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**SODIUM GRAPHITE REACTOR
QUARTERLY PROGRESS REPORT
JUNE-AUGUST, 1953**

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ABSTRACT

Engineering was continued on the development of sodium cooled, graphite moderated type reactors. General studies were carried out as well as studies specifically devoted to the following:

- a. Full scale power-only plant
- b. Thirty-megawatt pilot plant, the SGR
- c. Sodium reactor experiment, the SRE

This work consisted of theoretical analysis of various aspects of nuclear performance; economic investigations of different fuel element, cooling system and plant arrangements; and experimental investigations related to the properties of certain materials and to the development of components. Preliminary consideration was given to alternative reactor arrangements employing liquid hydrocarbon moderators and high temperature coolants other than sodium. In addition to a summary of the general design features of the SRE, a program was prepared outlining the proposed use of this installation.

This report is based upon studies conducted for the Atomic Energy Commission under Contract AT-11-1-GEN-8.

Previous SGR Progress Reports

NAA-227	SGR Quarterly Progress Report	September-November, 1952
NAA-260	SGR Quarterly Progress Report	December, 1952-February, 1953
NAA-274	SGR Quarterly Progress Report	March-May, 1953

I. INTRODUCTION

During this quarter, work continued on the development of the power-only, sodium cooled, graphite moderated reactor plant. Since for policy reasons the design of the SGR pilot plant was terminated, the principal emphasis was placed on aspects of the reactor system pertinent to the full scale plant and on planning a small scale reactor experiment referred to as the SRE.

Studies were completed on problems related to the nuclear performance of the sodium-graphite reactor such as temperature coefficients of reactivity, poison transients, neutron leakages, possible runaway during start-up, and the variation of conversion ratio with burnup. Some of these studies were specifically for the 30 megawatt SGR pilot plant.

In connection with the reactor engineering work, an investigation was carried out in order to optimize the cooling system and power plant conditions in the full scale plant. Also, the desirability of operating with two different fuel element diameters in a single reactor and of operating with more than one reactor as the heat source for a single power plant was evaluated. Design and experimental work was conducted in regard to problems in the use of stainless steel coolant tubes, an inner steel core tank to protect the graphite from the sodium coolant, and the alternative arrangement of zirconium cans containing cell-sized stringers of graphite. Materials work was initiated or continued on the joining of zirconium and stainless steel, the carburization of stainless steel, the weldability of stainless steel and nickel, mass transport by sodium in stainless steel, and the properties of uranium fuel alloys. The fuel rod loading apparatus was successfully used to prepare a seven-rod fuel element of uranium slugs in stainless steel jackets and bonded by sodium. The mock-up tower at the Santa Susana site was operated for the first time with the 12 foot long fuel element being loaded from the handling coffin into the sodium filled coolant tube assembly. Other large scale component testing work was initiated.

As alternate power reactor design studies the use of sulphur and diphenyl as coolants were investigated. Attention was also given to the possible use of some hydrocarbon liquid as the moderator. On the basis of present information, reactor arrangements making use of such materials do not appear capable of the high performance expected from the sodium-graphite combination.

A major integrated effort was placed on the preliminary design and proposed program for the Sodium Reactor Experiment (SRE). This is envisaged as a tank-type reactor with initial operation using graphite moderator canned in zirconium and with sodium as the coolant. The design stresses maximum experimental flexibility and has no provision for the generation of electricity. Its primary purpose is for development work on the technology of fuels, moderators, coolants, and general reactor components. While the SRE arrangement permits a variety of experiments to be performed relative to high temperature, low pressure reactor core concepts, the study of problems pertinent to the sodium cooled, graphite moderated arrangement will be emphasized.

II. REACTOR PHYSICS ANALYSIS

A. SGR Reactivity Coefficients (J. E. Garvey)

A study has been made to determine the approximate temperature coefficients of reactivity for the 30 megawatt SGR. Results of previous calculations of the reactor parameters as functions of metal, moderator and coolant temperatures, and of power level were used; these calculations evaluated reactivity changes between one equilibrium state and another.

The metal coefficient is primarily due to Doppler broadening in U^{238} , and the calculations are based upon the work of Mitchell, et al.¹ The graphite coefficient, a function of the temperature dependence of thermal utilization, (η), thermal diffusion length, age, and buckling, is strongly dependent upon the temperature variation of η ; we have chosen for natural uranium a value of $-5 \times 10^{-5}/^{\circ}\text{C}$ for $1/\eta \frac{d\eta}{dT}$ (at 20°C).

Defining the metal coefficient, C_m , as due to changes in fast effect and resonance escape probability and, as above, defining the graphite coefficient, C_g , as due to changes in all other parameters, then values for C_m and C_g may be determined. When the reactor is characterized by average temperatures of 400°C in the fuel, 400°C in the moderator, 350°C in the coolant and is operating with an average thermal flux of 3×10^{13} neutrons/cm²-sec in the fuel, then

$$\begin{aligned}C_m &= -1.1 \times 10^{-5}/^{\circ}\text{C} \\C_g &= -2.6 \times 10^{-5}/^{\circ}\text{C}\end{aligned}$$

The calculation further indicates that for the unpoisoned zero power operation with average temperature of the graphite less than 100°C , the value of C_g may be positive. At average graphite temperatures above 200°C and at more than one-tenth normal power level, C_g is negative. The value of C_m is negative under all conditions of operation. The values obtained by Borst² for the ORNL Graphite Reactor metal and graphite coefficients were:

$$\begin{aligned}C_m &= -1.04 \times 10^{-5}/^{\circ}\text{C} \\C_g &= -4.94 \times 10^{-5}/^{\circ}\text{C}\end{aligned}$$

While the quantities we define as C_m are not exactly those measured by Borst, the temperature coefficients of the two sets should agree as to sign, and should agree in magnitude to within a factor of two; such is the case. Thus, while the presence of water as coolant leads to a positive graphite coefficient for the Hanford type reactors, it seems clear from our calculations that the response of the SGR to temperature changes will be similar to that of the ORNL Graphite Reactor where the temperature coefficients of reactivity are negative and of the order of magnitude of $10^{-5}/^{\circ}\text{C}$.

B. Start-up Incident in 30 Megawatt SGR (J. E. Garvey)

A possible start-up incident in the 30 megawatt SGR is analyzed under the following imposed conditions:

1. The incident is initiated by the unmanaged withdrawal of a control rod.
2. The reactor reaches its normal full power level and attains prompt criticality simultaneously; the reactor period at this time is taken to be $1/3$ second.
3. The reaction is restrained by the metal temperature coefficient of reactivity.

The analysis is based upon the assumption of one group of free neutrons and one group of latent neutrons; the latter group is characterized by parameters obtained by suitably averaging the characteristic parameters of the six delayed groups. A set of four equations is then developed describing the density of free neutrons, the density of latent neutrons, the maximum fuel rod temperature

(or energy density), and the reactivity. The reactivity relation contains a term describing the contribution (negative) due to the metal temperature coefficient.

In terms of maximum fuel rod energy density (cal/gm) calculations were made for a control rod withdrawing at a constant velocity such that the rod will be completely withdrawn in 3 minutes and will contribute to the reactivity 2.6×10^{-2} dollars per second. The temperature coefficient is based upon the research of Mitchell¹ and calculations showed the average value (over the interested temperature region) to be -0.309 dollars per unit increase in maximum fuel rod energy density. Two cases were considered:

1. The control rod withdrawal speed is as mentioned above.
2. The control rod withdrawal speed is ten times as fast as that mentioned above.

The set of equations has been solved on the Nordsieck Differential Analyzer.

The results are shown in Figs. 1 and 2 where relative power (with respect to normal full power) and maximum fuel rod energy density are shown as functions of time after the reactor becomes prompt critical. The slower withdrawal speed allows about 37 seconds before fuel rod melting is attained. At this time the relative power has been reduced to 1.02 after having reached a maximum value of about 3.1. The faster withdrawal speed leads to fuel rod melting about 4.1 seconds after prompt criticality is reached. In the latter incident the relative power at 4.1 seconds has been reduced to 1.8 after having reached a maximum value of about 5.3

It can be concluded that a withdrawal speed slightly less than 3 minutes for complete withdrawal would, in the event of an incident of the nature considered, allow an operator time of the order of minutes to manually effect changes in the reactor operation before fuel rod melting occurred. Since the maximum relative power would be less than 3, one expects to retain the effective use of coolant throughout such an incident and to conclude that the metal temperature coefficient alone would provide a very high degree of intrinsic safety with respect to the prevention of fuel rod melting.

Fig. 1. Power Surge during Reactor Start-up Incident

Fig. 2. Maximum Fuel Rod Energy Density during Reactor Start-up Incident

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C. Poison Transients Following Power Cutback (C. Roderick)

The Xe^{125} and Sm^{149} transients due to rapid power cutback and the effects of these transients on reactivity were studied and presented in a general manner.

Xenon and samarium are important poisons and demand design consideration because of their large absorption cross sections for thermal neutrons. The poison problem is accentuated when the reactor suffers a sudden power cutback to some fraction of operating level or to zero power (scram) since xenon and samarium continue to be formed from fission product decay but are "burned out" at a slower rate. This results in the poison building up to a maximum before it ultimately decays. It is possible for such a poison build-up to make the reactor inoperative for a period of time if the reactivity available is insufficient to override the poison.

The reactivity after the power cutback can be expressed by the following equation:

$$P = 1 - \frac{\frac{1 + \sum_a R}{\sum_a U} + \frac{\sum_a P}{\sum_a U}}{\frac{\sum_a R}{\sum_a U} + \frac{\sum_a P_o}{\sum_a U}}$$

where $\sum_a U$ = absorption cross section of the fuel

$\sum_a R$ = absorption cross section of the moderator structure and coolant

$\sum_a P_o$ = absorption cross section of the equilibrium poison

$\sum_a P$ = absorption cross section of the poison at time t

The results of calculations for various values of $\sum_a R / \sum_a U$ when the reactor power is cut back to zero power and to 50 per cent power are plotted in Fig. 3.

D. Lattice Study (J. E. Garvey and F. L. Fillmore)

The two-group two-region reactor constants were evaluated for slightly enriched uranium, sodium cooled, graphite moderated lattices. A parameter study was made wherein enrichment was varied from 1.35 to 1.9 per cent $^{235}/^{238}$ atomic ratio. Temperatures ranged from 20 to 600° C, and the flux (which determines the equilibrium Xe and Sm poison) ranged from zero to 5×10^{13} neutrons/cm²-sec.

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NOTES:

Reactivity necessary to overcome poison transient due to rapid power cutback plotted vs time for the case where the average steady state flux is 2×10^{13} thermal neutrons/cm²-sec.

Fig. 3. Reactivity Change after Reactor Power Cutback

Eighteen different systems, each defined by a specific value of enrichment flux, and temperatures were considered, and the criticality condition was evaluated for each. A flux plot was made for a typical case. A curve of enrichment vs critical radius was obtained and is shown in Fig. 4. Figure 5 shows the effect of changing reactor conditions upon the critical radius for an enrichment of 1.35 per cent. As indicated under Remarks in Fig. 5, the variation in reactivity as evidenced by the core volume required to maintain criticality is shown, based on a "normal" condition of 30 megawatt operation at standard temperatures, for a clean reactor without coolant, for frozen sodium in the core, and for molten sodium in the core. The more gradual increase in volume required as the operating power level of the reactor is raised is shown by the broken line. An increase in the reflector thickness of 1 foot decreased the critical radius from 94.4 to 89.5 centimeters which amounts to about a 10 per cent savings in the core size or a reduction of about 0.035 per cent in the enrichment.

E. Neutron Leakage from 30 Megawatt Sodium Graphite Reactor (F. L. Fillmore)

A study of the fast and thermal neutron leakage from the reflector and thermal shield of an SGR reactor was made by three different methods. The results are shown in Table I. The core of the reactor studied is a square prism 12 feet high and 6 feet across each face. This is surrounded on all sides by a graphite reflector 2 feet thick. This is in turn surrounded by an iron thermal shield 8 inches thick. Two cells in each corner of the lattice will not be loaded with fuel.

In Method I, two calculations of the fast leakage from the reflector are made. One is based on solutions of the age equation in semi-infinite regions with an exponentially attenuated fast neutron source term in the nonmultiplying region and the application of corrections to take into account the finite geometry of the reactor. The other is based on an empirical formula.³ Fast leakage from the shield is obtained from solutions of the age equation.

Method II obtains the fast and thermal leakage from the reflector from a two-group two-region flux plot. The attenuation of these leakages by the thermal shield is obtained by the formula $e^{-x/L}$ where L is the appropriate diffusion length for the shield. For the fast leakage it is averaged over the energy spectrum of the leakage neutrons.

Fig. 4. Critical Radius vs Enrichment

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Fig. 5. Fractional Change in Core Volume Required to Maintain Criticality vs Reactor Condition

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Method III obtains the leakage from a bare reactor in terms of the well known leakage probabilities, $e^{-\tau B^2}$ and $L^2 B^2 / (1 + L^2 B^2)$. The attenuation of the leakage neutrons by reflector and thermal shield is then obtained by the formula $e^{-x/L}$.

The response of ionization chambers and fission chambers located outside the thermal shield is examined. Response to fast neutrons is assumed proportional to $1/\sqrt{E}$ and it is found that a thermal shield only 5 inches thick will provide ample leakage to operate compensated ionization chambers over their six decade range. However, it will be necessary to insert a fission chamber being used as a reactor start-up detector into the reflector, since the leakage flux at start-up will be too low by 10^5 to operate this instrument satisfactorily.

TABLE I
FAST AND THERMAL NEUTRON LEAKAGE FROM THE SGR

Leakage from the Reflector			
	Method I	Method II	Method III
J_f (61 cm reflector)	A. 6.5×10^{10} B. 5.7×10^{10}	7.4×10^{10}	1.4×10^{10}
J_f (84 cm reflector)	A. 4.7×10^9 B. 7.7×10^9	-	2.6×10^9
J_{th} (61 cm reflector)	--	8.1×10^{11}	3.5×10^{11}
J_{th} (84 cm reflector)	--	-	2.4×10^{11}
Leakage from an 8-inch Iron Shield			
J_f (61 cm reflector)	1.8×10^9	5.6×10^9	1.1×10^9
J_f (84 cm reflector)	7.7×10^7	-	9.5×10^7
J_{th} (61 cm reflector)	--	2.1×10^7	0.88×10^7
J_{th} (84 cm reflector)	--	-	0.61×10^7

F. Proposed Experimental Lattice Studies (F. L. Fillmore)

An experimental program on SGR lattices was proposed in order to obtain a check on some of the theoretical calculations. A study would be made of the

buckling, age, L^2 and disadvantage factors characteristic of a given lattice. These results would be used to evaluate the accuracy of the criticality calculations and to modify empirically those results if necessary so as to render them more reliable for reactor design purposes than they now are. Provision would be made for the insertion of a control rod into the lattice so that local flux disturbances in the neighborhood of the rod could be examined. These results could be used to evaluate the Δk of the rod.

The addition of graphite and iron slabs to the top of the lattice will permit flux measurements to be made which will be very useful in the evaluation of fast and thermal neutron leakages from the reflector and thermal shield of a reactor of this type.

G. Variation of Conversion Ratio with Burnup (C. Roderick, Jr. and E. Weisner)

A recent report (NAA-SR-216) presents several curves of changes in multiplication factor with burnup of the fuel for uranium and thorium-uranium systems. An assumption used in that report (and also used by W. B. Lewis in DR-21) is that the conversion ratio remains constant as burnup of the fuel progresses. A study has been carried out to check the validity of that assumption.

Using the assumption of a constant value of resonance escape, one can establish, and solve, the differential equations describing the buildup and decay of the various isotopes with exposure. Having determined the isotope concentration as a function of burnup, one can then compute the instantaneous value of the conversion ratio as a function of burnup.

The conversion ratio, C. R., is defined as

$$\frac{\text{amount of fertile material converted to fissionable material}}{\text{amount of fissionable material consumed during this conversion}}$$

When the fuel is a mixture of Pu^{239} , U^{235} , and U^{238} , for example, the conversion ratio can be expressed as

$$\text{C. R.} = \frac{[\Sigma_f(25)_v(25) + \Sigma_f(49)_v(49)] e^{-k^2 L_f^2} (1 - p) + \Sigma_a(28)}{\Sigma_a(25) + \Sigma_a(49)}$$

The assumptions which have been made in this study are:

1. The errors involved in neglecting the effects of Th^{233} and Pu^{241} are less than those induced by the errors in the cross section values.

Since Th^{233} has a relatively short half life (23.3 minutes), Pa^{233} is considered to be formed directly by neutron absorption in Th^{232} nucleus. The abundance of Pu^{241} is small enough in all cases considered that it can be neglected with essentially no effect upon conversion ratio.

2. The product $e^{-\rho^2 L_f^2}$ is equal to unity.

The departure of this product from unity increases somewhat as the reactor size decreases. The generality of the results is not impaired

however if (p) is interpreted as p' where $p' = 1 - e^{-\rho^2 L_f^2} (1 - p)$.

The variation in conversion ratio with burnup for natural uranium with no initial plutonium present is shown in Fig. 6 as a function of the initial conversion ratio. For relatively high values of the conversion ratio, there is a slight decline in conversion as burnup progresses. For relatively low initial values, there is a gradual improvement in conversion with burnup. The change in conversion ratio with burnup is less for the same initial conversion ratio as the enrichment is increased. Figure 7 shows the same variations when some plutonium is initially present. The initial presence of plutonium causes the system to behave similarly to a slightly enriched system.

The changes which occur in a system which initially has only thorium and U^{235} are shown in Fig. 8. There is a considerable steady increase in conversion ratio with burnup. This is due to the replacement of U^{235} by U^{233} which is a much superior fuel in thermal reactors.

The results for a system which initially contains only thorium and U^{233} are presented in Figs. 9 and 10 for fluxes of 1×10^{13} and 5×10^{13} respectively for 2 per cent U^{233} in thorium. The horizontal scale in these figures is given in units of time at constant flux rather than burnup.

Conclusion - For the U^{235} , U^{238} , Pu^{238} cycle with or without plutonium initially present, the variation in conversion ratio with burnup for the conversion ratios of practical interest is generally less than 10 per cent for burnups

Fig. 6. Conversion Ratio vs Burnup (No initial Pu)

Fig. 7. Conversion Ration vs Burnup (Pu; 235 = 1.00)

Fig. 8. Conversion Ratio vs Burnup (235: TH = 0.02)

Fig. 9. Conversion Ratio vs Time (233: $Th = 0.02$, $\phi = 1 \times 10^{13}$)

Fig. 10. Conversion Ratio vs Time (233: $Th = 0.02$, $\phi = 5 \times 10^{13}$)

up to 50 per cent. The multiplication factor variations presented in NAA-SR-216 would then appear to be slightly optimistic for conversion ratios greater than about 0.7 and slightly pessimistic for conversion ratios less than 0.7.

In the $\text{Th}^{232}\text{-U}^{235}\text{-U}^{233}$ cycle, the conversion ratio continually increases as the U^{235} is burned up. For 50 per cent burnup this increase in conversion ratio is approximately 35 per cent. This implies the curves for this cycle presented in NAA-SR-216 are quite pessimistic in predicting the burnup attainable for a given reactivity change.

In the $\text{Th}^{232}\text{-U}^{233}$ cycle, the variation in conversion ratio with time depends upon flux, initial concentration of U^{233} in thorium, and the initial value of conversion ratio.

III. REACTOR ENGINEERING

A. Optimum Cooling System and Power Plant (C. H. Robbins)

An economic study has been made to optimize the cooling system of the tank-type, single-flow reactor, with 19-rod fuel elements in the center region and seven-rod fuel elements in the outer region. Designs of power reactors using two different diameter fuel rods and different numbers of rods per element have certain advantages, discussed more fully in Section III, B. The purpose was to determine the best values for the following variables:

1. Primary sodium temperature into the reactor
2. Temperature drive between the primary and secondary coolants
3. Temperature approach (pinch point) in the economizer and evaporator between the secondary coolant and the boiler H_2O
4. Temperature approach between the steam leaving the superheater and the secondary coolant
5. The amount of regenerative feedwater heating that is desirable

In most parts of the study, the primary variable was the temperature of the sodium into the reactor since this has a considerable effect on the maximum reactor power. The effect of increasing the reactor power by relaxing some of the assumed temperature and velocity limitations was also investigated.

While the study is directed specifically toward the reactor mentioned above, the methods and some of its conclusions apply to other reactor systems. To determine the effect of sodium inlet temperature on the maximum reactor power, the following limitations were assumed:

1. Maximum uranium temperature, 1200° F
2. Maximum sodium velocity, 15 ft/sec
3. Maximum sodium temperature, 1000° F

By assuming these limitations, the maximum reactor power, sodium flow rate, and sodium outlet temperature become functions of the sodium inlet temperature. These functions are plotted in Fig. 11. It is apparent that the reactor power varies considerably with the sodium inlet temperature, but that the change in the sodium flow and outlet temperature is rather small.

1. Cost Analysis - The reactor configuration was assumed fixed and only the coolant temperatures and the steam cycle were varied. Power cost was assumed to be made up of four different groups of factors.

- a. Fixed investment costs or the equivalent which include reactor, fuel, site, and sodium system investments and also operation and maintenance.
- b. Costs varying directly with the reactor thermal power which comprise fuel preparation and chemistry less the amount received from the sale of plutonium.
- c. Investment costs which vary directly with the electric power produced. The turbogenerator plant was considered to be of this nature.
- d. Investment costs which vary directly with the reactor thermal power to some exponent. Heat exchangers were placed in this category.

2. Temperature Drives - The effect of temperature drives in the intermediate heat exchanger and the boiler on the cost of electric power was investigated. Considerably lower temperature drives than have been used in previous NAA designs appear justified.

3. Electrical Power vs Sodium Inlet Temperature - The steam pressure and electrical power were calculated as a function of the sodium inlet temperature with the amount of regenerative feedwater heating as a parameter. The results of the calculations are plotted on Fig. 12.

Fig. 11. Power Characteristics of Tank Type SGR vs Sodium Inlet Temperature

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Fig. 12. Electrical Power vs Sodium Inlet Temperature

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The following assumptions were made:

- a. The secondary coolant is 20° F colder than the primary sodium and the temperature range in the two circuits is the same.
- b. The temperature approach in the economizer and evaporator is 25° F.
- c. The temperature approach in the superheater is 75° F.

The following conclusions may be reached on the basis of Fig. 12.

- a. Electrical power is fairly insensitive to sodium inlet temperature and steam pressure over a wide range.
- b. The amount of regenerative feedwater heating does not have a major effect on the amount of electrical power obtainable. In Fig. 12 the maximum power was obtained with the least amount of feedwater heating. However, other calculations not shown indicate that this result depends on the assumptions used.

4. Possible Improvement in Power Cost by Raising Reactor Power - Considerable reduction in the cost of power appears possible if the reactor power is raised by such means as decreasing the diameter of the fuel rods, raising the limiting temperature of the fuel element, increasing the maximum sodium velocity or the ratio of coolant to fuel volume, and raising the maximum sodium temperature. The effect of increasing reactor power was investigated by assuming the fuel element modified in an unspecified manner to circumvent the limitation of the temperature of the fuel element. The reactor power was then raised by increasing the maximum sodium temperature from 1000 to 1150° F and the sodium velocity increased from 15 to 24 ft/sec.

5. Conclusions - The following conclusions rest on assumptions which appear reasonable now, but which can become unreasonable due to future developments. The recommendations are for a design in which the sodium is kept at less than 1000° F and 15 ft/sec.

- a. The sodium inlet temperature should be in the range 400 to 480° F.
- b. Steam pressure should be in the range of about 400 to 1000 psia.
- c. The temperature drive in the intermediate exchanger between the primary and secondary coolants should be about 20° F.
- d. The temperature approach in the economizer and evaporator between the secondary coolant and the water should be about 20 to 30° F.

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- e. The steam leaving the superheater should be in the neighborhood of 75° F lower than the entering secondary coolant.
- f. For minimum power cost, the amount of regenerative feedwater heating and the sodium inlet temperature should be chosen together. More regeneration appears justified as the sodium inlet temperature is increased.
- g. Considerable reduction in power cost appears possible if the reactor power is increased by such means as modifying the fuel element or raising its temperature limit, increasing the maximum allowable sodium temperature, and increasing the maximum sodium velocity.

B. Power Cost Study for Two-Diameter Fuel Rod Reactors (T. T. Shimazaki and E. F. Weisner)

The possibility of producing power with a two-diameter fuel rod reactor using a single flow cooling system at a cost comparable to that realizable with a single-diameter fuel rod reactor using a split flow cooling system has been investigated. For the split-flow arrangement (NAA-SR-227, page 32) the reactor core is divided into an inner and an outer coolant zone. This permits the use of different inlet and outlet coolant temperatures which in turn makes possible a higher electrical output when a limitation is taken in the form of a maximum fuel metal temperature. By using smaller diameter fuel rods in the central region of the reactor than in the outer region, as high or even higher power density can be attained with a single-flow cooling system than with a reactor having the larger diameter fuel rods throughout and employing a split-flow cooling system. The obvious advantages of the scheme are the mechanical and operational simplifications afforded by the single-flow cooling system. The principal disadvantages lie in the additional cladding material and increased difficulty of fuel element fabrication associated with the greater number of the smaller diameter fuel rods, which results in higher fuel feed costs.

The power cost study was made for five graphite moderated, sodium cooled reactors. Three of the reactors (Systems 2, 3, and 4) were two-diameter fuel rod reactors employing a single-flow cooling system, and the other two reactors (Systems 1 and 5) were single-diameter fuel rod reactors employing

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a split-flow cooling system. All reactor-power plant systems were sized for 150 megawatts of electrical power. The maximum coolant velocity in all reactors was limited to 15 ft/sec.

The results of the power cost study for these systems are tabulated in Table II. It is seen that the best two-diameter fuel rod reactor (19-rod elements with 0.455-inch diameter slugs in the central region and seven-rod elements with 0.75-inch diameter slugs in the outer region) with a single-flow cooling system is one having a coolant tube inside diameter of about 3-1/2 inches for the 19-rod elements and which has a sufficiently large central region of 19-rod elements to allow operation at the maximum power density permitted by the central fuel element. The power cost for this reactor (System 4) is comparable to the power cost for the one-diameter fuel rod reactors using a split-flow cooling system.

C. Economic Study of Multi-Unit Power Reactors (G. H. Syrov and E. F. Weisner)

It has been suggested that a reactor of a size required to produce 500 megawatt thermal power might be too large to lend itself to tank-type construction. The use of several smaller units which may or may not be coupled together into a common power plant and enclosed within a common shield and building would contribute to the efficacy of the tank-type construction. The question then arises: What is the effect on power cost of using multiple unit reactors?

A single unit of 500 megawatt thermal power has been selected as the basic reactor. The cases to be compared with this unit consist of two 250 megawatt reactors, three 167 megawatt reactors, and four 125 megawatt reactors. These cases will be studied under four different environments:

1. Each set of units feeding a common power plant
 - a. Load factor independent of number of units
 - b. Load factor dependent upon number of units
2. Each unit of a set is totally independent and separate from every other unit (except on same site)
 - a. Load factor independent of number of units
 - b. Load factor dependent upon number of units

TABLE II

POWER COST STUDY FOR TWO-DIAMETER FUEL ROD REACTORS

System	1	2	3	4	5
Reactor Investment	3.38	3.38	3.06	2.88	2.81
Power Plant and Cooling System	2.89	2.89	2.89	2.89	2.89
Site	1.29	1.29	1.29	1.29	1.29
Fuel Investment	1.13	1.28	0.91	0.77	0.72
Fuel Feed and Preparation	3.88	4.12	4.61	4.25	4.29
General Operation and Maintenance	2.00	2.00	2.00	2.00	2.00
Total	14.57	14.96	14.76	14.08	14.00
Pu Credit	3.98	3.87	3.82	3.51	3.51
Net Power Cost (mils/kw)	10.59	11.09	10.94	10.57	10.49

System 1 - All 7-3/4 inch diameter rods. Split flow. 30% eff.
All 2.77 inch ID coolant tubes. Central rod power 1680 kw.

2 - 19 - 0.455 inch diameter rods between $0 \leq r/R \leq 0.375$.
All 2.77 inch ID coolant tubes. 30% eff. Single flow.
Central rod power 1680 kw.

3 - 19 - 0.455 inch diameter rods between $0 \leq r/R \leq 0.70$.
All 2.77 inch ID. Single flow. 31.3% eff. Central rod
power 2510 kw.

4 - 19 - 0.455 inch diameter rods between $0 \leq r/R \leq 0.85$.
3.52 inch ID coolant tubes for 19-rod elements. 29.8% eff.
Single flow. Central rod power 3810 kw.

5 - All 19 - 0.455 inch diameter rods. 3.63 inch ID coolant tubes.
Split flow. 30% eff. Central rod power 4060 kw.

It is clear that results for other combinations (such as common shielding and building but separate power plants) will lie somewhere between the results of 1 and 2.

The assessment of values for load factors which are dependent upon the number of reactor units must be assumed where no operating experience is

available. Smaller units with less fuel element changes required per year should realize a higher load factor. The assumed load factors are indicated in Table III. These factors will apply whether each reactor has its own turbo-generator or whether all reactors feed a common generator. It is only necessary that there be a fuel charging machine available for each reactor shut down at the same time.

TABLE III
POWER COST - MILS PER KILOWATT-HOUR

Number of Reactors →	1	2	3	4
Power per unit in mw	500	250	167	125
Total Cost - Million \$	15.0	15.7	16.6	17.8
<u>For Pu Credited at \$40/gm</u>				
<u>1. Common Power Plant</u>				
a. 0.8 load factor all	9.83	10.13	10.56	11.31
b. Variable	9.83	9.90	10.24	10.90
Load Factor	0.80	0.85	0.87	0.88
<u>2. Separate Individual Units</u>				
a. 0.8 load factor all	9.83	10.84	11.72	12.77
b. Variable load factor	9.83	10.57	11.30	12.23
<u>No Pu Credit ($k_{ex} = 4\%$)</u>				
a. 0.8 load factor all	13.68	13.98	14.39	14.90
b. Variable load factor	13.68	13.75	14.05	14.49

It is to be noted that where no improvement in load factor is assumed that the power cost continually increases with subdivision into more and more units. This is, of course, to be expected since the only improvement (costwise) results from a more effective use of the reflector. The reflector saving is relatively independent of the dimension being reflected, so in the smaller units a higher average-to-peak ratio results, and consequently, a higher average over-all average-to-peak ratio for the combined units. This effect is manifested as a lower critical mass. The over-all investment cost for multi-units increases

steadily with subdivision although the increase in going from one unit to two is rather less than would be expected.

When improvement is assumed in the load factor with subdivision, there is very little difference in cost between one and two units whether plutonium is credited or not. Beyond two units the cost increase is more rapid. From Table III it may be concluded that a single unit will produce the most economical power unless a greater improvement in load factor can be realized than has been assumed here.

D. Critical Buckling of Coolant Tubes under External Pressure (W. C. Cooley)

It has been proposed in certain designs for the SGR that: (1) the core helium atmosphere be pressurized to prevent sodium from leaking through a single-wall coolant tube into the graphite, or (2) that the annulus between double-walled coolant tubes be pressurized to serve as a leak detector by observing any decrease in annulus pressure.

A study was made to determine the limiting pressure difference which a coolant tube can withstand without buckling, assuming no external restraint by an external tube or the graphite hole. In Table IV are tabulated the critical buckling pressures for 2.8-inch diameter 18-8 stainless steel and zirconium alloy tubes of various wall thicknesses. It is seen that an inward pressure difference of 42.2 psi will buckle a 0.025-inch wall stainless steel tube and 51 psi will buckle a 0.035-inch wall zirconium tube.

TABLE IV
CRITICAL BUCKLING PRESSURE FOR 2.8-INCH DIAMETER
STAINLESS STEEL AND ZIRCONIUM ALLOY TUBES

(Perfectly Circular, No Ovality)			
Mean Tube Radius (in.)	Wall Thickness (in.)	Buckling Pressure (psi)	
		Stainless Steel	Zirconium Alloy
1.41	0.010	2.7	1.2
1.41	0.015	9.3	4.0
1.41	0.025	42.2	18.3
1.41	0.035	117	51

In the SGR, the critical location in the coolant tube for buckling will occur at the upper end of the tube where the sodium pressure is a minimum, the collapsing pressure difference a maximum, and the tube temperature a maximum (about 900° F).

If the tubes have a slight amount of ovality, the critical buckling pressure is reduced. In this case the dangerous value of the pressure is that value at which yielding of the material begins.

Assuming an initial ovality of the coolant tube corresponding to a difference between the maximum and minimum diameters of 0.030 inch and taking properties of stainless steel and zirconium at 900° F, the critical pressure for yielding of a 0.025-inch stainless tube is calculated to be 36 psi and for a 0.035-inch zirconium alloy tube is 40 psi (Fig. 13).

E. Design Study of Reactor Core Tank

One type of design for the sodium-graphite reactor involves using the tank construction with a stainless steel inner core tank to protect the graphite from the sodium coolant. This design arrangement was discussed in Section IV of NAA-SR-274. It has a disadvantage of imposing difficult engineering problems in fabricating and handling the core tank if the reactor is designed for high power output and consequently requires a large core volume. In order to evaluate the feasibility of employing this type of design, the Consolidated Western Steel Corporation was requested to study a core tank for a sodium-graphite unit capable of producing 167 thermal megawatts. The Consolidated Western Steel Corporation prepared a practicable design believed capable of meeting the various requirements of strength, sodium tightness, and resistance to thermal shock. An estimate of engineering and construction costs was included in their design report.

The core tank assembly, as shown in Fig. 14, consists of an inner tank which contains the graphite moderator and vertical through-tubes which are the coolant tubes. Both the tube sheets and the shell of the inner tank are made of 1-inch thick Type 316 stainless steel because of its excellent high temperature properties and resistance to liquid sodium at elevated temperatures. The sodium coolant enters the bottom of the coolant tubes and flows upward into a sodium header tank located above the tank containing the graphite.

Fig. 13. Critical Coolant Tube Collapsing Pressure

Fig. 14. Reactor Core Tank Assembly

The outer tank acts as a support for the inner tank as well as a container for the coolant. A support ring welded around the inner tank rests on a step or shelf of the outer tank. Since only the inner side of the outer tank will be in contact with the sodium, the tank is made of 20 per cent Type 316 stainless steel clad plate with an A-285 Firebox quality backing. A supporting foundation and space for thermal insulation is provided below the outer tank.

The entire assembly will be of welded construction, and all applicable laboratory and shop tests such as X-ray, sonic, and other tests will be used. After the entire assembly has been completed and placed in its operating position, Consolidated Western Steel recommends the following installation tests: The core tank will be preheated using dry helium, sodium will be introduced, and heating will continue until a temperature of 1000° F is reached in the upper plenum. After operating for several days at this temperature the assembly will be subjected to a series of rapid thermal cycles between 1000 and 500° F.

Stainless steel bushings are welded into the tube sheets of the inner tank; test specimens have been fabricated and welded in order to select the best joint. The stainless steel coolant tubes (2.5-inch OD x 0.025-inch wall thickness) which pass through the graphite are joined at each end to the bushings in the tube sheets. Since a leaking coolant tube must be replaced, brazing was selected for joining the tubes to the bushings because it is more readily controlled from a remote position than welding. Also, brazed joints can be made and unmade by heat alone. Of several methods investigated to effect a brazed joint (torch, resistance, induction, and Thermit heating) the induction heating seems to offer the most promise of success. Two types of tube joints were successfully made using Nicrobraz and a nickel-manganese alloy. The joints were then reheated to determine if they could be unbrazed. The Nicrobraz joint would not come apart even though the stainless steel tube surface had started to melt. The specimen brazed with the nickel-manganese alloy separated readily on reheating and after cooling to room temperature was again brazed without adding any brazing alloy. The joint was brazed a total of four times and unbrazed three times without difficulty.

F. Coolant Tube Replacement (M. Mueller)

The problem of replacing a coolant tube in the tank-type design employing an inner stainless steel core tank may be especially difficult. A procedure was

suggested in which a coolant tube might be replaced by removing the fuel element and forcing the liquid sodium from only the affected tube through the use of a caisson. For this operation, a means of cutting out the coolant tube, replacing it by a new unit, and rewelding, all by remotely controlled apparatus, would have to be available. All of these operations, except for welding, can be performed by equipment and techniques developed by the petroleum industry. The means for joining the new coolant tube in place by means of remote welding has not been available, however. In order to explore the feasibility of developing apparatus for this operation, work was undertaken to devise heliarc equipment for remotely joining tubes.

The operation just described must be done in actual practice in a region of high radioactivity and a temperature of some 500° F. The basic limitation in the design of the welding tool is the inside diameter of the coolant tube. The tool must be lowered from a shielded coffin into the tube to the point where a weld must be made. The multichambered coffin which houses a coolant tube plug, cutting, grappling, joint dressing, and welding tools is centered over the coolant tube and sealed to the reactor face after the fuel element has been removed. Helium under pressure is admitted to the tube to force the sodium level in the tube below the tube bottom. The plug is then lowered and seated in the bottom where it acts as a stop and center for the cutting, grappling, dressing, and welding tools; each is lowered in succession as the coffin chambers are automatically rotated for each operation.

Heliarc fusion welding is considered to be the most feasible method, and the welding tool is primarily a modified heliarc torch (Fig. 15). The torch body is fixed at right angles to a 0.5-inch diameter hollow shaft and contains a chuck which holds the welding electrode. Argon is fed down through the hollow shaft of the torch to a series of small holes around the chuck; a stream of argon is directed at the point of weld to assure steady arcing in a pure argon atmosphere. At the lower end of the torch shaft is a floating center which engages the plug in the bottom of the tube. It assures concentric rotation of the welding electrode but allows it vertical freedom.

Above the chuck body is a 3-legged spider with wheels on the legs which performs the function of locating the electrode in the proper vertical position

Master Bellows for
Arc Gap Control

Positioning Spider

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Centering Plug

Electrode

SRP38-2B
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Fig. 15. Coolant Tube Remote
Welding Tool

by running on the lips that have been spun on the end of the new tube. It also transfers the weight of the torch to the lip and thereby assures concentric stability during rotation.

Controlled arc gap is necessary in order to obtain a sound weld. The arc gap is controlled by a master bellows which is hydraulically coupled to a slave bellows mounted on the electrode chuck. Pressure on the master bellows located outside the reactor is translated into linear motion of the electrode attached to the slave bellows. This hydraulic system may be operated by hand or by an electromechanical device which senses a very small arc voltage variation (due to change in gap length) and automatically adjusts the master bellows to correct the arc gap.

The torch shaft is rotated by a variable speed motor at speeds of 1/2 to 4 rpm, and welding current is supplied by a standard welding machine.

For simulating reactor conditions (except radiation), test samples to be welded are placed in a steel pipe which is lined with a cylinder of graphite representing the moderator. To simulate reactor shutdown temperatures (500° F) a sheathed cylindrical heater is placed around the graphite. Above the pipe is a glass cylinder 8 inches in diameter by 12 inches high capped by a blind flange; this arrangement forms a helium chamber in which the experiment may be observed. After the welding technique has been developed by visual observation the glass cylinder will be replaced by a 5-foot section of 6-inch pipe and the short torch shaft by a longer one with an intermediate spider to prevent deflection. This arrangement will permit completely blind operation.

The equipment is now assembled and ready for initial operation. The first experimental welds will be made on sections of stainless steel tubing 2.80 inches OD by 0.025-inch wall thickness and having 1/4-inch spun flanges. The fusion weld will be on the edge of adjoining flanges.

G. Zirconium Studies Related to the Canned Moderator (F. E. Bowman, W. Cockrell, and F. Burnett)

Problems related to the fabrication and operation of the zirconium canned graphite moderator are now being investigated. An evaluation of the design requirements and the zirconium alloys available has resulted in the selection of

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Zircalloy I for preliminary experimental studies. Tensile coupons of welded sheet have been fabricated and three specimens will be tested at each of four temperatures, room temperature, 450°, 700°, and 1000° F.

Equipment has been designed for an extensive program regarding sodium-zirconium compatability. A pump loop is to be constructed for dynamic corrosion studies of weld specimens as well as stress-rupture and galling studies. The sodium temperature in the specimen chamber will be 1000° F while in the cold leg of the loop it will be 500° F. This temperature difference will permit mass transfer measurements to be made.

Zirconium for the construction of two full size moderator elements has been ordered. The zirconium sheets (10 feet by 22 inches by 0.050 inch) have been rolled at the Bureau of Mines, and the tubing is being fabricated by Trent Tube Company.

The can head will be subjected to stress reversals caused by temperature differences between the can wall and the coolant tube. It is considered essential to establish the ability of the material and the design to withstand a number of such reversals, and tests are underway to determine the fatigue characteristics of full size can heads of various designs. Current tests are being conducted in air at room temperature, and preparations are being made to conduct the experiments in a combination sodium-helium environment at operating temperatures.

A series of tests have been planned to determine the effect of high temperature sodium on zirconium, graphite, and stainless steel. The first test assembly of a series has been tested. The test assembly consists of a graphite cylinder 2-1/2 inches long by 1 1/16-inch diameter which is encased in a zirconium tube jacket with a 1/16-inch clearance on the diameter. End caps welded into the zirconium tube contain 1/8-inch diameter holes to provide access for the sodium to the graphite. This assembly was placed in a stainless steel capsule filled with filtered sodium, and the entire system was then held at 1000° F for 720 hours. The zirconium can assembly was then removed from the steel capsule and the sodium was distilled out of the graphite by heating to 1100° F in vacuum. The zirconium can was then opened, and it was found that the graphite had cracked extensively and had swelled to form a tight fit in the can. There was no measurable distortion of the zirconium and no observable

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carburization of either the zirconium or the stainless steel. It was possible that some of the effects on the graphite resulted from the method used to open the can assembly and remove the sodium. Additional tests are being made in order to obtain a more quantitative indication of the effect of sodium, zirconium, graphite, and stainless steel on each other at temperatures approximating the reactor operating conditions.

Design drawings have been prepared for the fabrication of a prototype zirconium canned graphite moderator unit as shown in Fig. 16. The welding equipment and methods have been selected and tested, and the first zirconium can of 0.050-inch zirconium sheet is being fabricated. A graphite billet is being machined for the unit. Corrugated can heads to accommodate the flexing due to thermal cycling of the unit will be used.

Tolerances and clearances between adjacent moderator cans were determined for the SRE (see Section V), and dimples in the can surfaces are being considered for establishing the inter-can spacing. Experimental dimples which have been made to date contain cracks; the forming will now be tried as a coining operation. Dies are being made now to form dimples having a diameter of 0.375 inch.

H. Zirconium-Stainless Steel Joints (F. E. Bowman)

Flash welded joint specimens of 1-5/8 inch diameter tubing have been made and are being leak tested. Because the major portion of the eutectic material has been squeezed out during the welding process, these joints appear to have appreciably more ductility than those prepared by diffusion brazing. Specimens for tensile and bend tests as well as for thermal cycling, sodium corrosion, and hydraulic tests are now being prepared.

I. Weldability Studies of Type 304 L Stainless Steel vs Nickel (E. Reed)

In current reactor design studies nickel and Type 304 L stainless steel, in addition to other materials, are being considered as structural materials for the inner and outer tank assemblies. Some difficulties have been reported from post weld cracking of Type 347 stainless steel in 1-inch thick sections such as required for the outer tank. This difficulty is not expected with thick sections of nickel or Type 304 L stainless steel. In this connection some welding experiments were initiated to compare the weldability of nickel with Type 304 L stainless steel and to check for post weld cracks in thick sections.

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Fig. 16. Zirconium Canned Graphite Moderator Cell

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1. Nickel Welds - Figure 17 shows photographs of some welded nickel bend test specimens cut from 0.25-inch welded plate. Photomicrographs of nickel welds made by an inert gas arc and by a metallic arc are shown in Figs. 18 and 19.

In order to conduct sodium corrosion tests on nickel welds it was decided to weld together two vessels of 0.25 inch thick plate. Figure 20 shows both the component parts and a completed vessel. The cylindrical parts of the vessels are made from prebeveled 0.25-inch by 4-inch by 15-inch plates. These were metallic arc welded along one edge and then formed into cylinders. The first vessel completed was found to be leak tight when tested by a helium mass spectrograph. An X-ray examination indicated two small defective areas; these will be ground out and rewelded. Although the warped top and bottom plates indicate considerable residual stresses, no evidence of cracks has yet developed. Nickel tubes welded into the top of the vessels are for the introduction of liquid sodium.

Figure 21 shows 1-inch thick nickel sections made by multiple pass metallic arc welds. Material for welding thick sections was provided by using two halves of a 2-inch diameter by 3-inch nickel rod such as is shown at the top in the photograph. Both W6 and W8 specimens were restrained during welding by "tacking" the halves to 1-inch thick mild steel plates. Afterwards, the "tack" welds were cut away and the rough weld surfaces "milled" smooth. Both specimens exhibit sound welds by X-ray examination with no evidence of post weld crack development and very few inclusions. Weld slag was removed carefully after each pass both by "chipping" and grinding. In a few cases where "under-cutting" was evident, the weld surfaces were sandblasted before grinding.

2. Type 304 L Stainless Steel Welds - Two corrosion test vessels of Type 304 L stainless steel similar to the nickel vessels described above are to be made. Specimen welds on 0.25-inch and 0.5-inch steel plate were made by both the inert gas arc and the metallic arc methods. The weld metal appears to be sound, the porosity is low, and there are no indications of cracks. The photograph in Fig. 22 shows a typical inert gas weld area, and Fig. 23 shows an area where the metallic arc method was applied over an initial inert gas arc weld.

SRP47-24B

Argon arc welded using 20 cfh of argon with addition of 0.5 cfh hydrogen gas. Hydrogen was added in an effort to reduce the fine porosity observed in heliarc nickel welds, but apparently this is too much hydrogen since the welds were embrittled. Further tests with lower hydrogen additions are in progress.

SRP47-24B

Welded with arc shield composed of 30 cfh of argon plus 10 cfh of helium. This gas mixture resulted in a smoother arc and reduced porosity. No cracks are evident, although the weld sections have been elongated about 25 per cent.

SRP47-24B

Metallic arc welded using No. 131 coated welding electrodes. No cracks are evident and fine porosity in the weld appears to be less than with the inert gas arc method.

Fig. 17. Nickel Weld Bend Test Specimens

Inert Gas Arc
Nickel Weld

Parent Metal

Fig. 18. Inert Gas Arc Nickel Weld, 50x
20 cfh argon plus 10 cfh helium

Metallic Arc
Nickel Weld

Parent Metal

Fig. 19. Metallic Arc Nickel Weld, 50x
No. 131 Coated Electrode

SRP47-24E

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Fig. 20. Welded Nickel Vessel

SRP47-24B

Fig. 21. One-Inch Thick Nickel Weld Specimens

From the tests made to date a preliminary comparison of the weldability of nickel plate vs Type 304 L stainless steel indicates the following:

- a. Inert gas arc welding of nickel metal by ordinary techniques results in porous welds. A mixture of argon and helium reduces the porosity. Hydrogen is to be tested.
- b. Metallic arc welding of nickel gives a sound weld, but the flow characteristics are noticeably poorer than for stainless steel and the machine settings are therefore much more critical.
- c. Metallic arc welding of 1-inch thick nickel metal sections has been accomplished without evidence of post weld cracking.
- d. Inert gas arc welds of Type 304 L stainless steel using an argon flow are excellent.
- e. Machine settings for a metallic arc weld of Type 304 L stainless steel are not especially critical, and a good weld may be made. The slag formed on the stainless steel was much easier to remove by "chipping" than the slag from a metallic arc nickel weld.

Inert Gas Arc
304 L Weld

Parent Metal

Fig. 22. Inert Gas Arc Type 304 Stainless Steel Weld, 50x
35 chf argon gas

Metallic Arc
304 L Weld

Inert Gas
Arc Weld

Parent Metal

Fig. 23. Metallic and Inert Gas Arc Type 304 Stainless
Steel Weld
No. SW 262 uncoated electrode

(W. C. Ruebsamen and F. J. Shon)

For the first 15 microampere-hours of irradiation, the capsule was maintained in vacuum, at a residual air pressure of 10^{-4} millimeters. Because of the heating due to the beam, it was necessary to keep the beam current at 1 microampere or less. The capsule was thereafter maintained in an atmosphere of helium, in which higher currents could be tolerated. The specimen received a total of 290 microampere-hours of proton irradiation during 50.2 hours, for an average of 5.8 microamperes. The beam was non-oscillated. The energy of the protons incident on the graphite, allowing for their passage through the monitoring foils, was about 8.5 Mev. The range of such protons in graphite is almost exactly the thickness of the graphite disk used in the experiment, so that the irradiation power density was a maximum at the graphite-stainless steel interface. Discoloration of the stainless steel surface, however, indicated that the beam actually did enter the stainless steel.

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Fig. 24. Carburization Test Capsule.

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Photomicrographs were made of the irradiated stainless steel specimen and an unirradiated portion of the same specimen. No evidence of carburization is visible. The irradiation at the graphite-stainless steel interface is calculated to be equivalent to 2930 mwd/ct at Hanford. Based on this experiment it is felt that there is no possibility of carburization resulting from the use of stainless steel in contact with graphite in a reactor at 600° C.

K. Radioactive Stainless Steel Mass Transport (A. M. Saul)

Experiments on the mass transport of stainless steel by sodium were initiated at NAA in early 1953 in order to determine the shielding requirements and accessibility conditions for components of the coolant circuits external to the reactor core.

Type 347 stainless steel foils 2.0 inch by 0.5 inch by 0.005 inch were irradiated in the MTR for a total integrated thermal neutron flux of 1×10^{18} . The foils (Fig. 25) were degreased and lightly pickled before irradiation.

Each irradiated foil was attached to the end cap of a Type 347 stainless steel capsule which is made of 1 inch diameter by 12 inch long tubing having a 0.065-inch wall (Fig. 25). Each tube is fitted with an end cap containing an "L" bracket for positioning of the foil. This prevents the foil from touching the walls of the capsule during the experiment. The capsules and end caps were degreased and pickled before their use. The sodium used was filtered under high vacuum, but received no other treatment.

The end cap was inserted into the capsule, and the capsule was welded shut and leak tested. Each capsule was degassed by heating under vacuum for 24 hours at 800 to 900° C and then filled with 93 grams of filtered sodium. This is the calculated amount of sodium needed to fill the capsule at 700° C. Capsule No. 2 was inadvertently opened to air while filling with sodium, so that the oxide content was quite high.

The temperatures planned for the experiment were approximately 700° C on the bottom and 300° C at the top of each capsule. This approximate temperature profile was obtained by placing the base of the capsule in a Nichrome heater which was insulated with Asbestos-Cel, and leaving the upper 8 inches of the capsule exposed to room air. Five chromel-alumel thermocouples were spot-welded at 3-inch intervals along the length of each capsule, and continuous

Fig. 25. Type 347 Stainless Steel Mass Transport Foil and Capsule

temperature readings were recorded. Capsules No. 1 and 2 were heated for 330 hours, and capsule No. 3 was heated for 1464 hours.

After removal from the furnace, capsules No. 1 and 2 were opened and the sodium dissolved by methanol. The capsules were then sectioned into 2-inch lengths and a gross gamma measurement made with a vibrating-reed electrometer. The residual activity of the irradiated foil was also measured. The sections from capsule No. 1 did not exhibit any appreciable activity above background, so that further radiochemical study was not made on this capsule. Capsule No. 2, on the other hand, was quite active. Therefore, a radiochemical assay was made on each section of this capsule for iron, cobalt, tantalum, antimony, chromium, and manganese. The irradiated foil was also analyzed for residual activities.

From the two capsules which have been run to date, it is not possible to draw firm conclusions. However, several points are notable:

1. The presence of oxygen in the sodium greatly accelerates the transfer of activity.
2. Deposition of gross activity on initially inactive surfaces does not seem to be a function of the surface temperature, alone. (This verifies the observation of KAPL Memo CTU-3).
3. There is some evidence that the Ta-182 activity tends to concentrate at surfaces whose temperature is intermediate between the 300 and 700° C extremes of the present experiment.
4. There appears to be a manganese activity (probably Mn^{54}) in the Type 347 foil, which has a small tendency to transfer. This activity has not been previously reported.

It is planned to make detailed radiochemical assay of capsule No. 3 if gross activity measurements on the capsule indicate that such an assay is warranted.

L. Fuel Element Development

1. Dimensional Stability of Uranium Alloys under Thermal Cycling

(B. Hayward) - Uranium alloy specimens made by powder metallurgy methods were cycled 500 times between 100 and 500° C and were then cycled 500 times between 200 and 700° C. The specimens were approximately 1-inch

long and varied between 3/8-inch and 1/2-inch diameter. The changes which were observed are given in Table V.

TABLE V

THERMAL CYCLING OF URANIUM ALLOYS BETWEEN 200 AND 700° C

Material →	U-Cb 1.60 w/o Cb	U-Si 0.42 w/o Si	U-Cr 1.0 w/o Cr
Average change in diameter, %	-6.4 to +3.9*	+4.4	+4.2 to +3.6*
Length change, %	+5.0	+5.5	+1.7
Surface after test	Oxidized, rough	Oxidized, rough	Oxidized, rough
Warp in 1-inch length	0.007 inch	0.026 inch	0.024 inch
Hardness, DPH	+64	+5	+52

*These specimens were oval shaped.

Metallographic examination of the specimens indicated an increase in grain size in all specimens with apparent precipitation occurring in the U-Cr and U-Si alloys and definite spheroidization in the U-Cr alloy. Photographic comparisons of each alloy before and after these tests are shown in Figs. 26 and 27.

Additional uranium alloy samples have been received from Sylvania and are currently being cycled. These specimens include U-Nb, U-Bi, U-Cr, U-Al, U-Si, and U-Mo alloys prepared by hot-pressing.

Unalloyed uranium plate samples prepared by powder metallurgy were cycled between 100 and 500° C. The initial series of specimens became severely oxidized and measurement comparisons were not worthwhile, but visual observation after 392 cycles indicated the samples were relatively unwarped or distorted as compared with rolled plate specimens which distorted and warped considerably. A second series of plate samples were cycled 500 times between 100 and 500° C with the results shown in Table VI and Fig. 28. To further evaluate the effect on dimensional stability of an increase in the maximum uranium temperature (650° C, α - β transition) thermal cycling tests have been initiated in which

SRP47-11A

Specimens after additional 500 cycles (200 - 700° C)

SRP47-5

1.60 w/o Cb

0.42 w/o Si

Alloy specimens after 500 cycles (100 - 500° C)

Fig. 26. Thermally Cycled Uranium Alloy Specimens (1.60 w/o Cb and 0.42 w/o Si).

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Specimen after additional 500 cycles (200 - 700° C)

1.0 w/o Cr

Alloy specimen after 500 cycles (100 - 500° C)

Fig. 27. Thermally Cycled Uranium Alloy Specimen

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A

Sylvania
Electric
Products - F

Pressed at
11 tons/in.²

B

Sylvania Electric Products

C

Alpha Rolled

Note: Refer to Table VII

Fig. 28. Thermally Cycled Uranium Plates

TABLE VI

THERMAL CYCLING OF URANIUM PLATES BETWEEN 100 AND 500° C

Material →	A SEP-F Pressed at 11 ksi	B SEP	C d-rolled
Original size, inches	4 by 1-1/8 by 1/4	8 by 1-3/8 by 3/8	2 by 1-1/8 by 3/16
Average change in width, %	+0.4	+0.3*	- 5.0
Average change in length, %	- 0.4	+0.3	+14.0
Average change in thickness, %	+0.6	-0.3	- 5.8
Change in appearance	Slight warp	Warped on long axis	Bowed at center one end curved
Warp (in total length), inches	0.009 (on one corner)	0.044	0.034

*The change in width increased from zero at one end to +0.6% at the opposite end.

specimens are being cycled within a relatively short temperature range ($\sim 50^\circ\text{C}$) in (1) beta phase only, (2) bracketing the α - β phase transition temperature, (3) alpha phase only. The effect of rapid cycling with both large and small temperature increments via a fast quench (similar to recent Hanford work reported in HW-28305) is also being examined. Although these tests do not duplicate pile temperature conditions, the information gained will help to indicate both the effect of minor fluctuations into the beta phase and the deformation occurring within uranium which is continually changing phase.

Alpha rolled specimens and powder compacted specimens (5/8-inch diameter by 3 inches) were cycled 200 times between 630 and 680° C in a low heat capacity furnace with each cycle consisting of a 9 minute heating period and a 6 minute cooling period. All specimens had rough jagged surfaces and some showed considerable warp following the test. Two alpha rolled specimens were split lengthwise to examine the internal structure for cracks. The grain size was large with evidence of extensive strain; small voids were most numerous

parallel to the long axis and extended throughout the specimen. The powder compacted slug had a severely pimpled surface on which could be seen sharp radial cracks. The discrepancy in appearance between the surface, end, and warp characteristics of the two alpha rolled slugs may have been due to relative positions in the furnace which is known to have some temperature gradients. Results of these tests are shown in Table VII and Fig. 29.

TABLE VII
THERMAL CYCLING OF URANIUM BETWEEN 630 AND 680° C

Material →	α rolled	α rolled	α rolled	Powder Compacted
Size of specimen, inches	5/8 by 3	5/8 by 3	5/8 by 3	5/8 by 3
Average change in diameter, %	+7.4	+8.0	+3.0	+3.4
Change in length, %	-5.4	-4.8	-3.3	+1.0
Warp in total length, inches	0.083	0.157	0.020	0.057
Surface appearance	Very rough	Very rough	Wrinkled, one short long crack ends bulged	Very rough, radical cracks

2. Uranium-Zirconium Alloy Fuel Slugs (B. Hayward) - Conferences with FMPC and ANL have established the feasibility of making a production scale test for the manufacture of uranium-zirconium (2 w/o Zr) alloy slugs. The purpose of the test is to establish a source of supply and to develop processing techniques and cost data for future orders of a slightly enriched uranium-zirconium alloy. A formal proposal for this work is now being prepared.

The major problem which appears to require some development work is to obtain a homogeneous ingot. There are three or four approaches to this problem: (1) use a non-carbon containing reduction liner to prevent segregation due to zirconium carbide formation in the derby during co-reduction of both fluorides, (2) prepare master alloys which minimize the zirconium carbide

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Alpha
Rolled

Alpha
Rolled

Alpha
Rolled

Powder
Compacted

Uncycled

SRP47-26A

Note: These specimens were thermally cycled 200 times between 630 and 680° C (15 minutes/cycle).

Fig. 29. Thermally Cycled Alpha Rolled Uranium Specimens

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segregation which appears to reach a saturation, (3) use direct alloying methods in the casting furnace with ThO_2 coated crucibles.

The fabrication of the alloy ingot appears to be straightforward, with breakdown and rolling being done at about 650° C. The heat treatment following the rolling operation would be by either of two methods: (1) 775° C for 1 hour plus air cool, or (2) 775° C for 1 hour plus 1 hour at 500° C followed by a water quench. The preferred heat treatment will depend on ease of manufacture and metallurgical examination. The heat-treated rods would then be machined into slugs.

3. Thermal Conductivity of Uranium (J. J. Droher and F. A. Domingo) - The thermal conductivity of the fuel slug material at elevated temperature is of primary importance in any power reactor design. At present there are insufficient data for uranium at temperatures above 750° F. In order to provide some of this needed information, a program of elevated temperature thermal conductivity measurements has been undertaken. The apparatus which has been developed has been described in a previous quarterly progress report (NAA-SR-227, page 93). It is based upon measurement of the temperature gradient in a cylindrical disc for radial heat flow.

Measurements have been carried out with alpha rolled uranium in the "as received" and beta quenched condition. The data (66 values) obtained from the "as received" material between room temperature and 1220° F have been evaluated by the method of least squares. The following equation resulted from the evaluation:

$$K = 12.39 + 5.29 \times 10^{-3} T + 0.998 \times 10^{-6} T^2,$$

where K is the thermal conductivity expressed in Btu/hr sq ft °F per ft and T is the temperature expressed in degrees Fahrenheit. This quadratic equation is plotted in Fig. 30 and results in a curve that is slightly concave upwards.

To determine the effect of the beta phase on the conductivity several runs were conducted with gradients containing the transformation temperature of 1220° F (660° C). Heating curves were plotted from the thermocouple readings by a multipoint recorder. No appreciable discontinuity was observed when the transformation temperature was reached. - This could possibly be accounted for on the basis that the transformation proceeded slowly from the center outward

Fig. 30. Thermal Conductivity of Uranium

rather than a simultaneous transition of the entire mass. Conductivity values were calculated and plotted, but displayed no appreciable deviation from the extrapolation of the data observed in the alpha region.

Another set of specimens was heated to 1380° F (750° C), held at temperature for 10 minutes, and then water quenched. The method of least squares was also applied to points obtained with these beta quenched specimens and yielded the equation:

$$K = 13.06 + 5.45 \times 10^{-3} T + 0.0989 \times 10^{-6} T^2$$

for the range from 20 to 1220° F. This curve is also plotted on Fig. 30. It is similar to the curve for the "as received" material and is almost linear. Repeated temperature excursions of both the inner core and the entire specimen into the beta phase showed no appreciable effect on the conductivity values obtained at lower temperatures. This should answer the question raised by the Canadian investigators:⁴ "Is it possible to stabilize the conductivity in the alpha phase either before or after transitions from the beta phase?" The data indicate that the conductivity for the alpha rolled uranium is essentially stabilized, at any given temperature, both before and after transitions into the beta phase.

With the entire specimen in the beta phase, eleven equilibria readings were taken. The equation

$$K = 17.39 - 5.46 \times 10^{-3} T + 6.51 \times 10^{-6} T^2$$

for the temperature range 1220 to 1420° F (660 to 770° C) was obtained by the method of least squares. This equation is also plotted in Fig. 30, through the beta phase temperature range. There is a slight discontinuity between the conductivity curves for the alpha and beta temperature regions. This may not be a real difference due to the phase change, but may lie within the error which could be introduced in dividing the data into two groups. From the curve, it may be seen that in the beta region the conductivity continues to increase at an even greater rate with an increase in temperature.

All attempts to heat the specimens into the gamma phase have been unsuccessful due to heater burnouts. Size limitations on the uranium specimens in this radial method impose a limitation on the size of the internal heater which results in a severe heating load.

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4. Fuel Rod Assembly (H. Strahl, K. Johnson, and R. Crosgrove) - One of the fuel elements proposed for the sodium-graphite reactor is 151 inches long and contains seven individual rods. The rods are made up of stainless steel tubes 0.790 inch in diameter and with a 0.010-inch wall, each of which contains 24 uranium slugs, 0.750 inch in diameter and 6 inches long. The annulus between the slugs and the jacket is filled with sodium to permit maximum heat transfer rates.

The assembly of the fuel rod is complicated by the necessity of handling sodium in either a vacuum or an inert gas atmosphere. Operations which must be carried on under these conditions are:

- a. Outgassing of the uranium slugs and the jacket tube (700° F)
- b. Melting and introducing sodium into the jacket tube
- c. Introducing the slugs into the thin wall tube without rupturing or distorting it
- d. Placing and welding end closures
- e. Leak testing to insure tight welds

A cutaway drawing of the fuel rod loading mechanism is shown in Fig. 31.

In order to reduce the possibility of hot spots in the fuel slugs, a continuous bond of the slug to the jacket by sodium is desired. Several methods of treatment of the loaded rod to insure this condition are under investigation and work is being done on inspection techniques to reveal the presence of voids in the bond.

The loaded and inspected rods are assembled into clusters of 7 rods, and this unit is then called a fuel element. The center rod and each alternate rod is wrapped on a 12-inch pitch with a 0.090-inch diameter stainless steel wire to insure no line contact between rods and to permit the coolant sodium to flow between them.

Fourteen full size fuel rods have been fabricated; one element of seven rods is to be used in a hydraulic test loop. The other seven rods comprise an element for fuel handling experiments. The photograph of an assembled element is shown in Fig. 32.

The fuel element for the hydraulic test loop contains Type 304 stainless steel jackets. The purpose of this test is to establish the pressure drop and

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Fig. 31. Fuel Rod Assembly Apparatus

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Fig. 32. Seven-Rod Fuel Element

flow characteristics of the coolant around the fuel element using water to simulate the reactor coolant. Kerosene is used as a bonding agent in these rods in place of the sodium bond in the actual SGR fuel elements. Sodium was not used because:

- a. Of the hazard involved if the jacket should rupture with water present.
- b. The temperature using sodium would have to be about 700° F to simulate actual reactor conditions.

Kerosene was selected because:

- a. The viscosity of kerosene at 130° F is comparable to that of sodium at 700° F.
- b. Negligible corrosion is expected with uranium.

In order to study the heat transfer bond and to assist in interpretation of the results of nondestructive tests of the loaded fuel rod, four of the rods of the fuel element to be used in the fuel handling experiment were bonded with NaK while sodium was used in the remaining three. In addition, the uranium slugs were subjected to different surface treatments in an attempt to relate the surface condition to the wetting action of the bond. Of the total of 154 slugs used, 24 were etched in dilute nitric acid, 24 were polished with emery paper to remove surface oxide, and the remainder were used in the "as received" condition.

5. Nondestructive Testing of Fuel Element Components (J. Droher and R. Roush) - The SGR fuel element has three components: (a) uranium slugs, (b) tubular cladding, and (c) bonding material. It is important to detect any flaws in the physical structure of the fuel element before it is loaded into the reactor. Various nondestructive testing methods have been investigated for detecting flaws in the fuel element components. Results of bond tests have also been used to establish optimum time-temperature conditions for the fuel rod assembly operation. Primary effort has been expended on tests for bond voids using the Cyclograph eddy-current apparatus. Other methods including autoradiography, radiography, infra-red photography, ultrasonics, frost tests, and temperature sensitive paints have also been investigated.

a. Uranium Slugs - The uranium slugs are given a preliminary screening by the fabricator. Prior to assembling a fuel rod, the slugs are examined visually for surface defects, and dimensional checks are made. A

weight check of each slug gives an approximate measure of porosity and insures that there are no internal cavities. Finally, the slugs are passed through the test coil of the Cyclograph to detect any internal seams, inclusions or cracks.

b. Jacket Tubing - The thin wall tubing is examined visually for any obvious flaws or defects and is then checked for structural variations using the Cyclograph. The bottom cap is welded into the tube and the empty tube is then evacuated from the open end. A helium mass spectrometer leak-detector is used to find any submicroscopic pinholes in the tube wall or the bottom end-cap weld.

c. Bonding - After heat treatment to insure wetting of the uranium slug and stainless steel tubing by the sodium bonding, the fuel rod is cooled to room temperature. The bond is then checked for voids. The maximum permissible void sizes for the SGR fuel rod have been calculated as follows:

<u>Type of Void</u>	<u>Maximum Void Size</u>
360° circumferential void	0.080 inch wide
Axial void (extending rod length)	0.140 inch wide
Square void	0.34 inch by 0.34 inch

Voids larger than those tabulated above will offer a sufficient barrier to heat flow during reactor operation to cause localized overheating of the uranium.

d. Cyclograph Bond Testing - Initial development work toward adapting the Cyclograph for testing slugs and fuel rods was done at Argonne National Laboratory.^{5, 6, 7} With the cooperation of Argonne, their modifications were used as a starting point for testing the NAA fuel rod.

A set of test coils is desirable in order to cover a range of test frequencies, since the depth of penetration of the flux from the coils is inversely proportional to the square root of the frequency. Coils have been fabricated to cover the range of 2.0 to 324.0 kilocycles per second. Most of the coils have been fabricated to accommodate tubing since the major effort has been concentrated on void detection in the bond. Flux guides of Croloy No. 70, as introduced by Argonne, are used to restrict the length of specimen acted upon by the coil and hence to increase the resolution.

Before using the Cyclograph to detect defects it is necessary to calibrate its response to known "standard" defects. Known holes and grooves have been machined in slugs of uranium and stainless steel; simulated bonds 0.010 inch thick and 8 inches long were machined with known defects from Type 61-S aluminum tubing. This tubing could be slipped over the uranium slugs and into the thin wall stainless steel tubing, thus allowing the simulated bond defects to be measured. Cyclograph traces for one of the bond standards are reproduced in Fig. 33. Very sharp indications of all voids were noted. The indications increased in amplitude with both the sensitivity setting and the test coil frequency, and best results were obtained with coil frequencies of 2 and 4 kilocycles per second.

Cyclograph traces have been made on all regular uranium slugs which have been loaded to date and no serious defects have been found. Similarly, all tubing has been tested; defects have been found on certain tubes. The bonding of full length fuel rods prepared for the fuel handling experiment has also been tested.

Both sodium and NaK have been considered as bonding agents for the SGR fuel rod, and it is therefore desirable to evaluate their relative merits and disadvantages. The effect of sodium and NaK bonds were essentially alike. It was possible to detect the bond level, regardless of whether it was above or below the slugs. Voids and poorly bonded regions caused an increase in amplitude of the recorder trace. Well bonded regions cause a decrease in amplitude of the trace and the width of the trace variations gives some idea of the width of the defect. Since the coil spools are 0.25 inch thick, this limits the resolution of the instrument. Typical Cyclograph traces for a sodium bond are shown in Fig. 34.

e. Autoradiograph Tests (R. L. Koontz) - Since uranium is naturally radioactive it is possible to wrap a fuel rod with X-ray film and detect variations in the bonding by differences in exposure of regions of the film. It has been found that voids are detectable with autoradiography. Exposure times of 16 hours for Type K film, and 64 hours for Type M film have yielded best results to date. Exposure times can be decreased by slipping the fuel rods into an oversize hole drilled in a lead brick with the film slipped into the annular space. Back scattering by the lead decreases the exposure time but at the same time decreases the film resolution.

Fig. 33. Cyclograph Trace of Standard Defective Bond

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Fig. 34. Cyclograph Trace of Sodium Bonded Fuel Rod

Autoradiography is a valuable technique because it gives a direct indication of the size and shape of the voids existing at room temperature. It can be used in conjunction with the Cyclograph to aid in the proper interpretation of the recorded traces. Figure 35 shows a typical set of autoradiographs.

M. Fuel Handling Equipment

1. Tower Experiment (W. Hallett) - The arrangement and assembly of the prototype coolant tube which will be used for developing the methods for charging and discharging fuel elements has been previously described in some detail (NAA-SR-227, page 104 and NAA-SR-260, page 44). The installation and testing of all components of the system was completed, and the system made ready to be charged with the sodium coolant.

Preparatory to adding sodium to the system the coolant tube was cleaned with a hot alkaline solution, rinsed with distilled water, and dried with heat and vacuum. Purified helium was admitted, and filtered sodium was charged to the tube. The sodium was heated to 800° F and then drained in order to remove any foreign material that may have remained after the initial cleaning. The sodium was filtered again and recharged to the system.

A mock-up SGR fuel element, consisting of a typical cluster of seven solid stainless steel rods instead of uranium slug filled tubes was used in the initial fuel handling experiments. This cluster was inserted by means of the handling coffin into the static sodium contained in the coolant tube. A purified helium atmosphere was maintained during the whole operation. Upon removal from the sodium, which had been maintained at a temperature of 750° F, it was observed that the center rod of the fuel element had warped severely, making the element difficult to remove. The entire fuel element was uniformly coated with sodium indicating good wetting by the liquid metal. The element was washed with water in a helium atmosphere and disassembled. After the washing operation, oxidation of the stainless steel was observed, so prior to reassembly all parts were pickled. The center rod was straightened and detached from the bracket tying it rigidly to the remaining rods so that it could move freely and thus reduce the change of warping again.

During the time this work was underway, the fuel loading techniques had been developed to a point where the manufacture of a sodium bonded uranium

Autoradiograph Showing Defects
Machined into Simulated Bond of
0.010 inch thick 61-S Aluminum
Tubing

Autoradiograph Showing Voids in
Sodium Bond

Fig. 35. Autoradiographs of Fuel Rod Bond Defects

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slug prototype fuel element could be undertaken. The individual fuel rods already made were of sufficient flexibility that it was thought that no warping from thermal effects would be encountered. This fuel element was assembled as described in Section L, 4.

It was planned to measure the volume of water required for washing the sodium from the fuel element and analyze the solution of hydroxyl ion as a means of determining the quantity of sodium removed from the coolant tube upon removing a fuel element. The high mineral content of the well water used at the Santa Susana field laboratory resulted in a large quantity of material precipitating when the sodium went into solution. This masked the end point of the titration, and it was decided to install water softeners in the system.

After reassembly of the mock-up fuel element it was inserted in the coolant tube using the coffin as before. Prior to immersion in the sodium the element was suspended in the helium above the liquid metal in order to preheat it to 500° F. The temperature of the sodium was raised to 800° F after the element was immersed. After immersion in the sodium and subsequent removal, it was noticed that incomplete wetting of the stainless steel by the sodium had occurred and that the unwet areas were blackened by oxidation. The helium blanket over the sodium was analyzed and found to contain 13 parts per million of oxygen and so the system was flushed with NaK bubbled helium. The sodium was drained, filtered, and recharged to the coolant tube.

A prototype fuel element of uranium slug filled rods was assembled and immersed in the sodium following the usual procedure. Upon removal from the coolant tube, the conditions observed in the previous test, namely, incomplete wetting and severe blackening of the surface, were found to exist. Work is now underway to locate and eliminate the source of oxygen and to find a suitable means of removing the oxidation products from the surface of the stainless steel.

2. Shielding Requirements for Fuel Handling Coffin (R. L. Ashley) - The thickness of lead shielding required for the fuel handling coffin is determined by the activity of the irradiated fuel element and the hanger rod. In order to be conservative, calculations were based on the worst possible case - the transferring of the central fuel element of the 30 megawatt SGR. The power of this element is 1700 kilowatts, and the activity (for a 6-month irradiation) following

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various cooling periods was first calculated; the radiation levels at the surface of various thicknesses of lead shield were then calculated. A 2-inch air space (clearance) was assumed between the fuel element and the closest point of the coffin. Figure 36 was plotted using the gamma escape probability curves for a shielded cylinder.

In the choice of a coffin thickness, two factors must be borne in mind; (a) the data in Fig. 36 were calculated for the central element; (b) tolerance as stated by the Subcommittee on External Radiation of the National Committee on Radiation Protection is 300 mr/week. From Fig. 36 this corresponds to a thickness of from 11.3 to 12.2 inches, depending on the associated cooling period. A coffin thickness of 11.8 inches is recommended.

Hanger Rod Irradiation - The fuel elements of the SGR are supported from above by a hanger rod which is approximately 12 feet long and is made of 0.75-inch diameter rod of Type 347 stainless steel. Because of the inconvenience and long equipment required to handle a fuel element-hanger rod combination which is some 25 feet long, it will be desirable to develop a coupling between the element and the rod so they may be separated. The hanger rod must then either be connected to a new fuel element prior to charging into the reactor, or must be discarded. In either case, the activity of the hanger rod must be considered in the development of equipment and handling techniques. The radiation level of a hanger rod following a 6 month decay period is calculated to be 10.1 r/hr at a distance of 100 centimeters. A 1 year cooling period reduces this to 8.50 r/hr. From the calculated radiation levels it appears that access (at 3 feet) for longer than 6 minutes is not possible. Thus, it appears that any manual operation that required contact with the hanger rod by operating personnel is not safe if Type 347 stainless steel is used. Such a manual contact might be involved in operating the coupling or in any handling of the irradiated hanger rod. There are several alternatives:

- a. Discard the used hanger rod.
- b. Specify electrolytically-pure nickel for the hanger rods. This would reduce the cobalt activity to negligible proportions; for a 6-month exposure the radiation level for 6 and 12 month decay periods would be reduced by factors of about 10 and 30, respectively.

Fig. 36. Gamma Radiation Level at the Surface of Various Fuel Handling Coffins

- c. Specify a 1 per cent boron content for the steel. This would reduce the overall activity by a factor of at least 15; here, however, the effect of flux distortion in the reflector may be more important.
- d. Design the hanger rod so that a 2 foot length of the "hot" end (corresponding to the entire length in the reflector) is detachable with the fuel element.
- e. Look for another sodium-resistant material that has a lower nickel content and can still resist the high temperature encountered with the sodium coolant.

It should be mentioned that in three of these alternatives the shielding requirements on removal from the pile are the same. This is due to the fact that the 2.59 hour Mn^{56} isotope is the major contaminant up to about 15 hours of cooling. The exception is item c, in which the overall activity is reduced.

It appears that the problem of handling an active hanger rod may best be solved by the use of a simple coupling joint and remote handling equipment. The development of an adequate coffin and portable shielding for handling the hanger rod is quite feasible. A comparison of this approach with some of the alternatives which have been considered must yet be made.

3. Effect of Oxygen Contamination in Gas Blanket (W. R. Martini) - In order to simplify and thus reduce the cost of the fuel handling system for sodium-graphite reactors, the consequence of not using an entirely oxygen-free technique in charging fuel elements was investigated for the SRE (see Section V).

If complete absence of oxygen is not required, the use of pure helium is an unnecessary expense. Helium costing about \$9/100 CF can be replaced with nitrogen containing 0.3 per cent oxygen costing \$0.75/100 CF. If this commercial nitrogen is used in the inert gas blanket above the liquid sodium the 0.3 per cent of oxygen will eventually combine with sodium and be removed to the cold trap. The same thing will happen to any oxygen that is introduced through the coffin when a fuel element is changed. The question then is: During the life of the reactor will such sodium contamination overload the capacity of the cold trap?

These calculations are based on the assumed design constants for the SRE shown in Table VIII.

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TABLE VIII
ASSUMED DESIGN CONSTANTS FOR THE SRE

Sodium in Radioactive System

(a) Volume	550 ft ³
(b) Average temperature	700° F
(c) Average density	53.7 lb/ft ³
(d) Weight	28,500 lb

Fuel Handling Coffin

(a) Inside dimensions	1 ft by 1 ft by 25 ft
(b) Volume	25 ft ³
(c) Temperature	400° F
(d) Pressure	1 atm
(e) Lb mol gas capacity	0.0398

Gas Blanket

(a) Dimensions	10 ft dia. by 2 ft
(b) Volume	157 ft ³
(c) Temperature	900° F
(d) Pressure	1 atm
(e) Lb mol gas capacity	0.158

The volume of cold trap required to take care of the contamination during the life of the reactor was calculated using the following assumptions:

- a. The oxygen is present in the cold trap as Na₂O.
- b. The free volume of the cold trap can contain as much as 20 per cent Na₂O by weight.
- c. The density of the Na-Na₂O mixture is approximately 60 lb/ft³.
- d. The Na-Na₂O mixture is held on steel net that has 90 per cent void space.

The calculated volume of a cold trap required to hold all oxygen contamination from the fuel element handling coffin during the life of the reactor (1000 element changes) is 1.6 ft³. The volume required if air were used in the coffin is

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REF ID: A65710

N. Sodium Pump Loop (R. Cygan and R. Crossgrove)

To reduce costs, a commercial hot oil pump was selected which has a record of long time service in refineries at the desired temperature (900° F). These units are designed with long stuffing boxes which makes them more readily adaptable for modification to use a freeze-seal. To facilitate replacement a vertical shaft design was adopted.

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Fig. 37. Solubility of Sodium Oxide in Sodium

pressurizing the area behind the freeze-seal with helium. A rotating face seal is used on the upper end of the shaft.

The basic pump specifications are: capacity 1200 gpm of sodium at 900° F, 160-foot head of sodium at rated flow, rated speed 1750 rpm, drive motor 75-hp DC continuously variable from 45 to 1750 rpm.

The loop is of sufficient length to accommodate the instrumentation necessary to evaluate pump performance. This includes devices for measuring sodium pressure, flow, and level. A dump tank is used for storage and cleanup of the sodium. Oxide content will be checked using a KAPL Type II plugging indicator. Flow control is obtained by varying pump speed. The operating head is changed by throttling the flow with a modified angle valve equipped with a freeze-seal stem.

The arrangements of components is shown schematically in Fig. 38. The installation will be made at the Santa Susana site during the spring of 1954. Design of the system is now 80 per cent complete, and all purchased parts are on order. The first subassemblies are scheduled for delivery about January, 1954.

IV. ALTERNATE REACTOR DESIGN STUDIES

A. Hydrocarbon Moderated Power Reactor (E. F. Weisner)

Previous studies of power producing reactors have considered units employing graphite, beryllium, beryllium oxide, and light water as moderating materials. A preliminary study using a hydrocarbon, para-terphenyl, as the moderator has been completed.

The system selected for study consists of the usual seven rod element using 0.75-inch diameter rods clad in 0.020-inch zirconium. The fuel element is contained within a 0.050-inch zirconium tube 2.80 inches ID to channel the sodium flow. A second 0.050-inch zirconium tube surrounds the coolant tube. This is principally to provide a gas gap of about 0.050 or 0.060 inch to insulate the moderator from the sodium coolant. In this manner the moderator temperature can be kept considerably below the coolant temperature if desired. The moderator is circulated through an external circuit to exchange its heat to the

Fig. 38. Sodium Pump Loop

sodium coolant just before the sodium enters the reactor. The system is shown schematically in Fig. 39. This arrangement allows the moderator to operate within a temperature range where its vapor pressure is low and still its heat (between 6 and 8 per cent of the total heat power) is utilized by heating the sodium from 500 to approximately 520° F. The average moderator temperature can be controlled over a small temperature range (500 to 700° F) by controlling its flow rate. The sodium temperature can be controlled by orificing each coolant tube so that the mixed mean outlet temperature will be a maximum within the physical limitations existing for the materials used.

The hydrocarbon selected for this study is para-terphenyl which melts at 213° C (415° F) and boils under atmospheric pressure at approximately 420° C (790° F). Using a critical temperature of 798° C and the appropriate expansion factors, one may compute the expected density variation with temperature in the liquid range. Using an average moderator temperature of 400° C, the density is 0.865 gm/cm³. The hydrogen density for this material then can be determined as 3.17 x 10²² atoms/cm³. Since there exists no experimental nuclear data on this material, one must resort to other means of estimating such quantities as age, thermal diffusion length, etc. The following constants are calculated for C₁₈H₁₄:

$$\begin{aligned} \Sigma_a &= 0.00659 \text{ cm}^{-1}, & L^2 &= 44.9 \text{ cm}^2, \\ \text{"}\Sigma_a\text{"} &= 0.690 \text{ cm}^{-1}, & \text{"}\lambda\text{"} &= 0.36 \text{ cm}^{-1} \end{aligned}$$

The dependence of materials buckling, k^2 , multiplication factor, and conversion ratio on enrichment and cell spacing were determined. The cost of power was determined for several values of cell spacing and amounts of excess k. The minimum power cost occurred for the minimum spacing considered; namely, 4.0 inches on a square lattice (or 4.2 inches on a triangular lattice). Para-terphenyl was assumed to cost 30¢/lb. Approximately 675 ft³ (or 4100 lb) are required in the reactor. Assuming an equal volume external to the reactor, the total hydrocarbon cost is only \$2500.

A breakdown of the estimated costs follows for three 167 megawatt reactors feeding one 500 megawatt power plant.

Fig. 39. Sodium Cooled Hydrocarbon Moderated Reactor

Reactor investment	2.08 mils/kwh
Power plant investment and site development	4.18
Fuel investment	1.91
Fuel feed	4.68
General operation and maintenance	<u>2.00</u>
	14.85
Credit for Pu at 40 \$/gm	<u>4.05</u>
Net cost of power	10.80 mils/kwh

Under the same assumptions, a graphite moderated and reflected system was estimated to produce power for 10.56 mils/kwh.

Each reactor would require a tank 10.4 feet in diameter by 8 feet high to enclose the core and a 1-foot reflector. There would need to be about 492 fuel elements 6 feet long. The enrichment required is slightly more than 1 per cent U²³⁵. The conversion ratio is 0.75. Twenty-seven per cent burnup of the U²³⁵ (2950 mwd/t) could probably be realized. A 1-foot reflector of the same material as the moderator has been assumed. This is not a very good reflector since the thermal diffusion length is so small, and consequently, only a small (~8 centimeters) saving can be expected. A graphite reflector would provide a larger saving but would need to be thicker. The average-to-peak power ratio could be improved, and hence, fuel investment reduced, by the use of a graphite reflector, but the over-all reactor size would be increased somewhat.

From such a preliminary study it is not possible to make many conclusions. It appears that a hydrocarbon moderated reactor could produce power for approximately the same cost as a graphite moderated unit. The cost of the moderator is a very small item, so a relatively high decomposition rate could be tolerated with little effect on cost, providing the performance is not affected. Hydrocarbon moderated units can be quite small, and accordingly, might find application as package power units.

B. Engineering Study of Hydrocarbon Moderator (J. R. Wetch)

A preliminary study of the engineering characteristics of the above sodium-cooled, terphenyl moderated reactor was made based on the available data regarding the hydrocarbon. The following design assumptions were used:

1. Double-walled zirconium coolant tubes of 2.80 inches ID by 0.050 inch wall.
2. Seven 6 feet by 0.75 inch uranium (0.0128 enrichment) rods per fuel element.
3. Approximately 500 fuel elements for 167 megawatt total reactor power with 700 kilowatt center element heat output.
4. Approximately 6 to 8 per cent of the total reactor heat is generated in the moderator and thermal shield. The moderator may cool the thermal shield.
5. The lattice spacing is 4 inches.
6. The sodium outlet temperature from the economizer is 450° F.
7. A tank about 10 feet in diameter and 8 feet high is required which includes a 1-foot terphenyl reflector.

This study is based on meta-terphenyl because more data regarding the physical properties are available than for para-terphenyl. Even so, all the physical properties of meta-terphenyl that are needed to calculate heat transfer coefficients have not been measured. Estimates of the thermal conductivity at normal liquid temperature were made by employing semi-empirical methods. The heat capacity was obtained by assuming that the variation of terphenyl heat capacity with temperature corresponds to the variation of biphenyl and benzene heat capacities with temperature.

1. Thermal conductivity	$\approx 0.06 \text{ Btu/hr-ft-}^\circ\text{F at } 600^\circ \text{ F}$
2. Heat capacity	$\approx 0.46 \text{ Btu/hr-ft-}^\circ\text{F at } 600^\circ \text{ F}$

A generalized viscosity vs temperature curve for terphenyl is plotted in Fig. 40. The effect of reactor irradiation on viscosity has been measured for terphenyl at temperatures up to 700° F and for exposures of 9×10^{18} nvt. The viscosity of the irradiated liquid has been assumed to vary with temperature similarly to the variation experienced by normal liquids. The results of these extrapolations are also shown on Fig. 40.

If the average moderator temperature is 575° F, the rate of change of viscosity vs exposure is plotted on Fig. 41. The mass of the moderator inside the reactor, including the reflector is estimated to be 24,000 pounds, and the mass outside of the reactor is estimated to be 10,000 pounds. Based upon an

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Fig. 40. Viscosity of Terphenyl vs Temperature and Irradiation

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Fig. 41. Viscosity of Terphenyl at 575° F vs Irradiation

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average thermal flux in the reactor of 3×10^{13} neutrons/cm²-sec, the calculated average viscosity of the hydrocarbon as a function of the purification rate is shown in Fig. 42.

In order to prevent boiling of the hydrocarbon moderator at the hottest spot in the reactor, namely, at the outlet end of the coolant tube wall, it is necessary to provide an insulating medium between the tube and the moderator. The annular gap between the coolant tube and the concentric outer tube may be designed to satisfy this requirement. Based upon a 1 per cent side stream purification of the moderator and a helium atmosphere in the annular gap, the maximum moderator temperature is as follows:

<u>Sodium Exit Temperature</u>	<u>Moderator Temperature</u>		
	<u>in</u>	<u>out</u>	<u>max</u>
800° F	550° F	600° F	687° F

These temperatures are calculated upon the basis of the following conditions:

1. The maximum sodium temperature attainable, based on 1200° F maximum uranium temperature and 15 ft/sec sodium velocity is about 1050° F.
2. The boiling point of meta-terphenyl is 714° F. Thermal stability tests indicate that the maximum temperature that may be practically maintained in terphenyl lies between 700 and 850° F. It is assumed that 750° F is the upper limit for design study purposes, thus 8 psig must be maintained to prevent boiling.

A preliminary study of the requirements of the terphenyl to sodium heat exchanger has been made. The sodium will flow on the shell side and the moderator on the tube side in order to minimize sodium pressure drop and heat exchanger construction costs. This comes about because the sodium flow rate is about twice the moderator flow rate and the conductivity of sodium is 45 Btu/hr-ft °F whereas the conductivity of the moderator is 0.06 Btu/hr-ft °F. Thus, the moderator must flow in a small tube at a high velocity in order to provide good heat transfer.

The overall heat exchanger dimensions are about 12.5 feet tube length and about 27-inch shell ID. The total heat transfer surface based on 5/8 inch OD tube is 1450 square feet including a 25 per cent safety factor.

Fig. 42. Average Viscosity of Terphenyl Moderator System vs Temperature

The rates of moderator make-up of interest are near 1 per cent per hour and would correspond to a moderator cost of 2 mils/kwh not including handling and disposal of 34,000 pounds of spent moderator every 4 days. This assumes that no purification is used and that new (30¢/lb) terphenyl is used as the make-up material. This high moderator cost can be reduced by a factor of 2.5 by going to make-up rates of only 0.4 per cent per hour. This alternative, however, would increase the external heat exchanger size by a factor of 2 and the required pumping power would increase by a factor of 37. Thus, it seems obvious that some sort of purification and recycling of the moderator will be required.

At the present time it appears that a fractionating column would be employed in the system. The light fractions, toluene, alkylbenzene, etc. must be removed in order to maintain low system pressures. The heavy fractions must be removed to hold the viscosity to reasonable values for good heat transfer in the external heat exchanger and to maintain low wall temperatures within the reactor itself.

The information presented on Fig. 42 may serve as a guide to determine the necessary by-pass stream flow rates to the fractionating column, if a middle fraction can be obtained that has the same viscosity as the initial charge. If a different viscosity is attained by the middle fraction, Fig. 42 can be revised using that initial viscosity.

The absolute make-up rate to the entire system including the fractionating column cannot now be calculated because the fraction of heavies and lights and their viscosities are not presently known.

The side stream flow rate into the distillation column need only be 5.7 lb/min or 0.75 gpm. Thus, the column will be small and relatively inexpensive.

The pumping cost for the organic amounts to less than 0.01 mil/kwh so it does not seem prohibitive.

The liquid moderator would permit the use of a carbon steel core tank except for the tube sheets.

The possibility of merely plugging a ruptured coolant channel presents itself when a non-sodium reactant liquid moderator is used.

The thermal shield will not need a separate cooling system; it can cool directly to the moderator.

It appears that the use of meta-terphenyl, or a similar fluid, as a sodium cooled power reactor moderator may be feasible. The advantages of a liquid moderator seem very much worth considering. However, in order to accurately determine the design and performance of an organic moderated power reactor, more complete data are required concerning the physical properties of the organics at the temperatures of interest. A knowledge of the percentage of decomposition products in irradiated terphenyl as a function of viscosity and exposure is required. This information is necessary to determine make-up rates and the cost of maintaining the organic moderator. The cost of make-up in turn will determine to a large extent the economic feasibility of using these fluid moderators in a power producing reactor.

C. Reactor Coolant Studies

1. Sulfur as a Reactor Coolant (T. T. Shimazaki) - Sulfur melts at 235° F, boils at 832° F, and has a specific heat of 0.27 Btu/lb °F at 700° F. A viscosity-temperature curve is given in Fig. 43. Due to the high viscosity of liquid sulfur in the temperature range of interest, the coolant velocity is necessarily low for a reasonable pressure drop. Due to the combination of high viscosity and low velocity, the flow is laminar. Consequently, the heat transfer coefficient and, therefore, the heat output of the fuel element are low. A comparison between sodium and sulfur operating at the same inlet and outlet temperatures is shown in Table IX where μ is the mean viscosity in lb/ft-sec.

It is seen that for sulfur the pressure drop is approximately five times that for sodium and the heat output is approximately one-seventh that for sodium. By increasing the flow area in the case of sulfur the pressure drop can be reduced, but this will also reduce the heat output.

Since liquid sulfur as a coolant has attractive physical properties (high boiling point and relatively high volumetric specific heat) except for its viscosity, and since it is possible to suppress, partially at least, the increase in viscosity which sulfur undergoes above 310° F by means of an additive (e. g., 2 per cent sulfur chloride), it is of interest to investigate the effect of reducing the viscosity on the effectiveness of sulfur as a coolant. Results for viscosities of 0.005 lb/ft-sec

TABLE IX
COOLING COMPARISON OF SULFUR VS SODIUM

Coolant →	Sodium	Sulfur		
		(untreated) $\mu = 1.00$	$\mu = 0.005$	$\mu = 0.0005$
Inlet temperature, °F	500	500	500	500
Outlet temperature, °F	805	805	805	805
Velocity, ft/sec	15	1.29	5.6	7.7
Pressure drop, psi	18	80	10.2	10.8
Heat transfer coefficient, (Btu/hr-ft ² °F)	14,900	126	1200	3850
<u>Coolant thermal resistance</u>	0.0492	0.862	0.396	0.170
<u>Total thermal resistance</u>				
Heat output per fuel element, kw	1505	219	951	1308

and 0.0005 lb/ft-sec are tabulated in Table IX. These viscosities are within the realm of theoretical possibility as can be seen in Fig. 43 by extending to the higher temperatures the viscosity curve between 250 and 300° F. For a viscosity of 0.005 lb/ft-sec, the heat output is above 63 per cent of that for sodium. For a viscosity of 0.0005 lb/ft-sec, the heat output increases to 87 per cent of that for sodium. In both cases, the pressure drop and velocity are less than the allowable values so that the flow area can be reduced and the heat output thereby increased somewhat. The results obtained using sulfur as a coolant and the coolant tube and fuel element design currently being developed for a power reactor are shown in Fig. 44.

It appears that liquid sulfur is a possible reactor coolant only if a suitable additive can be found which will reduce the viscosity of sulfur to less than 0.005 lb/ft-sec in the temperature range of 500 to 900° F. For sulfur to become comparable to sodium, as a coolant, the viscosity of sulfur must be reduced to less than 0.0005 lb/ft-sec in the above temperature range.

2. Diphenyl as a Reactor Coolant (T. T. Shimazaki) - A study has been made to determine the optimum operating conditions for a specific power reactor when diphenyl is employed as the reactor coolant. The results for diphenyl are

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Fig. 43. Viscosity of Sulfur

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Fig. 44. Sulfur as a Reactor Coolant

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also approximately valid for other organic compounds, such as Dowtherm A, whose heat transfer properties (specific heat, density, viscosity, and thermal conductivity) do not differ appreciably from those of diphenyl. The known heat transfer properties of the terphenyls are similar to those of diphenyl, so that the results of diphenyl will be approximately valid for the terphenyls.

The study is based on the following data:

Fuel element	Seven 0.75 inch by 6 foot uranium rods
Fuel rod jacket	Zirconium - 0.020 inch
Conductivity of uranium alloy	16 Btu/hr-ft °F
Maximum fuel temperature	1140° F
Coolant tube ID	2.80 inches
Maximum coolant temperature	750 °F
Coolant velocity	20 ft/sec max
Diphenyl:	
Density	49 lb/ft ³
Specific heat	0.65 Btu/lb °F
Viscosity	1×10^{-4}
Thermal conductivity	0.08 Btu/hr-ft °F

The results are shown in Fig. 45. The maximum electric power is obtained when the steam pressure is 600 psia. The coolant temperatures for this steam pressure are 512° F in and 600° F mixed-mean out. Due to the small temperature rise of the coolant, a reheating, regenerative steam cycle is found to be the best cycle to use. The results of Fig. 45 are based on a single reheat cycle with 4 stages of regeneration. The small coolant temperature rise is due to a combination of short fuel element, relatively low coolant heat transfer coefficient and a relatively high fouling factor. For a coolant velocity of 20 ft/sec the pressure drop in the coolant tube is 13 psi.

Table X shows a comparison between diphenyl and sodium as coolants. The fuel element electric power density for diphenyl is seen to be about two-thirds that for sodium.

a. Effect of Coolant Upper Temperature Limitation - The limiting upper temperature for some organic compounds, such as the terphenyls, are higher than 750° F. If there were no upper temperature limitations on the

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Fig. 45. Diphenyl as a Reactor Coolant

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TABLE X
COOLING COMPARISON OF DIPHENYL VS SODIUM

	Diphenyl	Sodium
Central fuel element thermal power, kw	507	706
Power plant efficiency, %	28.2	30
Central fuel element electric power, kw	142	212

coolant, the average coolant temperature rise for the coolant inlet temperature range shown in Fig. 45 would be increased by only about 3 per cent. Thus these results are approximately valid for organic compounds having an upper temperature limitation greater than 750° F, provided their heat transfer properties are similar to those of diphenyl.

b. Effect of Fuel Rod Diameter - Due to the relatively low heat transfer coefficient and high fouling factor associated with diphenyl, the thermal resistance of a 3/4-inch diameter fuel rod in such a system is only about 50 per cent of the total resistance, whereas the rod constitutes about 80 per cent of the total resistance in a sodium cooled system. It can be seen then that if the fuel rod diameter is reduced, the increase in power density will be less, percentage-wise, in the diphenyl system than in the sodium system.

c. Effect of Coolant Tube Inside Diameter - If the coolant tube inside diameter is decreased, then for a given coolant velocity, the coolant heat transfer coefficient and, therefore, the overall heat transfer coefficient are increased. If the mean coolant temperature and the coolant velocity are held fixed, decreasing the coolant tube diameter will increase the thermal power of the fuel element. The 2.80 inch ID coolant tube assumed in this study is considered to be the minimum practicable diameter from the standpoint of desirable clearances for the fuel element.

V. SODIUM REACTOR EXPERIMENT (SRE) PROPOSAL

A preliminary engineering design study has been completed for a proposed reactor facility intended for development work on basic reactor features of

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liquid metal cooled, high temperature power reactors. A program of development work using the facility has also been outlined. The proposed facility has been designated the SRE, which stands for "Sodium Reactor Experiment." It is not intended as a pilot plant for any specific reactor. The facility is designed for maximum flexibility in conducting engineering development work on fuel, moderator, and coolant materials under various operating conditions. The reactor cooling system is designed for removal of 10 megawatts of heat but includes no provision for electric power generation.

A. Reactor Design

The reactor unit of the SRE facility is recommended to be of the "tank-type" design, in which the entire core is considered replaceable. A large stainless steel tank surrounded on the sides and bottom by permanent shielding would be installed with its top at ground level. All entry to the reactor core for operations such as control rod actuation and fuel handling would be made through a top shield, which itself would be removable. The equipment required for circulation of the reactor coolant would be mounted in a compartment adjacent to the reactor structure. Pipes connected into the base of the large tank would serve as inlets for the coolant, and other pipes connected into the tank at an elevation above the core would serve as outlets. This arrangement can be made capable of considerable flexibility, permitting modifications in the core materials as well as their geometry. The characteristics of the proposed initial installation are given in Table XI.

The major design features of the SRE with its initial core are shown in Fig. 46. A concrete foundation is to be installed which will place the entire reactor unit below ground level. Some piping is built into this concrete near its inside surface to provide for the circulation of a coolant and the removal of the small amount of heat generated there. Inside of the concrete is a steel liner which also serves as a form during pouring of the concrete. Refractory blocks for thermal insulation are installed next to this liner and the thermal shield is placed adjacent to the insulating blocks. The thermal shield in this position absorbs little heat and there is no necessity to provide special cooling. This is because the heavy walled, stainless steel container tank removes the major amount of radiation contributing to heating effects in the shielding. This

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TABLE XI
INITIAL CHARACTERISTICS OF THE SRE

Materials

Moderator: Graphite canned in 0.040 inch Zr
Coolant: Na
Coolant Tube: 0.080 inch Zr, 2.800 inch ID
Cladding: 0.020 inch Zr
Bonding: 0.010 inch Na
Fuel rod diameter: 0.750 inch
Fuel element geometry: Seven rods per element
Fuel: 0.0180 U²³⁵ in U

Lattice Constants

$\bar{\eta}$ (clump): 1.593
 ϵ : 1.047
p (hot): 0.820
f (hot): 0.924
k (hot, poisoned): 1.264
 L_f^2 (hot): 353 cm²
 L^2 (hot): 212 cm²
 B^2 (hot, poisoned): 423 x 10⁻⁶ cm⁻²

Power Distribution

Reflected length/unreflected length: 0.648
Central/average rod power: 1.37
Peak/average thermal flux: 2.12

Reactor Assembly

Active fuel rod length: 6.0 ft
Total number of elements: 37
Active core volume: 161 ft³
Overall core plus reflector diameter: 10 ft
Fuel weight: 2.78 tons
Moderator weight: 3.9 tons
Center to center element spacing: 11 in.

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TABLE XI (Continued)Data for Central Fuel Element

Power: 730 kw
Specific power: 578 kw/kg U^{235}
Maximum uranium temperature: 1200° F
Coolant inlet temperature: 500° F
Coolant outlet temperature: 874° F
Coolant velocity: 6 ft/sec
Coolant flow rate: 21,500 lb/hr
Coolant pressure drop: 5 psi

Operating Conditions

Total reactor power: 20 mw
Average specific power: 435 kw/kg U^{235}
Average thermal flux in U: 1.6×10^{13}
Peak thermal flux in U: 3.4×10^{13}

container tank is, of course, cooled by the sodium and is suspended within the cavity formed by the thermal shield from three anchor points. Double walled piping connects into the bottom and sides of the tank to provide coolant inlets and outlets. The space between the container tank and the thermal shield, as well as within the tunnels provided for the coolant system, would be filled by helium gas. The coolant pipes pass into shielded compartments just below floor level and adjacent to the reactor structure. In these compartments, heat exchangers, pumps and auxiliary equipment required for the sodium system are installed. The particular configuration just described has the advantage as far as safety aspects of the reactor are concerned, that no accidental break in the entire sodium system can lead to draining coolant from the reactor core.

For the initial SRE installation it is proposed that the sodium system equipment be designed for the removal of 10 megawatts of heat. A 10 megawatt heat exchanger and associated equipment is to be located in one compartment adjacent to the reactor. In a second compartment, a heat exchanger and associated equipment is to be installed for 1 megawatt of heat removal. These two

Fig. 46. SRE Arrangement

The fuel elements in the form of slightly enriched uranium rod clusters are supported on hanger rods attached to plugs placed in the top shield. These fuel elements are cooled by sodium flowing up through the coolant tube from a plenum chamber below the core to the reservoir of sodium above. This reservoir above the core has a depth of about 6 feet and as a result of its free surface, can serve as a surge volume for the system. The handling equipment for removing fuel elements or components of the control or safety system, consists of a shielded coffin operating above the top shield. The coffin would be used

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only during reactor shutdown and would rely on circulation of helium gas to remove the afterglow heat of the irradiated elements.

The parts of the reactor which are intended to be modified as required by the experimental program, are the core and the matching top shield. For initial operation of the SRE, the moderator consists of graphite blocks assembled into zirconium cans, 11 inches on a side and 10 feet high. Each can is penetrated along its axis by a zirconium tube welded at the ends. These cans are separated from each other by a gap of approximately 1/16 inch which permits a small circulation of sodium to prevent the creation of excessive temperatures along the can walls. The cell size moderator units are supported and located by a heavy steel grid which in turn is attached to a supporting cylinder suspended from a ledge on the inside of the large container tank. Provision is made for a small amount of sodium to leak through this lower grid in order to circulate past the can walls, while most of the sodium flow takes place within the coolant tubes. As just described, each of the graphite cells is square in cross section and protected by a thickness of perhaps 0.040 inch of zirconium. Another alternate arrangement, which would be satisfactory and which would make use of less zirconium, is one in which the cell size blocks are hexagonal in cross section. In any case, these zirconium canned graphite cells would be utilized throughout the active core and the reflector. Those units serving only in the reflector would not be provided with an axial coolant tube.

The top shield for the reactor is composed of deep beams, 22 inches wide and placed in a parallel array at the top of the large container tank. At the base of each shield beam, there are two layers of steel, each about 2 inches thick. These layers are separated by a series of thin steel plates, serving as thermal insulation. The remainder of the shield consists of approximately 4 feet of heavy concrete with an additional steel plate at the top. The sides of each shield beam are also encased by a thin steel plate. In order to form a gas seal for the helium atmosphere which is retained above the sodium reservoir, a flexible strip of stainless steel sheet is welded between adjacent pairs of the shield beams. Certain of these welds must be cut and remade on occasions which require the removal of any of the shielding beams. A pattern of holes, lined by steel tubing, penetrates the top shield where required to provide access to the tubes for the fuel elements, control and safety rods. The assembly for each

of these elements provides an adequate shielding plug within this top shield, when inserted into the reactor. In the case of the fuel elements, the hanger rod for their support is made integral with the shielding plug. In order to measure the sodium exit temperature at the top of each coolant tube, a thermocouple is located at the bottom of the hanger rod and the leads are brought up inside this rod and through the shielding plug to connections at the upper surface of the top shield. Pipes are installed inside the concrete of the shield beams in order to remove the heat generated there. An appropriate coolant such as toluene will be circulated through these pipes as well as through those cooling the concrete at the sides and bottom of the reactor.

B. Experimental Studies Using the SRE

1. Fuels - The fuels giving long life operation and lowest cost power are metal elements in which no diluent (not including fertile material) is present except in small percentages which might increase dimensional stability. No comparison is intended here with homogeneous and fluid fuel reactors, because of their unproven technology and uncertain costs. The basic SRE configuration calls for a heterogeneously loaded core in which the fuel elements are hung vertically in the coolant tubes. This fuel element configuration is considered to be excellent for any high performance power reactor, and in addition, is one which readily lends itself to use in a reactor unit designed for experimental flexibility.

a. Fuel Element Geometry - The fuel element recommended for first use in the SRE consists of a seven rod cluster of slightly enriched uranium rods approximately 3/4 inch in diameter. This geometry appears good from the standpoints of ease of fuel element preparation and effectiveness of heat transfer. Variations of fuel element geometry which may give better performance depending upon fuel element fabrication techniques to be developed in the future are as follows:

1. Fuel elements of the type just described but having 19 or 37 rods.
2. Hollow fuel elements can be fabricated by powder metallurgy methods. Instead of varying the number of rods per element hollow elements can be prepared using one or more concentric hollow cylinders.

3. Flat plate assemblies of the fuel material can be fabricated in which the number of plates per assembly and their thickness can be varied.

b. Low Enrichment Uranium Fuel - Uranium of minimum U^{235} enrichment is the cheapest form of fuel presently available for power reactor operation. Low costs are also dependent upon obtaining a high conversion ratio and good dimensional stability under long time irradiation. This latter factor is dependent upon the method of fuel preparation, alloy additions, metallurgical processing and subsequent heat treatment, and may be dependent upon the type of bonding and cladding used. The magnitude and effect of these various factors will be experimentally determined by the use of the SRE facility.

c. Thorium-Uranium Alloy Fuel - Fuel elements prepared by alloying a small percentage of uranium in thorium are certain to play an important role in power reactor operation of the future. This concept leads to use of another fuel material and utilizes the Th- U^{233} cycle to an extent dependent upon the reactor conversion ratios which can be achieved. These fuel elements must be tested for radiation and temperature stability. It is expected that the thorium-uranium alloy will be much better than uranium as a fuel material, since thorium has superior physical properties to those of uranium including good crystalline structure, lack of phase change, high melting point, and high thermal conductivity. In addition to contributing to good dimensional stability, these factors will make the use of higher specific reactor powers and higher power conversion efficiencies possible, since higher coolant temperatures will be permitted.

2. Coolants - The SRE is designed as a high temperature, low pressure reactor, particularly adapted to liquid metal cooling. While the use of liquid metal cooling could be advantageous for reactors designed for any purpose, its great attraction lies with reactors intended for the production of power. This conclusion has been among those reached during the course of a 2-1/2 year study conducted by North American Aviation, Inc. More specifically, this study showed for the purpose of electric power production from large stationary reactor plants, that of all reactor systems within the scope of immediate construction feasibility, a thermal reactor using slightly enriched uranium fuel, sodium coolant, and graphite moderator would produce the lowest cost electric power. The basic advantages of high temperature and low pressure operation,

leading to high thermodynamic conversion efficiencies and simplified structural arrangements, are fundamental to a high performance plant.

The SRE is designed initially to employ liquid sodium as the reactor coolant. It is planned that, if desired at some future date, appropriate changes could be made to use some other liquid as the coolant. This modification would be simple if the new liquid were compatible with the stainless steel used in the original reactor tank and piping installation.

Analyses of steam cycles and power plants associated with power producing reactors have indicated that optimum reactor coolant inlet temperatures lie between about 400 and 550° F, and that the coolant outlet temperatures should be above 850° F. Sodium and the sodium-potassium alloys are especially suitable within this temperature range. In addition they are outstanding as heat transfer media. The sodium-potassium alloys, however, have some disadvantages relative to sodium, such as higher costs, from four to eight times that of sodium, and poorer cooling properties due to lower specific heat, lower density and poorer film coefficients for the same velocity. The fact that sodium is not liquid at room temperature makes for an advantage in favor of some of the sodium-potassium alloys, but this is not of great importance and applies only during start-up periods.

While liquid metals appear especially attractive as reactor coolants as a result of their good heat transfer characteristics, low vapor pressure and perfect radiation stability, there are other liquids which may eventually have applications in high temperature, low pressure reactors. In this connection there are a number of organic compounds which are thermally stable and have low vapor pressures in the range of temperatures of interest. These materials are basically cheap and possess other advantages in being chemically inactive and of low neutron absorption cross section. For these reasons, such organic liquids are also of interest as possible moderating materials. The application of an organic liquid as a coolant would be especially attractive in a system using the same material as moderator.

3. Reactor Components - A secondary objective of the SRE program will be to design, develop and test such items as the control rod system, safety rod system, thermal insulation layer, biological shield, instrumentation system,

fuel handling equipment, and fuel storage facilities. Studies will be conducted to develop organic liquid shield coolants which will not react chemically with the primary sodium coolant.

Neutron flux measurements and various studies concerned with the nuclear characteristics of the reactor components will be made. In-pile irradiations may be made either by placing specimens in a coolant tube or by substituting a test capsule assembly for a zirconium canned moderator unit.

4. Moderators

a. Graphite - On the basis of a number of reactor performance and cost studies, graphite appears to be the best moderator material now available for use in a high temperature thermal reactor. There is a considerable background of existing technology in the use of graphite, and it is relatively inexpensive. At the temperatures existent in a power reactor, there should be no problem of radiation stability. The thermal expansion coefficient of graphite is small, and its strength increases with increasing temperature. A disadvantage exists in the use of graphite as a moderator, especially with sodium as a coolant, due to the fact that if the coolant is allowed to contact the graphite it can be absorbed into the pores of the graphite, increasing the parasitic neutron losses. In addition, this may result in chemical effects producing a deterioration of the graphite or a carburization of surrounding metal surfaces, depending upon the temperatures which are present. Below about 600° C, no difficulty from such reactions is expected. However, it is important in the design of the core structure to insure that the sodium, or any other liquid coolant, will not contact the graphite moderator. Several designs have been suggested to achieve this requirement, and among those which are pertinent to the "tank-type" structure with sodium coolant are the following:

1. The graphite may be contained in a single, stainless steel core tank with stainless steel coolant tubes penetrating the moderator to provide for a heterogeneous configuration of fuel elements in the coolant stream. The disadvantages of this design are principally due to the fact that a single leak may result in admitting coolant to the entire moderator and reflector structure, and that the coolant tubes must necessarily be thin walled.

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2. Another design consists of separate sections of the graphite moderator and reflector within zirconium cans. These might consist of individual cell sized cans, each penetrated along its axis by a heavy walled zirconium coolant tube. If the tube or the can wall developed a leak, only one cell sized section of graphite could be flooded, and this could be replaced during some future convenient shutdown period. Such a design would result in a core having high neutron performance and good engineering feasibility and is the basis for the first suggested operation of the SRE.
3. Another design combines the stainless steel core tank with thick walled zirconium coolant tubes penetrating the moderator. Such a configuration has the advantage of high coolant tube integrity and high neutron performance, but has the disadvantages of exposing all of the graphite to a single coolant leak and the engineering difficulty of the core tank construction. This combination of steel and zirconium also depends upon the development of strong leak-tight joints between zirconium and stainless steel.

b. Beryllium - A general evaluation of power producing reactors indicates that a beryllium moderator must be considered as a possible competitor with graphite. This is especially true in the case of units designed for the production of small blocks of electric power.⁸ On the basis of recent cost figures for beryllium of \$50 to \$60 per pound, the cost of power is estimated to be only a few per cent greater than that of an optimum graphite moderated system.

The great advantage of beryllium in high temperature reactor applications is that there is no necessity for protecting this moderator from the coolant. Here, it is assumed that a coolant compatible with beryllium would be chosen, and it is especially important to point out that liquid sodium fulfills this requirement throughout the temperature range of interest. An attractive core design consists of beryllium blocks stacked on zirconium coolant tubes. The sodium coolant would penetrate the interstices of this structure in addition to streaming through the coolant tubes. This results in a very simple arrangement on which no maintenance problems should be expected.

c. Beryllium Oxide - Assuming the price of BeO to be \$40 per pound, general cost studies⁸ on large scale electric power producing reactor plants

have indicated that designs utilizing BeO moderator and reflector can produce power at perhaps 10 per cent greater cost than in optimum graphite systems.

Hot-pressed BeO would have the same advantage as beryllium metal in that it should not be necessary to prevent contact with liquid sodium in the temperature range of interest, and a core design similar to that just described for beryllium could be utilized. In order to prevent excessive thermal stresses in a high power density reactor, a moderator matrix made up of relatively thin sections of BeO is required. In addition, BeO suffers from some effects of radiation damage, although this is not believed to be a limiting design factor.

d. Organic Liquids - The use of a hydrogenous material as moderator in a reactor design for low pressure operation and liquid metal cooling has the advantage of small core size and the disadvantage of increased enrichment (see Section IV, A). There are several reactor arrangements which permit the use of a hydrogenous liquid moderator operating at low pressure, together with a high temperature coolant. Economic analyses based on presently estimated investment and operating costs indicate that such systems are capable of power production costs only slightly higher than for the case of an optimum graphite moderated system.

1. An organic liquid as moderator, in which a thermal barrier is placed between this moderator and the coolant tube carrying liquid sodium. Here, it is assumed that a separate cooling system would be used to remove heat generated in the moderator and prevent its temperature from exceeding some selected low value, in order to assure no thermal decomposition or high vapor pressures.
2. An organic liquid as moderator, in which no thermal barrier exists between it and the coolant tube. In this case, it is assumed that the organic liquid selected as moderator would be capable of operation at sufficiently high temperatures that no auxiliary cooling system for the moderator would be required. The heat generated in the moderator would be transferred directly through the coolant tube walls to the reactor coolant.
3. An organic liquid as both moderator and coolant, in which no separate coolant tube need exist unless required to channel the flow of this liquid for heat transfer reasons. Such an arrangement

would be structurally simple, but would necessitate a primary coolant heat transfer system employing this organic liquid. While this arrangement loses the advantages of the exceptionally good heat transfer properties of liquid metals, it offers a simpler, one-fluid system and has negligible radioactivity associated with the external coolant circuit. The liquid must be a compound having good thermal stability and low vapor pressure in the required temperature range.

There are a large number of organic liquids which will be screened by thermal and radiation stability tests now in progress in NAA and elsewhere. Biphenyl may be a good choice for this application in view of its high radiation stability, low cost, and moderately high boiling point. Para-terphenyl, which boils at 389° C and has good thermal stability up to 450° C, may be the best choice for the highest operating temperatures. Other compounds which now appear promising from the standpoint of thermal stability, radiation stability, and cost, are meta-terphenyl, ortho-terphenyl, triphenylmethane, and silicon tetrabenzyl. Operation of the SRE with any of the solid or liquid materials serving as moderators would answer technological questions regarding their performance. The type of questions to be answered are: the extent of success achieved by the zirconium cans or the steel tank assembly in protecting the graphite from the reactor coolant, the stability of BeO against thermal stresses set up during conditions of steady state operation and reactor scram, and the decomposition products formed when some chosen organic liquid is operated as the reactor moderator.

For the initial operation of the SRE, a core comprised of cell size blocks of graphite canned in zirconium is recommended. Experiments on other moderator arrangements could be performed by replacing one or more of these cell size units. In the case of a liquid moderator, a separate tank whose over-all dimensions correspond to those of one or more cells could be installed in the reactor core for the period required for a test. The design of such a test assembly is shown in Fig 47.

If desired, the entire core tank containing the reflector could be replaced to carry out a full scale test on some moderator configuration of special

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Fig. 47. Liquid Hydrocarbon Moderator Test Cell Assembly

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interest. The fact that the reactor top is removable and that, if necessary, the large stainless steel containing tank can be filled with water for shielding purposes, makes the execution of such operations feasible. It seems likely that most of the tests carried out on moderator materials would best be conducted by substituting only a part of the core at one time.

C. Reactor Building Design

The building for the SRE is divided into two parts, one housing the reactor and its auxiliary equipment, the other housing the facilities for the handling and examination of irradiated core components. This division into two structures is desirable for reasons of cost, since the structure housing the reactor is to be of more expensive construction than that for housing the facilities. In addition, it is planned that water be used as required in the facilities building, but that no water outlets be provided in the reactor building in order to minimize any possible hazard from reaction with the sodium coolant. A layout of the reactor and facilities buildings has been prepared. Within the reactor building a large high bay area is provided for the reactor, its primary and secondary cooling systems, and storage of new fuel. Adjacent rooms provide space for instrument storage and repair, reactor control, helium equipment, and sodium handling equipment. The entire floor area is at one level. New fuel elements are stored in vertical pipes located in the ground. The primary heat exchanger compartments are covered by shielding blocks whose top surfaces are at floor level. Similarly, the pumps and surge tanks of the secondary system are below ground, but pipes carry sodium-potassium to the air heat exchangers and blowers located on a second floor at one end of the building. The air intake is through the end of the building, and the exhaust is forced vertically through venting stacks on the roof. Appropriate storage area for the reactor top shield is to be provided on the main floor, which is serviced by a 50-ton bridge crane. This crane traverses the high bay areas in both the reactor and facilities buildings and is to be used to support and locate the coffin during fuel handling operations. In addition, for purposes of disposing of highly radioactive components which may be removed from the reactor tank, the bridge crane would be used to transport these assemblies to the passageway between the reactor and facilities buildings. For transportation beyond this point, a truck or rail car would be used.

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In the facilities building, vertical cylindrical cells installed below floor level are provided for cleaning fuel and moderator elements which may then be stored under water in an adjacent pond. For post irradiation examinations, a fuel element may be disconnected from its hanger rod in a specially equipped cell and transported underground to a large hot cell. Space is provided surrounding the hot cell for both contaminated and uncontaminated experimental work areas, and a maintenance shop is provided where a few machine tools subject to contamination can be operated. Other rooms within the facilities building include lavatories, an office for supervision, and an instrument room for health physics.

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