CORE II PHYSICS TESTS ON THE
SODIUM REACTOR EXPERIMENT

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ABSTRACT

A series of experiments were performed during the operation of the SRE-Core II to determine some of the basic nuclear parameters for that core. The reactor transfer function for the second fuel loading was measured with a pile oscillator. Measurements at-power were first performed while the core had a positive prompt power coefficient. After the fuel clusters were constrained, the transfer function at 10, 15, and 20 Mwt clearly demonstrated the absence of a positive coefficient. The combined fuel temperature coefficient was found to be $-0.12\text{¢}/^{\circ}\text{F}$ by the flow ramp analysis. The inlet temperature coefficient was determined, by the use of an inlet temperature ramp, to be $-0.23\text{¢}/^{\circ}\text{F}$. The xenon parameters previously determined have been verified. The results of the shim rod No. 3 calibration check show that the present shim rod worth is 94% of that initially determined. The integrated isothermal temperature coefficient was found to be 72.7 cents as compared to 81.9 cents previously.

I. INTRODUCTION

The Sodium Reactor Experiment (SRE) is a 20-megawatt (thermal) reactor constructed to study and improve the technology associated with sodium-cooled graphite moderated reactors. Since the main intent was to build a facility for experimentation in all phases of reactor development, many of the design features of the reactor are similar to those in a full scale power reactor. Upon installation of the second core in the SRE, a test program was initiated to experimentally determine the nuclear parameters of the core and to check the core stability. These parameters are employed in the calculation of the SRE reactivity history. Accurate calculation of the excess reactivity in the core, when compared to measured values as indicated from control rod positions, provide a monitor for abnormal behavior. In addition, these tests were designed to provide information on the method and techniques for determining reactor parameters for future large sodium graphite reactors.

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II. OSCILLATION MEASUREMENTS

Oscillation measurements have been used extensively at the SRE to determine the reactor transfer function. A detailed description of the theory and techniques used are contained in NAA-SR-7264. The reactor transfer function for the second fuel loading was measured with a pile oscillator¹. Measurements at power were first performed while the core had a positive prompt power coefficient associated with fuel rod bowing. After the clusters were constrained to prevent bowing, the transfer function clearly demonstrated the elimination of the positive coefficient and associated effects.^{2, 3, 4}

Additional measurements were made during recent power operation. These measurements at 10, 15 and 20 Mwt verify the absence of a positive coefficient. The results are presented graphically in Figures 1, 2, and 3. The scatter present in the data is due to the flux oscillations present in the SRE core. Repeated measurements showed that no real resonant peaks were present.

The data obtained at 20 Mwt was used to calculate the power coefficient. Using the expression

$$G_{20} = \frac{G_0}{1 + G_0 H_{20}} \quad \dots (1)$$

where

- G_0 = the zero power transfer function
- G_{20} = the 20 Mwt overall transfer function
- H_{20} = the 20 Mwt feedback transfer function.

The feedback transfer function was calculated. In this analysis the feedback transfer function is equal to the power coefficient divided by the power level. Due to the scatter present in the data, a band was chosen and the extremes of this band were analyzed. The transfer function analysis indicated that the power coefficient was $-2.25 \pm 1.75 \text{ } \$/\text{Mwt}$. This compares reasonably with the value measured by power ramp of $-1.5 \text{ } \$/\text{Mwt}$. It is apparent that unless a precise determination of the transfer function can be made, the power coefficient cannot be measured accurately by this technique.

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Figure 1. 10 Mwt Transfer Function

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Figure 2. 15 Mwt Transfer Function

NAA-SR-9510

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Figure 3. 20 Mwt Transfer Function

In addition to the determination of the transfer function, valuable information was obtained about the measuring system. The at-power measurements verified the flexibility of the system. Compact portable Donner analogs were used. These inexpensive consoles provide a simple and reliable flux correlation circuit, as shown in Figure 4.

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Figure 4. Flux and Temperature Correlation Circuitry

III. FLOW RAMPS

Flow ramps were initiated in an attempt to isolate the combined fuel and sodium temperature coefficients. These were the only fast temperature coefficients of any magnitude, and theoretical calculations indicated that they were essentially independent of power or temperature over the range of interest. During the initial testing of SRE core II, flow ramps were used extensively to determine the effect of wire wrapping the fuel elements.⁵

After completion of the initial testing and fuel element modifications, flow ramps were run to determine the final core II coefficients. Two negative flow ramps were performed at different main primary flows with the reactor at 5 Mwt. These ramps were performed in a manner which provided data suitable for analysis by simulation on an analog computer. The method was identical to that used for power ramps.⁶

The position of shim rod No. 3 for criticality following the ramp was converted to reactivity after correcting for inlet temperature changes. The result is plotted in Figure 5 as a function of time. Analysis of these data was performed with the simulator to factor out the effect of delayed neutrons. The combined fuel and coolant temperature coefficient was found to be $-0.12 \text{ } \beta / \text{F}^\circ$ for the SRE Core II. This agrees very well with the sum of the calculated Doppler and coolant coefficients.

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Figure 5. Flow Ramp Reactivity

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IV. INLET TEMPERATURE RAMP

At the SRE there exists a temperature coefficient associated with the expansion of the lower grid plate (stainless steel). This expansion of the grid plate results in an increase in the gap between moderator cans. The grid plate coefficient has been calculated to be $-0.22 \text{ } \$/\text{F}^\circ$.

In order to experimentally verify this number, an inlet temperature ramp was performed by ramping the feedwater flow down, thus increasing the reactor inlet. During this time, the shim rod position was recorded and converted to reactivity using shim rod calibration curves. Subsequent analysis with the analog simulator revealed that the length of time required for this ramp introduced a large number of time lags. Since an exact treatment of the system would be very complicated, it was decided to confine the analysis to a simple model. Using this method, a grid plate coefficient of $-0.25 \text{ } \$/\text{F}^\circ$ was obtained (Figure 6).

The model used simple lag loops for fuel, moderator, and a lag associated with the lower sodium plenum. This simple model is not sufficient for the analysis because the change in inlet temperature was very slow and resulted in an interaction between the reactivities associated with core temperatures. Under these conditions the exact transient response of the effects becomes very important, and the simple lag loops used successfully for faster ramps were not sufficiently accurate. Therefore, the future measurements of the grid plate coefficient will require a fast inlet temperature ramp. This will be accomplished by ramping the secondary flow and not the feedwater flow. The ramp will be of a duration such that temperature effects due to the moderator will not interfere with those due to fuel and the grid plate.

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Figure 6. Inlet Temperature Ramp

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V. XENON TRANSIENT

The xenon transient reactivity data has been used satisfactorily to provide a check of the accuracy of the calculated transient.⁷ This information has been incorporated in the method of calculation to provide better analysis for future conditions.

The initial xenon transient test was performed by reducing the power level of the SRE from 5 Mwt to 0.05% on the Log N as quickly as possible. The flow rates were simultaneously reduced to prevent lowering of the core temperature at a rate exceeding 60°F/hour. The xenon reactivity was followed by measuring the critical position of shim rod No. 3 at 30-min intervals for 91 hours. The reactivity absorbed by xenon was then determined by using the measured calibration curve of shim rod No. 3. Core temperatures were monitored to permit correction for temperature effects.

The results of this analysis are shown in Figure 7. The data were corrected for changes in reactor inlet temperature but not for core ΔT changes. The measured curve agrees well with theory after the initial 5 hr of the transient. The apparent initial offset is due to the many temperature changes occurring in the core.

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Figure 7. 5 Mwt Xenon Transient

Because of the difficulty involved in the analysis, a second transient was performed. In this case, care was taken to maintain constant temperatures. The ramp involved a reduction in power from 12.5 to 6 Mwt, maintaining reactor inlet and outlet temperatures by lowering flow. The method of analysis remained the same. The data from the second transient (constant temperatures) were more readily adaptable to analysis and resulted in the curve of Figure 8.

The information gained in these transient studies was applied to the on-line xenon computer and resulted in reliable xenon worths for operational use.

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Figure 8. 12 Mwt Xenon Transient

VI. SHIM ROD CALIBRATIONS

After completion of the wire wrapping of 40 Th-U fuel clusters, a shim rod calibration curve was determined for Core II. The resulting differential curve is shown in Figure 9. The total rod worth is 1% higher than originally. This agrees with theory since wire-wrapping the fuel reduced the relative flux in the fuel. The resulting decrease in the effective cell thermal absorption cross section increases the effectiveness of the control rods. The shim rod worth was determined by the period technique reported in NAA-SR-6585.

Further changes in Core II necessitated a check of these shim rod calibration curves to determine changes in the worth of the shim rods caused by changing the core loading from 40 to 43 fuel elements, and by replacing two moderator cans. This check was required to obtain accurate calibration curves for reactivity histories and for analysis of experimental data.

This check was accomplished with period measurements⁸ on shim rod No. 3 at 27, 35, and 42 inches. The gang was calibrated by moving it, noting the change in shim rod No. 3 position to maintain criticality, and converting the change in position to reactivity. To determine the difference due to shadowing between the total gang worth and the sum of the individual worth curves, data from the previous shim rod calibration was used. Results of the shim rod No. 3 calibration check show that the present shim rod worth is 94% of that determined in the previous calibration (Figure 9). To obtain the worth of the gang, the sum of the individual gang member worths should be further reduced by 3%. The decrease is associated with the larger core size, and agrees with theoretical predictions.

To further verify the accuracy of the above, a test was run to determine the effect of temperature on shim rod worth. The period measurement technique was again utilized. An expected worth was calculated using the corrected curves in

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Figure 9. Core II Shim Rod Calibration (40 elements)

Figure 9 to obtain $\rho(T_0)$ in the expression $\rho(T) \approx \rho(T_0) [T/T_0]^{1/2}$, where T is in degrees absolute; this expression assumes that the flux distribution is independent of the moderator temperature.

The shim rod worth does not change with temperature as expected over the temperature range of 300°F to 665°F. The reason for this discrepancy between theory and experiment is not known.

For the period runs, three channels were used. Two channels received signals from fission chambers as previously, while the third channel used a compensated ionization chamber which produces a dc current signal proportional to the incident neutron flux. A high precision electrometer detects this signal and provides a proportional dc voltage output. This is converted to periodic pulses by means of an analog to digital converter, and pulses are counted by a scaler.

The results given by this system were superior to those of the fission chamber system. A representative period was allowed to continue to 0.18% of full power. The period data obtained plots as an exponential function as expected (a straight line on semilog paper). Comparison of the data obtained using both counting systems is shown in Figure 10. Note that the period data obtained using the ion chamber exhibits no upward curvature in the higher count rate region of the measurement. This upward curvature has always been experienced in the fission chamber system. Over the range of measurements made, the calculated period for the ion chamber system is consistently lower than that calculated for the two fission chamber systems (254 sec vs 237 sec for data in Figure 10).

The upward curvature present in the fission chamber system causes the apparent asymptotic period to be steeper in slope, hence faster than the actual. The deviation at the lower end of the curve in the ion chamber system is due

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Figure 10. Period Data

to the normal approach to an asymptotic period. Therefore, by use of the ion chamber system in period measurements, more accurate period data can be obtained and hence more accurate shim rod calibrations.

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VII. ISOTHERMAL TEMPERATURE COEFFICIENT

The isothermal temperature coefficient for the SRE second core loading was determined during the low-power physics test prior to wire wrapping.^{9,10} Subsequent core changes necessitated additional determinations.

The starting temperature was 600°F. The reactor temperature was lowered in 40°F steps to 320°F. Electrical heat was then used to raise the temperature from 340°F to 660°F in 40°F steps. Nuclear heat was then required to raise temperature from 660°F to 700°F. After holding at 700°F, the temperature was lowered in steps to 680°F, 640°F, and 600°F. A 70-min waiting period was observed, after reaching a particular temperature, before readings were taken, to allow the moderator to approach thermal equilibrium with the core. The "isothermal" temperature was an average of several temperature readings (reactor inlet, two fuel exits, two fuel meats, and two Be probe temperatures).

Moderator flow of 100 gpm, main primary flow of 1000 gpm, auxiliary primary flow of 40 gpm, and core cover gas pressure of 0.6 ± 0.1 psi were maintained during the test. Except where nuclear heat was used, the reactor power was 0.005% on the log N scale.

Corrections were made for the introduction of Xe by nuclear heating. The corrected points, which determine the integrated isothermal temperature coefficient, were subjected to a least-squares fit. The second degree polynomial fit shown had a standard deviation of only 1.084 from the original data points. This was almost as small as the sixth degree standard deviation of 1.062. The second degree fit should be accurate enough for most calculations.

A test performed after the 40 element core had been wire wrapped resulted in the expression (which gives $\Delta\rho$ above 327.6°F)

$$\Delta\rho(\%) = -147.8 + 0.524T - (2.23 \times 10^{-4})T^2 \quad \dots (2)$$

where T equals isothermal temperature, °F. The test was repeated after the replacement of two reflector cans and the addition of three fuel clusters. The expression for the integrated isothermal temperature coefficient was

$$\Delta\rho(\%) = -126.3 + 0.3996T - 4.3 \times 10^{-5}T^2 \quad \dots (3)$$

The coefficient, integrated between 340°F and 600°F, is now 72.7¢ as compared to 81.9¢ in the previous case as plotted in Figure 11.

The above expressions were then differentiated, to obtain expressions for the differential isothermal temperature coefficient. The shape of the differential curve was rather sensitive to the degree of fit. Figure 12 shows the comparison of the sixth degree with the second degree fit for the 40 fuel clusters, wire wrapped case.

The differential worth curve for the wire wrapped case is higher than that obtained before the fuel clusters were constrained (Figure 13). This is due to the fact that the fuel is now held in a lower flux region, increasing the disadvantage factor. This increases the positive moderator coefficient making the overall coefficient a greater positive value.

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Figure 11. Integrated Isothermal Temperature Coefficient

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Figure 12. Differential Curve Degree of Fit

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Figure 13. Differential Worth Curve

The differential coefficient for the 43 fuel cluster case (Figure 13), is lower than that for 40 clusters. This is contrary to theory, since increasing the core size should decrease the negative coefficient associated with leakage, and thus make the total coefficient more positive. The reason for this anomaly has not been determined.

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VIII. CONCLUSIONS

The experimental techniques discussed in this report are the results of five years experience in sodium graphite reactors. These techniques have been developed in an effort to simplify the experiment without compromising the results.

The results reported here have substantiated the confidence placed in the experimental methods, and have verified the stability of the SRE Core II. On the basis of these results some firm decisions can be made concerning requirements for future test programs. To determine the stability of the reactor over the frequency range required, oscillation measurements and power ramps should be performed at selected power levels.

The performance of a reactivity history program necessitates the accurate determination of core parameters. These parameters will be measured by flow ramps, inlet temperature ramps, shim rod calibrations and xenon transients similar to those discussed in this report. The methods described in this report will yield sufficient accuracy to permit a useful reactivity history program.

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