu ²³⁹
10 ²⁴ X N		0.0828	0.022	0.0847	0.0473	0.0473	0.0403
Σ_{tr}	1	0.120	0.0387	0.167	0.184	0.184	0.157
	2	0.220	0.0741	0.188	0.237	0.237	0.202
	3	0.320	0.0928	0.294	0.378	0.378	0.322
	4	0.359	0.1496	0.593	0.662	0.662	0.564
	5	0.368	0.0640	0.949	1.183	2.081	1.773
	6	0.383	0.0640	0.949	0.747	27.907	41.1
Σ_c	1	---	0.0000055	0.00212	0.00237	0.00355	0.00524
	2	---	0.0000103	0.00263	0.00530	0.00639	0.00484
	3	---	0.000022	0.00347	0.01017	0.01599	0.00605
	4	---	0.00011	0.00635	0.0284	0.05463	0.0685
	5	0.000005	0.000132	0.00635	0.993	0.568	0.677
	6	0.00033	0.0088	0.19481	0.1277	4.26	12.45
Σ_{in}	1	0.0190	0.00506	0.0618	0.1045	0.0568	0.0343
	2	0.0265	0.00528	0.0203	0.0237	0.0142	0.0085
	3	0.0248	0.00352	0.00483	---	---	---
	4	0.0116	0.00242	0.00407	---	---	---
	5	0.0071	0.00064	0.00390	---	---	---
	6	---	---	---	---	---	---
Σ_f	1	---	---	---	0.0265	0.0591	0.0746
	2	---	---	---	0.0019	0.0579	0.0705
	3	---	---	---	---	0.0799	0.0766
	4	---	---	---	---	0.1561	0.137
	5	---	---	---	---	1.14	0.9672
	6	---	---	---	---	23.2	28.21
ν					2.5	2.5	2.9

*All inelastic scattering is into the next lower group except for Group 1 in U²³⁸, where the transfer coefficient $C_{1 \rightarrow 2} = 0.6$, $C_{1 \rightarrow 3} = 0.4$; and Group 1 in U²³⁵ and Pu²³⁹, where the transfer coefficient $C_{1 \rightarrow 2} = 0.67$, $C_{1 \rightarrow 3} = 0.33$ were used. Any elastic moderation has been lumped into the elastic cross section.

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In the first problem (IBM-101), the reactor consists of a 52.34 cm radius core, containing 30% U, 15% Fe and 55% Na by volume, with an 18 in. thick, depleted uranium blanket and an outside graphite reflector. The calculated critical mass was 589 kg, with the initial breeding ratio 1.353. If no charge is made for loss of blanket U^{235} . (The breeding ratio becomes 1.332 if an equal charge is made for blanket U^{235} .)

A detailed neutron balance is given in Table II, while the six group fluxes are plotted in Figure 1. The results are normalized on the basis of one fission neutron born in the core per cc of core.

Table II

IBM PROBLEM NO. 101

Critical Mass: 589 kg

Breeding Ratio: 1.353

	<u>Core</u>	<u>Blanket</u>	<u>Reflector</u>
Outer Radius, cm	52.34	98.52	129.3
	<u>Composition, %</u>		
Na	55.	20.	
Fe	15.	10.	
U^{238}	24.71	69.86	
U^{235}	5.29	0.14	
C	---	---	100.
	<u>Neutron Balance</u>		
U^{235} fission	0.334	0.00588	
U^{235} capture	0.0621	0.00117	
U^{238} fission	0.0658	0.0386	
U^{238} capture	0.190	0.347	
Na capture	0.00097	0.00026	
Fe capture	0.0393	0.0170	
Leakage out	0.308	0.00857	0.0084

In the second problem (IBM-102), the reactor is the same except that the outer graphite reflector has been omitted. The critical mass increased slightly to 597 kg, while the breeding ratio dropped to 1.300. The detailed neutron balance in Table III shows that more neutrons are leaking out of the blanket than in problem 101; thus, the multigroup calculation agrees

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FIG. 1
SIX-GROUP FLUX PLOT FOR
IBM PROBLEM NO. 101

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with previous one-group estimates¹ in that there will be significant leakage through 18 in. of a 70% U blanket. The fluxes for problem 102 are plotted in Figure 2.

Table III

IBM PROBLEM NO. 102

Critical Mass: 597 kg

Breeding Ratio: 1.30

	<u>Core</u>	<u>Blanket</u>	<u>Reflector</u>
Outer Radius, cm	52.34	98.52	--
	<u>Composition, %</u>		
Na	55.	20.	
Fe	15.	10.	No
U ²³⁸	24.64	69.86	Reflector
U ²³⁵	5.36	0.14	

Neutron Balance

U ²³⁵ fission	0.334	0.00523	
U ²³⁵ capture	0.0619	0.00103	
U ²³⁸ fission	0.0656	0.0393	
U ²³⁸ capture	0.186	0.328	No
Na capture	0.00097	0.00022	Reflector
Fe capture	0.0397	0.0162	
Leakage out	0.311	0.0329	

The third and fourth problems were designed to check the effect of reflector control. In the third problem (IBM-103) a 6 in. reflector, consisting of 90% Fe and 10% Na, was placed between the core and the 18 in. thick blanket. No outer graphite reflector was used. The critical mass went up slightly to 614 kg, while the breeding ratio dropped severely to 0.913. Of this drop, perhaps 0.11 can be attributed directly to the loss in fast fission bonus. The inelastic scattering of the iron reflector markedly reduces the U²³⁸ fissions in the blanket. The remaining 0.33, primarily due to capture in the iron reflector, is a considerably less definite effect, since fairly pessimistic capture cross sections have been chosen for iron in the absence of

¹ Reactor Engineering Division Quarterly Report, ANL-5060, June 15, 1953, p. 11.

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FIG. 2
SIX-GROUP FLUX PLOT FOR
IBM PROBLEM NO. 102

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definitive data. In previously reported² two-group calculations, the use of an iron reflector next to the core lowered the critical mass, while in these calculations a slight rise was noted. This is believed due to the use of higher capture cross sections for iron, plus certain modifications of the U^{238} cross sections used.

The neutron balance for problem 103 is given in Table IV; the fluxes are plotted in Figure 3.

Table IV

IBM PROBLEM NO. 103

Critical Mass: 614 kg

Breeding Ratio: 0.913

	<u>Core</u>	<u>Reflector</u>	<u>Blanket</u>
Outer Radius, cm	52.34	67.73	110.8
	<u>Composition, %</u>		
Na	55.	10.	20.
Fe	15.	90.	10.
U^{238}	24.48	--	69.86 70
U^{235}	5.52	--	0.14
	<u>Neutron Balance</u>		
U^{235} fission	0.335		0.00258
U^{235} capture	0.0632		0.00056
U^{238} fission	0.0650		0.00916
U^{238} capture	0.186		0.177
Na capture	0.00102	0.000142	0.00013
Fe capture	0.0404	0.126	0.00784
Leakage out	0.309	0.183	0.0154

In the fourth problem (IBM-104), the iron reflector has been removed and the space filled with sodium. The critical mass now rises to 636 kg, but the breeding ratio is higher than in problem 103, being 1.26. The neutron balance and fluxes are given in Table V and Figure 4, respectively.

²Ibid., p. 8.

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FIG. 3
SIX-GROUP FLUX PLOT FOR
IBM PROBLEM NO. 103.

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Table V

IBM PROBLEM NO. 104

Critical Mass: 636 kg

Breeding Ratio: 1.26

	<u>Core</u>	<u>Inner Blanket</u>	<u>Outer Blanket</u>
Outer Radius, cm	52.34	67.73	110.8
	<u>Composition, %</u>		
Na	55.	100.	20.
Fe	15.		10.
U ²³⁸	24.88		69.86 70
U ²³⁵	5.72		0.14
	<u>Neutron Balance</u>		
U ²³⁵ fission	0.336		0.00485
U ²³⁵ capture	0.0630		0.00103
U ²³⁸ fission	0.0644		0.0312
U ²³⁸ capture	0.177		0.325
Na capture	0.00095	0.00132	0.00023
Fe capture	0.0377		0.148
Leakage out	0.321	0.320	0.0306

Possibly there is an undesirable aspect associated with the use of the outer graphite reflector in the first problem. The moderation therein will result in a higher $\alpha (= \frac{\sigma_c}{\sigma_f})$ for the plutonium produced in the outer portions of the blanket. Of course, the blanket thickness itself could be increased to reduce neutron leakage and to do away with the need of an outer reflector. However, since some structural material is likely out here anyway, it was decided to investigate the effect of substituting Fe for the graphite in the outer reflector of problem 101. The solution of this problem (IBM-101-A) gave a critical mass of 595 kg and a breeding ratio of 1.347. The neutron balance and fluxes are given in Table VI and Figure 5, respectively.

The assistance of James H. Alexander and Jean F. Hall (Argonne Physics Division) in programming the multigroup setup for the IBM is gratefully acknowledged.

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FIG.4
SIX-GROUP FLUX PLOT FOR
IBM PROBLEM NO. 104

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Table VI

IBM PROBLEM NO. 101-A

Critical Mass: 595 kg

Breeding Ratio: 1.347

	<u>Core</u>	<u>Blanket</u>	<u>Reflector</u>
Outer Radius, cm	52.34	98.52	129.3
<u>Composition, %</u>			
Na	55.	20.	
Fe	15.	10.	100.
U ²³⁸	24.65	69.86	
U ²³⁵	5.35	0.14	
<u>Neutron Balance</u>			
U ²³⁵ fission	0.334	0.00550	
U ²³⁵ capture	0.0620	0.00109	
U ²³⁸ fission	0.0658	0.0400	
U ²³⁸ capture	0.187	0.347	
Na capture	0.00097	0.00024	
Fe capture	0.0396	0.0170	
Leakage out	0.310	0.0167	0.00465

2. Plutonium Quality in Graphite-Reflected Blanket - D. Okrent

As shown in the results of IBM problem 101, the graphite-reflected blanket will have a much lower neutron energy spectrum in its outer portion. Some calculations were made to investigate the effect of the energy degradation on the plutonium production in such a blanket. The flux distributions of IBM problem 101 (Figure 1) and the cross sections of Table I were adopted for this analysis. The reactor under question has the composition and dimensions specified in Table II.

Calculations were first made on the variation of α through the core and blanket for both U²³⁵ and Pu²³⁹ (where $\alpha = \frac{\bar{\sigma}_c}{\bar{\sigma}_f}$) as a function of position. The results are given in Figure 6.

Next the production and quality of plutonium were examined under the conditions when one begins with a brand new reactor and runs the core to 2% burnup. Only average properties were found for the core, but

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FIG. 5
SIX-GROUP FLUX PLOT FOR
IBM PROBLEM NO. 101-A

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FIG.6
 VARIATION OF $\alpha = \frac{\sigma_c}{\sigma_f}$ FOR U^{235} AND Pu^{239} WITH POSITION
 IN CORE AND BLANKET. IBM PROBLEM NO. 101

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spatial variation was computed for the blanket. In Figure 7 the plutonium quality ($\text{gm Pu}^{240}/\text{gm Pu}^{239}$) is plotted as a function of position, while in Figure 8 the concentration of Pu^{239} (gm/cc) is plotted as a function of position.

The blanket was arbitrarily divided into spherical annuli or shells of equal thickness, and the total Pu^{239} production and average was computed for each region. The results are listed in Table VII; region 1 is the innermost blanket section, region 5 the outermost.

Table VII

Shell	Radius, cm		Grams Pu^{239} Produced in 2% Core Burnup	Average Quality for 2% Core Burnup $\frac{(\text{Grams Pu}^{240})}{(\text{Grams Pu}^{239})}$
	Inner	Outer		
0 (core)	-	52.34	31,000	0.0074
1	52.34	61.58	24,000	0.0031
2	61.58	70.81	16,200	0.0014
3	70.81	80.05	9,700	0.0006
4	80.05	89.01	5,400	0.0003
5	89.01	98.52	4,500	0.0017

Next, plutonium quality was computed for the case where the various sections of the blanket are run for different times, so that all may have approximately the same concentration of Pu^{239} when removed for processing. The average concentration in the blanket was set at such a value that the Pu^{239} fissions in the blanket equalled 10% of the core fissions. Actually the blanket material would be removed when the concentration reached twice the average figure (for a continuous process). The concentration at removal was estimated to be $0.288 \text{ gm Pu}^{239}/\text{cc}$, or roughly $20,000 \text{ gm/T}$. Using this number and the average concentration found in each shell for 2% core burnup, the number of cores each shell sees burned out before it is removed for processing is readily computed. The results follow:

Shell	Cores
1	4.54
2	9.06
3	19.71
4	44.21
5	65.71

The above numbers will only give approximately the same average concentration in each shell, since the Pu^{239} burnout rate differs from shell to shell.

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FIG. 7
VARIATION OF PLUTONIUM QUALITY (gm Pu^{240} /gm Pu^{239}) WITH
POSITION IN BLANKET FOLLOWING 2% BURNUP OF ONE CORE.
(AVERAGE QUALITY IN CORE = 0.0074)
IBM PROBLEM NO. 101.

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FIG.8
 VARIATION OF Pu^{239} CONCENTRATION WITH POSITION
 IN BLANKET FOLLOWING 2% BURNUP OF ONE CORE.
 (AVERAGE CONCENTRATION IN CORE = 0.052)
 IBM PROBLEM NO. 101.

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The production (allowing for burnout) and the average quality for each shell were now found, applying the above exposure periods. The results are listed in Table VIII. It is seen that the quality in shell 5 is particularly poor. However, the average quality, weighted with production rate, is only 0.019 for the entire blanket.

Table VIII

Shell	Exposure in No. of Cores	Grams Pu ²³⁹ Produced	Average Quality
1	4.54	100,500	0.0140
2	9.06	132,000	0.0127
3	19.71	170,700	0.0126
4	44.21	214,700	0.0133
5	65.71	215,300	0.1045

If one processed the outer shell more frequently, the quality herein could be improved. The quality is almost linear with exposure time for these exposure periods. If one were to run the outer shell for 1/8 the time previously listed, i.e., 1/8 (65.71) = 8.21 cores, the average quality in this shell would drop to 0.0141, and the average quality for the entire blanket would become 0.0134. This would not be achieved without cost, however. The average time before processing for the entire blanket would be dropped by a factor of 1.9.

Since the quality is given to a first approximation by $Q = \frac{N_{\text{Pu}^{240}}}{N_{\text{Pu}^{239}}} \approx \frac{\sigma_c^{\text{Pu}^{239}} \phi t}{2}$, the quantity $\frac{\text{Pu}^{239}}{\sigma_c}$ is an important quantity in making quick estimates. For the flux of Figure 1 and the cross sections of Table I, it is found to vary as shown in Figure 9.

C. Fuel Elements and Fuel Media

1. Gas Entrapment in Sodium - Uranium Ball Columns - E. L. Currier

Preliminary experiments concerning the effect of gas entrapment when gas passes up a column packed with steel balls in liquid sodium were reported previously.³ Subsequent tests confirmed the tentative

³Reactor Engineering Quarterly Report, ANL-5060, June 15, 1953.

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FIG.9
VARIATION OF σ_c Pu^{239} WITH POSITION
IN BLANKET. IBM PROBLEM NO. 101

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conclusion that partial entrapment of a gas does occur in the case of 1/16 in. diameter steel balls packed in a 3/8 in. ID column, 10 in. high, containing sodium at a temperature of 600C, operating conditions being static.

In order to check the above conclusions using conditions more closely approximating an actual fuel element, the test equipment was revised by the addition of a thermal convection arm. In actual service the projected element would be composed of small spherical particles of fuel elements immersed in sodium-potassium alloy or in liquid sodium, sealed in a small diameter stainless steel tube. Coolant circulates outside the tube. It is expected, therefore, that thermal convection currents will exist in the sealed tube due to the temperature gradient existing between the tube center and the tube wall. This internal thermal convection current is simulated by the addition of the aforementioned side arm connected to the test section at the top and the bottom, the side arm being held at a lower temperature than the test section. Thermal convection in the test section is thus established on the basis of sodium density differences.

Considerable work was performed to demonstrate the circulation of sodium in the system. Proof of circulation was shown by the large temperature fluctuations exhibited at the two side arm thermocouple well locations following cooling of the test section.

In the following tests, 1/16 in. diameter uranium balls were used in the test section. The pickled balls were prepared from swaged wire and formed by cold heading. The inside areas of the test equipment were pickled, evacuated, and degassed. All tests were performed at one atmosphere of helium pressure. Liquid sodium was used as the heat transfer agent. After the test section and the side arm temperatures were stabilized, a small measured amount of helium gas was released through a needle valve into the bottom of the test section. Volumetric measurements of the gas in the liquid sodium in the test section were made with an electric probe at the surface.

Averaged data for five tests are as follows:

	Temperature, C	
	Start	Finish
Test Section	596	596
Upper Arm	499	489
Lower Arm	490	482
Time, min.	cc Helium at 600C	
0 60	0.59	0.48
5 120	0.59	0.48
18 180	0.59	0.48
20	0.48	

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The value of time in minutes shows elapsed time after gas is admitted into the test section. The value of cc helium shows that the quantity initially admitted after reaching temperature was constant up through 10 min of elapsed time and that the quantity remaining in the test section was reduced approximately 20% to a constant value for the test. The results show gas entrapment by the fact that the helium volume trapped in the test section remains constant over a long period of time.

Additional tests were made under a reverse temperature condition, i.e., with the test section held at 500C and the side arm held at 600C. This condition generates a downward flow of sodium in the test section. Probing the liquid level after admitting helium into the column demonstrated gas entrapment similar to the above results.

2. Sodium Filling and Cleaning of Fast Exponential Experiment Fuel Cans - M. Sivetz

Two steel cans, 3 in. x 3 in. x 24 in. with tack-welded, dished heads soldered vacuum tight, were successfully filled with molten sodium at 150C. With a surrounding water cooling coil, the cans then were brought to room temperature. Next, the cans and sodium were reheated to 150C and were drained by forcing with nitrogen gas under pressure. Finally, a rinse recycle was accomplished on both cans to remove any residual sodium. Rinsing of the first can, which contained seven natural uranium plates, 0.075 in. thick by 2.7 in. wide, and seven similarly sized steel plates, together with about a pound of residual sodium, was tried using 10C ethanol, but the rinse reaction got out of control, and the temperature rose to 180F; however, the plates were unharmed. Rinsing of the second can, which contained 15 natural uranium plates, and which was more thoroughly drained of sodium initially, was successfully accomplished in a few minutes using 10C isopropanol. These experiments indicate that the steel cans prepared and sealed as above could be satisfactorily filled and emptied of sodium and could be rapidly cleaned with isopropanol followed by air drying.

The isopropanol (or ethanol) circulation apparatus had a paper filter incorporated into it, and the filter collected 2 gm of impurities from clean-up of the second can which, when ashed at 600C, showed the following major constituents:

Element	U	Mg	Cu	Zr	Pb	Ca	Fe	Si	Al	Others
Wt %	15	18	8	8	7	5	3	2	1	<1

The spectrochemical method of analysis employed is probably accurate to within a factor of about two. This technique of dissolving sodium in isopropanol (or ethanol) and analyzing the filter residue eliminates bulk sodium interference on the sensitivity of spectrochemical analysis of other elements. Laboratory filtration of 10 gm of du Pont and National Distillers sodium brick dissolved in ethanol showed the following major impurities:

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Element	Mg	Cu	Pb	Ca	Fe	Al	K
du Pont, ppm	40	80	4	120	50	10	50
Nat. Dist., ppm	20	10	<3	40	20	2	20

These values appear to be those expected from published analyses. It is seen that most of the sodium impurities listed occur in the filter residue analysis described above.

Bare air-rolled, oil-rolled, and electropolished uranium samples, .075 x 0.5 x 3 in., subjected to the above procedure, showed 0.1 to 1% weight loss, while 0.1 mil nickel-plated uranium was unaffected by the sodium pour, melt, or clean-up process.

3. Thermal Stability Tests - W. S. Fagan, J. H. Monaweck

Six normal uranium cylinders, 0.36 in. diameter by 1-7/8 in. long, were prepared by Sylvania Electric Products using powder metallurgy techniques. The six pieces were machined from a single slug 1.37 in. diameter by 4 in. long which was made by hot pressing -80 mesh uranium powder at 12 ton/sq in. for 10 minutes at a temperature of 580C in vacuum.

Each sample was sealed into a capsule 5 in. diameter by 2.5 in. long containing NaK. No precautions were taken to eliminate oxygen. These samples were thermocycled as noted in the tabulation.

Sample	Cycles	Temperature, C		Change in Length, in.	Appearance
		High	Low		
1	425	600	200	0	Good.
2	400	725	230	+1/4	Rough. Restrained by capsule.
3	150	825	230	+1/8	Rough. Restrained by capsule.
4	117	725	600	0	Rough.
5	125	825	600	-1/8	Slightly rough.
6	125	825	725	-1/32	Rough.

These data on powder metallurgy samples indicate:

a. Thermocycling from a comparatively cool temperature to a temperature below the alpha-beta transformation range does not produce appreciable longitudinal growth.

b. Thermocycling from a cool temperature to a temperature above the alpha-beta or the beta-gamma transformation range does produce considerable longitudinal growth.

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c. Thermocycling from a temperature just below the alpha-beta transformation to either a temperature in the beta range or in the gamma range did not produce longitudinal growth.

These experiments are to be repeated using cylinders pressed to the exact dimensions given above.

D. Reflector and Structural Materials

Mass Transport Studies - C. H. Scheibelhut

Construction of the dynamic system (loop) designed for the purpose of studying mass transport in liquid metal systems is virtually complete. Design details of the loop were previously reported.⁴ Shakedown operation is expected to begin on September 1, 1953. The radioactive metal samples (type 347 and type 304 stainless steel) to be installed in the loop for the first transport tests have been fabricated and are currently being irradiated in CP-3'.

An additional test section has been added to the loop, the objective being to gain information on the practicability of employing Schedule 5 piping in liquid metal dynamic systems. If feasible, use of Schedule 5, instead of the heavier piping (usually Schedule 40) normally used in large-scale experiments or prototype reactors, would effect considerable economy. The test section consists of two 2 in. lengths of 2 in. NPS, Schedule 5, type 347 stainless steel pipe joined by a heliarc butt weld and heliarc welded into the loop at a point where maximum temperature and pressure expected are 600C (1112F) and 75 psig, respectively.

Work on study of mass transport in static systems has been indefinitely deferred in order to concentrate effort on the dynamic system study.

E. Reactor Coolant, Heat Transfer, and Power Generation

1. Liquid Metal Sampling and Analysis - J. R. Humphreys

Effort is continuing on the evaluation of 3 analytical methods for determination of oxide in sodium or sodium-potassium alloy.

The first method is, basically, the Williams and Miller⁵ modification of the Pepkowitz and Judd⁶ method of mercury amalgamation separation of the oxide from the metal. The second method being considered

⁴Reactor Engineering Division Quarterly Report, ANL-5060, June 15, 1953, p. 23.

⁵D. D. Williams, R. R. Miller, "Modified Apparatus for the Determination of Sodium Monoxide in Sodium," Anal. Chem., 23 1865 (1951).

⁶L. P. Pepkowitz, W. C. Judd, "Determination of Sodium Monoxide in Sodium," Anal. Chem., 22 1283 (1950).

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is that of White, Ross, and Rowan,⁷ in which separation of the oxide is accomplished using the Wurtz reactions (in which the alkali metal reacts with n-butyl bromide to form the alkali bromide and octane). As the alkali oxide does not react with n-butyl bromide, it may be determined by titration after the addition of water. The third method is one in which the sample is contained in an open cup under vacuum and heated until all alkali metal has been removed from the oxide by vacuum distillation. The residual oxide in the cup is then dissolved in water and quantitatively determined by titration. The method used for obtaining samples for analysis by any of the three methods is such that the sample is introduced directly into the glass or metal reaction vessels (see ANL-5060), thus eliminating the necessity of removing the sample from small capsules or cups while under vacuum.

The necessary analytical apparatus and chemicals are on hand for conducting analyses by all three methods. Preliminary investigation of the mercury amalgamation method is now underway and work is to be started immediately on the other methods.

The portable (bellows type) liquid metal sampler has been mounted on its carriage and has been fitted with the vacuum pump and inert gas systems. Service testing of this sampler has been started.

The alternate type of sampler ("O"-ring sealed) has been received from the shops but has not yet been assembled or leak tested.

2. Thermal Analysis of the Reactor Core - L. L. Kintner,
R. R. Rounsley, W. R. Simmons

Temperatures in an equilateral triangular fuel element were obtained by means of an electrical analogue. Two configurations of fuel in the element were considered: (1) six fuel pins inside the triangular stainless steel can with a liquid sodium bond; and (2) a triangular fuel rod with a liquid sodium bond between the fuel and stainless steel can. In each configuration, 91 fuel elements are arranged in a hexagonal subassembly made of 1/32 in. stainless steel. Spacing of the elements is accomplished by means of longitudinal fins at the apex of the triangular element. The elements are cooled by forced convection flow of sodium on the outside surface of the elements.

Fuel element and coolant temperatures for full power operation at 500 mw output (core power) with 25 fps coolant velocity are given in Table IX. The maximum to average heat flux in the core was assumed to be 1.45 as in other analyses of the core. No allowance was made for deviations from the ideal fuel element in this preliminary estimate of temperatures. Therefore more realistic coolant temperatures would be somewhat lower than those indicated in Table IX.

⁷J. C. White, W. J. Ross, R. Rowan, Jr., "The Determination of Oxygen in Sodium," ORNL-1286, Declass., April, 1952.

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Table IX

COMPARISON OF TRIANGULAR FUEL ELEMENT AND COOLANT TEMPERATURES

	1	2	3
Fuel alloy shape	Round pin	Round pin	Triangular rod
Fuel alloy dimension, in.	0.0625 (dia.)	0.0938 (dia.)	0.2245 (side of triangle)
Triangular container side length, in.	0.250	0.366	0.250
Fin length, in.	0.055	0.0706	0.055
Container thickness, in.	0.010	0.010	0.010
No. of fuel elements/assembly	91	91	91
No. of assemblies/3 ft by 3 ft reactor	137	71	137
Core composition			
Volume per cent of fuel alloy	22.4	26.5	26.5
Volume per cent of stainless	17.2	12.9	17.2
Volume per cent of sodium coolant	45.3	47.7	45.3
Volume per cent of sodium bond	10.5	10.8	6.4
Volume Per cent of sodium between assemblies	4.6	3.1	4.6
Inlet coolant temperature, F	815	430	930
Average coolant temperature rise, F	430(3)	410(3)	430(3)
Outlet coolant temperature, F	1245	840	1360
Maximum cooling surface temperature, F	1320	945	1440
Maximum sodium bond temperature, F	1620(1)	1620(1)	1523
Maximum fuel temperature, F	1620	1620	1870(2)
Estimated core pressure drop, psi	24	18	24

(1) Boiling point of sodium at 1 atmosphere pressure.

(2) Melting point of 90% uranium - 10% plutonium alloy.

(3) Assuming equal temperature rise in all subassemblies.

A comparison of columns 1 and 2 of Table IX shows that with the same maximum temperature limitation, the smaller element allows the highest coolant temperatures. However, the amount of fuel alloy is less and stainless steel is greater in the smaller element than in the larger unit, and nuclear considerations may indicate a compromise between the two sizes. The element of column 3 is the same size as the small element (column 1), but because the sodium bond is confined to the outside of the element, the boiling point of sodium (1620F) does not limit this design. Instead, the melting point of 90% uranium - 10% plutonium (1870F) is considered the limiting temperature. Thus, even higher coolant temperatures can be obtained in the solid triangular rod element than in the small element containing pins. The percentage of fuel in this element is the same as in the large element containing pins (26.5%).

The maximum surface temperature at any cross section of the triangular element occurs at the mid-point of a side. The effect of the fin touching the side of an adjacent element is negligible since a good thermal bond between the can and the touching fin is effected by the sodium coolant. The maximum temperature variation around the periphery of the element is about 150F. The proposed fuel element should be tested with this temperature gradient imposed on it to see that it does not cause severe buckling of the sides and/or failure of the corner seam weld.

Temperatures were calculated for another type of core geometry, consisting of tubes in a casting of fuel alloy (similar to Detroit Edison design). Coolant flows inside the tubes and the fuel alloy (U-Cr eutectic) is outside the tubes. The stainless steel tubes (wall thickness = 0.020 in.) were assumed to be spaced on an equilateral triangular pitch. For a 1/8 in. space between tubes, 45% coolant volume, 25 fps maximum coolant velocity, and 1.45 maximum to average heat flux ratio, the tube inside diameter is about 13/32 in. The volume percentage of fuel alloy in this geometry is 45%, leaving 10% stainless steel in the core. If the maximum fuel temperature is limited to the melting point of U-Cr eutectic (1600F), the estimated coolant temperatures for 500 mw core power are 560F inlet and 1000F outlet (assuming equal coolant temperature rise in all channels). Higher coolant temperatures result from smaller spacing between tubes but the problems of fabricating the core increase considerably. Slightly higher coolant temperatures are calculated for a lower percentage coolant but the temperature rise of the coolant also increases, causing greater problems in heat exchangers, etc.

3. Transient Coolant Temperatures - W. S. Flinn

Preliminary calculations were made to estimate the transient coolant temperatures which might occur if the reactor were suddenly scrammed after prolonged operation at full power. Knowledge of these temperature changes during this transient is necessary in order to determine possible resulting thermal stresses in reactor structural components.

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For these calculations, the following was assumed:

- a. Initial inlet and discharge core coolant temperatures (full power) of 520F and 920F, respectively.
- b. Full power coolant flow exists in the primary and secondary systems during the time intervals considered - 14,000,000 lb/hr Na and 20,100,000 lb/hr NaK.
- c. A core discharge plenum chamber was assumed to be 7 ft in diameter and 2 ft high with perfect mixing of the core discharge in this plenum.
- d. Four 16 in. diameter discharge ducts were considered.

The calculations showed that the core discharge plenum temperature changes rapidly; within two seconds after scram, the temperature dropped from 920F to about 600F. If the 16 in. reactor discharge pipes had 1/4 in. thick walls, the thermal stresses imposed by this rapid drop do not appear excessive, although it probably would be advisable to use thermal sleeves or liners to lower these stresses to a more conservative value.

The effect of these temperature changes on the various heat exchangers is not known since the details of their designs are not established at the present time. Since boiler, preheater, and superheater must contain higher pressures than the intermediate exchanger, they are probably more sensitive to coolant temperature changes because of their stronger or more massive designs.

The same size plenum was assumed in both the NaK and the Na circuits. With such an arrangement, the NaK side of the superheater changed from 860 to 715F in about a 2 sec period. (Calculations indicated that it takes about three seconds after scram before the NaK temperature in the superheater starts to drop from 860F.)

The thermal inertia effect of the plenum chambers is beneficial in minimizing thermal shock. However, at the present stage of design, much more thermal inertia is required to dampen or delay the temperature change to the point where it becomes obvious that the rates of temperature drop are permissible from a thermal stress standpoint. Just what the permissible rates are can not be determined without calculations of a more detailed and specific nature, further design information, and thermal cycling information on the materials involved.

The results of these calculations indicate that a sudden shutdown of a PBR-type reactor operating with a coolant rise of 400F could create thermal stresses. Reduction of coolant flow during shutdown, the use of more or larger plenum chambers, and the use of thermal sleeves, may be necessary to minimize these transient conditions and/or permit greater core coolant temperature rises.

4. Heat Exchangers - L. L. Kintner, R. R. Rounsley

A plate-type heat exchanger was analyzed for use as a Na to NaK, low pressure heat exchanger. The exchanger design assumes 0.025 in. thick, stainless steel, flat plates, welded at the sides to form coolant passages, with a Na and NaK header at each end. The Na channels were assumed to be 0.100 in. and the NaK channels 0.160 in. thick. The estimated size required to transfer 500 mw of heat energy at 10 fps Na and NaK velocity and 31F temperature difference between fluids is 5 ft by 5 ft by 12.5 ft. The calculated over-all heat transfer coefficient is 2340 Btu/(hr)(sq ft)(F) and the total heating surface is 23,300 sq ft.

The size and performance is comparable to a tubular exchanger using 1/4 in. tubes. However, over 35,000 tubes are required for the tubular exchanger; whereas only 385 plates are required for the plate type. Since the volume per unit surface area is less in the plate-type exchanger, the material cost should be less than the tubular type. Methods and costs of fabricating such a plate-type exchanger are being investigated.

5. Apparent Conductivity Tests - L. L. Kintner, K. D. Kuczen, R. R. Rounsley

An apparatus has been designed (Figure 10) and is being constructed to determine the apparent conductivity of solid metal aggregates in liquid metals. It is a cylindrical conductivity apparatus with a d-c electrical heater (6 kw capacity) at the center and radial heat flow through an annulus containing metal spheres submerged in a liquid metal. The heater has been constructed, tested, and found to be satisfactory. Temperatures will be measured on each side of the annulus near a central cross section. The data will be used to calculate the temperatures in fuel elements containing fuel particles in a liquid metal bond. Data will be taken for the following range of variables:

Sphere material: steel, nickel, copper.

Sphere diameter: 1/64 in. to 1/8 in.

Liquid metal: mercury, sodium-potassium, sodium.

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L.L.KINTNER:S.H., 12-11-53

POWER REQUIRED: 6kw d-c (600 amp, 10V)

FIG. 10
SCHEMATIC OF APPARENT
CONDUCTIVITY TEST APPARATUS

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F. Instrumentation

Electromagnetic Flowmeter "Wetting" - F. A. Smith

Calibration against a standard orifice flowmeter has been accomplished on 8 electromagnetic, liquid metal flowmeters scheduled for general experimental use in the laboratory. The resulting data appear to provide further insight into the phenomenon of "wetting" of such meters.

Past experience with a single electromagnetic flowmeter installed in a loop has indicated that the time required for wetting (at a particular temperature) might be a function of gas content or oxide content of the liquid metal in the loop or of the degree of cleanliness of the inner pipe wall surface in the region of the flowmeter electrode. In the present calibration loop, which consists of four 4 in. and four 2 in. diameter electromagnetic flow meters in series with a pump, a throttling valve, an orifice flowmeter, and an expansion tank, no attempt was made to clean initially the loop components or to blanket the expansion surface of the liquid metal (NaK) with an inert atmosphere. The data presented in Figure 11 show that during the first 2 days of operation, August 4 and 5, only 6 of the 8 flowmeters developed a finite output voltage (indicating a certain degree of wetting). Not until August 6 did the remaining two meters indicate any wetting. The fact that all 8 flowmeters did not begin to wet at the same time appears significant. If wetting were a function of gas or oxide content of the liquid metal, all meters logically should have begun to wet at the same time, since the same liquid metal flowed through each meter. Consequently, the observed behavior strongly suggests that the surface condition of the pipe is the primary criterion for wetting.

To further investigate this phenomenon, a test is being started in which the relative speeds of wetting (using the same liquid metal) will be determined for 2 flowmeters, one of which will employ a cleaned, electropolished pipe section, and the other an uncleaned section.

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F.A. SMITH:H.C., 9-4-53

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CENTRAL STATION WATER REACTOR (CSWR)

A. Physics and Critical Assembly Work

The Conversion Ratio Experiment - J. C. Carter, J. A. Thie

The purpose of this experiment was to determine the conversion ratio of a heavy water, natural uranium reactor.

The experiment was conducted in the cold, unpoisoned ZPR-II reactor without control rods. Criticality was attained by varying the D_2O level. Tests of approximately one-hour duration at 800w were run, using various loadings of 1 in. and 1.3 in. OD natural uranium fuel elements.

The value of the conversion ratio was established by two different methods, namely, the moderator normalization and fuel normalization. In both methods enriched and depleted uranium foils were used to measure the fission of U^{235} and the neptunium content of both bare and cadmium-covered foils.

In both methods, the plutonium production rate was measured by a scintillation spectrometer. Corrections were made for natural background and for fission product activity. The foils used were depleted uranium (0.491% U^{235}).

The U^{235} fission rates were measured in U^{235} -aluminum alloy foils. Two alloys were used: 4% U^{235} -aluminum and 6% U^{235} -aluminum.

The initial U^{235} content of the enriched foils was determined by counting 180-kev U^{235} gammas. Fissions were measured by counting gammas of over 450 kev. These measurements were corrected for natural U^{235} background.

The thermal flux and resonance flux in the reactor were measured by using a copper wire technique and by using cadmium-covered gold foils, respectively.

In building up a set of foils for a test, an effort was made to compile units of foils similar in geometry, weight, and isotopic content. Typical foil units and their relative positions are shown schematically in Figure 12.

1. Moderator Normalization Method

This method requires that 2 bare foil units be placed in a vertical fuel rod and that one bare and one cadmium-covered foil unit be placed in the moderator on a parallel vertical center line. This center line must be equidistant from the 3 adjacent fuel elements, and the corresponding rod and moderator foils must be in the same horizontal plane.

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J. CARTER: S.R.H., 12-3-53

FIG. 12
DIAGRAM OF FOIL UNITS AND THEIR ARRANGEMENT
FOR MODERATOR NORMALIZATION TYPE TEST

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Essentially, the method of Towler and Untermeyer⁸ is used. To obtain a conversion ratio from the irradiated foils described above, it is necessary to determine the neptunium content of the depleted foils and the fission product content of the enriched foils. Let N_m , N_c , and N_r equal the number of neptunium atoms/cm³ of depleted foils located, respectively, in the moderator (without cadmium), within the cadmium covers, and in the fuel rods. The corresponding quantities for enriched foils are F_m , F_c , and F_r .

The conversion ratio (CR) is defined as

$$CR = 137.8 \frac{\frac{\sigma_{Ia}^{U^{238}} + \sigma_{IIa}^{U^{238}}}{\sigma_{Ia}^{U^{235}} + \sigma_{IIa}^{U^{235}}}}$$

where subscripts I and II describe cross sections averaged over thermal and epicadmium fuel spectra, respectively. (1 and 2 will be used for corresponding moderator spectra.) This is readily expressed in terms of the N 's and F 's if one utilizes the virtual equality of the moderator's thermal α and the fuel's total α .

$$\begin{aligned} CR &= 137.8 \frac{\sigma_{Ia}^{U^{238}}}{\sigma_{Ia}^{U^{235}}} \times \frac{\sigma_{Ia}^{U^{238}} + \sigma_{IIa}^{U^{238}}}{\sigma_{Ia}^{U^{238}}} \times \frac{\sigma_{If}^{U^{235}}}{\sigma_{If}^{U^{235}} + \sigma_{IIIf}^{U^{235}}} \\ &= 137.8 \frac{\sigma_{Ia}^{U^{238}}}{\sigma_{Ia}^{U^{235}}} \times \frac{N_r}{N_m - N_c} \times \frac{F_m - F_c}{F_r} \end{aligned}$$

a. Counting and Its Corrections

The scintillation counter has been found applicable for measuring the above quantities by Axtmann and Stutheit.⁹ $(F_m - F_c)/F_r$ is determined by counting the foils in a NaI counter biased to count only γ -rays above 475 kev. To determine $(N_m - N_c)/N_r$, the depleted foils are counted in another NaI counter having a 10% energy window at 100 kev. K X-rays at 103.69 and 99.50 kev are observed after higher energy γ -rays of neptunium decay have been internally converted. Evidence that neptunium was being counted was obtained by averaging the slopes of ten decay curves of N_c (Test Set 3). The half life determined was 2.34 days.

⁸S. Untermeyer, private communication, May 13, 1953.

⁹R. C. Axtmann, J. S. Stutheit, "Scintillation Counting of Foils for Flux Measurement in Natural Uranium Piles," ANL-4867, July, 1952.

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A number of corrections must be applied to the raw counts; these include:

- (1) Background of counter and foil before irradiation;
- (2) Slight differences in foil weights which also affect the self-absorption of the foil;
- (3) Decay of foils when the quantities in a ratio are counted at different times.

In addition, it was found that approximately 6 % of the activity of foils, N_m and N_r , can be ascribed to fission products. Some of their decay γ -rays fall within the energy window. However, higher energy γ -rays make their presence known by ionizing uranium atoms which then emit 98.4 and 99.6 kev X-rays. The expression for the number of counts to be subtracted from an N_m or N_r count previously corrected for (1), (2), and (3) is:

$$\frac{\left[A(t) \right] \left[\text{(Counts-background) of an enriched foil in the neptunium counter} \right] \left[\text{Number of fission products in the depleted foil} \right]}{\left[\text{Number of fission products in the enriched foil} \right]}$$

The fission product ratio is determined by counting the 2 foils in the fission counter. $A(t)$ is an "amplification factor" which takes into account the fact that γ -rays above 100 kev in a 96% aluminum foil cannot produce 100-kev X-rays as they do in a pure uranium foil. Figure 13 shows $A(t)$ as determined by Redman¹⁰ who compared the activity of a group of 1-mil enriched foils with their amplified activity when interleaved in a group of 1-mil unactivated, depleted foils.

The optimum time for detection of neptunium in the presence of fission products occurs when the above correction becomes a minimum, percentagewise. This can be shown to occur when the slopes of fission product and neptunium decay curves are equal, i.e., at 90 hr after activation for this type of effective fission product spectrum. In practice, a typical run's counting schedule was once daily for 5 days, starting at 70 hr.

b. Cross Section Basis

Since $\sigma_{1a}^{U^{238}}$ and $\sigma_{1a}^{U^{235}}$ have slightly different energy dependence, each must be averaged with the moderator spectrum. The latter was taken to be gaussian up to E_j , and $1/E$ beyond this. The experimental cadmium ratios, F_m/F_c and N_m/N_c , can be calculated from this spectrum and cross section curves. Within the accuracy of the latter, both cadmium

¹⁰ W. C. Redman, private communication.

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W.C. REDMAN: E.A.K., 12-11-53

FIG. 13
 TIME DEPENDENCE OF THE
 URANIUM "AMPLIFICATION FACTOR"

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ratios can be calculated using only the one value $E_j = 0.172$ ev. $\sigma_{1a}^{U^{238}}$ is then $2.77 \times 0.8862 \times 1.0052$ barns when the 2200 m/sec value¹¹ is corrected for special distribution. The spectrum's $1/E$ tail and the distortion of a $1/v$ cross section by the 6.7 ev resonance gives the 0.52% increase. $\sigma_{1a}^{U^{235}}$ is $684 \times 0.8862 \times 0.980$ barns if the 2200 m/sec value is similarly corrected. Here the non- $1/v$ effect is 2%. Thus, all results listed below are based on the value

$$137.8 \frac{\sigma_{1a}^{U^{238}}}{\sigma_{1a}^{U^{235}}} = 0.5753$$

2. Fuel Normalization Method

This method requires a bare foil unit and a cadmium-covered foil unit (Figure 14) placed in the same vertical fuel rod and spaced equidistant from the center line of the active core. The conversion ratio is found by the same formula used in the moderator normalization method if one sets $N_r = N_m$ and $F_r = F_y$ and uses the cross section $\sigma_{1a}^{U^{238}} / \sigma_{1a}^{U^{235}}$.

The program of tests conducted are tabulated.

Test Set	Test No.	Normalization Method Employed	Type of Enriched Foil	Fuel Rod OD, in.	Critical Ht, in.	Fuel & Foil Arrangement Fig. Ref.
1	1-6	Fuel	4% U^{235} - Al	1	68.4	15
2	7-18	Moderator	4% U^{235} - Al	1	68.4	15
3	19-24	Moderator	4% U^{235} - Al	1.3	82.	16
4	25-28	Moderator	4% U^{235} - Al	1.3	103.4	17
5	29-31	Fuel	6% U^{235} - Al	1.3	103.4	17
6	32-34	Fuel	6% U^{235} - Al	1.3	82.	16

The power during all tests was 800w. The duration of each test was approximately 1 hr. The cover plates were of 0.003 in. depleted uranium (4% U^{235}). The depleted foils were 0.010 in. depleted uranium (4% U^{235}) except in Test No. 9 in which the depleted foils used were 0.003 in. depleted uranium (0% U^{235}).

¹¹ Reactor Science and Technology, 3, 291, (1953).

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FIG. 14
DIAGRAM OF FOIL UNITS AND THEIR ARRANGEMENT
FOR FUEL NORMALIZATION TYPE TEST

RE-6-12000-A
J. CARTER: S.R.H., 12-3-53

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LEGEND

ALL Q-TUBES LOADED WITH 1 in. OD FUEL ELEMENTS EXCEPT THOSE MARKED WITH AN X

CRITICAL HEIGHT OF ACTIVE CORE: 68.4 in.

FOILS LOCATED 8 in. ABOVE AND BELOW CENTER LINE OF ACTIVE CORE

FUEL ELEMENT SPACES EVERYWHERE EXCEPT WHERE FOILS ARE LOCATED

FIG. 15
CONVERSION RATIO EXPERIMENT. PLAN
VIEW OF REACTOR - LIGHT LOADING

LEGENDS

ALL Q-TUBES LOADED WITH 1.3 in. OD FUEL ELEMENTS ARE MARKED 1.3

ALL Q-TUBES LOADED WITH 1 in. OD FUEL ELEMENTS ARE MARKED 1

FOILS LOCATED 8 in. ABOVE AND BELOW CENTER LINE OF ACTIVE CORE

CRITICAL HEIGHT OF ACTIVE CORE: 82 in.

FIG. 16
CONVERSION RATIO EXPERIMENT. PLAN
VIEW OF REACTOR - INTERMEDIATE LOADING

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LEGEND

ALL Q-TUBES LOADED WITH 1.3 in. OD FUEL ELEMENTS ARE MARKED 1.3

ALL Q-TUBES LOADED WITH 1 in. OD FUEL ELEMENTS ARE MARKED 1

FOILS LOCATED 8 in. ABOVE AND BELOW ACTIVE CORE

CRITICAL HEIGHT OF ACTIVE CORE: 103.4 in.

FIG. 17
CONVERSION RATIO EXPERIMENT. PLAN
VIEW OF REACTOR - HEAVY LOADING

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3. Results

Analysis is sufficiently complete for the moderator normalization tests to give preliminary results. It is expected that the fuel normalization runs will confirm these. The bucklings were measured in the region of interest by flux traverses. The average conversion ratio of the moderator normalization runs are tabulated.

Test Set	$B^2, \text{cm}^{-2} \times 10^6$	Conversion Ratio
2	530	0.94 ± 0.03
3	430	1.08 ± 0.03
4	300	1.14 ± 0.03

The errors given are relative probable errors. In addition, one has the systematic errors of 1.5% in $\sigma_{1a}^{U^{235}}$, and 2% in $\sigma_{1a}^{U^{238}}$.

B. Fuel Elements and Fuel Media

Experiments have been continued to evaluate fuel materials for high temperature, water-cooled reactor applications. Corrosion tests have been made on uranium-niobium alloys; the alloys have been tested either unclad or clad in a manner to simulate possible perforation of protective material. Preliminary tests were made to determine corrosion resistance of welded samples of several zirconium alloys. Other tests were initiated to study the use of low melting point metals as bond between clad and fuel material, realizing that these materials might have an application as a secondary or "self sealing" protective layer to retard corrosion.

1. Corrosion of Uranium-Niobium Alloys—W. W. Galbreath, Jr., A. H. Roebuck

Static autoclave corrosion tests in 500F distilled degassed water were made on clad and unclad uranium alloys melted by the Argonne Metallurgy Division.¹² Improved corrosion resistance was obtained in alloys containing either 3 or 6% niobium; less improvement was obtained in alloys containing either 1-1/2% zirconium - 1-1/2% niobium or 6% zirconium. The niobium alloys exhibit highest corrosion resistance after heat treating by quenching from temperatures above 1470F; alloys in the as-rolled condition or after aging at temperatures above 600F show lower corrosion resistance. Rates of weight change for normal uranium compared to the several uranium alloys in distilled degassed water at 500F are shown as follows:

¹²J. E. Draley, "Preliminary Report on Low Columbium-Uranium Corrosion Resistant Alloys," ANL-5078, June 24, 1953.

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	Rate of weight change, mg/(cm ²)(mo)
Unalloyed natural uranium	-550,000
U - 1-1/2% Zr - 1-1/2% Nb, heat treated	-35,000
U - 3% Nb, as rolled	-3,400
U - 6% Zr, heat treated	-2,900
U - 3% Nb, heat treated	-135
U - 6% Nb, as rolled	-80
U - 6% Nb, heat treated	-30

a. Unclad Alloys

Figure 18 shows corrosion data for the 3 and 6% niobium alloys at several temperatures. The 500F data for the 3% alloy, heat treated by quenching from temperatures above 1470F, show a weight change rate of about -135 mg/(cm²)(mo); this rate is relatively constant throughout the accumulated 1600 hr testing time. The 570F data for the quenched 3% alloy show slight increases in corrosion with time, followed by breaking of sample into several pieces after 943 hr. Other data, not plotted, show that the quenched 3% alloy has a still higher weight change rate, resulting in failure in less than 400 hr at 600-620F.

Data for the heat treated uranium - 6% niobium alloy show lower weight change rates at 500 and 600F than for the 3% alloy. Some 680F data, not plotted, show failure of 6% alloy in less than 200 hr. The 500F data for the as-rolled 6% alloy rolled at 850F show a weight change of approximately twice that observed for the heat treated 6% alloy and only one-half of that observed for the heat treated 3% alloy. In a 66 hr test in 500F degassed water, the rate of weight change for the as-rolled 6% alloy was -92 mg/(cm²)(mo), and the as-rolled 3% alloy disintegrated.

Table X shows the effect of aging on corrosion resistance. Increasing the aging temperature appreciably decreases the corrosion resistance of the uranium-niobium alloys. These results show that heat treated 3 and 6% niobium alloys of uranium do not age appreciably at 500F for the time tested. Corrosion effects observed at temperatures of 570 to 620F indicate lower corrosion resistance that may be due to either effect of temperature on the corrosion process or to thermal aging. Thermal aging at temperatures of 680F and above is apparently the primary factor involved in destroying the corrosion resistance obtained by rapid quenching from above 1470F. Other factors that may contribute to lower corrosion resistance are non-homogeneity of the alloy and the general impurity content of the alloy.

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Table X

EFFECT OF AGING URANIUM-NIOBIUM ALLOYS

Aging Temperature		Aging Time, Hr	Method of Aging	High-Temperature Corrosion Behavior of Alloys after Aging
F	C			
<u>U - 3% Nb</u>				
930	500	30	Glass ampule, evacuated	Disintegrated in 72 hr at 570F.
660	350	76-80	Glass ampule, evacuated	Two samples disintegrated in 72 hr at 570F.
600	316	48	Glass ampule, air filled	In 16 hr at 500F sample corroded 75 mg/cm ² ; 3400 mg/(cm ²)(mo).
570	300	950	Autoclave test	In 943 hr at 570 sample corroded 708 mg/cm ² ; 542 mg/(cm ²)(mo). See 570F data in Fig. 18.
500	260	1600	Autoclave test	In 1600 hr at 500F sample corroded 286 mg/cm ² ; 130 mg/(cm ²)(mo). See 500F data in Fig. 18.
<u>U - 6% Nb</u>				
930	500	40	Glass ampule, evacuated	Disintegrated in 72 hr at 570F.
680	360	160	Autoclave test	Disintegrated in 72 hr at 680F.
660	350	78	Glass ampule, evacuated	In 72 hr at 570F sample corroded 23 mg/cm ² ; 230 mg/(cm ²)(mo).
600	315	327	Autoclave test	In 327 hr at 600F sample corroded 222 mg/cm ² ; 490 mg/(cm ²)(mo).
500	260	1600	Autoclave test	In 1600 hr at 500F sample corroded 55 mg/cm ² ; 24.5 mg/(cm ²)(mo). See 500F data in Fig. 18.

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QUENCHED FROM ABOVE 1470F UNLESS NOTED

FIG. 18
CORROSION OF URANIUM-NIOBIUM ALLOYS
IN DEGASSED DISTILLED WATER

A. ROEBUCK: G.K., 9-25-53

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b. Clad Samples with Known Defects

Two types of tests were made to simulate the corrosion of canned uranium alloys following clad perforation. In both types of tests, water access to the uranium alloy was allowed through a "known defect" drilled hole. The first type of test, the capsule type, was made on alloy samples enclosed in, but not bonded to, capsules of stainless steel; capsules were placed in a visual sight apparatus for continuous short time observation or in static autoclaves with periodic long time observation. The second type of test was made on alloys bonded to the clad material.

Capsule Tests: Several capsule type tests at 450-460F were made in a visual sight apparatus on samples of normal uranium to study the swelling and rupture process. During the first portion of the test (about 2 hr) bubbles evolved rapidly through the drilled hole in the capsule; no outward signs of capsule deformation such as swelling were apparent. During the second phase (about 1 hr), few or no bubbles were evolved through the hole, and no swelling was apparent. The third and last phase (about 1 hr) of the test was initiated by a rapid evolution of bubbles followed by swelling and rupture. The rupture was not explosive in nature, but occurred from what appeared to be a localized buildup of corrosion products in the capsule. The observed weight change accompanying swelling and rupture were more rapid than could be accounted for by the average corrosion rate of unclad uranium at the temperature of test.

Samples of uranium - 3% niobium alloy quenched from above 1470F were tested in capsules and showed increased corrosion resistance, resulting in considerable lengthening of the time before swelling and rupture over that observed for normal uranium. One sample tested for 445 hr at 500F showed little or no corrosion during the first 300 hr of test; rapid corrosion leading to swelling and rupture followed in the last part of the test. Another sample was tested for 667 hr before any appreciable corrosion took place; this sample has started to swell and is undergoing further tests.

Analysis of the corrosion data shows that samples of partially clad uranium alloys exhibit higher weight change rates just prior to rupture. Comparison of the weight change rates of the bare versus clad uranium alloys shows that, although the initial rates are not greatly different, corrosion proceeds much faster in the capsule immediately prior to rupture. The corrosion process for capsuled uranium is thought to involve formation of substantial amounts of uranium hydride as an intermediate corrosion product during an induction period.

Similar capsule tests are being run on uranium - 6% niobium alloy at several temperatures. The data to date show that this alloy has increased resistance with respect to the uranium - 3% alloy. The induction period corrosion phenomena is also apparently involved in the corrosion of capsuled 6% alloys.

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These tests have shown that uranium alloys, having improved corrosion resistance, delay the failure of a simulated perforated can. However, the indication of an induction period corrosion phenomena suggests that, even if the corrosion resistance of an alloy is improved, acceleration of the corrosion process due to partial containment of corrosion products may still be anticipated if clad perforation is experienced. Factors such as the hydride reaction are apparently still important in the corrosion of clad and unclad alloys and should be considered.

Bonded Plate Tests: A test on a plate with one "known defect" drilled hole and containing uranium - 3% niobium roll bonded to Zircaloy 2 has been started. In 450 hr at 500F no swelling of the clad has occurred. Preliminary observations indicate that drilling small holes to simulate points of clad failure do not accelerate 500F water corrosion of core alloy bonded to clad, within existing test times up to 450 hr. However, sufficient test time for evaluation at 500F has not been accumulated.

2. Corrosion of Welded Zirconium Alloys - J. W. Frank, W. W. Galbreath, Jr.

Preliminary tests were made to study the effect of water corrosion at 500F on seam welds of zirconium and some zirconium alloys. A series of plates approximately 1 in. x 2 in. x 1/8 in. were cut from the following materials:

Material	Source
Grade I Crystal Bar	1/8 in. section sawed from billet (transverse specimen)
5% Sn-Zr Alloy	1/8 in. section sawed from billet (transverse specimen)
Zircaloy 2	Rerolled by ANL Metallurgy (longitudinal specimen)
Zircaloy 2 (75 ppm N)	1/8 in. section sawed from billet (transverse specimen)
0.1% Ta-Zr Alloy	1/8 in. section sawed from billet (transverse specimen)

Duplicate pairs of samples of each material were placed side by side so as to form a square. Using a heliarc process in the open air with a flow of about 30 cu ft/hr of argon, a seam weld was made without supplying any filler metal; the flow of argon was continued until the plates were no longer at a visible red heat (ca 900F). About 135 amp were used, except for the crystal bar and the rerolled Zircaloy 2 where the plates were somewhat thinner and the current was about 100 amp. The plates were reversed, and a weld put on the opposite side in exactly the same way. The

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plates were machined on one side to clean up and obtain a general surface flatness; due to warpage, much of the weld on one side was removed. Visual inspection showed weld penetration to the surface exposed except in the case of the one specimen of high nitrogen Zircaloy 2 (marked 102). Helium leak tests after welding showed no leakage except in the high-nitrogen sample.

After degreasing and pickling, specimens were wrapped with two turns of stainless wire to prevent large common areas of contact and placed in a single autoclave with degassed water. This autoclave was held at 500F for two weeks. Figure 19 shows the unmachined and machined faces, respectively, after the final two weeks of test. It is to be noted that the weld areas of the 5% tin alloy and the Zircaloy 2 specimens showed little more corrosion (white spots) than the other areas of the plates. Further, the areas that show these white spots are generally where the wire was in contact. The weld area of the crystal bar and the tantalum-zirconium alloy specimens were completely covered with the white corrosion product on the unmachined side, and rather badly attacked on the other side. A second helium leak test showed no leakage in any sample. Apparently the opening in the faulty plate was filled with corrosion product, thus becoming imper-vious to helium.

The plates were again degreased, lightly pickled, rewrapped with wire and corrosion tested for an additional 30 days under the same conditions. The appearance of the plates was little changed, except that in the case of the crystal bar and tantalum alloy, the corrosion product was somewhat heavier. There was no apparent change in the appearance of the other alloys.

The helium leak test will be repeated, and the plates will be retested for a period of 4 months.

3. Secondary Layer Protection - J. W. Frank, W. W. Galbreath, Jr.

An alternative or auxiliary method of protection against catastrophic failure of fuel elements in high temperature water may be the use of a secondary intervening layer probably, but not necessarily, metallic, to protect the fuel in case an opening should occur in the clad envelope. Low melting metals, reasonably low in nuclear cross section, offered a quick experimental approach. Lead (σ_a - 0.18 barns; mp-662F); bismuth (σ_a - 0.015 barns; mp-520F); tin (σ_a - 0.06 barns; mp-449F), and zinc (σ_a - 1.0 barn; mp-787F) were chosen. Samples of unclad metal and of metal contained in Type 347 stainless steel capsules having drilled holes for access of water were corrosion tested in 500F water. Unclad samples of lead, bismuth and tin showed very good corrosion resistance; lead and bismuth samples contained in the stainless steel were unsatisfactory. Tin melted in both tests, but the melted button showed a bright, clean surface, indicating no substantial amount of oxidation. The unclad sample of zinc reacted completely with water, thus eliminating the metal from further consideration.

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FIG. 19
 APPEARANCE OF WELDED ZIRCONIUM ALLOYS EXPOSED
 FOR TWO WEEKS IN 500 F DEGASSED WATER

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In view of the favorable corrosion results with lead and bismuth, specimens shown in Figure 20 were made using natural uranium cores, and the following combination of can and bond materials.

<u>Specimen</u>	<u>Can</u>	<u>Intermediate Metal</u>
A	Zr	Pb
B	Zr	Bi
C	Zr	Pb
D	Zr	Bi
E	Type 347	Bi
F	Type 347	Pb
G	Type 347	Bi
H	Type 347	Pb

Specimens were prepared as follows: the clad cylinder with welded bottom plug and drilled hole was placed vertically, and the natural uranium was inserted on the centering plug; molten metal was poured through a funnel and centering device to the level indicated; top plug was inserted and welded. Prior to final top closure, radiographs were taken with the side drilled hole in line and at 90° to the film, using a megavolt machine, to determine whether troublesome voids were present, and to verify the central position of the core. On the first trial, only three specimens were satisfactory: B, D, and F, as shown in Figure 21. Specimens A, C, E, and G were reprocessed, and, as shown in Figure 22, produced satisfactory specimens.

Each specimen was placed into an individual autoclave with degassed water and held at 500F for one week. Weight changes were erratic. In the case of specimen C (Zr-Pb), the weight loss was high, and there was some evidence of lead extrusion through the drilled hole. Radiographs were taken after test on a 250-kv machine to observe local corrosion effects in the neighborhood of the hole. The detail obtained on the radiographs was unsatisfactory, and the radiographs are not reproduced. Within the resolution obtained, no local effects were noted.

The specimens were replaced in autoclaves and held under the same test conditions for an additional 2 weeks. During this period, the furnace temperature rose above the melting point of bismuth. On removal there were droplets of bismuth in the bottom of the autoclaves. In addition, much extrusion of lead from can C (Zr-Pb) and a slight extrusion from can A (Zr-Pb) was found. No extrusion was found in the case of cans F and H (SS-Pb).

Radiographs were taken of the eight cans placed in the same two positions used previously and are shown in Figure 23. In can C, it is to be noted that the lower end of the uranium is corroded away. There has been

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J.W.FRANK :T.L., 12-10-53

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FIG. 20
SCHEMATIC OF TEST SPECIMEN

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B C D E F

DRILLED FACE

B C D E F

DRILLED FACE
REVOLVED 90 DEG (←)

SPECIMEN	B	C	D	E	F
CAN MATERIAL	Zr	Zr	Zr	TYPE 347	TYPE 347
BOND MATERIAL	Bi	Pb	Bi	Bi	Pb

FIG. 21
RADIOGRAPH OF SPECIMENS BEFORE CORROSION TEST

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H G E C A

DRILLED FACE

H G E C A

DRILLED FACE
REVOLVED 90 DEG (→)

SPECIMEN	H	G	E	C	A
CAN MATERIAL	TYPE 347	TYPE 347	TYPE 347	Zr	Zr
BOND MATERIAL	Pb	Bi	Bi	Pb	Pb

FIG. 22
RADIOGRAPH OF SPECIMENS BEFORE CORROSION TEST

NO
EXTRUSION
OF LEAD
OBSERVED

A B C D E F G H

DRILLED FACE

A	B	C	D	E	F	G	H
---	---	---	---	---	---	---	---

DRILLED FACE REVOLVED 90 DEG (→)

SPECIMEN	A	B	C	D	E	F	G	H
CAN MATERIAL	Zr	Zr	Zr	Zr	TYPE 347	TYPE 347	TYPE 347	TYPE 347
BOND MATERIAL	Pb	Bi	Pb	Bi	Bi	Pb	Bi	Pb

FIG. 23
APPEARANCE OF SPECIMENS
EXPOSED FOR EIGHT WEEKS IN 500F
DEGASSED WATER

some attack on the uranium specimens A and F slightly below the linear center. Specimen H is apparently unattacked. The specimens containing bismuth show the presence of material more dense than the can in spite of the loss of the bismuth; this is probably the oxide of uranium.

It would have to be a threefold coincidence if the attack on the uranium protected by lead stopped with the end of the test period. A more tenable explanation would be that the expansion of the uranium on oxidation extruded the lead and thereby sealed the unit against access of more water, thereby terminating corrosion.

It is realized that the $3/32$ in. thickness of lead used to expedite fabrication and to get experimental work started is unreasonable in reactor application. Further, the can thickness of $1/16$ in. is more than is desirable for reactor application. Of course, there is a lower limit to lead thickness, where insufficient lead for extrusion is provided; and also a lower limit to can thickness wherein the can would bulge rather than extrude the softer material. Experiments are continuing to further evaluate lead and to study the wetting properties of bismuth.

C. Reactor Coolant, Heat Transfer, and Power Generation

1. Heat Transfer Experiments

a. Boiling Density Tests - E. A. Wimunc

A detailed description of the natural circulation boiling density test procedure was previously reported.¹³

Three direct-resistance heaters were tested for burnout and density determinations. Two identical heaters were fabricated from an 0.0312 in. thick stainless steel Type 304 sheet 24 in. high and 16 in. long and formed into an accordion-shaped rectangle 5 in. by $1-3/32$ in. by 24 in. with $1/4$ in. channels between folds. Two copper plates ($1/4$ in. by 5 in. by 30 in.) were brazed to the stainless sheet and $1-1/2$ in. diameter copper current-carrying bars were fixed to these plates so that the current would be well distributed to the stainless steel sheet. A third test section differed only in that the length was 10.8 in. and the channels were $1/4$ in. on each side of a $1/2$ in. channel.

Heater No. 1: A series of density readings versus power input was made; the results are plotted in Figure 24. The lowest power input to the heater was limited by the rectifier characteristics; the curve is extrapolated from this point to zero power input.

¹³ Reactor Engineering Division Quarterly Report, ANL-5060, June 15, 1953.

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FIG. 24
 POWER INPUT PER UNIT
 VOLUME VS PER CENT VOIDS

RE-7-12049-A
 E.A. WIMUNC: S.H., 12-14-53

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A similar subassembly fabricated from "A" Nickel had been previously tested, and the results are shown in Figure 25. A comparison of Figure 24 with the corresponding atmospheric run, Figure 25, indicates a wide variation in the measured voids between the two similar test sections. This discrepancy in steam volume fraction can only be attributed to the major differences in the two test conditions.

The stainless unit with the smaller number of channels had a correspondingly larger downcomer area than the nickel unit. To facilitate flow metering, the copper plates were extended 6 in. at the bottom section of the stainless unit. A pitot tube was inserted between the copper extensions and just below the stainless steel folds. The pitot varied between 0.8 - 2.0 cm differential of water at the various power levels.

The heater burned out at 109 kw/liter, which corresponds to a heat flux of 140,000 Btu/(hr) (sq ft). Burnout was indicated by a fluctuating voltage, and the power to the heater was immediately turned off just as cherry-red steel particles erupted from the boiler. The partially disassembled burned out heater is shown in Figure 26.

Heater No. 2: A second heater was tested to burnout, and the results are shown in Figure 27. This test was similar to the one reported above except for the channel spacing. The results of a previous test on similar subassembly made of "A" Nickel, with ten 1/2 in. channels, are shown in Figure 28 for comparison.

The first run of this test was made on July 3, with density measurements being determined near the top of the heater. No burnout occurred during this run. On July 7 a similar test was conducted with density determinations being made at the mid-height of the heater. The heater burned out just as the power level reached 124 kw/liter, which corresponds to a heat flux of 232,000 Btu/(hr) (sq ft). Considerable "puffing" of steam could be heard erupting from the heater just prior to burnout. The lateral displacement of the copper plates was approximately 1/8 in. The burned out heater is shown in Figure 29.

Heater No. 3: This heater was identical to heater No. 1 and was tested to obtain supplemental data at points other than near the top of the heater.

The maximum voltage limitation of the rectifiers prevented a higher power input, and thus burnout was not attained. From the previous test it might be assumed that the heater would have burned out with a slight increase in power, since the curves were in close agreement.

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E.A.WIMUNC:E.F., 8-20-53

FIG. 25
POWER INPUT PER UNIT VOLUME
VS PER CENT VOIDS

RE-7-11514-A

FIG. 26
PARTIALLY DISASSEMBLED TYPE 304 RESISTANCE
HEATER FOLLOWING BURNOUT AT 109 kw/LITER

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RE-7-12048-A
 E.A. WIMUNC: S.H., 12-14-53

RE-7-12048-A

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E.A.WIMUNC:E.F., 8-21-53

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FIG. 28
POWER INPUT PER UNIT VOLUME
VS PER CENT VOIDS

RE-7-11518-A

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FIG. 29
PARTIALLY DISASSEMBLED TYPE 304
RESISTANCE HEATER FOLLOWING
BURNOUT AT 124 kw/LITER

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Two separate tests were made: one on July 27 and the second on July 29. Since burnout did not occur on the first test, higher power input tests were made to determine densities at 7-1/2 in. and 15 in., respectively, below the top of the heater. The results are given in Figures 30 and 31. At present there is no explanation for the discrepancy of the two tests on the same heater, except that the water may not have been of the same quality.

The test apparatus has been dismantled and is being replaced with a larger unit.

b. Large-Scale Boiling Heat Transfer Test - E. A. Wimunc

A large-scale test apparatus has been designed and is in the process of being assembled.

Tests will be made to determine the amount of heat that can be removed from the heater by natural convection of the water, the density of the fluid in the heater, water velocity, plate temperatures, film coefficients, effect of pressure, etc.

The apparatus has been designed for a working pressure of 250 psig and a corresponding saturation temperature of 406F. Test sections having an active length of 5.6 ft can be used.

The first heater to be tested will be a shrouded 7-plate stainless steel active section 5 ft long with inactive copper plate lengths on each end. The plates are 1/8 in. thick by 3.42 in. wide with a channel spacing of 1/2 in. between plates.

c. Single Pulse Heating Experiment - H. H. Hooker, R. J. Schiltz
Apparatus and Instrumentation

In order to determine the approximate dimensions of a suitable test section and to test the performance of an electrical pulsing device, preliminary tests were made utilizing a small 10 kva transformer and an existing test section. As a result of satisfactory performance during these tests, a tubular copper test section 6 in. long, 0.312 in. ID with 0.015 in. wall, was fabricated for use with the 130 kva transformer. Suitable terminals and connecting bus bars were designed and fabricated.

Voltage is measured between the ends of the test section. Current is measured by utilizing a short length of connecting bus bar as an ammeter shunt in such a manner as to minimize inductive pickup. A fast-response, copper-constantan thermocouple, with wires welded directly to the outside surface of the test section, is used for temperature measurement. Thermocouple pickup of the heating voltage pulse is balanced out using a bridge circuit.

FIG. 30
 POWER INPUT PER UNIT
 VOLUME VS PERCENT VOIDS

RE-7-12052-A
 E.A.WIMUNC:S.H., 12-11-53

RE-7-12052-A

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FIG.31
 POWER INPUT PER UNIT
 VOLUME VS PERCENT VOIDS

RE-7-12053-A

RE-7-12053-A
 E.A. WIMUNC: S.H., 12-11-53

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In order to correlate internal steam pressure with time of application of the power pulse, a pressure transducer is coupled to the lower end of the test section. A direct-coupled amplifier was designed and constructed so that pressure transducer signal may be used to drive one of the oscillograph galvanometers.

In certain tests the time interval between application of the power pulse and the expulsion of water from the upper end of the test section is measured. For this purpose a probe is situated approximately 2 mm above the upper end of the test section. An oscillograph galvanometer is connected in series with the probe, a battery, and the test section.

High-speed moving pictures (approximately 4000 frames per second) were made of the expulsion of water during some tests. For these tests, a glass extension was added to the upper end of the test section. An aluminum float at the top of the water column was used as an aid to visibility. The initiation of the power pulse was made visible by means of a flashlight bulb which, when connected to the heating transformer, flashed and burned out in less than one millisecond after pulse initiation.

Transformer flux resetting is accomplished by momentarily connecting a 6-v storage battery to the transformer primary with polarity opposite to that of the power pulse.

Results

A series of test runs was made in which electrical input, length of power pulse, and initial water temperature was varied. Some difficulty with double pulsing was experienced. This has, to a large extent, been eliminated.

The results of typical tests are summarized below as taken from oscillograph recordings.

Run No.	38	40*	68	69
Pulse length, ms	8	8.75	6.4	6.4
Volts, peak	23.1	35.2	25.7	25.3
Amperes, peak	38,500	43,200	38,700	40,900
Kilowatts, approximate peak	820	1700	920	950
Time delay before pressure surge, ms	3.6	2.6	2	3.3
Time delay before water expulsion, ms	6.1	5.4		
Initial water temperature, C	27	27	90	28
Maximum surface temperature, C	525	750	600	498

*Test section burned out after approximately 8.75 ms.

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D. Instrumentation and Control

1. Instrumentation

Speed of Thermocouple Response - W. Kropp (du Pont), A. J. Ulrich

Parameters involved in the speed of thermocouple response were investigated as a basis for the design of thermocouples useful in the transient phase of the Boiling Reactor Experiment. The speed of thermocouple response to a temperature step function was determined as a function of wire radius. These results make possible an estimate of the thermocouple wire size necessary to follow a rapid temperature variation.

Thermocouple response is much improved if the material whose temperature is being measured is a series element in the thermocouple circuit. Then there is no time lag while heat is conducted into an adjacent thermocouple junction. Unfortunately, the attached thermocouple wires allow diffusion of heat to or from the material, so that the junction temperature may differ from the temperature of the main body of the material.

In the present experiment, a molten tin-bismuth alloy is the body whose temperature is being measured. A chromel wire is always immersed in the alloy so that the emf produced at this junction is constant. A copper wire at room temperature is suddenly plunged into the molten material initiating a thermal transient in the wire, the junction and the region of the bath around the junction, and at the same time completing the thermocouple circuit (Figure 32). The emf due to this copper bismuth-tin temperature transient appears on the oscilloscope added algebraically to the constant emf due to the chromel-bismuth-tin junction. When the two junctions reach the same temperature the combined emf is equivalent to the emf of a copper-chromel thermocouple at the temperature of the hot body because the emfs due to the bismuth-tin alloy cancel. With the circuit shown, the fact that the copper is initially cooler than the alloy produces a thermoelectric signal of the same polarity as that of the copper-chromel equilibrium signal for the alloy temperature. Thus the observed signal decreases from a maximum to an equilibrium value. The transient for which the response time is determined is observed as the difference between the initial signal and the equilibrium signal.

The surface temperature transient of an infinitely long cylinder subjected to a sudden change in ambient temperature is given by¹⁴

$$\frac{\theta_s}{\theta_i} = \sum_{v=1}^{v=\infty} \frac{2}{X_v} \frac{J_0(X_v) J_1(X_v)}{J_0^2(X_v) + J_1^2(X_v)} e^{-\frac{X_v^2 \alpha T}{S^2}} \quad (1)$$

¹⁴M. Jakob, "Heat Transfer," (New York: John Wiley and Sons, Inc., 1949), Vol. I, p. 277, Eq. (13-70).

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FIG. 32
THERMOCOUPLE CIRCUIT DIAGRAM

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where:

θ_i = the initial difference between the cylinder and ambient temperatures, C

θ_s = difference between surface and ambient temperatures, C

α = thermal diffusivity of cylinder, $\frac{\text{cm}^2}{\text{sec}}$

S = radius of cylinder, cm

τ = time from beginning of transient, sec

X_v = the v 'th root of the following equation:

$$\frac{ns}{bs} = \frac{J_0(ns)}{J_1(ns)}$$

where

b = logarithmic derivative of θ at the surface, and n is a separation variable, $0 \leq n < \infty$

The first term of Equation (1) is sufficient to express the behavior for $\tau \geq 10^{-3}$ sec, which is the region of interest. Keeping only the first term and defining

y = transient component of trace deflection, cm
 $= k\theta_s$, where k = proportionality constant, $\frac{\text{cm}}{\text{C}}$

Equation (1) becomes

$$y = k\theta_i \frac{2}{X_1} \frac{J_0(X_1) J_1(X_1)}{J_0^2(X_1) + J_1^2(X_1)} e^{-\frac{X_1^2 \alpha \tau}{S^2}}$$

Defining y_0 = initial transient component of trace deflection (cm), Equation (1) simplifies to

$$y = y_0 e^{-\frac{X_1^2 \alpha \tau}{S^2}}$$

or

$$\ln y - \ln y_0 = -\frac{X_1^2 \alpha \tau}{S^2}$$

In terms of common logarithms, $\log y - \log y_0 = -0.4343 \frac{X_1^2 \alpha \tau}{S^2}$

The circuit is shown in Figure 32. The copper wire of the thermocouple circuit and wire connected to the trigger circuit are plunged into the bath in the same motion. The time delay between the two contacts is controlled by the relative position of the wire tips. The depth of submersion is fixed at 3 to 6 wire diameters by a stop. Traces are rejected

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if freezing of the alloy occurs. The sweep speed used in the calculations is the nominal one. The transient deflections are determined from photographs of the oscilloscope screen. A typical picture is shown in Figure 33. A set of data derived from such traces is plotted in Figure 34. The measured and calculated results are summarized.

Copper Wire Size No.	ΔT for 1 Decade ⁽¹⁾ of Decay, sec.	Slope of Semi-log Plot, sec ⁻¹	S, cm	$X_1^{2(2)}$	Remarks
14	0.212	- 4.72	0.0813	0.059	X_1^2 calculated from S and ΔT
20	0.080	- 12.5	0.0406	0.039	X_1^2 calculated from S and ΔT
26	0.024	-41.0	0.0202	0.032	X_1^2 calculated from S and ΔT
40	0.001	$\sim 10^3$	0.0039	0.029	ΔT calculated assuming X_1^2 and S

(1) Also ΔT = time for 90% response to step function

(2) $\alpha = 1.22 \text{ cm}^2/\text{sec}$ for copper at the temperatures which occur in this experiment.

The curvature shown by the points plotted in Figure 34 is probably due to the combination of radial conduction and conduction into the part of the wire above the bath. Because the ratio of effective heat transfer surface to volume of metal for conduction up the wire is less than that for the infinite cylinder, a slower response is to be expected. The error due to the finite depth of immersion of the wire is minimized by using the initial slope for the calculation of X_1^2 . However, the curvature is sufficiently small that conclusions about over-all response time for thermocouples drawn from X_1^2 values are approximately correct even though conduction is not that of an infinite cylinder.

The emf produced in a thermocouple is the average of the local emfs produced at each element of area junction, but these local emfs are weighted in the average in favor of those elements near the lead out wire. Consequently, the dominant emf measured here is that due to the element of contacting area nearest the surface of the bath.

Wetting of the copper by bismuth is improved appreciably by the addition of tin, but variation in the slope of the transient due to poor contact is evident among the plots. The fastest response observed in 4 or 5 measurements with a given wire size is used in calculations because it represents the case with the best thermal contact and the least oxide coating.

The data for No. 40 copper wire agree with the calculations, but only qualitatively, because disturbances due to multiple contacts and finite immersion time occur during the first few milliseconds.

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FIG.33

TIME INCREASES FROM RIGHT TO LEFT. THE HORIZONTAL GRID LINES ARE 1 cm APART, THE VERTICAL ONES 0.5cm APART. FOUR TRACES APPEAR FOR EACH RUN. THE TOP AND BOTTOM ARE CALIBRATION LINES AT 0mv AND 15 mv. THE CURVED TRACE IS THE TRANSIENT, AND THE REMAINING TRACE IS OF THE THERMOCOUPLE AFTER THERMAL EQUILIBRIUM IS ESTABLISHED. THE NOMINAL SWEEP SPEED IS 10ms/cm. RUNS (a) AND (b) IN FIG.34 ARE SHOWN HERE.

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FIG. 34
TRANSIENT COMPONENTS OF THERMOCOUPLE
EMF AS A FUNCTION OF TIME AFTER
CONTACT FOR 20-GAUGE COPPER WIRE

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Information more pertinent to peened thermocouples is provided by similar experiments in which an iron point at room temperature is suddenly brought into firm contact with a hot aluminum plate. Slopes on a semi-log plot are about $-2.7 \times 10^3 \text{ sec}^{-1}$, the fastest being $-6.5 \times 10^3 \text{ sec}^{-1}$. The areas of contact, determined by an optical comparison of pit marks and wires, are of the order of the cross section of a 33 to 40 gauge wire.

A slope of $-4 \times 10^3 \text{ sec}^{-1}$ with 36-gauge wire ($S = 0.00635 \text{ cm}$) gives $X_1^2 = 1.83$. For 90% response in 10^{-3} sec ($\Delta\tau = 10^{-3} \text{ sec}$) and $X_1^2 = 1.83$, the wire size needed is No. 30 gauge ($S = 0.0127 \text{ cm}$). The improved response over the case of copper-bismuth-tin, where No. 40 gauge wire is required, can be explained in terms of the thermal conductivities and diffusivities of the materials and the improved thermal contact at the junction.

These experimental observations indicate that for a peened junction of thermocouple materials having heat transfer properties comparable with iron and aluminum, No. 30 gauge wire is small enough to follow a step function change in temperature to 90% in 0.001 sec.

Welded junctions should provide thermocouples with somewhat faster response because of the better thermal contact at the junction.

One of the temperature transients of most interest is a temperature excursion where power is produced exponentially in a fuel plate. There follows a calculation of the response of a thermocouple to this type of excursion when the response to a step function is known. When the response time for a step function is 0.001 sec, and the period of the exponential is 0.01 sec, the indication is too small by 9%.

Power per cc is of the asymptotic form $P(t) = P_0 e^{t/\tau}$, where t = time, P_0 = constant, τ = period of the exponential. The energy generated per cc is

$$\begin{aligned} E(t) &= \int_0^t P(t') dt' \\ &= P_0 \tau \int_0^t e^{t'/\tau} \frac{dt'}{\tau} = \left[P_0 \tau e^{t'/\tau} \right]_0^t = P_0 \tau \left[e^{t/\tau} - 1 \right] \end{aligned}$$

If ρ = density, c = specific heat, $\Delta T(t)$ is the temperature of the body minus the initial temperature as a function of time, and if no heat is lost

$$\rho c \Delta T(t) = P_0 \tau \left[e^{t/\tau} - 1 \right]$$

then

$$\Delta T(t) = \frac{P_0 \tau}{\rho c} \left[e^{t/\tau} - 1 \right] = T_0 (e^{t/\tau} - 1)$$

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The response to a step temperature change is known to be of the form $\theta = \delta (1 - e^{-at'})$. The response to an exponential excursion.

$\Delta T = T_0 \left(\frac{t'}{e^\tau - 1} \right)$ can then be calculated using the method of superposition of step functions (Figure 35).

θ_i = increment in temperature indication of thermocouple occurring at time t ($t = n\Delta t$), due to step function change in temperature δ_i occurring at time t' ($t' = i\Delta t$).

t = time for which the response for the exponential is to be calculated.

$t - t' = (n - i)\Delta t$ = time of response for the i th step.

δ_i = incremental step at time $i\Delta t$.

$a = \frac{X_1^2}{S^2} (\text{sec}^{-1})$ = response coefficient of the thermocouple calculated from experiment.

FIG. 35

$$\theta_0 = \delta_0(1 - e^{-an\Delta t}) = 0, \text{ since } \delta_0 = 0$$

$$\theta_1 = \delta_1(1 - e^{-a(n-1)\Delta t})$$

$$\theta_i = \delta_i(1 - e^{-a(n-i)\Delta t})$$

$$\theta_{n-1} = \delta_{n-1}(1 - e^{-a\Delta t})$$

$$\theta_n = \delta_n(0) = 0$$

Substituting for δ_i in θ_i gives

$$\theta_i = \frac{T_0}{\tau} e^{\frac{i\Delta t}{\tau}} (1 - e^{-a(n-i)\Delta t}) \Delta t$$

$$\theta = \sum_{i=0}^n \theta_i = \sum_{i=0}^n \frac{T_0}{\tau} e^{\frac{i\Delta t}{\tau}} (1 - e^{-a(n-i)\Delta t}) \Delta t$$

$$\Delta T = T_0(e^{t'/\tau} - 1)$$

$$d\Delta t = T_0 e^{t'/\tau} \frac{dt'}{\tau}$$

Letting $d\Delta T$ and dt' become finite gives

$$\delta_i = T_0 e^{\frac{i\Delta t}{\tau}} \frac{\Delta t}{\tau}$$

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Letting $\Delta t \rightarrow 0$ and $n \rightarrow \infty$ gives

$$n\Delta t \rightarrow t, i\Delta t \rightarrow t', \Delta t \rightarrow dt'$$

$$\begin{aligned}\theta &= \int_0^t \frac{T_0}{\tau} e^{\frac{t'}{\tau}} (1 - e^{-at} e^{at'}) dt' \\ &= \int_0^t \frac{T_0}{\tau} e^{\frac{t'}{\tau}} dt' - \int_0^t \frac{T_0}{\tau} e^{-at} e^{\frac{t'}{\tau} + at'} dt' \\ &= T_0 \int_0^t e^{\frac{t'}{\tau}} \frac{dt'}{\tau} - \frac{T_0 e^{-at}}{\tau(\frac{1}{\tau} + a)} \int_0^t e^{(\frac{1}{\tau} + a)t'} (\frac{1}{\tau} + a) dt' \\ &= T_0 \left[e^{\frac{t'}{\tau}} \right]_0^t - \frac{T_0 e^{-at}}{\tau(\frac{1}{\tau} + a)} \left[e^{(\frac{1}{\tau} + a)t'} \right]_0^t \\ &= T_0 \left\{ \left[e^{\frac{t}{\tau}} - 1 \right] - \frac{1}{a\tau + 1} \left[e^{\frac{t}{\tau}} - e^{-at} \right] \right\}\end{aligned}$$

For $\tau = 0.01$ sec and $\frac{1}{a} = 0.001$ sec

$$\theta = T_0 \left\{ \left[e^{100t} - 1 \right] - \frac{1}{11} \left[e^{100t} - e^{-1000t} \right] \right\} \approx T_0 \frac{10}{11} e^{100t}$$

2. Control Rod Drive Development - J. N. Young

Studies are being directed toward the development of improved and simplified control rod drive mechanisms. Current emphasis is on a solenoid-operated step-motion linear drive mechanism.

The design (Figure 36) consists of 3 solenoid mechanisms: one for holding or braking, one for lifting, and one for scrambling the rod. The mechanisms clutch a 1-3/16 in. diameter extension of the control rod. A short magnetic section at the top serves as the armature of the magnetic position indicator.

The lift solenoid has a movable armature and movable grip mechanism, the distance of travel being approximately 1/8 in. When the solenoid is not energized the movable armature is held down by a Belleville spring and by gravity; the ball gripper is inactive. When the solenoid is energized, the movable armature starts upward, and the ball gripper engages the control rod. Since the brake armature is also holding the rod, no movement results until the brake solenoid is de-energized. When the brake gripper is released, the lifting solenoid completes its travel, lifting the rod approximately 1/8 in. Re-energizing the brake solenoid again secures the rod in a fixed position.

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The brake solenoid has a movable armature and a fixed gripper mechanism. When the solenoid is not energized, the ball gripper is inactive. When the solenoid is energized the movable armature is raised slightly and the balls, forced inward by the tapered cup, bear against the gripper shoes which hold the rod suspended.

To lower the rod, the lift solenoid is held de-energized, and the brake solenoid is momentarily de-energized to permit the rod to fall freely in predetermined small increments of travel.

The scram mechanism is shown in the partially scrambled position. When the lifting mechanism raises the rod, the latter carries the movable scram armature by virtue of the ball gripper at its lower end, compressing the scram spring. Once the movable scram armature has been raised to contact its mating fixed armature, it is held by the magnetic field set up by the scram solenoid coil. Simultaneously, the gripper unloading sleeve disengages the ball clutch at the lower end. This represents the normal position of the scram solenoid mechanism.

The rod is scrambled by de-energizing the coil which permits the scram spring to force the movable scram armature downward. After 3/16 in. of travel the ball clutch is re-engaged, gripping the control rod and forcing it downwards.

The energizing of the solenoids in the desired sequence of lifting, holding, lowering or scrambling is accomplished by a motor-driven switching device.

The lift and brake solenoid mechanisms have been built. The inside magnetic components are of USS Stainless W; the balls and shoes are Type 440 stainless steel; and the control rod extension shaft is bronze.

Both mechanisms were tested for 104 hr in 120F water during which time the lift and brake solenoids, powered by an external iron flux path, operated 200,000 and 500,055 times, respectively, in lifting and lowering a 75-lb load (85 lb weight in water) with a travel of 35 to 45 in. During operation, the lift coil used 360w and the brake coil 220w. The lifting was positive, resulting in 1/8 in. movement of the rod for each lifting cycle. The lowering movement, however, was found to be erratic; the increments of travel varied to such an extent as to necessitate redesign of the mechanism.

A completely new mechanism has been designed (Figure 37) which includes the following modifications:

- a. A step-motion linear motor which will give precise step motion in both directions.
- b. Elimination of the release springs by reversing the polarity of the magnet coils.
- c. A spring collet device to replace the ball shoes, retainers, and retainer spring.

REACTOR RESEARCHNEUTRON PRODUCER (CP-6)Corrosion of Savannah River Fuel Slugs - E. A. Wimunc

A stainless steel water circulating loop, shown schematically in Figure 38, was built so that corrosion effects on CP-6 canned uranium slugs with known defects could be studied by electrically heating these slugs to simulate pile conditions.

Four categories of slugs were prepared:

- (1) Faulty welds and leakage paths to uranium;
- (2) Severe Al-Si penetration;
- (3) Uranium metal of poor structure, cracks, fissures, inclusions, etc;
- (4) Unwetted uranium areas.

Five slugs were welded in series with aluminum electrodes welded on each end so that the current could be applied for resistance heating through these electrodes. Figure 39 shows an assembly ready for testing in the loop. The slug housing (Figure 40) in the loop consisted of a 1-7/16 in. ID Pyrex tube surrounded by two larger lucite tubes acting as shields. This was a transparent housing so that examination of the slugs was possible at all times.

In the first series of tests, in which slugs numbered, from top to bottom, 6, 7, 19, 20, and 50 were tested at a power equivalent to 29 kw/ft which corresponds to a thermal heat flux of 359,000 Btu/(hr)(sq ft) and with 25 fps coolant velocity in the 1/8 in. annulus, the test was terminated after only 36 hours due to faulty operation of system safety interlocks.

Normally, as soon as the pump fails or a control switch shuts off the pump, the power to the slugs is automatically cut. In this particular instance when a control switch cut off the pumps, it was not realized that the power to the slugs was not cut off until it was observed that the slugs had reached a bright cherry red to orange glow. About 15 sec elapsed from the time the pump turned off to the time the power to the slugs was cut manually. The observed glow corresponded to about 760C (1400F) when compared with a heat color chart. Most of the aluminum clad from three slugs was melted away with large layers of uranium oxide up to 1/16 in. thick peeling away from the slug.

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FIG. 40
SLUG HOUSING

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Between the time the pump stopped and the time the power to the slugs was manually turned off, the water around the slugs was rapidly turned into steam, and the pressure build-up was relieved by the surge tank which was open to the atmosphere. The glass shattered, but the lucite tube shield retained the glass, aluminum, and uranium oxide and prevented the entrance of air. Figure 41 shows the results of the over-heating.

A second group of five slugs were tested for 1021 hours prior to a can failure. The test conditions were:

Power per slug, kw	25	35
Corresponding heat flux, Btu/(hr)(sq ft)	435,000	610,000
Calculated maximum peak core temp., F	509.1	642.3
Calculated average core temp., F	421.8	520
Average water velocity, fps	25	25
Average inlet water temp., F	173	168
Average outlet water temp., F	185	185
Specific resistivity of water, megohm-cm	4-6	4-6
Water pH	5.5-6.5	5.5-6.5
Average inlet pressure, psig	33	33
Average outlet pressure, psig	5	5

The slugs tested were numbered from top to bottom: Nos. 46, 24, 10, 23, and 14. Slugs Nos. 10 and 14 had defective uranium, that is, surface cracks, seams, and striations; Nos. 23 and 24 each had a non-wetted area about 1/2 in. in diameter; No. 64 had a simulated deep Al-Si penetration.

The slugs were run at 25 kw/slug for 451 hours followed by 35 kw/slug for 570 hours before slug No. 10 ruptured near the end cap (Figure 42). This slug was intentionally offset 1/8 in. from the other slugs to simulate a "cocked" slug in-pile. Erosion was first observed after 173 hours of operation. Measurements showed no lengthwise growth of the slugs. The only other indicative sign of a flaw occurred on slug No. 24 which had a non-wetted area on the uranium. All the slugs had a light rust-colored film over most of the aluminum surfaces even though the system had an ion-exchange resin column and filter bypass. The film could easily be rubbed off and the above mentioned flaw was located by a bright aluminum circle within the rust-colored deposit and is shown in Figure 43.

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FIG. 41
APPEARANCE OF SLUGS FOLLOWING
ACCIDENTAL BURNOUT

FIG. 42
RUPTURE OF SLUG NO. 10 AFTER 1021 HR
(451 HR AT 25 kw/SLUG; 570 HR AT 35 kw/SLUG)

A black and white photograph of a long, cylindrical metal object, possibly a pipe or a tool, lying horizontally. The object has a textured, weathered surface and is secured by a band or clamp at each end.

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A third series of slugs was also tested under similar conditions, at 35 kw/slug. The following slugs were tested, listed from top to bottom: Nos. 49, 25, 12, 39, and 18. Slug No. 12 was intentionally offset $1/8$ in. from the vertical axis of the slugs.

After 234 hours of testing, the end-cap stub of No. 18 melted away, probably due to a poor electrical bond between the slugs. The slug was then replaced with slug No. 17 and run an additional 194 hours before the glass housing broke. The break probably occurred because the slug may have been very close to the glass. Figure 44 shows slug No. 17 with about a square inch of aluminum melted away and exposing the Al-Si covered uranium. Slug No. 17 was then replaced with slug No. 8.

Slug No. 39 must have touched the glass after 431 hours, for the glass broke and some of the aluminum from this slug fused to the glass. An area of approximately $1/4$ in. by $3/4$ in. on the aluminum had melted away but did not expose the uranium.

A corrective measure to prevent the slugs from touching the glass was performed by using 4 spacer wires around the slugs instead of 3. The test was continued until 528 hours had accumulated on No. 39 before the flaw opened, blistered, and exposed the uranium (Figure 45).

Slug No. 17 was canned with a $1/2$ in. diameter non-wetted area $1-15/16$ in. from the bottom (outside measurement); the affected area began $2-1/4$ in. from the bottom.

Slug No. 39 was canned as a simulated Al-Si penetration of the can wall. A hemispherical defect was machined in the inner wall approximately $4-1/2$ in. from the bottom (outside measurement). A slightly pitted hole at this location and about $1/16$ in. diameter is noted on this slug and is shown in Figure 45.

The following tabulation lists the total hours of testing at 35 kw/slug.

<u>Slug Number</u>	<u>Defect</u>	<u>Hours</u>
49	Al-Si penetration	525
25	Non-wetted areas	525
12	Defective U	525
39	Al-Si penetration	528
18	Non-wetted area	234
17	Non-wetted area	194
8	Defective U	97

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FIG. 45
APPEARANCE OF SLUG NO. 39
TESTED FOR 528 HR AT 35 kw/SLUG

REACTOR PHYSICS

Rod Drop Experiments on ZPR-I - M. Ash, O. A. Schulze

Introduction

The rod drop affords a quick simple method for determining the worth of control rods in a critical reactor. Not only can it be used for calibrating a single rod, but it can also be used to determine quickly the worth of several rods taken together - a very tedious task by the positive period method.

As the name implies, a control rod is dropped into a critical reactor. From a proper recording of the delayed neutron production, the negative reactivity can be determined.

In the following it is shown that if the reactivity is introduced into the pile instantaneously and that if the mean effective neutron lifetime is zero, the neutron density will immediately drop by the factor $1/(1 + \Delta k)$, where Δk is the negative reactivity in dollars.

Consider the pile kinetics equations for thermal neutron density

$$n'(t) = \left(\frac{\rho - \beta}{\ell^*} \right) n(t) + \sum_{i=1}^6 \lambda_i C_i(t) \quad (1)$$

$$C_i'(t) = -\lambda_i C_i(t) + \frac{\beta_i n(t)}{\ell^*} \quad i = 1, 2, \dots, 6 \quad (2)$$

where

$$\rho = \text{reactivity} = \frac{k_{\text{eff}} - 1}{k_{\text{eff}}}$$

ℓ^* = mean effective neutron lifetime.

Assuming that the pile has run long enough prior to the drop that the delayed neutron emitters are in equilibrium, and considering very short times after the reactivity has been instantaneously introduced, the concentration of the i 'th neutron precursor will not differ appreciably from its equilibrium concentration; hence, in Equation (2),

$$C_i'(0) = 0$$

and

$$C_i(0) = \frac{\beta_i}{\lambda_i \ell^*}, \text{ letting } n(0) = 1 \quad (3)$$

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Substituting Equation (3) into Equation (1)

$$n'(t) = \left(\frac{\rho - \beta}{\ell^*} \right) n(t) + \frac{\beta}{\ell^*}$$

where

$$\beta = \sum_{i=1}^6 \beta_i$$

The solution to this differential equation is

$$n(t) = \frac{1}{\rho - \beta} \left[\rho e^{\frac{\rho - \beta}{\ell^*} t} - \beta \right]$$

If $\ell^* = 0$ and ρ is negative, the exponential term drops out. Denoting neutron density by n_1 ,

$$n_1 = -\frac{\beta}{\rho - \beta} = \frac{1}{1 - \frac{\rho}{\beta}}$$

but

$$\Delta k = -\frac{\rho}{\beta} = \text{negative reactivity in dollars}$$

therefore

$$n_1 = \frac{1}{1 + \Delta k}$$

One scheme for analyzing rod drop data is to extrapolate the decay curve back to $t = 0$ to determine n_1 ; however, for the data taken on the ZPR the accuracy of this scheme is questionable. Other methods for analyzing the data have been used and are explained in later sections of this report. All the methods, however, depend upon the solution of the pile kinetics Equations (1) and (2).

Purpose

Although the rod drop method has been used extensively by other groups, notably KAPL, so far as it is known the method has not previously been applied in calibrating the comparatively slow moving rods of the ZPR. Furthermore, since it had been decided to calibrate the rods of the Boiling Reactor Experiment in this manner, it was felt that a thorough-going experiment on the ZPR would help to establish the validity of the method as compared to the usual method of positive period measurement.

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Another important purpose of the experiment was to establish the method as a means of obtaining the worth of a large number of rods such as the 6 xenon rods in the ZPR.

Experimental Setup

A schematic of the ZPR core is shown in Figure 46. The central rod is at 1, while the xenon rods are labelled 2 through 7. The 12 shim rods were completely withdrawn from the core. The instrument used to detect the neutron signal was a Wattenberg chamber which was strapped to the outside of the plastic tank and located about 6 in. from the bottom of the core. Since this chamber was not gamma compensated, lead bricks were lowered into the water reflector to form a 4 in. gamma shield in front of the detector.

The signal from the detector was fed into a specially built preamplifier and was then transmitted to a scale changer in the control room. The changer made possible a scale amplification of 5 for accurately recording weaker signals. The signal was then fed into a Brush d-c amplifier which drove one of the pens of the Brush recorder. The other pen recorded a signal provided by the depression of the button which scrambled the central rod. An additional signal recorded the instant the tip of the central rod passed a limit switch located about 12 cm from the bottom of the core. At about this position the central rod began to decelerate due to the action of the shock absorbers.

An additional method for obtaining data on the rod drops was provided in the form of a lead-shielded detector attached to a counter. The counter was triggered by the passage of the central rod past a limit switch located near the bottom of the core. However, only 2 runs were completed satisfactorily using this equipment before difficulty occurred in the triggering circuits. For the remainder of the runs the counter was triggered manually.

The ZPR was in a particularly favorable condition at the time this experiment was run, inasmuch as the water was poisoned with approximately 6.5 kg of boric acid as a part of the experimental program. This allowed the 12 shim rods to be completely withdrawn, the reactor going critical with the remaining rods 70.4 cm from the bottom of the core.

Results

Positive period measurements to determine the differential worth of the central rod were taken. The results, including the integrated rod worth, are given in Figure 47.

A series of 7 runs were made in which the central rod was dropped from heights of 20 to 90 cm and the neutron level recorded. In each particular run the 6 xenon rods were maintained at the same height. Figure 48 is a reproduction of a portion of the Brush recording for run No. 5.

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FIG. 46
PLAN VIEW OF EXPERIMENTAL APPARATUS

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FIG. 47
CENTRAL ROD CALIBRATION. POSITIVE PERIOD METHOD.

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FIG. 48
REPRODUCTION OF BRUSH
RECORDING OF RUN NO.5

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In addition, 3 runs were made in which the central rod plus 1, 3, and 6 xenon rods were dropped. A repetition of the last run was made with the gamma shielding removed from the chamber and the counter.

The decay curves for the 10 runs are normalized and replotted in Figures 49 and 50.

Table XI lists the time measurements obtained for those runs in which the lower limit switch was operative.

Table XI

| <u>Run</u> | <u>Total
Distance
of Drop</u> | <u>Col. 1</u> | <u>Col. 2</u> |
|------------|---------------------------------------|---------------|---------------|
| 2 | 30 cm | 0.14 sec | 0.30 sec |
| 3 | 40 cm | 0.13 sec | 0.37 sec |
| 4 | 50 cm | 0.11 sec | 0.43 sec |
| 5 | 60 cm | 0.10 sec | 0.48 sec |

Column 1 = Time elapse between scram signal and a detectable fall-off in neutron flux.

Column 2 = Time elapse between scram signal and tripping of limit switch.

Analysis

All of the following methods for analyzing the rod drop data are dependent upon solutions to the pile kinetics Equations (1) and (2); however, there is a considerable variation in the approach to such solutions.

Comparison Method: The rod drop data is compared to the solution of the kinetics equation for instantaneous drops some time after the drop occurs. Figure 51 is a plot of negative reactivity versus neutron density ratio for times of 10 and 15 sec after the drop occurs. Knowing the experimentally obtained ratio, the reactivity is readily obtained from the graphs. This method is fairly accurate even though the rod does not drop instantaneously because even at 10 or 15 sec the decay curve is reasonably flat.

M. Ash Method: For convenience, Equations (1) and (2) are repeated with appropriate initial conditions.

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FIG. 49
NEUTRON DENSITY AFTER SHUTDOWN
(NORMALIZED TO STEADY-STATE = 1)

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FIG. 50
NEUTRON DENSITY AFTER SHUTDOWN
(NORMALIZED TO STEADY-STATE = 1)

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RE-7-12040-A

FIG. 51
 REACTIVITY VS NEUTRON DENSITY
 FOR INSTANTANEOUS ROD DROP
 $\ell^* = 10^{-4}$ sec

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$$n'(t) = \left(\frac{\rho - \beta}{\ell^*} \right) n(t) + \sum_{i=1}^6 \lambda_i C_i(t) \quad n(0) = 1, \quad n'(0) = 0 \quad (1)$$

$$C_i'(t) = -\lambda_i C_i(t) + \frac{\beta_i n(t)}{\ell^*} \quad C_i'(0) = 0 \quad i = 1, \dots, 6 \quad (2)$$

where:

$$\rho = \text{reactivity} = \frac{k_{\text{eff}} - 1}{k_{\text{eff}}}$$

ℓ^* = modified prompt generation time for thermal neutrons in an un-reflected pile. $\cong \frac{T_0}{k_{\text{eff}}(1 + L^2 B^2)} = \text{constant } (10^{-4} \text{ sec})$

$\frac{T_0}{1 + L^2 B^2}$ = prompt generation time for finite pile

$T_0 = \frac{1}{v \Sigma_a \text{ total}}$ prompt generation time for infinite pile

v = thermal neutron velocity

Integrating Equation (2) and substituting for $c_i(t)$ (the i 'th delayed neutron emitter concentration) in Equation (1) yields:

$$\sigma n'(t) = \kappa(t) n(t) - \int_0^t n'(\tau) f(t - \tau) d\tau \quad (3)$$

where:

$\kappa(t)$ = reactivity in dollars

$$\sigma = \ell^* / \beta \approx 0.0132$$

and $f(t)$ is the empirically determined fit of the delayed neutron yield curve.¹⁵ This curve is attributed to 6 groups of delayed neutron emitters, which form the basis for the 6 equations represented by Equation (2).

¹⁵D. J. Hughes, et al., "Delayed Neutrons from Fission of U^{235} ," Phys. Rev. 73, 2, (1948), pp. 111-124.

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The actual fit is:

$$f(t) = \sum_{i=1} \nu_i e^{-\lambda_i t} \quad f(0) = 1 \quad (4)$$

where:

$\nu_i = \frac{\beta_i}{\beta}$; and λ_i = decay constant of the i 'th delayed neutron emitter.

$f(t)$, ν_i , and λ_i are tabulated in Table XII.

Table XII

| t ,
sec | $f(t)$ | t ,
sec | $f(t)$ |
|--------------|---------|--------------|---------|
| 0.00 | 1.00000 | 8.00 | 0.29449 |
| 0.01 | 0.99148 | 9.00 | 0.27412 |
| 0.05 | 0.96328 | 10.00 | 0.25670 |
| 0.10 | 0.93672 | 11.00 | 0.24156 |
| 0.50 | 0.81648 | 12.00 | 0.22829 |
| 1.00 | 0.71167 | 13.00 | 0.21638 |
| 2.00 | 0.57936 | 14.00 | 0.20575 |
| 3.00 | 0.49323 | 15.00 | 0.19614 |
| 4.00 | 0.43134 | 20.00 | 0.15694 |
| 5.00 | 0.38458 | 30.00 | 0.11350 |
| 6.00 | 0.34809 | 50.00 | 0.06493 |
| 7.00 | 0.31872 | 100.00 | 0.01945 |

$$f(t) = \sum_{i=1}^6 \nu_i e^{-\lambda_i t}$$

$\nu_1 = 0.11067$	$\lambda_1 = 1.61$
$\nu_2 = 0.31621$	$\lambda_2 = 0.456$
$\nu_3 = 0.27668$	$\lambda_3 = 0.154$
$\nu_4 = 0.22398$	$\lambda_4 = 0.0315$
$\nu_5 = 0.03426$	$\lambda_5 = 0.0125$
$\nu_6 = 0.03821$	$\lambda_6 = 14.$

A central problem of pile kinetics is the analytical determination of solutions of Equation (3) for prescribed $\kappa(t)$'s. No generally satisfying method has yet been found. For each type of $\kappa(t)$, therefore, a particular scheme must be devised. Such a device is given herewith, for one type of $\kappa(t)$, which will further the intended purpose of this report.

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Consider a $\kappa(t)$ such that the neutron density does not fluctuate wildly over the range of interest. An integration by parts of the integral in Equation (3) yields:

$$\sigma n'(t) = [\kappa(t) - 1] n(t) + f(t) - \int_0^t n(\tau) f'(t - \tau) d\tau \quad (5)$$

Now $f'(t)$ has a rather sharp cutoff as can be seen from Table XIII. If $n(\tau)$ varies smoothly over the small region of time before $f'(t - \tau)$ cuts off, then in the integral of Equation (5) $n(\tau)$ can be replaced, to a good approximation, by the first two terms of its expansion about the upper limit. The result of this procedure converts Equation (5) to the following equation.

$$n'(t) [C_1 - t f(t) - g(t)] - n(t) [\kappa(t) - f(t)] = f(t) \quad (6)$$

where

$$C_1 = \sigma + \sum_{i=1}^6 \frac{\nu_i}{\lambda_i} = 12.42598$$

and

$$g(t) = \sum_{i=1}^6 \frac{\nu_i}{\lambda_i} e^{-\lambda_i t}$$

which is tabulated in Table XIII. From the definition of σ , it is seen that the dependence of Equation (6) on ℓ^* is practically non-existent. The solution to Equation (6) can now be written directly.

Equation (6), as well as being easily integrated, formally at least, for given $\kappa(t)$'s also provides here a convenient formula for determining $\kappa(t)$ when $n(t)$ is given. The rod drop experimental data gives the required type of $n(t)$ (Figures 49 and 50) for the range of times of interest. Turning Equation (6) around then yields:

$$\kappa(t) = -f(t) \left[\frac{1 - n(t)}{n(t)} \right] + \frac{n'(t)}{n(t)} [C_1 - t f(t) - g(t)] \quad (7)$$

Knowing $n(t)$ for a particular drop (with Tables XII and XIII), Equation (7) gives a straightforward means for determining $\kappa(t)$. The first term of Equation (7) is the preponderant one so that the possible inaccuracies involved in graphically determining the derivative are only secondarily reflected in $\kappa(t)$.

Equation (7) was used to determine the amount of reactivity introduced into the assembly for each of ten drops. The calculated reactivities are plotted in Figure 52. Those drops that consisted of only the central rod are

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FIG. 52
CALCULATED REACTIVITY AS A
FUNCTION OF TIME AFTER SHUTDOWN

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compared in Table XIV with the positive period determination of the reactivity of the rod. As can be seen, the agreement is quite good. This is explained in the following way.

Assume that the pile has initially experienced a change of reactivity and that now the neutron density is approaching its variation along a stable period. Then, $n(t) \simeq e^{\lambda t}$. Substituting this into Equation (3) gives:

$$\sigma\lambda = \kappa(t) - \lambda \sum_{i=1}^6 \frac{\nu_i}{\lambda_i + \lambda} \left[1 - e^{-(\lambda_i + \lambda)t} \right] \quad (8)$$

If the reactivity is positive ($\lambda > 0$), then Equation (8) becomes the in-hour equation after a relatively short time has elapsed. This is manifest experimentally when the central rod is calibrated using positive periods. For negative reactivities, $\lambda < 0$ and $|\lambda| \leq 0.0125$: this number is the decay constant corresponding to the delayed neutron group with the longest decay period. The condition on $|\lambda|$ must hold so that Equation (8) has the proper asymptotic behavior. The pile takes a very long time to reach a subcritical stable period corresponding to a $|\lambda| \leq 0.0125$, even after which more time must elapse as it approaches the inhour equation. This can be seen from the exponential term of Equation (8), the data curves (Figures 49 and 50), and the values of λ_i given in Table XII. Also, for long times, $n \approx \lambda^2 n$ is very small, which is the condition under which Equation (7) was established. In other words, the inhour equation is the asymptotic form of both Equations (7) and (8). Then determining the reactivity by using the positive period method, or Equation (7) should give close agreement.

However, Equation (7) is more widely applicable and easier to use, especially for negative reactivities, since it is not necessary to wait until the stable period is established. In practice, the procedure does not have to be as elaborate as given in Figure 52. Three or four spaced points on the neutron density curve taken after the prompt "jump" should suffice for a particular reactivity calculation.

The curves of calculated reactivity (Figure 52) for small values of time should not be taken too seriously, because the conditions under which Equation (7) is obtained are not valid there. It is doubtful that even Equation (3) holds for small times, especially for the larger (fast) reactivity drops, because of the generation of higher space modes¹⁶ (subsequently attenuated) of the neutron density which the kinetics Equations (1), (2), and (3) assume are always absent. So that numerically integrating Equation (3) to solve, as well, for the manner of introduction of $\kappa(t)$ into the pile may lead to erroneous results.

¹⁶A. Weinberg, H. C. Schweinler, "Theory of an Oscillating Absorber in a Chain Reactor," Phys. Rev. 74, pp. 851-863, October 8, 1948.

The fast cutoff of $f'(t)$ in Equation (5) means that the "memory" of Equation (5) (or Equation (3)) for the past history of the neutron density is poor. Or, after long times, the dependence of neutron density on how the reactivity was initially introduced is virtually erased. One can compare, therefore, neutron density curves for a given drop with the solutions of Equation (3) for constant (negative step function) reactivities which are well known.

Table XIII

| $t,$
sec	$ f'(t) $	$g(t)$
0.00	0.90740	12.41281
0.01	0.84711	12.40301
0.05	0.63363	12.36375
0.10	0.48321	12.31646
0.50	0.29868	11.96958
1.	0.17684	11.59028
2.	0.10687	10.95113
3.	0.07412	10.41833
4.	0.05483	9.95828
5.	0.04219	9.55180
6.	0.03346	9.18711
7.	0.02726	8.85476
8.	0.02268	8.54867
9.	0.01925	8.26511
10.	0.01662	8.00031
11.	0.01455	7.75168
12.	0.01288	7.51744
13.	0.01152	7.29559
14.	0.01037	7.08481
15.	0.00941	6.88438
20.	0.00621	5.94009
30.	0.00359	4.66523
50.	0.00178	2.93981
100.	0.00044	1.08994

$$f'(t) = - \sum_{i=1}^6 \nu_i \lambda_i e^{-\lambda_i t}$$

$$g(t) = \sum_{i=1}^6 \frac{\nu_i}{\lambda_i} e^{-\lambda_i t}$$

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C. KAPL Method¹⁷

The ratio of the integrated neutron density before and after the drop is correlated to the negative reactivity in dollars by the formula:

$$\Delta k = C (R-1)$$

where

$$R = \frac{\int_{-15}^0 n \, dt}{\int_0^{15} n \, dt} \quad \text{drop occurs at } t = 0, \text{ (t is in seconds)}$$

The factor C depends upon the decay properties of the delayed neutrons and upon the ratio R. It has been calculated from the kinetics equations assuming instantaneous drop time. C is monotonically decreasing from 0.409 at $R = 1$ to 0.367 at $R = \infty$. Figure 53 is a plot of reactivity vs. the ratio R.

In this experiment the ratio R was determined in two ways:

1. Integration

The ratio was obtained by numerically integrating the data of Figures 49 and 50. In order to compensate for the non-instantaneous drop time the curves were extrapolated back to some reasonable value of n_1 , and R was obtained by integration of this curve. From the Δk obtained, a new value of n_1 was calculated from the relation $n = 1/(1 + \Delta k)$ and the process repeated. Usually just one iteration sufficed.

2. Counter

As can be seen from Figures 49 and 50, a considerable error will be included in the denominator of R if the counter is started at the time the rod starts to drop. Hence, the counter was rigged to be triggered by the passage of the tip of the central rod past a limit switch located near the bottom of the core. Unfortunately this triggering mechanism was only operative for runs 4 and 5. The succeeding runs were triggered manually and are probably in error.

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¹⁷ "Reactor Physics; Progress Report November, December, 1951, January, 1952," Knolls Atomic Power Laboratory, KAPL-706, p. 18.

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FIG. 53
REACTIVITY VS THE RATIO R

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The results of these various methods are presented in Table XIV.

Table XIV

Run No.	Rods Dropped	Dist. Drop, cm	Pos. of Xe rods, cm	Pos. Period	Per Cent $\Delta k/k$				
					(A)		(B)	(C)	
					Comparison		M. Ash	KAPL	
					10 sec	15 sec		Int.	Counter
1	Central	20.0	94.6	0.17	0.15	0.15	0.15	0.15	---
2	Central	30.0	89.4	0.38	0.42	0.42	0.37	0.41	---
3	Central	40.0	83.1	0.72	0.77	0.78	0.72	0.78	---
4	Central	50.0	77.3	1.20	1.25	1.30	1.18	1.23	1.27
5	Central	60.0	72.8	1.48	1.60	1.52	1.50	1.50	1.65
6	Central	70.0	70.5	1.70	1.78	1.80	1.70	1.69	1.74
7	Central	90.0	69.2	1.82	1.88	1.90	1.80	1.78	2.12
8	Central
#2 Xe	70.4	70.4	---	2.94	3.00	2.95	2.89	4.25
9	Central
#2, 4, 6 Xe	70.4	70.4	---	7.10	6.85	6.90	6.81	7.58
10	Central
#2, 3, 4, 5,
6, 7 Xe	70.4	70.4	---	11.40	12.40	11.80	11.22	13.99
10*	Central
#2, 3, 4, 5,
6, 7 Xe	70.7	70.7	---	6.60	6.50	---	7.10	11.41

*No gamma shielding.

Conclusions

The good agreement between the methods of analyzing the rod drop data and the agreement between the rod drop and the positive period measurement clearly indicate the worthiness of the rod drop method for calibrating control rods in a reactor with free-falling control rods.

The analytic methods of determining reactivity from the rod drop data give approximately the same results except for the counter method which was not given a fair trial in this experiment. However, the counter method is probably the easiest and quickest to use, while the comparison method runs a close second.

The success of the method depends greatly upon an accurate gamma-free indication of the neutron density.

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FUEL ELEMENT DEVELOPMENT

Properties of Fuel Alloys - A. E. Dwight

A theory has been developed in which the effect of solute elements on the allotropic transformations in titanium, zirconium, and uranium is shown to vary in a periodic manner with the atomic number of the solute. A thermodynamic quantity serves as the index which determines whether a given solute will raise or lower the transformation temperature. The theory is discussed in detail in ANL-5091.¹⁸

COOLANTS

A. Organic Coolant Studies - W. K. Anderson

Tests have been initiated to evaluate the operational problem in utilizing diphenyl as a coolant for a thermal reactor. Three aspects of this problem considered first are: (1) the pyrolytic destruction of diphenyl; (2) its radiolysis; and (3) an engineering study of costs chargeable to coolant in a prototype reactor using diphenyl as a coolant.

Pyrolytic studies are under way at 500F, 600F, and 700F. Judged by melting point, no perceptible damage was inflicted to the diphenyl after one-month exposure at 500F and 600F. A sample of diphenyl containing 5% by weight of some of the more probable pyrolysis products was heated in an autoclave for two weeks at 700F. A great deal of the sample was lost by leakage, but examination of the residual material and of the walls of the autoclave revealed no caking of carbonaceous material and no sedimentation of solid material in the previously molten diphenyl.

Reasonable theoretical considerations lead to the expectation that radiolytic and pyrolytic destruction products of diphenyl should not differ too greatly. In an effort to test this hypothesis a sample of diphenyl, previously subjected to an nvt of 10^{18} in a CP-3' converter, was carefully fractionated in a small vacuum still. Approximately 0.06 gm of tarry material remained in the flask as a residue from the distillation of 10 gm of the irradiated diphenyl. A determination of the melting point gave 69.4C, thus indicating that the distillate was reasonably pure diphenyl.

As a source of pyrolytic material, several pounds of the tarry residues from the diphenyl purification stills at the Anniston, Alabama plant of Monsanto Chemical Company were procured. Infrared spectrograms

¹⁸A. E. Dwight, "Allotropic Transformations in Titanium, Zirconium, and Uranium Alloys," September, 1953.

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were run on carbon tetrachloride solutions of both the radiolytic tar described above and of the pyrolytic material obtained from Monsanto. Table XV shows the spectra obtained. The peak heights presented in this table are corrected by calculation to the same concentration. Examination of Table XV shows that although the strengths of the various absorption lines are not the same, there is enough basic similarity of the 2 spectra to lend considerable strength to the assumption that the two materials are of quite similar nature. In the cases of the lines at wave numbers of 1265 and 2950 cm^{-1} , where strong lines are recorded for the radiolytic material and definite zeros recorded for the pyrolytic material, there is strong reason to suspect that the radiolytic tar had suffered oxidation, since these lines are at wave lengths which are strongly absorbed by hydroxyl and other oxygen-containing functional groups.

These data at least lend some support to the belief that pyrolytic and radiolytic destruction products of diphenyl will be similar and that studies of systems consisting of mixtures of diphenyl with its pyrolysis products should yield information that would closely parallel that which should be obtained by studies of diphenyl which has been partially deteriorated by radiation. With this as a basis the behavior of mixtures of diphenyl and the pyrolytic tars were studied from the standpoint of their melting points and viscosities.

Data on melting points of mixtures of diphenyl and its pyrolysis products are plotted in Figure 54. The viscosities, in centistokes at 130°C , of similar mixtures are given in Figure 55. The results of these tests are encouraging since they indicate that quite high concentrations of the deterioration products of diphenyl can be tolerated without danger of deposition of solids at temperatures of interest and without decrease in pumping efficiency due to high viscosity of the deteriorating fluid coolant.

In order to obtain some more fundamental information on behavior of diphenyl mixed with probable products of its radiolysis, melting points of mixtures of diphenyl with the three isomeric diphenylbenzenes were also studied. Figures 56, 57, and 58 show melting points of the systems composed of diphenyl in admixture with o-diphenylbenzene, m-diphenylbenzene and p-diphenylbenzene, respectively. Only the p-diphenylbenzene system exhibits high melting points. Viscosities of these systems are being studied.

Work on development of an analytical method for determination of the amount of products of deterioration in diphenyl is progressing. Two methods are under investigation. The most promising is the use of a small vacuum distillation apparatus operated at about one millimeter of mercury absolute pressure. Although the method is being used on a tentative basis, further work is necessary before final specifications for the method can be issued.

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FIG.54
MELTING POINTS OF THE
DIPHENYL - TAR SYSTEM

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FIG.55
VISCOSITY IN CENTISTOKES AT 130C AS A FUNCTION OF COMPOSITION
OF DIPHENYL AND PYROLYTIC TARs

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FIG. 56
MELTING POINTS FOR THE SYSTEM
DIPHENYL-O-DIPHENYLBENZENE

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FIG. 57
MELTING POINTS FOR THE SYSTEM
DIPHENYL-m-DIPHENYLBENZENE

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FIG. 58
MELTING POINTS OF THE SYSTEM
DIPHENYL-p-DIPHENYLBENZENE

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Table XV

COMPARISON OF THE INFRARED SPECTRA OF RADIOLYTIC
AND PYROLYTIC MATERIALS

(Determined by Argonne Chemistry Division)

| Wave Numbers,
cm ⁻¹	Height of Absorption Peak (Relative)	
	Radiolytic	Pyrolytic
673	24	8
695	120	83
895	4	13
1005	*	15
1030	*	7
1075	*	13
1265	40	0
1410	1	15
1450	10	16
1480	40	50
1575	1	10
1600	20	32
1800	0	5
1875	1	6
1950	2	5
2950	20	0
3020	24	30

*Peak probably present but masked by solvent.

Tests for solid carbonaceous deposits are carried out by filtration through weighed, sintered glass filters held at 100C with subsequent copious washing of the filters with acetone prior to drying and reweighing.

Work on the radiation chemistry of diphenyl and comparisons of its radiolytic and pyrolytic products with that of o-diphenylbenzene deteriorations are in progress. Samples of both of these materials have been irradiated in a CP-3⁴ converter. Analysis of the products awaits satisfactory standardization of analytical methods.

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Calculations on the costs chargeable to coolant were made for a diphenyl cooled reactor. A prototype reactor was selected for which the core contained 25 tons of diphenyl and which would produce 300 mw of electrical power based on an assumed thermal efficiency of 25%. A differential equation expressing the relationship between R , the fraction of diphenyl in the coolant at time t , the fraction of diphenyl destroyed per 24-hour day (k_1), and the reaction of the total coolant reprocessed per 24-hour day (k_2) was shown to be:

$$\frac{dR}{dt} = -k_1 R + k_2(1-R) \quad (1)$$

Solution of this equation gives

$$R = \frac{k_2 + k_1 e^{-(k_1 + k_2)t}}{k_1 + k_2} \quad (2)$$

R. O. Bolt¹⁹ has reported considerable data on the pile-irradiation of diphenyl. Analysis of these data indicate that the rate of destruction of the material may not rise above 15% per 24-hour day in a reactor operating at a flux of 10^{14} n/(cm²)(sec). Assuming 0.10 for k_1 , the rate of diphenyl destruction, and 0.25 k_2 , the fraction of the coolant processed per 24-hour day, calculation using Equation (2) above shows that the system reaches a steady-state in diphenyl concentration by about the tenth day of operation. The concentration at this steady state can be determined by equating $\frac{dR}{dt}$ to zero in Equation (1). This operation yields:

$$R = \frac{k_2}{k_1 + k_2} \quad (3)$$

as the equilibrium value for the diphenyl concentration.

Distillation was selected as the most successful method for purification of the coolant.

Under the assumption that radiolysis would yield similar products to pyrolysis, it was recognized that the coolant could probably be used with relatively high concentrations of the destruction products carried in solution.

Calculation of the size of the reactor system based on thermal calculations of heat exchanger sizes indicated that the system should contain approximately 49 tons of diphenyl.

¹⁹R. O. Bolt, "Research on the Radiation Stability of Organic Fluids," California Research Corporation: AEC Report No. 4, January 9, 1953.

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Costs of the distillation plant based on an assumed design were estimated. Heat and material balances were used to calculate heat requirements for the process.

Costs of the plant and its maintenance and operation are shown in Table XVI. Design for values of k_2 less than 2 yielded such small plants that the minimum design size was set at this value of the reprocessing rate.

Table XVI

ESTIMATED COST OF PURIFICATION PLANT

| Reprocessing
Rate, per
24 hr, k_2	Plant Cost	Operating Cost, mill/kwh			
		Amortization	Maintenance	Labor	Total
2	\$ 910,000	0.035	0.004	0.017	0.056
3	1,130,000	0.043	0.004	0.017	0.064
4	1,350,000	0.052	0.005	0.017	0.074
5	1,570,000	0.060	0.006	0.017	0.083

The following equation was derived expressing the cost, C_d , for diphenyl replacement, as a function of k_1 and R :

$$C_d = 2.09 k_1 R \quad (4)$$

Another equation expressing the total cost for heat, C_h , to operate the purification unit is given as

$$C_h = 0.0083 k_1 R \quad (5)$$

Equations (4) and (5), and the plant cost data of Table XVI, were used to compute the data shown in Tables XVII through XXI. A summary of the total cost of purification for different values of R and k_1 is also given in Figure 59.

In brief, it may be concluded that:

1. Distillation is a reasonable method for constant purification of organic coolant used in a nuclear reactor. Cost of construction, maintenance, and operation of the plant is not prohibitive, and heat costs for its operation are insignificant.

2. The principal item of cost in maintaining the diphenyl coolant in a nuclear reactor will be that of replacement of the coolant destroyed by radiation and pyrolysis.

3. If rates of destruction indicated by preliminary data are not exceeded and if concentrations of tars approaching 47% can be tolerated, then the cost of maintenance of coolant should be below 0.25 mill per kilowatt-hour. Where higher destruction rates are found or where maintenance of higher purity coolant is demanded costs will be correspondingly higher.

Table XVII

COST DATA FOR MAINTAINING DIPHENYL CONCENTRATION AT
53% IN COOLANT

| Fraction
of Diphenyl
Destroyed
per day | Fraction of
Coolant Cir-
culated Through
Purifier per day | Weight Loss,
tons/day | Cost, mill/kwh | | |
|---|--|--------------------------|----------------|---------|--------|
| | | | Replacement | Heat | Total |
| 0.05 | 0.057 | 1.35 | 0.056 | 0.00023 | 0.1122 |
| 0.10 | 0.112 | 2.60 | 0.109 | 0.00043 | 0.1654 |
| 0.15 | 0.170 | 4.02 | 0.168 | 0.00066 | 0.2247 |
| 0.20 | 0.236 | 5.19 | 0.217 | 0.00093 | 0.2739 |
| 0.25 | 0.274 | 6.52 | 0.274 | 0.00107 | 0.3311 |
| 0.30 | 0.340 | 7.79 | 0.329 | 0.00122 | 0.3862 |
| 0.35 | 0.395 | 9.11 | 0.381 | 0.00156 | 0.4386 |
| 0.40 | 0.449 | 10.44 | 0.432 | 0.00181 | 0.4898 |
| 0.50 | 0.562 | 12.94 | 0.541 | 0.00216 | 0.5992 |
| 0.60 | 0.673 | 15.58 | 0.645 | 0.00262 | 0.6657 |

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Table XVIII

COST DATA FOR MAINTAINING DIPHENYL CONCENTRATION AT
60% IN COOLANT

| Fraction
of Diphenyl
Destroyed
per day | Fraction of
Coolant Cir-
culated Through
Purifier per day | Weight Loss,
tons/day | Cost, mill/kwh | | |
|---|--|--------------------------|----------------|---------|--------|
| | | | Replacement | Heat | Total |
| 0.05 | 0.075 | 1.47 | 0.062 | 0.00025 | 0.1183 |
| 0.10 | 0.15 | 2.94 | 0.144 | 0.00049 | 0.2005 |
| 0.15 | 0.275 | 4.41 | 0.187 | 0.00075 | 0.2438 |
| 0.20 | 0.30 | 5.88 | 0.254 | 0.00099 | 0.3110 |
| 0.25 | 0.375 | 7.35 | 0.312 | 0.0012 | 0.3692 |
| 0.30 | 0.45 | 8.82 | 0.373 | 0.00144 | 0.4304 |
| 0.35 | 0.525 | 10.29 | 0.435 | 0.0018 | 0.4928 |
| 0.40 | 0.60 | 11.76 | 0.490 | 0.0020 | 0.5480 |
| 0.50 | 0.75 | 14.70 | 0.628 | 0.0025 | 0.6865 |
| 0.60 | 0.80 | 17.64 | 0.747 | 0.0030 | 0.8060 |

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Table XIX

COST DATA FOR MAINTAINING DIPHENYL CONCENTRATION AT
70% IN COOLANT

| Fraction
of Diphenyl
Destroyed
per day | Fraction of
Coolant Cir-
culated Through
Purifier per day | Weight Loss,
tons/day | Cost, mill/kwh | | |
|---|--|--------------------------|----------------|---------|--------|
| | | | Replacement | Heat | Total |
| 0.05 | 0.116 | 1.72 | 0.071 | 0.00029 | 0.1273 |
| 0.10 | 0.233 | 3.43 | 0.144 | 0.00057 | 0.2006 |
| 0.15 | 0.35 | 5.15 | 0.217 | 0.00093 | 0.2739 |
| 0.20 | 0.467 | 6.86 | 0.287 | 0.00117 | 0.3442 |
| 0.25 | 0.585 | 8.56 | 0.360 | 0.00144 | 0.4174 |
| 0.30 | 0.700 | 10.29 | 0.431 | 0.00180 | 0.4888 |
| 0.35 | 0.822 | 12.01 | 0.502 | 0.00204 | 0.5600 |
| 0.40 | 0.935 | 13.72 | 0.574 | 0.00238 | 0.6324 |
| 0.50 | 1.165 | 17.15 | 0.720 | 0.00288 | 0.7789 |
| 0.60 | 1.400 | 20.58 | 0.860 | 0.00349 | 0.9195 |

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Table XX

COST DATA FOR MAINTAINING DIPHENYL CONCENTRATION AT
80% IN COOLANT

| Fraction
of Diphenyl
Destroyed
per day | Fraction of
Coolant Cir-
culated through
Purifier per day | Weight Loss,
tons/day | Cost, mill/kwh | | |
|---|--|--------------------------|----------------|---------|--------|
| | | | Replacement | Heat | Total |
| 0.05 | 0.20 | 1.96 | 0.081 | 0.00033 | 0.1373 |
| 0.10 | 0.40 | 3.92 | 0.162 | 0.00065 | 0.2187 |
| 0.15 | 0.60 | 5.88 | 0.245 | 0.0010 | 0.3020 |
| 0.20 | 0.80 | 7.84 | 0.318 | 0.00132 | 0.3753 |
| 0.25 | 1.00 | 9.80 | 0.410 | 0.00168 | 0.4667 |
| 0.30 | 1.20 | 11.76 | 0.491 | 0.00204 | 0.5490 |
| 0.35 | 1.40 | 13.72 | 0.572 | 0.00230 | 0.6303 |
| 0.40 | 1.60 | 15.68 | 0.658 | 0.00265 | 0.7167 |
| 0.50 | 2.00 | 19.60 | 0.810 | 0.00336 | 0.8694 |
| 0.60 | 2.40 | 23.52 | 0.990 | 0.00398 | 1.0530 |

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Table XXI

COST DATA FOR MAINTAINING DIPHENYL CONCENTRATION AT
90% IN COOLANT

| Fraction
of Diphenyl
Destroyed
per day | Fraction of
Coolant Cir-
culated Through
Purifier per day | Weight Loss,
tons/day | Cost, mill/kwh | | |
|---|--|--------------------------|----------------|---------|--------|
| | | | Replacement | Heat | Total |
| 0.05 | 0.45 | 2.21 | 0.092 | 0.00037 | 0.1484 |
| 0.10 | 0.9 | 4.42 | 0.184 | 0.00075 | 0.2408 |
| 0.15 | 1.35 | 6.62 | 0.277 | 0.00111 | 0.3341 |
| 0.20 | 1.80 | 8.82 | 0.374 | 0.00144 | 0.4294 |
| 0.25 | 2.25 | 11.03 | 0.461 | 0.00192 | 0.5209 |
| 0.30 | 2.70 | 13.23 | 0.554 | 0.00230 | 0.6193 |
| 0.40 | 3.60 | 17.64 | 0.738 | 0.00300 | 0.8100 |
| 0.50 | 4.50 | 22.05 | 0.922 | 0.00370 | 1.0037 |
| 0.60 | 5.40 | 26.46 | 1.21 | 0.00445 | 1.3015 |

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FIG.59
TOTAL COST AND DAILY REPLACEMENT WEIGHTS AS
FUNCTIONS OF THE DIPHENYL DESTRUCTION RATE

RE-7-12022-A

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B. Temperatures Produced by Sodium-Mercury Reactions -
J. W. Frank

Several experiments were made to study the temperatures produced by accidental mixing of sodium and mercury that might occur in a sodium-mercury-water heat transfer system. It is known that sodium and mercury react exothermically and that a solid amalgam exists to about 350C.²⁰ A literature search produced the heats of formation of a number of sodium amalgams,²¹ specific heats of two sodium-poor amalgams at relatively low temperatures,²² density and specific heats of sodium and mercury at various temperatures.²³ In the absence of data on the heat of fusion of amalgams, solid at 25C, experiments were made to obtain values for calculating temperature rises accompanying the reaction.

In the method used, mercury, at a known temperature, was mixed with sodium, either liquid or solid at a known temperature, in a stainless steel beaker provided with thermocouples at the top and bottom and insulated at the bottom and sides. The bottom temperature was assumed to be uniform, and the side wall temperature was assumed to vary linearly between measured temperatures at the top and bottom. Measurements were made on a fast-response pyrometer, manually switched.

A correction, made by dividing the calculated heat content of the beaker by the specific heat of the mass of the amalgam, was added to the observed temperature of the reaction mixture. Temperature readings were taken for several minutes after the reaction occurred, but agreement between samples was found only for first readings, apparently due to heat losses.

Two series of tests were made, the first with room temperature reactants and the second with the metals and container heated to about 100C. The results are given in Tables XXII and XXIII.

It is noted that the heat liberated by the potassium-mercury compound and by the sodium-rich amalgam is less than that from the sodium-poor amalgam.

In Experiment 15 there was a loss of sodium. Since all effects tend to promote heat losses, greater weight should be given to the larger temperature rises in the tables.

²⁰E. W. Washburn (Editor), "International Critical Tables," (New York: McGraw-Hill Book Co., 1948), Vol. II, p. 436.

²¹Ibid., Vol. V, p. 202.

²²Ibid., Vol. V, p. 119.

²³Liquid Metals Handbook, NAVEXOS P-733 (Rev.), Table 2.2, p. 40ff.

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Table XXII

EXPERIMENTAL TEMPERATURE RISE -
ROOM TEMPERATURE TESTS

<u>Compound</u>	<u>Experiment</u>	<u>Weight % Na</u>	<u>Temperature Rise, C</u>
NaHg ₂	2	5.16	578
	3	5.30	591
	4	5.25	572
	5	5.33	580
	6	5.38	585
	7	5.25	575
		Average	580
Na _{2.2} Hg	8	20.4	314
	9	20.2	298
		Average	290
KHg ₂	10	9.10 (%K)	120

Table XXIII

EXPERIMENTAL TEMPERATURE RISE - ELEVATED
TEMPERATURES FOR NaHg₂

<u>Experiment</u>	<u>Weight % Na</u>	<u>Temperature, C</u>		<u>Temperature Rise</u>
		<u>Mercury</u>	<u>Sodium</u>	<u>observed plus</u>
<u>temperature correction,</u>
<u>C</u>
11	5.45	79	119	616
12	5.30	94	124	651
13	5.11	93	132	526
14	5.31	103	106	523
15	5.	116	116	590
			Average	581

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If the latent heat of fusion of the sodium is taken into account, the heats of room temperature reaction would be 29C higher. Adding this figure to the unweighted average room temperature rise of 580C for the NaHg_2 compound, the calculated rise is about 610C. The unweighted average of all elevated temperature measurements shows about 30C less rise in temperature than that calculated from the room temperature results. If greater weight is given to the larger experimental values, this difference is diminished or may even vanish. One can extrapolate that, on mixing sodium and mercury at about 250 - 300C and in about the proportions required by the formula NaHg_2 , a temperature rise of no more than 500C is to be expected.

It would appear that if a mixing of hot liquid sodium and mercury could possibly occur, it would be preferable to have the mercury flow into a large volume of sodium than for the reverse to occur, both from the standpoint of temperature rise and the possible production of high melting amalgams.

INSTRUMENTATION

Flowmeter - W. W. Galbreath, Jr.

The investigation²⁴ directed toward finding a reliable meter for measuring slow flow rates and which is capable of operating continuously at high temperatures and pressures, has continued.

Two Potter electronic flow cells* were tested in a dynamic corrosion loop. The cells were 7/16 in. by 1 in. nominal diameter. These cells operate with generation of an a-c signal by a rotating magnet, and this signal is amplified and converted to d-c and read on a d-c milliammeter. The flow rate meter reading was compared to the flow rate readings of two orifice meters placed in parallel and located downstream in series with the flowmeter. The loop flow rate was too great for the range of the 7/16 in. cell; thus all the data collected are for the 1 in. meter.

The tests were conducted under the following conditions: (1) 500F, 1000 psi, 257 hr; (2) 550F, 1500 psi, 25 hr; and (3) 600F, 1800 psi, 20.5 hr. After the 500F run was completed, the meter was removed and examined. It was found that the downstream bearing housing had slipped, allowing an excessive amount of play to the rotor movement. It was assumed that this condition was caused by the hydraulic shock when the loop pumps were turned on. In all other respects the meter behaved satisfactorily, and showed consistent readings as compared with the orifice meter readings. It is believed that the shock effect from the pumps can be eliminated by mounting the meter vertically and directing the flow downward through it.

²⁴Reactor Engineering Division Quarterly Report, ANL-4925, September 15, 1952, p. 28.

*Manufactured by the Potter Aeronautical Company, Newark, New Jersey.

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