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**SODIUM GRAPHITE REACTOR
QUARTERLY PROGRESS REPORT
APRIL-JUNE, 1955**

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Bureau of Ships	34
Bureau of Yards and Docks	35
California Patent Group	36
Carbide and Carbon Chemicals Com	37-44
Chicago Operations Office	45
Chicago Patent Group	46
Chief of Naval Research	47
Department of the Navy-Op-362	48
duPont Company, Augusta	49-52
duPont Company, Wilmington	53
Duquesne Light Company	54
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Engineer Research and Development Laborator	56
General Electric Company (ANPD)	57-60
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Hanford Operations Office	69
Headquarters, Air Force Special Weapons Center	70-71
Iowa State College	72
Knolls Atomic Power Laboratory	73-76
Los Alamos Scientific Laboratory	77-78
Massachusetts Institute of Technology (Benedict)	79
Materials Laboratory (WADC)	80
Mound Laboratory	81
National Advisory Committee for Aeronautics, Cleveland	82
National Advisory Committee for Aeronautics, Washington	83
Naval Research Laboratory	84-85
New York Operations Office	86
New York University	87
Nuclear Development Associates, Inc.	88
Nuclear Metals, Inc.	89
Patent Branch, Washington	90
Phillips Petroleum Company (NRTS)	91-97
Powerplant Laboratory (WADC)	
Pratt & Whitney Aircraft Division (Fox Project)	
San Francisco Operations Office	
Sylvania Electric Products, Inc.	
USAF Headquarters	100
USAF Project RAND	103
U. S. Naval Postgraduate School	104
U. S. Naval Radiological Defense Laboratory	105
University of California Radiation Laboratory, Berkeley	106-108
University of California Radiation Laboratory, Livermore	108-109
Vitro Engineering Division	110
Vitro Laboratories	111
Walter Kidde Nuclear Laboratories, Inc.	112
Technical Information Service, Oak Ridge	113-327
File	328-400

3

TABLE OF CONTENTS

SECTION A

TECHNOLOGY OF THE SODIUM GRAPHITE REACTOR

SECTION A EDITOR: A. B. MARTIN

	Page No.
I. Full Scale SGR	11
II. Determination of Power and Temperature Coefficients of Reactivity	14
III. Nuclear Parameters of Sodium Graphite Lattices	15
A. Introduction	15
B. Experimental Measurements	15
C. Buckling and Average Thermal Neutron Fluxes	16
IV. Metallurgy of SGR Fuels	25
A. Experimental Fuel Program	25
B. MTR Irradiation of SGR Fuels	27
V. Zirconium Behavior in Sodium Systems	27
VI. Engineering Evaluation of Graphite	30
A. In-Pile Thermal Conductivity of Graphite (MTR-NAA-3 Experiment)	30
B. Irradiation of High Density Graphite Types (MTR-NAA-16 Irradiation)	30
C. Experiments in the SRE	30
VII. Organic Coolant Studies	31

TABLE OF CONTENTS

SECTION B

SODIUM REACTOR EXPERIMENT

SECTION B EDITOR: GUY M. INMAN

	Page No.
VIII. Nuclear Engineering, Physics	35
A. Nuclear Engineering.	35
B. Reactor Physics	35
1. SRE Core Data	35
2. Reactivity Studies	35
IX. Land, Utilities, Buildings	36
A. Land, Utilities	36
B. Reactor Building	36
C. Engineer Test Building.	39
X. Fuel Elements	39
A. Fuel Rod Jackets	39
B. Fuel Rod Assembly	42
XI. Moderator and Reflector Cells	52
A. Design	52
B. Zirconium Procurement	52
C. Fabrication and Welding	52
D. Graphite Procurement	54
E. Gas Evolution	56
F. Heat and Vacuum Treatment	57
G. Sodium Diffusion Experiment.	58
XII. Reactor Structure	59
A. Core Tank Assembly and Internal Structure	59
B. Outer Tank Assembly and Supporting Structure.	63
C. Thermal Shield Rings	63
D. Core Cavity Liner	63
E. Thermal Insulation	63
XIII. Heat Transfer	68
A. Sodium System	68
1. Design	68

TABLE OF CONTENTS (continued)

	Page No.
2. Engineering and Tests on SRE Components	69
3. Engineering Tests of Sodium Flow Under SRE Conditions	72
B. Toluene System	74
1. Development	74
2. Design	74
XIV. Instrumentation and Control	75
A. Control Rod System	75
1. Control Ring Development	75
2. Control Rod Lead Screw Development.	77
B. Safety Rod System	81
1. Latch	81
2. Torque Tube	81
3. Snubber Test	81
4. Solenoid	82
C. Instrumentation	82
1. Development	82
2. Production	85
XV. Shielding	85
A. Top Shield	85
B. Control Rod	87
C. Thimble Shield Plugs	88
D. Fuel Handling Coffin	88
1. Radiation Levels Within Coffin	88
2. Fuel Coffin Shielding	89
3. Radiation Level of Surface of Coffin	89
E. Hot Cell	90
1. Auxiliary Coffin Shielding	90
2. Shielding For Hot Cell Doors	90
3. Hot Cell Shielding	91
F. Primary Gallery Shielding.	92
G. Specific Activity of Primary Sodium	94
H. Disposable Cold Trap System	94
I. Liquid Radioactive Waste Disposal System	96

TABLE OF CONTENTS (continued)

	Page No.
J. Gaseous Radioactive Waste Disposal System	96
K. Shield Concrete Tests	97
L. Gallery Seals.	97
XVI. Reactor Services	98
A. Inert Gas Systems	98
1. Design.	98
2. Purification of Inert Gas.	100
B. Fuel Handling Equipment.	103
C. Radioactive Liquid Waste System	109
XVII. Reactor Operation	109
A. Systems Analysis	109
B. Coordination of Experimental Program	109
C. Reactor Personnel Training	111
References	113

LIST OF TABLES

I. Experimental Buckling and Average Thermal Flux Values	17
II. General Specifications for SRE Experimental Fuels	26
III. Decomposition and Metallic Corrosion by Gamma Irradiation At 150° F	33
IV. Tensile Test Data Type 304 Stainless Steel Cap Welded to Type 304 Stainless Steel Tube	43
V. Gases Released by 1700° C Vacuum Degassing	57
VI. Specific Activity of Primary Sodium.	95
VII. Decay Time Required to Stay Below Given Fuel Rod Temperature	110

LIST OF FIGURES

	Page No.
1. Amount of U-235 in Th-U <u>vs</u> Initial Conversion Ratio	12
2. Amount of U-233 in Th-U <u>vs</u> Initial Conversion Ratio	13
3. Material Buckling of Exponential Assemblies	19
4. Neutron Distribution in a Cell	21
5. Neutron Distribution in a Cluster.	22
6. Neutron Distribution on Outer Rod Surface	23
7. Neutron Distribution in Outer Rod	24
8. Capsule Components	28
9. Decomposition <u>vs</u> Gamma Dosage at 150° F (MTR Gamma Facility)	32
10. Three 1000 KVA Transformers at SRE Site Substation	37
11. General View of Hot Cell Floor Plates, Moderator Storage Cells, Fuel Storage Cells, and Fuel Washing Cells.	38
12. General View of Reactor Building Area.	40
13. Metallographic Analysis of Stainless Steel Fuel Rod Jacket Tubing	41
14. Typical Specimens of End Caps Welded to Fuel Rod Jackets After Tensile Strength Tests	44
15. Outgassing Retort	45
16. Fixtures for Outgassing Retort	46
17. Fuel Element Upper Guide and Spider	48
18. Old and New Lower Guide Designs	49
19. Assembled Fuel Element	50
20. Plug and Ring Gage on Moderator Cell Can	53
21. Moderator Cell Can Assembly and Side Panels	55
22. Sodium Diffusion Apparatus	58
23. Core Tank Galling Test Capsule	60
24. Core Tank Galling Test Apparatus	61
25. Typical Galling Test Specimens	62
26. Completed Core Cavity Liner	64
27. Core Cavity Liner After First Pour of Biological Shield Concrete	65
28. Instrument Thimble Stubs Passing Through Core Cavity Liner	66

LIST OF FIGURES (continued)

	Page No.
29. Core Cavity Liner After Installation of Instrument Thimbles	67
30. Cold Trap and Economizer.	71
31. Heat Transferred Through Control Rod Rings <u>vs</u> Thimble Temperature	75
32. Brinelled Type 316 Stainless Steel Ball Screw After 3000 Cycles at Room Temperature	78
33. Chromized Rex M ₂ Steel Screw After 1500 Cycles at Room Temperature	79
34. Chromized Type 1040 Steel Screw and Nut After 1500 Cycles at 1050 F.	80
35. Snubber Test Before and After Release	83
36. Pull Tube, Piston, and Cylinder of Snubber Test Rig	84
37. Radiation Level <u>vs</u> Shield Density	93
38. Cerrobend Metal Alloy Gallery Seal.	99
39. Neoprene Gasket Gallery Seal	99
40. Flow Rate <u>vs</u> Oxygen Concentration, Inert Gas Purification Tests.	101
41. Rate of NaK Entrainment in Helium Flow Through 1 1/2 inch ID Bubbler	102
42. Upper End of Process Cylinder with Coffin Locating Ring in Place	104
43. Lower End of Process Cylinder, Sodium Loading Connection, Diffusion Cold Trap, and Sodium Return Line	105
44. Plugging Loop, Electromagnetic Pump, and Flow Meters	106
45. Interlock Region of Coffin with Pickup Device Extended	107
46. Dry Storage and Fuel Washing Cells.	108

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NAA-SR-1292	OCTOBER-DECEMBER, 1954, MAY 15, 1955
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10

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SECTION A

TECHNOLOGY OF THE SODIUM GRAPHITE REACTOR

I. FULL SCALE SGR

E. F. Weisner

The full scale sodium graphite power plant study during the past quarter has been concerned principally with certain details of the design reported in NAA-SR-1300.¹ Also some very preliminary work has been started in regard to advanced or improved SGR type reactor cores.

The build-up of uranium isotopes in recycled Th-U²³⁵ fuel and the amount of additional U²³⁵ necessary to restore the original reactivity has been studied. It was assumed for this study that the SGR plant would start operation with an alloy of Th-U²³⁵ fuel which would be irradiated for 10,000 Mwd/ton (40 per cent burn-up of the original U²³⁵). The fuel would be reprocessed, refabricated and reloaded into the reactor. If the initial conversion ratio is not high enough, some additional U²³⁵ will need to be added to the fuel (depending upon the resonance escape probability). The amount of U²³⁵ required in Th-U fuel at the beginning of each cycle and the amount left at the end of each cycle is shown in Fig. 1. The difference in the amount at the end of the nth cycle and the amount needed at the beginning of the n+1 cycle is the amount which must be added during reprocessing or refabrication. This amount of U²³⁵, ΔN_5 , can be represented approximately by the following expressions:

$$\text{After 1 cycle through reactor: } \Delta N_5 \cong -0.437C_1 + \frac{0.823 p}{1 + p}$$

$$\text{After 2 cycles through reactor: } \Delta N_5 \cong -0.45C_1 + \frac{1.7p}{1 + 3.2 p}$$

$$\text{After 3 cycles through reactor: } \Delta N_5 \cong -0.47C_1 + \frac{4.2 p}{1 + 9.3 p}$$

$$\text{After many cycles through reactor: } \Delta N_5 \approx -0.48C_1 + 0.40$$

where C_1 = initial conversion ratio

p = resonance escape probability

The amount of U²³³ present at the beginning and end of each cycle is shown in Fig. 2. As can be seen from Fig. 1, the amount of U²³⁵ needed becomes independent of p after about four cycles. Also, if the initial conversion ratio can be greater than about 0.85, there is no need to add U²³⁵; that is, the system will

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Fig. 1. Amount of U-235 in Th-U vs Initial Conversion Ratio

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Fig. 2. Amount of U-233 in Th-U vs Initial Conversion Ratio

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become a breeder as the U^{233} builds up. To achieve an initial conversion ratio of 0.85 or greater in the type of design considered requires that the reactor be large and deliver a large block of power.

A study was begun to investigate the inherent safety features of the full scale SGR. In the initial phases of this study the uranium fueled reactor was considered. It is planned to extend the study to include Th-U fuel later. The response of the reactor and surrounding material environment to nuclear incidents, ranging in magnitude from those resulting from equipment malfunction and operational errors to those of explosive magnitude, was studied. It was concluded that the present design will contain all the products of a 2 per cent Δk step function incident. No conceivable means of introducing 2 per cent excess reactivity as a step function has yet been found short of deliberate sabotage.

II. DETERMINATION OF POWER AND TEMPERATURE COEFFICIENT OF REACTIVITY

G. W. Rodeback

Previous quarterly progress reports^{2,3,4,5} have described the experimental procedure and preliminary results of a measurement of the temperature coefficients of the effective resonance integrals of U^{238} and Th^{232} . The temperature coefficient for uranium was determined by measuring the relative activations of a special U^{238} foil (highly depleted in U^{235}) irradiated at different temperatures. For the case of thorium the same procedure was applied to an iodide thorium foil. The U^{238} foil was irradiated while in place in a 3/4 inch diameter slug of normal uranium and the thorium foil was similarly placed in a 3/4 inch diameter slug of Ames thorium. The placement and shape of the foil, in either case, was such that ideally its activation was a measure of the effective resonance integral for a 3/4 inch diameter fuel slug in a graphite moderator. The facilities at the SUPO reactor at Los Alamos were used to carry out the experiment.

As given in the previous report⁵ the preliminary values obtained for the temperature coefficients for the effective resonance integrals of 3/4 inch diameter cylinders are:

for U^{238} (meas. from 20° to 660° C) $+0.9 \pm 0.2\%$ per 100° C

for Th^{232} (meas. from 20° to 775° C) $+2.0 \pm 0.3\%$ per 100° C

These were quoted subject to further evaluation of errors and uncertainties involved in the experiment.

In the last quarter measurements have been made to determine temperature corrections and thermal expansions in the slugs at elevated temperatures. Also, neutron flux measurements have been made to determine the departure of the experimental foil activations from those of an ideal case. The resulting errors and uncertainties are such as to leave the above results unchanged.

III. NUCLEAR PARAMETERS OF SODIUM GRAPHITE LATTICES

W. W. Brown, B. A. Engholm, W. J. Houghton, and R. A. Laubenstein

A. INTRODUCTION

The measurement of thermal and epithermal neutron flux distributions in exponential lattices of 4- and 7-rod fuel elements of uranium in graphite is continuing. The exponential assembly consists of fuel elements inserted in regularly spaced vertical holes in a graphite stack mounted on top of the thermal column of the NAA Water Boiler Reactor. The assemblies are 64 inches high and the holes for the fuel elements form a square array which can be arranged with a 7, 9.5, or 12 inch spacing by properly stacking the graphite. The fuel clusters consist of aluminum cylinders which are 2.9 inches in diameter and have vertical cylindrical holes through their full length to contain the fuel rods. One type has four 1-inch diameter holes with centers on a 1.59 inch diameter circle about the axis and is loaded with 0.9 per cent enriched uranium rods. The other has a central 0.75 inch diameter hole surrounded by six similar holes with centers on a 1.81 inch diameter circle. Both natural and 2.78 per cent enriched uranium rods are available at NAA for use in the 7-rod fuel elements. Some lattices have been loaded with a 6-rod fuel elements, formed from the 7-rod holder by removing the central uranium rod and replacing it with a 0.75 inch diameter graphite rod. Further details of the assembly and its components are given in a previous progress report.⁴

B. EXPERIMENTAL MEASUREMENTS

The measurements involve the determination of the relative activities of indium and dysprosium foils that have been exposed within a given lattice. The

indium foils are exposed along the vertical axis of a lattice. Except near the top and bottom, the distribution of activity is of the form $\sinh \nu(h-z)$ where z is the height above the bottom of the lattice. The value of ν is determined by a numerical analysis of the experimental data. The overall distribution of neutrons in a horizontal plane is basically of the form $\cos \mu x \cos \mu y$, but with superimposed depressions in the vicinity of the fuel rods. Here x and y are distances measured from the lattice axis along lines parallel to the sides of the assembly. From measured activities of indium foils exposed in the horizontal plane of a solid graphite stack of the same external dimensions as the fuel loaded lattice, the value of μ is obtained. The buckling, B^2 , of the lattice is then given by $B^2 = \mu^2 - \nu^2$.

The dysprosium foils are exposed in the uranium, aluminum, and graphite of one of the cells near the assembly center. The measured activities of these foils give the detailed thermal neutron flux distribution in each component of a lattice cell. This information is necessary for the calculation of the disadvantage factors and thermal neutron utilization for each component of the lattice and is useful for obtaining the power generation distribution in the fuel element.

C. BUCKLING AND AVERAGE THERMAL FLUX VALUES

The lattices studied to date together with their measured bucklings are listed in Table I. The bucklings obtained with the fuel containing 2.78 per cent U^{235} are somewhat higher than was expected on the basis of preliminary calculations. For safety reasons, measurements with this fuel on a 9.5 inch lattice spacing were made in a lattice of 5 cells by 5 cells (25 cells total) rather than on the 6 by 6 lattice as originally planned. With a 6 by 6 lattice assembly, the geometrical buckling of the assembly in the horizontal plane would be less than the material buckling and the neutron flux distribution in the vertical direction would have been a sine rather than a sinh function. In Fig. 3 the measured bucklings are plotted as a function of lattice spacing for both the natural and the 2.78 per cent enriched fuel loadings of the 7-rod element.

Listed in Table I are the relative average thermal neutron flux values over various volumes and surfaces within a lattice cell. These averages are based on integrations of the measured flux distributions at a given horizontal level divided by $\cos \mu x \cos \mu y$. Division by this factor effectively removes the effect of the finite size of the lattices and makes the average flux values listed in the table

17
TABLE I

EXPERIMENTAL BUCKLING AND AVERAGE THERMAL FLUX VALUES

Type of Cluster			Exponential Assembly Dimensions			Buckling
No. of Rods	Rod Size (inch)	U-235 Fuel Enrichment (Wt. %)	No. of Cells	Width (Inches)	Spacing (Inches)	M-2
4	1	0.9	36	57	9.5	1.86 ± 0.15
4	1	0.9	16	48	12	2.20 ± 0.15
7	0.75	Natural	36	42	7	-2.17 ± 0.15
7	0.75	Natural	36	57	9.5	-0.34 ± 0.15
7	0.75	Natural	16	48	12	0.53 ± 0.15
7	0.75	2.78	36	42	7	7.7 ± 0.3
7	0.75	2.78	25	47.5	9.5	9.6 ± 0.2
7	0.75	2.78	16	48	12	8.6 ± 0.2
6	0.75	2.78	16	48	12	8.3 ± 0.2

18

TABLE I (Continued)

EXPERIMENTAL BUCKLING AND AVERAGE THERMAL FLUX VALUES

Relative Average Thermal Neutron Fluxes							
No. of Rods*	<u>Central Rod</u>		<u>Side Rod</u>		<u>Aluminum</u>		Graphite
	Volume	Surface	Volume	Surface	Volume	Graphite Interface	
4	-	-	1.20	1.44	1.67	2.05	3.38
4			1.29	1.57	1.82	2.23	3.99
7	1.07	1.23	1.36	1.48	1.66	2.00	2.97
7	1.07	1.15	1.37	1.50	1.72	2.09	3.35
7	1.09	1.17	1.40	1.55	1.78	2.19	3.80

*These five numbers under "No. of Rods" are repeated from the preceeding page.

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Fig. 3. Material Buckling of Exponential Assemblies

applicable to a cell in the central region of a very large lattice. For cells loaded with 7-rod element the average flux values are normalized to 1.00 at the axis of the central fuel rod. For 4-rod element normalization to 1.00 is made at that point on the surface of a fuel rod that is closest to the cluster center. These average flux values are used in the calculation of thermal neutron utilization and diffusion length in the lattice. Neutron age in the lattice, η , and ϵ are calculated and from these quantities together with the measured buckling, the resonance escape probability is calculated. "Experimental" values of p found in this manner are compared with theoretical values since it is believed that the calculation of p introduces the largest uncertainty into theoretical calculations of reactor lattices of the type being investigated here. For the natural uranium 7-rod cluster lattices at 7, 9.5, and 12 inch spacings experimental values of p have been found to be 0.913, 0.817, and 0.720, respectively.

In Fig. 4, 5, 6, and 7 are plotted the thermal neutron flux distributions along the horizontal lines within a cell as indicated by the insert diagram accompanying each graph. These distributions are for 6- and 7-rod fuel elements of uranium enriched to 2.78 per cent U^{235} and the data were obtained in a 4 cell by 4 cell assembly with a 12 inch lattice spacing. Together with similar distributions to be plotted for the 9.5 inch spaced lattice, they will be of use in calculations of temperature distributions within the fuel cluster in the sodium reactor experiment as well as for evaluation of the relative merits of 6- and 7-rod fuel elements.

The one lattice of 6-rod fuel element which has been investigated to date indicates that the 6-rod fuel element compares favorably in reactivity with the 7-rod fuel element. For the 12 inch lattice spacing, the buckling of the 6-rod fuel element is only slightly lower than that of the 7-rod fuel element, and for a smaller lattice spacing it is expected that the buckling of the 6-rod fuel element will more closely approach or exceed that for the 7-rod fuel element. Use of a 6- instead of a 7-rod fuel element may enable the critical loading of uranium in the SRE to be decreased and the average power density in the uranium to be increased because of a more favorable distribution of heat generation in the uranium.

Fig. 4. Neutron Distribution in a Cell

Fig. 5. Neutron Distribution in a Cluster

Fig. 6. Neutron Distribution on Outer Rod Surface

Fig. 7. Neutron Distribution in Outer Rod

IV. METALLURGY OF SGR FUELS

B. R. Hayward, P. J. Pankaskie, L. E. Wilkinson and G. G. Bentle

A. EXPERIMENTAL FUEL PROGRAM (B. R. Hayward)

One of the principal purposes of the SRE is to develop uranium and thorium fuels for operation at high temperature and for long burnup. Alpha-rolled, beta-heat treated, unalloyed uranium was selected as the primary fuel material for the SRE, as it was the only material available in quantity production with reasonably known behavior history. Slugs of 3/4 inch diameter by 6 inches long have been produced and are currently on site; they will be loaded into the fuel rods in the immediate future. In addition, an extensive experimental fuel program has been formulated to study fuels for these more severe conditions. Specifications (see Table II) have been completed and the procurement phase is in progress on the various alloys and fuel element designs. There are four types of experimental fuels being studied. All of the experimental uranium fuels will be slightly enriched and the Th-U alloy will contain fully enriched uranium; each will carry its own weight reactivity-wise.

1. Powder compacted uranium silicon alloy containing 0.5 w/o Si and powder compacted uranium molybdenum alloy containing 1.2 w/o Mo. These alloys will be cold pressed and sintered by Sylvania into 0.750 by 1.00 inch slugs for loading into standard fuel tubes.
2. Centrifugal casting of three uranium alloys at National Lead is currently in the equipment shakedown stage of fabrication. Two of these 3/4 by 6 inch alloy slugs are of the same composition as under item 1; the third slug consists of uranium-zirconium containing 2 w/o Zr. These alloy slugs will be used in standard fuel tubes. A development and evaluation program with unenriched U will precede the actual SRE slugs.
3. Large hollow cylinders made by hot pressing methods (powder metallurgy) at Sylvania represent a major alternate fuel design. These unalloyed pieces are approximately 2.4 inches OD by 1.35 inches ID by 4 inches and contain the same amount of uranium as a standard 7-rod fuel element.
4. Cast and extruded thorium uranium alloy containing 5 per cent U-235

26

TABLE II

GENERAL SPECIFICATIONS FOR SRE EXPERIMENTAL FUELS

	Large Hollow Cylinders	Powder Compacted Alloys	Cast Alloys	Thorium Alloy
Material	Unalloyed Uranium	U-Si, U-Mo	U-Zr, U-Si, U-Mo	Ta-U
Alloy Content	None	0.5w/o Si, 1.2w/o Mo	2.0w/o Zr, 0.5w/o Si, 1.2w/o Mo	5.4% U
Method of Fabrication	Powder compacting by hot pressing	Powder compact by cold pressing and sintering	Centrifugally chill cast	Cast and extruded
Isotopic Content w/o U-235	2.78	2.78	2.78	~93.0
Chemical Composition	Reactor grade	Reactor grade Si and Mo $\pm$ 0.1	Reactor grade Si, Zr, Mo $\pm$ 0.1	Reactor grade U $\pm$ 0.1
Length	4.000 in. $\pm$ 0.010 in.	1.000 in. $\pm$ 0.002 in.	6.000 in. $\pm$ 0.010 in.	6.000 in. $\pm$ 0.010 in.
Diameter	2.400 in. $\pm$ 0.003 in. ID 1.350 in. $\pm$ 0.003 in. OD 0.015 in. taper/ft on ID permitted	0.750 in. $\pm$ 0.002 in.	0.750 in. $\pm$ 0.002 in.	0.750 in. $\pm$ 0.002 in.
End Radius	Break and deburr	Break and deburr	Break and deburr	Break and deburr
Warp	0.003 in. max. single throw	0.003 in. max. single throw	0.003 in. max. single throw	0.003 in. max. single throw
Surface Condition	250 RMS max.	250 RMS max.	250 RMS max.	250 RMS max.
Density	18.86 g-cm ⁻³ min.	18.45 g-cm ⁻³ U-Mo 17.9 g-cm ⁻³ U-Si	-	-
Grain size	40 μ max.	100 μ max. U-Mo 400 μ max. U-Si	-	-
Metallurgical Quality	* no heat treat	*	*	*
Inspection	100% visual† and non-destructive	100% visual† and non-destructive	100% visual† and non-destructive	100% visual† and non-destructive
Identification	**	**	**	**

* Representative samples to be checked for grain size, structure, and general metallography.

† Only light imperfections acceptable; no end cracks.

* All pieces will be identified to relate each piece to processing details and chemical analysis.

NOTE: Complete chemical and metallurgical histories will be available on all pieces.

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will be fabricated by Nuclear Metals. These will be 3/4 by 6 inch slugs and will be loaded into standard SRE fuel rods.

All of these experimental fuels have delivery dates in late 1955 in order to be inserted into the initial full power loading of the SRE. These alloy fuels will be loaded into mixed clusters; that is, each rod will contain one alloy but the seven rods in a cluster may contain different alloys. The mixed clusters will be removed at specified intervals and the various fuel materials will be examined in the SRE hot cells. This will provide relative performance data on all of the experimental materials, as well as permit monitoring of the primary fuel material, the unalloyed beta-heat treated uranium.

B. MTR IRRADIATION OF SRE FUELS (L. E. Wilkinson)

Since there have been little or no data on irradiated fuel materials at SRE temperature conditions, capsule tests have been designed to irradiate in the MTR small specimens of each of the experimental alloys, with internal temperatures at 1200° F and surface temperatures at 900° F. The capsules (see Fig. 8) are stainless steel, welded in series of six, with two thermocouples per capsule. Each capsule contains a specimen 3/8 inch by 1-3/8 inches with a 1/2 inch thermocouple well at one end. The initial series will contain three alpha-rolled beta-heat treated uranium and three U-Zr cast specimens. The other alloy materials are being fabricated for subsequent test series. Dummy capsules have been loaded, tested non-destructively, and assembled in the manner proposed for the test unit, with satisfactory results. Capsules for the test unit have been loaded and are undergoing non-destructive testing at present. MTR insertion date is planned for mid-July. All tests will be irradiated to 0.5 per cent burnup of total atoms, which, since the specimens contain 10 per cent U-235, will require a 15 week irradiation. Post irradiation examination will be done in the SRE hot cell and will include a test for fission gas evolution from the capsules. Apparatus for specimen decanning is being procured and modified for use in the hot cell.

V. ZIRCONIUM BEHAVIOR IN SODIUM SYSTEMS

F. E. Bowman, D. J. Zigrang, R. L. Eichelberger

The first run of the sodium corrosion loop was completed in mid-April, after 35 days at 1000° F. Failure of the plugging indication prevented a continuous

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Fig. 8. Capsule Components

monitoring of the oxygen level. The zirconium specimens were, however, rather severely oxidized (weight gains of about 0.60 mg/cm^2). An increase in surface hardness to a depth of about 0.001 inches was noted, indicating oxygen diffusion. Specimens were found to be brittle at room temperature in bend tests. Elevated temperature bend tests (1000° F) revealed ductility essentially equivalent to the "as-received" material. Eight to ten reversals of a 150° bend over a 3T radius were readily obtainable.

During the loop shut-down several changes and improvements were made. A specimen changer was added to the test pot permitting the addition and removal of specimens during a run without disturbing the system. The plugging indicator was improved and heater repairs made.

New zirconium specimens were placed in the system after cold trapping had reduced the oxygen level to 40 ppm as determined by the plugging indicator. Further running apparently reduced this level to less than 10 ppm since the indicator would no longer plug. Three specimens were removed after ten days. These showed weight gains of about 0.30 mg/cm^2 , surprisingly high if the oxygen content of the sodium had been accurately determined. It is possible, however, that these weight gains were effected during the short time the specimens saw 40 ppm sodium. No loss in room temperature ductility or hardness increase was noted, suggesting that the time at temperature was insufficient to effect any oxygen diffusion. The loop is now being stabilized at an oxygen level of 30 ppm and new specimens will be placed in test at this level. KAPL results indicate satisfactory behavior under these conditions.

Capsule tests have been conducted to determine the effectiveness of possible methods of lowering the oxygen content of hot sodium to a value below that at which zirconium is adversely affected. Results to date indicate that calcium added to sodium at 1200° F has a deleterious effect on the room temperature mechanical properties of zirconium immersed in it for 30 days: this, despite the fact that thermodynamically the available oxygen content should be reduced appreciably by the calcium. Spectrographic analysis of zirconium so treated indicates some calcium pickup and an increased iron content. A metallographic examination of a zirconium specimen after thirty days in calcium in a stainless steel capsule at 1600° F gives evidence of the Fe-Zr eutectoid.

Tests conducted with zirconium-titanium 50 mole per cent alloy as a hot trap getter material show that pure zirconium is protected by the preferential oxidation of the alloy when present.

A circulating sodium loop has been designed to test the various counter-measures against oxygen and hydrogen embrittlement of zirconium. This loop is now under construction. First tests in it will give an evaluation of the effectiveness of zirconium-titanium alloy turnings as a hot trap getter in a rheological system.

In addition to metallographic and microhardness investigations the specimens from the various loops will be subjected to elevated temperature flexure tests to determine the notch sensitivity and general resistance to cyclic stresses.

VI. ENGINEERING EVALUATION OF GRAPHITE

D. J. Klein

A. IN-PILE THERMAL CONDUCTIVITY OF GRAPHITE (MTR-NAA-3 Experiment) (R. E. Durand, D. J. Klein)

This experiment was carried out in order to examine the behavior of moderator graphite in the temperature range, 300° to 700° C, covering the conditions of the SRE. Lattice spacing measurements by X-ray diffraction have confirmed similar measurements at Brookhaven National Laboratory, showing that the irradiation resulted in no detectable change in lattice parameter.

The topical report has been written and is being prepared for publication.

B. IRRADIATION OF HIGH-DENSITY GRAPHITE TYPES (MTR-NAA-16 Irradiation) (W. J. Greening)

The samples of high-density graphite (density ~2.0) have been irradiated and await shipment to Downey for measurement of length, strength; resistivity and volume of evolved gases, in order to determine possible suitability in sodium-graphite type reactors.

C. EXPERIMENTS IN THE SRE (R. R. Eggleston, W. K. McCarty)

Experimental facilities consisting of a full-size moderator element (zirconium-clad graphite) and a smaller experimental moderator element model

(stainless steel-clad graphite) have been designed. The design provides for the circulation and monitoring (for chemical composition) of inert gas, and the measurement of temperature in the graphite, in several positions. Particular attention will be given to the full-scale moderator element, as it provides an indication of the behavior of the remaining moderator elements of SRE. Comparison with the stainless steel-clad element will indicate the role of the zirconium.

Design of the remainder of the experiments and irradiations under this program will be initiated during the next quarter. They are intended to investigate in more detail the structural, thermal and chemical properties of various grades of graphite under consideration for a Sodium Graphite Reactor, under conditions of temperature and surroundings anticipated for such a reactor.

VII. ORGANIC COOLANT STUDIES

E. L. Colichman

Investigations dealing with development work in the organic coolant program are no longer considered a part of the SGR program. In the future, developments related to hydrocarbon coolants will be reported separately. Concluding work on the development of an alternate shield coolant for the SRE was carried out during the last quarter and is reported here.

Both xylene and toluene have been previously considered as suitable for use as the SRE shield coolant. It was suggested that tetralin (1, 2, 3, 4 - tetrahydronaphthalene, $C_{10}H_{12}$) would be more suitable than either of these two because of its considerably higher flash point and lower toxicity.

In order to further determine the suitability of tetralin, its radiation resistance was measured. Quartz tubes containing tetralin in the absence and presence of iron and aluminum metal were irradiated at $150^{\circ}F$ in the MTR gamma facility over the dosage range 0.5 to 3.6×10^8 roentgen. The radiation resistance was evaluated on the basis of gas evolution (see Fig. 9). Corrosion of iron and aluminum submerged in tetralin while in the radiation field was also determined (see Table III).

The gas evolution data indicate that tetralin is substantially as radiation resistant as toluene at dosages up to about 0.8×10^8 roentgen. This is compared

Fig. 9. Decomposition vs Gamma Dosage at 150° F (MTR Gamma Facility)

33

TABLE III

DECOMPOSITION AND METALLIC CORROSION BY GAMMA IRRADIATION AT 150° F

Compound	Dosage, R	Gas Evolution	Corrosion		For 3 Year Exposure		
		(ml/gm)	Wt. Change, mg. (mg.)		Gas Evol. (ml/gm)	Penetration (Mils/3 yrs.)	
			Iron	Al		Iron	Al
Tetralin	2.1×10^8	1.05	-0.5*	+0.1	0.15	0.01	0.01
Toluene	2.1×10^8	0.75	1.3	+0.6	0.12	0.03	0.04

* Both billets exposed in the same sample of tetralin to check bimetallic corrosion

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to an estimated accumulative dosage, equivalent to 0.5×10^8 roentgen, that will be acquired by the shield coolant fluid when operated for a three year period in the SRE. Corrosion and radiation induced corrosion effects are seen to be negligible and are even lower than that for toluene.

It is concluded that on the basis of its physical and hydraulic properties and low toxicity that tetralin is more suitable for use as the SRE shield coolant than either toluene or xylene.

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SECTION B

SODIUM REACTOR EXPERIMENT

VIII. NUCLEAR ENGINEERING, PHYSICS

A. NUCLEAR ENGINEERING (W. E. Abbott)

Control rod calculations have been carried out and the position of the rods for the hot and equilibrium poisoned core has been determined. The axial flux distribution obtained for this case is being used to re-evaluate the temperature rises in the fuel elements. The radial flux distribution, in conjunction with the axial distribution, is being used to determine the proper orificing for the SRE core.

A study of the change in SRE power capabilities for various insertions of the control rods is also in progress. Preliminary results indicate that the reactor power will increase as the control rods are inserted until the rods are at least half way into the core.

The compositions of several experimental fuels have been studied to determine their effects on the reactivity of the SRE. Fuels have been chosen which yield the same multiplications as the standard SRE fuel.

B. REACTOR PHYSICS (F. L. Fillmore)

1. SRE Core Data - In the previous quarterly report,⁵ it was noted that criticality under three different core conditions-dry, wet, and hot poisoned-had been studied and the core size for the hot poisoned condition had been determined. The two-group, reflected-reactor problem has now been coded for machine calculation and as soon as the code is thoroughly checked, the critical core size for the dry and wet critical assemblies will be computed.

2. Reactivity Studies - The effect of various reactor perturbations on reactivity is being evaluated by means of two group perturbation theory. The perturbations being considered are: (1) the dummy fuel elements in the extra coolant tubes, (2) a dummy fuel element in place of a normal one, (3) the stainless steel control rod thimbles, (4) the stainless steel reflector cans, and (5) a small change in enrichment.

The reactivity controlled by the control rods is being investigated by means of two group perturbation theory. The effect of a rod located on the axis of the

reactor is being re-examined, as well as the combined effect of the proposed four control rods.

The reactivity change caused by a fuel element dropping into the reflector was calculated by perturbation theory and was found to be approximately +0.15 dollars. This would result in a stable period of about 60 seconds.

IX. LAND, UTILITIES, BUILDINGS

A. LAND, UTILITIES

The SRE site water distribution system has been completed. The welded 4-inch steel water line, approximately 2000 feet long, from the water well to the 50,000 gallon storage tank, has been installed and the 8-inch fire water main has been completed except for the installation of the fire pumps.

Installation of the electrical power distribution system at the site has been completed. Three 1000 kva transformers for this system have been installed at the Substation (see Fig. 10) and are now in use.

The security fence surrounding the SRE site and the yard lighting are the major improvements remaining to be done, but these items are being deferred so as not to interfere with construction progress.

B. REACTOR BUILDING

All excavation in the reactor building area for the various sumps and drain lines has been completed. The three moderator can storage cells and 96 sonovoids (heavy cardboard forming tubes) for the irradiated fuel element storage cells have been positioned and the pouring of concrete around these units has been started. The three liners for the fuel cleaning cells have also been placed in position; these will be imbedded in earth backfill. Liquid waste drain lines, the stainless steel floor plates for the fuel handling and the fuel evaluation hot cells have been positioned. The concrete foundations have also been poured. The hot cell floor plates, moderator storage cells, fuel storage cells, and fuel cleaning cells are shown in Fig. 11.

The walls for the hot cell room have been formed and the first 12-foot lift of concrete has been poured. Necessary forming pads for the fabrication of the tilt-up concrete wall panels have been prepared. The pouring of concrete has been

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37

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Fig. 10. Three 1000 KVA Transformers at SRE Site Substation

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Fig. 11. General View of Hot Cell Floor Plates, Moderator Storage Cells, Fuel Storage Cells, and Fuel Washing Cells.

started for the reactor building, the handling bridge column footings, and the vaults for the primary sodium fill tank and the disposable cold trap. Figure 12 shows the present status of the reactor building area

C. ENGINEERING TEST BUILDING

The engineering test building has been completed and is now occupied by NAA personnel. Equipment has been set-up in the building. The uranium storage vault has been completed and the SRE fuel slugs will be transferred from the Downey Plant when required. The security fence surrounding the engineering test building has been completed and the security arrangements have been inspected and found satisfactory by AEC representatives.

X. FUEL ELEMENTS

A. FUEL ROD JACKETS (J. Droher, H. Strahl)

An order of 5600 feet of Type 304 stainless steel tubing for jacketing the fuel rods was received from the vendor. The tubing was fabricated from stainless steel strips, 1-1/2 inches wide by 0.65 inches thick. It was shaped, welded, and then drawn to its final dimensions of 0.790-inch OD by 0.010-inch wall. Of the 700 pieces of tubing received, 44 pieces were found to be defective and were accepted for experimental use only.

Metallographic analysis (see Fig. 13) showed the tubing to be very clean with very few stringers, oxides, or other inclusions. The chemical and spectographic analyses are listed below:

Chemical Analysis

C - 0.07
Mn - 1.58
Cr - 17.70
Ni - 10.28
P - 0.027
S - 0.007
Si - 0.58

Spectographic Analysis

B-trace	Pr-nil	Tm-trace
Cu-low	Co-nil	Yb-nil
Cd-nil	Gd-trace	Lu-nil
Mo-low	Dy-nil	Ti-nil
Cb-nil	Ho-nil	Pm-*
Ce-trace	Er-nil	Tb-*

*not determinable

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Fig. 12. General View of Reactor Building Area

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0.010 inch

TYPE 304 STAINLESS STEEL TUBING
(100X)

Fig. 13. Metallographic Analysis of Stainless Steel Fuel Rod Jacket Tubing

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Eleven tensile specimens showed yield strengths ranging from 79,000 to 125,000 psi and ultimate strengths ranging from 105,000 to 148,000 psi. Elongation varied from 27 to 39 per cent. Hydraulic strength tests indicated that the tubing would not burst until internal pressures ranging from 2300 to 3100 psi were applied. The above tests were conducted at room temperature.

Tensile tests were performed to determine the mode of failure and the ultimate strength of the weld securing the fuel rod end cap to the tubing. Specimens were tested in air at room temperature, 1000 F, and at 1200 F. The results are given in Table IV. Three of the eight specimens tested at room temperature failed either at or adjacent to the weld. All four specimens tested at elevated temperatures failed in the tubing probably indicating that in each case the weld was stronger than the tubing. Since the welding current and atmosphere were varied to determine the optimum welding conditions, a failure either on or next to the weld indicates unfavorable conditions. In all cases the load at which failure occurred was from 70 to 100 times greater than the 20-pound load which is exerted in normal operation. Typical failures are shown in Fig. 14.

B. FUEL ROD ASSEMBLY (J. Draher, H. Strahl)

A vacuum retort for outgassing uranium slugs and heat treating fuel rods has been fabricated, assembled, and leak checked and heaters have been installed (see Fig. 15). Holding fixtures for outgassing the slugs in 21 batches of 12 slugs each have been fabricated and delivered. These fixtures are shown in Fig. 16. A fixture for heating 21 fuel rods has also been fabricated and delivered and is shown in Fig. 16 with one fuel rod in place. The outgassing and the heat treating procedures are discussed in the previous progress report.⁵

A 21-rod fuel loader tank and lid which were fabricated during the last quarter were rejected because of deviations from specifications on the part of the vendor. Fabrication of a new tank and lid have started in NAA facilities. The internal racks, weld head, variable drive transmission, bond loader, and other miscellaneous parts have been fabricated and delivered.

The installing of equipment in the fuel loading area of the Engineering Test Building has been started. The fuel storage vault has been completed and is ready for use. The fuel loading pit has also been completed and the outgassing retort has been placed in position in the pit. Overhead hoists have been installed and are awaiting completion of electrical work in the building by the contractor.

43

TABLE IV

TENSILE TEST DATA: TYPE 304 STAINLESS STEEL CAP WELDED TO
TYPE 304 STAINLESS STEEL TUBE

Spec. No.	Type Cap	Cap Material	Tube O.D. in.	Type Tubing	Wall Thick in.	Welding Current amps.	Test Temp. °F	Atm.	Load lbs.	Ult. Stress ksi†	Remarks
9	Outer	Old	0.793	Exp.	0.0112	40	RT	He	1735	64.9	Failed adjacent to weld
10	Outer	Old	0.793	Exp.	0.0111	40	RT	He	2200	80.7	Failed adjacent to weld
11	Center	New	0.791	SRE	0.0102	9	RT	He	2780	111.1	Failed in tube
12	Center	New	0.791	SRE	0.0101	29	RT	He	1915	77.3	Failed in & adj. to weld
13	Center	New	0.790	SRE	0.0102	12	RT	He	2745	110.9	Failed in tube
15	Outer	New	0.791	SRE	0.0102	40	RT	He	2735	109.3	Failed in tube
16	Outer	New	0.790	SRE	0.0101	40	RT	He	2760	111.5	Failed in tube
17	Outer	New	0.790	SRE	0.0101	40	RT	He	2735	110.5	Failed in tube
3	Center	Old	0.793	Exp.	*	12	1000	He	1710	69.6*	Failed in tube
4	Center	Old	0.793	Exp.	*	12	1000	He	1710	69.6*	" " "
5	Outer	Old	0.793	Exp.	*	18	1000	He	1690	68.8*	" " "
6	Outer	Old	0.793	Exp.	*	20	1000	Ar	1690	68.8*	" " "
1	Outer	Old	0.798	Exp.	*	18	1200	He	1515	61.6*	" " "
2	Outer	Old	0.798	Exp.	*	20	1200	He	1515	61.6*	" " "
7	Center	Old	0.793	Exp.	*	12	1200	-	1385	56.3*	" " "
8	Center	Old	0.793	Exp.	*	12	1200	-	1385	56.3*	" " "

*Wall thickness taken as 0.011 in. for stress calculation (experimental tubing has 0.011-inch wall thickness).

Specimens at 1000 and 1200° F had a cap welded on each end of the tube (load applied through caps).

Room temperature specimens had a cap on one end of tube only.

† 1000 x psi

Fig. 14. Typical Specimens of End Caps Welded to Fuel Rod Jackets After Tensile Strength Tests

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Fig. 15. Outgassing Retort

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Fig. 16. Fixtures for Outgassing Retort

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Final design of the end caps and other fuel element hardware has been completed. The new upper guide design, shown in Fig. 17, provides a positive locking action on the end caps which will prevent the rods from vibrating excessively, from rotating in place, and from being inadvertently forced out of the fuel element. Wrench flats have been added to the center rod end caps to prevent undue torsional stresses on the end cap welds during assembly and disassembly. The orifice plate has been moved downward on the lower guide and has been provided with positioning pins to orient holes in the plate with inner channels in the fuel element. The finned plates on the lower guide have been replaced with open wires, as shown in Fig. 18, to permit cross flow and mixing. An order has been placed for the end caps, bottom spiders, spacers, orifice plates, and lower guides required for the standard fuel elements in the first SRE loading.

The upper plug assembly which will be used in fuel element handling experiments has been fabricated and delivered to the tower at Santa Susana. Concrete will be omitted from the plug until the mix for high density concrete has been established. Until then steel shot will be used to give the plug the required weight during handling tests.

During preliminary testing, solid stainless steel rods will be used in place of fuel rods. These will be replaced with regular fuel rods of uranium encased in thin-walled stainless steel tubing which will be assembled during the initial operation of the 21-rod loader. An assembled fuel element is shown in Fig. 19.

Bonding void tests have been performed at elevated temperatures using a simulated fuel rod with known voids in a solid aluminum bond. The usual Bakelite form was replaced by a Lavite coil form and the Croloy flux guides were omitted. Tests were carried out at 160 F, 355 F, and at 465 F to detect the effect of temperature on coil sensitivity. Although there was a loss in power because of the increased impedance of the coil and the specimen, it was still possible to detect the voids at the test temperatures.

Removable end caps to be used on special unbonded rods will be required for the dry critical experiments. Threaded joints cannot be used in the 0.010-inch wall tubing. A screw-pin cap joint has been fabricated and tested for shear strength. It failed at 480 pounds, which is well above the 20-pound load requirement.

TOP END OF
FUEL ELEMENT

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Fig. 17. Fuel Element Upper Guide and Spider

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(9693-D5158)

Fig. 18. Old and New Lower Guide Designs

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Fig. 19. Assembled Fuel Element

Coolant mixing is obtained primarily by the rotational flow induced by the wire wraps which are helically wound around each of the 6 outside rods.

The degree of coolant mixing has been experimentally determined. At the upstream end of the fuel element, a chemical solution was injected into the coolant stream flowing between the rods. Then at various distances downstream, measurements were made of the chemical concentration in the coolant flowing between the outside of the rods and the coolant tube wall.

The results of this experiment showed that the degree of coolant mixing was sufficient. The experimental data also indicated that the flow was not symmetrical with respect to axis of the fuel element. This was probably because of the non-symmetrical orientation of the wire wraps on the fuel element used in the experiment. Apparently the wire wrap orientation was also responsible for the undesirable lateral flow across the inner channel of the fuel element. In the previous quarterly progress report,⁵ this action was ascribed to slight longitudinal bends in the fuel element, but subsequent data showed that this interpretation was not a correct one.

When the distance between two adjacent wire wraps at the throat of a coolant passage which joins the inner and the outer channel is relatively small, the flow between the two respective channels is enhanced and consequently the amount of mixing is increased.

This experimenter recommends that wire wraps on the fuel element should be symmetrically oriented with about 1/8-inch minimum space between wires in the alternate passages between rods.

b. Flow Distribution in Core Tank Plenum - To determine hydraulic pressure distribution in the SRE coolant outlet plenum, a 1 to 3.5 scale model has been constructed with water as the circulating fluid. The model was designed so that relative coolant velocities are the same as in the prototype, achieving kinematic as well as geometric similarity. In addition, the design permits the testing of various flow vanes and baffles in the inlet plenum. Other experiments are also planned for the model; these include an examination of operational practices associated with removal and replacement of fuel elements.

Mechanical and instrumentation installation has been completed and check-out runs are nearly concluded. Experimentation will begin shortly.

XI. MODERATOR AND REFLECTOR CELLS

A. DESIGN

Several design changes affecting the moderator and reflector cells have been made during this report period. First, the number of coolant channels in the SRE core has been reduced from 55 to 43. Second, breather tubes have been added to all moderator and reflector cells to prevent pressure build-ups within the cells. These are snorkel-type tubes which vent the insides of the cells to the helium atmosphere above the core sodium pool. Special weld fittings have been designed to support the breather tubes and reduce bending loads on the tube welds.

Fabrication drawings for the moderator and reflector cells have been released and test cells have been designed for installation in the SRE. Two of the test cells will be provided with thermocouples for temperature distribution measurements and access tubes for gas sampling, pressure control, and sodium leak detection. The problem of gaining access to the cells through the loading face shield and the sodium pool has not yet been fully solved.

B. ZIRCONIUM PROCUREMENT (W. L. Cockrell)

Difficulties met in converting zirconium billets to sheet and plate have resulted in relatively low yields. To mitigate this situation, an effort has been made to salvage cell side panel strip by welding short lengths end to end. In this operation, maintaining longitudinal alignment has been difficult.

Two more zirconium billets are being procured, one large billet for strip and a smaller billet for tube weld fittings.

C. FABRICATION AND WELDING (R. C. Brumfield, W. L. Cockrell)

Twist must be eliminated from moderator and reflector cells so that interference will be avoided when the cells are installed in the reactor. Twist can be detected by measuring the relative angular displacement between two pairs of ring and plug gages locked on opposite ends of the cell (see Fig. 20). To eliminate twist, a weld-stake jig will be used for the last longitudinal weld on the hexagonal cell walls and the three graphite blocks making up the cell stringer will be doweled together.

The 2-7/8-inch OD coolant tubing procured for the development program was found to be defective. Welds were relatively hard, had cracked during swaging,

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Fig. 20. Plug and Ring Gage on Moderator Cell Can

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and had caused the tubes to bow about one inch per ten feet. During a visit to the vendor's plant technical information, weld samples, and test data were exchanged. More systematic welding controls will now be enforced and a better product will undoubtedly result. Two short lengths of 2-7/8-inch OD tubing have been made at NAA by brake-forming the two halves and then welding them longitudinally. This method will be available in case the vendor fails to fabricate the tubing satisfactorily. The major hazards are possible setback in schedule and loss of tube strip.

End expansion of the coolant tubes has resulted in some tearing in the welds and cracking of the parent material. Cracking can be detected by dye-penetrant checks. In some cases, these difficulties can be avoided by heating the tube end in liquid solder (400 to 500° F). Further improvement can be effected by heating the expansion dies.

The first 0.035-inch wall cell assembly was completed by the middle of June. Also, the fabrication of six hexagonal zirconium (0.035-inch thick) cans for development purposes was completed successfully except for excessive twist. The fact that these are twisted will not affect their usefulness in the development program.

Several cell panels and a two-scallop cell can are shown in Fig. 21.

D. GRAPHITE PROCUREMENT (R. L. Carter)

A vendor has been selected to supply the moderator graphite for the SRE and an order has been placed for graphite of TSP grade. The original gas content specifications established by NAA required that less than 0.03 cm³ (NTP) gas per cm³ graphite be released at 1000 C and less than 0.15 cm³ per cm³ be released at 1800 C. Also, for protection against the atmosphere during shipment, it was prescribed that the finished graphite pieces be wrapped in 0.001-inch aluminum foil within resin-sized paper wrapping.

Production of the first batch of material was completed early in June and machining was to begin on June 16. The vendor's evaluation of the gas content, however, was so unfavorable that the vendor has refused to accept the specifications until more experience has been accumulated. To maintain a reasonable delivery date and avoid extended holding of the unprotected raw graphite, the specifications have been relaxed for the first batch of 36 stringers to accept 0.09 cm³ gas per cm³ graphite at 1000 C and 0.24 cm³ per cm³ at 1800 C. This first batch will be employed to machine E- and F-type graphite stringers only. After the raw

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Fig. 21. Moderator Cell Can Assembly and Side Panels

material for the remaining stringers is purified, the pieces will immediately be wrapped in two-layer foil and resin-sized paper for protection during shipment to the machine shop.

It is expected that, as the vendor gains experience, the original more stringent specifications will be re-established for the fabrication of the more critical stringers.

E. GAS EVOLUTION (R. L. Carter, W. H. Greening)

The practice of measuring pressures during graphite degassing with a thermocouple gage has been modified. To enable comparison with National Carbon Research Laboratory results, pressure measurements are being made with a McLeod gage. Calculations of the relative, expected, and apparent pressures were made for gas of approximately the composition observed in mass spectrometer analyses of desorbed gas. These were in agreement with the pressure rise curves measured by thermocouple gage and by the McLeod gage during degassing of identical samples of graphite. This pair of samples had been previously standardized by degassing to 2000 C, then permitting readsorption of room air for 75 days.

The effectiveness of cooling in helium following purification of the graphite was studied by degassing a set of samples prepared by the National Carbon Research Laboratory. These were immersed hot in helium and cooled to 1300, 600, and 200 C, respectively, before being exposed to air. The results, when compared with results of similar measurements at National Carbon, indicated, at most, a weak dependence on helium letdown temperature, but suggested a strong dependence on time of exposure to room air. This was verified by a comparison of the gas content for degassed graphite samples as a function of time. Possible increased effectiveness of argon gas as a means of retarding room gas sorption was tested and yielded negative results.

To evaluate various means of wrapping finished pieces of graphite in order to protect them from gas pick-up, four samples, 12 inches in diameter by 1 inch thick, were wrapped for shipment in brown paper, resin-sized paper, polyethylene plastic, and 0.0015-inch aluminum foil, respectively. Gas evaluation measurements indicated that aluminum foil wrapping gave significantly better protection than the others.

Mass spectrometric analyses (see Table V) show that the principal component of the gas released from the purified graphite is invariably hydrogen (70-80 mole per cent). The balance is almost entirely carbon-monoxide. In the case of the

TABLE V
GASES RELEASED BY 1700° C VACUUM DEGASSING
(MOLE PER CENT)

Component	Purified GBF Cooled in He To 1300° C	Purified GBF Cooled in He To 200° C	Purified TSP
Hydrogen	70.8	77.5	81.5
Carbon Monoxide	25.6	15.7	16.9
Carbon Dioxide	1.7	0.1	0.1
Nitrogen	0.7	1.5	1.4
Helium	--	4.6	0.02
Methane	0.5	0.4	0.1

sample which was cooled in helium to 200 C, a quantity of helium was trapped in the pore volume, but not a large enough quantity to substantially reduce the amount of active gases occluded.

F. HEAT AND VACUUM TREATMENT

A new stainless steel vacuum retort (15-inch ID by 132 inches long) has recently been placed in service. With the retort interior at a temperature of 1200 F absolute pressures in the neighborhood of 3×10^{-5} mm of mercury can easily be obtained. The maximum operating temperature is 1500 F.

The first zirconium cell fabricated (0.050-inch wall, electrode graphite) was placed in the retort to test the corrosive influence of outgasses on zirconium pressure buildup and to check the possibility of holding a longitudinal temperature gradient on the cell. The top furnace coil was set to 1200 F; the lower coils were not turned on. The bottom end of the cell eventually reached a temperature of about 700 F, thus duplicating the temperature gradient planned for the SRE. This cell was placed in the retort on May 21, 1955 and will be left there until a new cell is fabricated with a 0.035-inch wall and either Speer or TSP graphite. The next test will add a sodium source with the furnace kept at a high temperature as long as possible to gain information on the behavior of the cells under simulated SRE operating conditions. At a later date, pressure fluctuations will be superimposed to simulate structural loads and breathing action of the helium atmosphere. The experiments

will then be repeated with stainless steel reflector cells. In this way, it is hoped that potential difficulties with SRE moderator and reflector cells can be anticipated on the basis of high temperature operating experience.

G. SODIUM DIFFUSION EXPERIMENT

A sodium diffusion experiment was started on February 15, 1955. An exploded view of the apparatus is shown in Fig. 22. A temperature of 1200 F was established in the sodium container and around the breather tube, dropping off linearly to 750 F at the bottom end of the condenser tube. The experiment was continued for 0.136 years and, at the end this time, 2.2 gm of sodium had diffused to the lower end of the tube. At this rate, a 25 cubic inch cup would fill in 21.6 years. This diffusion rate is 5.1 times higher than the rate calculated, although theoretical and experimental models were not identical.

The diffusion apparatus has been rebuilt and put back into operation for a check run. In addition, a similar apparatus will be mounted on top of moderator and reflector cells during high temperature tests in the retort.

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Fig. 22. Sodium Diffusion Apparatus

XII. REACTOR STRUCTURE

A. CORE TANK ASSEMBLY AND INTERNAL STRUCTURE

Fabrication of the 304 stainless steel core tank has been started at the vendor's shop. The flanged bottom head has been formed and is now being welded to the rings which form the sides of the tank. Since this tank will be in contact with hot sodium, the specifications and inspection requirements are stringent. Although there have been some delays due to design changes, lack of qualified welding personnel and laboratory testing of the materials, NAA inspectors have accepted the work performed to date. Because of the delays, a revised delivery date, approximately three months behind the original schedule, is now anticipated. Every effort is being made to minimize the effect of this delay upon the reactor installation schedule.

Tests have been made to determine the amount of galling and sliding friction between the type 304 stainless steel core tank bottom and several alloy steels considered for the support plate on which the core tank will rest. The test apparatus is shown schematically in Fig. 23 and 24. The test was run in a helium atmosphere at 750 F. The metal specimens were cylinders, machined so that the area of contact was one square inch (see Fig. 25) and the pressure applied to the specimens was 100 psi. Rotary motion was applied by means of a hand torque wrench.

Two tests were run of type 304 stainless steel vs type 4130 steel and one test was run of type 304 stainless steel vs type 1340 steel. Each test consisted of 600 cycles of rotation through a 36-degree arc with a one-minute pause between cycles. After the test, the type 1340 steel specimen was found to be badly rutted; galling was also expected because of a jerkiness encountered during the cycling process. Inspection of the type 4130 steel specimen after the test showed very little wear and there was no indication of galling. As a result of these tests, type 4130 steel has been specified for the core tank support plate.

The fabrication drawings of the stainless steel plate which forms the core tank liner have been completed and the steel was ordered. Layout drawings of the core clamps have been approved and detail drawings are now being prepared.

Fig. 23. Core Tank Galling Test Capsule

Fig. 24. Core Tank Galling Test Apparatus

62

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Fig. 25. Typical Galling Test Specimens

B. OUTER TANK ASSEMBLY AND SUPPORTING STRUCTURE

The contract has been awarded for the fabrication of the steel outer tank and all material obtained. The transition flange of the tank has been welded to the 3-inch thick (149.75-inch diameter) bottom plate. This assembly has passed inspection and the top and bottom surfaces of the bottom plate will now be machined parallel.

C. THERMAL SHIELD RINGS

Fabrication of the cast steel thermal shield rings is now underway at a vendor's shop. Progress is satisfactory and no fabrication difficulties have been encountered. An effort is being made to reduce the fabrication time so that all rings will be delivered to the reactor site by August 26, 1955.

D. CORE CAVITY LINER

The field assembly and welding of the Core Cavity Liner was completed on June 7 (see Fig. 26).

Thermocouples to measure the temperature on the outside of the liner and in the concrete (which will be poured around the liner) have been installed and tested. The toluene cooling pipes have been welded on the outside of the liner and the inlet and outlet toluene manifolds installed. Shoring has been installed in the liner to maintain concentricity during the pouring of the concrete and concrete has been poured to the elevation of the sodium pipe bustles (see Fig. 27). Cooling coils have been mounted on the sections of the instrument thimbles which protrude through the wall of the core cavity liner (see Fig. 28) and the instrument thimbles attached to these sections as shown in Fig. 29.

E. THERMAL INSULATION

To prevent flaking of the Superex block insulation which surrounds the core tank assembly, coating the insulation with sodium silicate has been considered. To test this coating against the action of molten sodium, two cylindrical insulation specimens, three inches in diameter by two inches thick, were first baked for 12 hours at 1000 F, then brushcoated with a sodium silicate solution prepared by dissolving sodium metasilicate in boiling water. After three hours of air drying, the specimens were baked for 15 hours at 1000 F. Then, both specimens, loaded

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Fig. 26. Completed Core Cavity Liner

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Fig. 27. Core Cavity Liner After First Pour of Biological Shield Concrete

66

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Fig. 28. Instrument Thimble Stubs Passing Through Core Cavity Liner

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67

6-24-55

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Fig. 29. Core Cavity Liner After Installation of Instrument Thimbles

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with a one-inch cube of sodium, were placed in a vacuum pot and baked for three hours at 1000 F. Examination showed the specimens to be intact though badly honeycombed with glass-like slivers.

XIII. HEAT TRANSFER

A. SODIUM SYSTEM

1. Design

a. Piping and Instrumentation (A. M. Stelle) - Several design changes have been made in the sodium systems during this report period.

The main sodium circuit pipes have been rerouted in the main galleries to lessen the stress imposed on the heat exchanger nozzles and to improve pipe support locations. This routing change has added several feet of pipe to the system but has not affected the circuit pressure drops.

In the auxiliary primary system, an orifice has been installed at the entrance to the reactor core tank assembly to increase the system pressure drop by 5 psi. The additional pressure drop limits flow changes in the auxiliary primary system to 20 per cent when the flow in the main system is varied from 10 to 100 per cent of full flow.

In the auxiliary secondary system, a circulating cold trap has been replaced by a diffusion cold trap which has been installed on the expansion tank.

Pressure drop and pump horsepower calculations for the secondary circuits have been revised to account for changes in pressure drop and circuitry of the air-blast heat exchangers. These changes have increased the total horsepower required in the secondary circuits.

Piping specifications, piping and instrumentation diagrams, and process descriptions for the sodium circuits have been released. Piping material has been ordered and most of it is already on hand.

b. Sodium Pumps (W. D. Alderson) - The fabrication of four vendor-supplied sodium pumps is 40 per cent complete. The vendor has experienced some difficulty with stainless steel castings. Two castings were rejected for low tensile strength and it was necessary to make new ones. Items manufactured in NAA shops such as case extensions, floor plates, etc., are 60 per cent complete for the primary

pumps and 15 per cent complete for the secondary pumps. The pump assembly stand is about 30 per cent complete.

c. Heat Exchangers (R. Holmes) - Approved engineering drawings of the sodium system heat exchangers have been returned to the vendor. Final design changes include reorienting of certain nozzles, changes in support structures, and strengthening of some nozzles to withstand expected pipe loads. The welding procedures of the vendor have been approved with certain exceptions which are presently being negotiated. No further design changes in these units, except minor ones, are anticipated.

d. Sodium Cold Traps (R. Holmes) - Drawings have been released for fabrication of three boiling toluene circulating-type sodium cold traps. The design differs only slightly from the cold trap which is now undergoing development tests in the pump test loop. A disposable circulating-type cold trap from which the center stainless steel mesh section can be removed has been designed and is being evaluated. The diffusion-type cold traps which will be installed on the secondary system fill tank and in the auxiliary secondary system have been designed and released for fabrication.

e. Electromagnetic Drain Pumps (W. D. Alderson) - A specification has been issued to a vendor to supply three electromagnetic pumps for use in the sodium service system. Assembly drawings and wiring diagrams of the vendor's pump have already been received.

2. Engineering and Tests on SRE Components

a. Sodium Pump Development (R. Cygan) - Owing to system modifications and mechanical failure of two pump shaft freeze seals, there has been a relatively small amount of running time on this project.

Pump shaft seals of a new design are now being fabricated with emphasis on structural soundness and low cost. Past failures have been at brazed joints primarily and, therefore, the new designs eliminate or replace these joints with welded construction.

The lower pump stabilizing bracket has been moved upward so that it is now located above the case seal cooling jacket. Preliminary tests indicate that this bracket acts as a cooling fin. It appears to reduce the load on the case seal coolant system by about one-third.

Difficulties have been experienced in removing oil from the pump in the test loop. To permit a concentrated effort on lubrication problems without interfering with other vital tests on the pump loop, a separate lubrication test unit has been built. This unit will incorporate an SRE auxiliary pump bearing assembly and slingers.

A full-size cold trap system for boiling toluene has been added to the pump test loop. This system includes a sodium economizer, toluene reservoir, cold trap, and toluene condenser. Initial tests of this unit are now in progress. Start-up operations have indicated that when the cold trap is relatively dirty, the operating temperature must be lowered gradually to avoid plugging the economizer.

The results of a group of flow decay tests which have been run on the pump test loop will be extrapolated to apply to the SRE system. The curves obtained are in good agreement with calculated data. The tests have shown that more rapid flow decay takes place when the pump speed is reduced by means of the control rheostat instead of by opening the motor contactor. This was to be expected since opening the contactor allows the pump to coast freely to a stop whereas using the rheostat takes advantage of the dynamic braking action of the drive motor.

A circulating toluene system for cooling the various sodium freeze seals has been installed in the pump test loop. The system consists of a toluene reservoir, a toluene cooler (evaporative), flowmeters, and a toluene pump. The system has been checked out and is now ready to be tied in to the freeze seals.

b. Cold Trap Development (D. Condotta) - A prototype sodium oxide cold trap has been installed in the pump test loop (see Fig. 30). A series of tests have been run to obtain general operating information such as flow characteristics and heat transfer data. The cold trap appears to function satisfactorily and only minor design changes should be necessary. The following summarizes the information obtained so far.

1. The design flow of 3 gallons of sodium per minute can easily be obtained. Flows as high as 15 gpm were pushed through the cold trap with pressure drops of about 10 psi.
2. The heat transfer information which is available has not yet been completely evaluated. However, the cooling capacity seems to be more than adequate and fairly close to the design specifications.

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Fig. 30. Cold Trap and Economizer

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3. Good temperature control over the minimum cold trap temperature can be maintained by varying the level of the toluene in the jacket. The boiling toluene system provides the cold trap with automatic temperature control and permits low sodium temperatures with little danger of freezing.
4. On one test run, the cold trap lowered the oxide content of the sodium in the pump test loop from 80 parts per million to less than 10 parts per million. This oxide determination was made after 9 hours of operation. The rate of removal, however, may have been much faster than this would indicate.
5. The tests have shown that care must be taken to prevent oxide plugging in the economizer during cold trap start-up. This is especially true if the sodium contains a large amount of oxide (70 or 80 ppm).

c. Wedgeplug Valve Tests (J. C. Flint and J. P. Charles) - A 6-inch Wedgeplug valve was installed on a high temperature sodium test vessel and operated at 500, 700, 950, and 1250 F. Tests were made with pressures varying from a few psig to 30 psig. No difficulty was encountered in operating the valve at the temperatures and pressures of interest. The torque required to break away or to reseal the valve was not excessive even when the freeze seal was operated with full water flow.

The tests have indicated, however, the freeze seal design can be improved. Leakage across the valve was quite high at all temperatures. Also, the time required to break the freeze seal when the water flow in the seal was turned off varied from 5 to 30 seconds. After the tests, measurements made on the valve plug and on the valve seat showed variations between the two which prevented proper valve seating. Modifications are now underway which are expected to prevent these difficulties.

3. Engineering Tests of Sodium Flow under SRE Conditions (W. J. Freede)

a. Determination of Coolant Mixing in SRE Fuel Element - The SRE fuel element (Fig. 19) is a cluster of 7 rods constructed so that a center rod is surrounded by the remaining 6. The entire element is positioned vertically in a coolant tube through which sodium flows. To maintain adequate cooling within the fuel element, a definite amount of mixing of the coolant (sodium) flowing between the rods with that flowing around the outside of the rods must take place.

Coolant mixing is obtained primarily by the rotational flow induced by the wire wrap which are helically wound around each of the 6 outside rods.

The degree of coolant mixing has been experimentally determined. At the upstream end of the fuel element, a chemical solution was injected into the coolant stream flowing between the rods. Then at various distances downstream, measurements were made of the chemical concentration in the coolant flowing between the outside of the rods and the coolant tube wall.

The results of this experiment showed that the degree of coolant mixing was sufficient. The experimental data also indicated that the flow was not symmetrical with respect to the axis of the fuel element. This was probably because of the non-symmetrical orientation of the wire wraps on the fuel element used in the experiment. Apparently the wire wrap orientation was also responsible for the undesirable lateral flow across the inner channel of the fuel element. In the previous quarterly progress report,⁵ this action was ascribed to slight longitudinal bends in the fuel element, but subsequent data showed that this interpretation was not correct one.

When the distance between two adjacent wire wraps at the throat of a coolant passage which joins the inner and the outer channel is relatively small, the flow between the two respective channels is enhanced and consequently the amount of mixing is increased.

This experimenter recommends that wire wraps on the fuel element should be symmetrically oriented, with about 1/8-inch minimum space between wires in the alternate passages between rods.

b. Flow Distribution in Core Tank Plenum - To determine hydraulic pressure distribution in the SRE coolant outlet plenum, a 1 to 3.5 scale model has been constructed with water as the circulating fluid. The model was designed so that relative coolant velocities are the same as in the prototype, achieving kinematic as well as geometric similarity. In addition, the design permits the testing of various flow vanes and baffles in the inlet plenum. Other experiments are also planned for the model; these include an examination of operational practices associated with removal and replacement of fuel elements.

Mechanical and instrumentation installation has been completed and check-out runs are nearly concluded. Experimentation will begin shortly.

B. TOLUENE SYSTEM

1. Development (H. E. Richter) - An investigation was made of shield coolants other than toluene for the SRE. The object of this investigation was to learn whether materials providing less fire hazard were available to permit greater flexibility in the use of electrical equipment. Tetralin was selected as sufficiently promising to warrant further study. The irradiations were made in capsules utilizing the Materials Testing Reactor at Arco, Idaho. These tests, preliminary in nature, indicate radiation stability and corrosion resistance comparable to that of toluene. Pumping power and other design criteria indicate that tetralin may be substituted in a system designed for toluene with no changes in equipment.

A toluene cooling system has been designed and installed as part of the sodium pump test loop. This system will be used to gain operating experience and determine whether design modifications of the system will be necessary for the SRE.

2. Design (A. M. Stelle) - The toluene lines have all been relocated to accomodate shielding and manifolding requirements around the reactor. The lines running from the reactor building to the toluene pad have been relocated underground to allow unlimited overhead clearance for the truck entrance.

Pressure drop calculations have been completed and valving selected to balance the manifolding and the flow in all the various branch systems.

The 400 gpm, evaporative-type, toluene cooler will be replaced by two smaller coolers of the same type. Each unit will be capable of cooling 200 gpm of toluene from 125 to 95 F. This capacity enables either cooler to handle the total heat load so that should one cooler fail or require servicing, the system can continue to operate. Specifications have been written and the units are now being manufactured by a vendor. Delivery is scheduled for August.

The forced air coolers in the sodium pipe galleries have been replaced by finned tubing coils placed at the ceiling. The gallery atmosphere is circulated by convection past the coils to keep the galleries below 150 F. This change removes all of the rotating machinery from inside the shielded area.

The piping and instrumentation diagrams, system description, and piping specification for the toluene system have been released. Valves and special components have been ordered.

XIV. INSTRUMENTATION AND CONTROL

A. CONTROL ROD SYSTEM

1. Control Rod Ring Development (J. D. Howell) - Heat transfer tests on centrifugally cast nickel-boron rings have been completed. Rings with outside diameters of 2.500 and 2.487 inches were tested. Corresponding nominal annulus sizes, calculated for an average ring temperature of 1100 F and a thimble temperature of 725 F, are 0.0127 inches and 0.0195 inches, respectively. On the basis of these tests, a ring diameter of 2.487 inches was selected for the SRE control rods.

The above tests were made with the modified apparatus mentioned in the previous quarterly progress report.⁵ Besides these modifications, radiation shields of tantalum were installed to prevent the thermocouples and leads from "seeing" the heater. It was suspected that an error was introduced by the transfer of heat down the leads to the thermocouples embedded in the rings. A significant lowering of the apparent ring temperatures was observed after the shields were installed.

Figure 31 shows the results of the final heat transfer tests with 2.487-inch diameter rings. Results are plotted as allowable heat load in the rings vs thimble temperature. The parameter is maximum ring temperature.

The most unfavorable heat transfer condition in an SRE control rod occurs at the tip of a rod inserted halfway into the reactor core. Calculations indicate that the power transfer rate at the tip of a rod in this position is 8.3 kw/ft. and that the thimble temperature is 730 F. Based on these calculations and on the test data shown in Fig. 31, the maximum ring temperature should be not more than 1400 to 1450 F for the 2.487-inch ring. The maximum temperature gradient from one side of the ring to the other, when eccentric and touching the thimble tube, will not exceed 400 F.

During reactor operation the control rod rings will experience thermal cycling due not only to vertical movement of the rods, but to radial movement of the rings inside the thimble. The diameter as well as the thickness of the ring in a fixed size thimble determines the maximum attainable gradient across the ring.

Fig. 31. Heat Transferred Through Control Rod Rings vs Thimble Temperature

With a ring size for the control rods selected, and expected conditions established, it was necessary to investigate the resistance of the rings to warping and cracking under thermal cycling. The conditions selected for the cycle were:

- 1) Upper limit, 1500 F high side; 1100 F low side
- 2) Lower limit, 700 F high side; 450 F low side

One ring has been cycled 250 times between these limits. Measurements taken before and after the test showed no significant change in dimensions or ovalness. No cracks were visible or disclosed in dye-penetrant tests. It is planned to log an additional 250 cycles on this ring.

2. Control Rod Lead Screw Development (E. Phillips) - A type 316 stainless steel ball nut lead screw and nut were cycled without lubrication over a one-foot test section to determine the life of a control rod lead screw. A 60-pound weight was hung from the nut to simulate the weight of the boron steel control rod rings. The screw was cycled 1500 times at room temperature and then inspected for damage. No damage to the screw was visible although graphite from a bushing had fallen into the threads.

The screw was again cycled 1500 times at room temperature after which the threads in both screw and nut were Brinelled. Figure 32 shows the end of the test section and the end of the Brinelled thread. The damage was probably accelerated by the foreign matter retained on the threads. Further tests of the ball screw have been deferred until suitable materials for screw and nut are found.

A similar test was run on a one-foot section of an Acme screw and nut made of Rex M₂ tool steel, surface hardened by chromizing, a process which yields a thin case of chromium carbide. The screw and nut were cycled 1500 times in air at room temperature. No lubrication was used and a 60-pound weight was suspended from the nut. After 1500 cycles, none of the chromized case was left on the mating surfaces of the screw and nut and the parent material was badly torn and galled. The cycled screw is shown in Fig. 33.

A chromized Acme screw and nut made of type 1040 steel were cycled 1500 times in an argon atmosphere at 1050 F. The screw was lubricated with MoS₂. After 1500 cycles, the thread in the nut was smoother than before the test but there was no apparent wear on the screw. The screw and nut after test are shown in Fig. 34.

78

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Fig. 32. Brinelled Type 316 Stainless Steel Ball Screw After 3000 Cycles at Room Temperature

79

Fig. 33. Chromized Rex M₂ Steel Screw After 1500 Cycles at Room Temperature

Fig. 34. Chromized Type 1040 Steel Screw and Nut After 1500 Cycles at 1050 F

The Acme screw and nut (1040 steel) were cycled an additional 3350 times. At this point the test rig became very noisy and the operation was stopped. This test was performed in an argon atmosphere at a temperature of 1100 F. Inspection revealed that the threads in the nut were damaged to such an extent that metal had been displaced from the bottom sides of the threads to the top sides. The displacement was great enough to cause physical contact between the top sides of the nut threads and the bottom sides of the screw threads. However, the screw and nut were still in condition to raise and lower the 60-pound weight, but accompanied by a great deal of noise.

Another Acme screw and nut were tested. This time the screw was made of annealed Sichromo (Timken alloy 17-22) and the nut of Rex M₂ tool steel hardened to a Rockwell "C" of 64. After 1430 cycles in an argon atmosphere at a temperature of 1100 F with a 60-pound weight suspended from the nut, the nut was stuck fast to the shaft which was badly galled.

The above tests were performed in the ball nut test rig described in a previous quarterly progress report.⁴

B. SAFETY ROD SYSTEM

1. Latch (A. E. Miller) - A safety rod latch has been fabricated from type 316 stainless steel in accordance with the original latch design. During its assembly, a slight modification was required in the dogs which support the latch plate.

A torque of 7.5 lb-in. is required to release the latch under a load equivalent to the weight of the pull tube and the safety rod rings. Under this load, the latch has been sealed in a helium atmosphere and is being maintained at a temperature of 1100 F for 500 hours. At the end of this time the latch will be examined for "self welding".

2. Torque Tube (D. Whitlock) - A torque tube and its test rig were fabricated during the report period. When rotated, the torque tube releases the safety rod latch. The instrumentation required to measure the angular deflection of the tube as a function of torque and temperature is being installed.

3. Snubber Test (A. E. Miller) - The snubber is a piston and cylinder device which decelerates the safety rod when the rod falls to scram the reactor. A rig for snubber tests has been constructed and is shown before and after release in

Fig. 35. The pull tube and the weight which simulates the load of the safety rod rings can be seen in the figure. When released, the weight falls until it contacts the retaining nut. Then the piston begins to move downward compressing the gas within the cylinder and decelerating the weight and pull tube until they come to rest. The piston, the cylinder, and the pull tube are shown in Fig. 36.

4. Solenoid (E. B. Hecker) - During normal operation, the safety rod solenoid will be continuously energized in an ambient temperature of 105 F. The solenoid must be capable of operating under these conditions without overheating.

Tests to resolve the suitability of the two-coil solenoid have been run with the stroke limited to 5/16 inches under a 26-lb load. The ambient temperature was 105 F. The following results, which are considered satisfactory, were obtained:

- 1) Pull-in - 95 volts, 4.6 amperes
- 2) Hold - 102 volts, 0.050 amperes
- 3) Drop-out - 28 volts, 0.016 amperes
- 4) Temp. rise - 13.4° F

C. INSTRUMENTATION (E. Matlin)

1. Development - A sodium capsule test was performed to decide the effectiveness of using stainless steel sheathed thermocouple material as a sodium leak detector. The test indicated that thermocouple material might be suitable at temperatures above 350 F.

A thermocouple-type sodium level detector test unit has been designed and built. It consists of a stainless steel Calrod assembly with stainless steel encased thermocouples attached at intervals along its length. The Calrod assembly will be inserted in the liquid sodium and raised to a temperature slightly above the sodium temperature. The different voltages between adjacent thermocouples then indicates the position of the liquid-gas interface. Test results are expected shortly.

The inductance-type sodium level detector test unit which used a No. 30 copper conductor in the coil failed at high temperatures. The conductor became embrittled resulting in an open coil. The cause of embrittlement has not been definitely established. Outgassing or contact with the Lavite coil are possible causes. The coil, rewound with No. 30 silver wire for expedience, is being tested again.

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6-7-55 (9693-5542)

Fig. 35. Snubber Test Before and After Release

Fig. 36. Pull Tube, Piston, and Cylinder of Snubber Test Rig.

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2. Production

a. High Bay Area - Sheathed electrical heaters for preheating the reactor structure have been designed. These are to be installed on the bottom of the outer tank assembly. A vent and gas purge system has been designed for the north and south instrumentation pits and associated thimbles. Radioactive gases can now be expelled from these areas. Drawings have been completed for the core tank preheaters. The preheaters are needed for the dry and the wet criticality tests and also for heating sodium in the tank during operational shutdowns. The design arrangement embedding instrumentation wiring in the loading face shield has been completed.

b. Control Room - The design specification for the control console has been released and bids are being received from interested vendors. A control schematic has been made for electrical system design information. Preliminary layouts have been made for all sodium system instrumentation panels, including the two control room panels, the service panel, the main primary panel, the auxiliary primary panel, and the secondary area panel. The preliminary layout has also been made for the inert gas and vent panel.

c. Electrical Power - Design specifications have been released and bids are being accepted for the following items:

- 1) 240 volt and 480 volt distribution, heater, and motor control centers
- 2) Outdoor transformers and primary switchgear
- 3) Emergency electrical power system which includes batteries, motor generator sets, diesel engine-driven alternator, and automatic control
- 4) Main sodium system pump motors and control.

XV. SHIELDING

A. TOP SHIELD (R. L. Ashley)

The design of the top shield has been reviewed in detail and several of recommendations have been made.

The void created by the cerrobend seal next to the rotating plug casing is a weak point in that portion of the shield. This weakness is aggravated by the fact that neutron streaming through the rotating shield plug wall and casing and the 1/4 inch gap between them is expected to cause high radiation levels. It would

be desirable to add shielding above the cerrobend seal. In this case, the steel plate covering the rotatable portion of the top shield should end at the wall of the rotating plug (instead of at the casing for the ring shield) to make the seal accessible. A borated plastic was recommended as the most suitable shielding material to be added above the cerrobend seal.

The voids created by the cutouts for the roller drive assemblies present three serious weak spots in the top shield. It was recommended that 2 inches of steel be added on the sides and bottom of each of the roller drive assembly boxes.

The design of the 40-inch plugs is such that straight paths containing 405 stainless steel and gap, or just 405 stainless steel are present. This type of path may permit excessive neutron streaming. The addition of sleeves to the lower portions of all plugs has only added to the seriousness of this situation.

A solution to this problem would be to add shielding to the top of the reactor; the amounts required must be determined by measurement after the reactor attains full power.

Steel shielding should be added to all plugs in the top shield which now contain a 5-1/4 inch void above the plug.

All concrete plugs penetrating the top shield should contain extra heavy concrete down to the prescribed depth. Extra heavy concrete is required to compensate for the voids created in the top shield by the presence of the electrical wells. The minimum density of this concrete should be $4.44 \text{ gm} \cdot \text{cm}^{-3}$ (277 pounds/ft³).

Neutron streaming through the 1/2 inch gap between the ring shield and the core cavity liner may produce high radiation levels above the cerrobend seal and over the circular trench surrounding the ring shield. To reduce these levels, the depth of the circular trench should be increased, and/or the toluene and electrical lines in the trench should be lowered. This will make it possible to add lead shielding beneath the cover plate for the trench. A borated plastic shield should also be added above the cerrobend seal to reduce neutron streaming up through the gap and the 3/4 inches of steel comprising the ring shield outer liner and core cavity liner.

The location of the coolant wells in the 20 and 40 inch plugs is such that during fuel transfer operations excessive radiation levels will exist. It was

calculated that one inch of lead should be added to the sides and cover plates of the coolant wells in the 20 inch and the outermost 40 inch plug, while 1/2 inch of lead should be added to the sides and one inch to the top cover plates of the coolant wells located in the central 40 inch plugs. As an alternative these wells could be filled with steel or lead shot (the minimum diameter of which should be greater than the gap between the 40 inch plug wall and casing).

Unless the coolant lines in the loading face and ring shield are bent, to prevent gamma ray streaming during fuel transfer operations, it should be specified that all coolant lines must be full during such operations. Were this not the case, very high radiation levels would exist in portions of the reactor room. Several of the electrical conduits including the portions in the ring shield should also be bent to avoid excessive gamma streaming during fuel transfer operations.

In order to further alleviate the problem of neutron streaming through the 1/2-inch gap and 3/4 inches of steel comprising the core cavity and outer ring shield liners, it was recommended that as much of the uppermost portion of the core tank insulation as possible be replaced by graphite.

Regarding the extra heavy concrete, it was recommended that an experimental program be started, the goal of which should be to achieve a minimum concrete density of $4.44 \text{ gm} \cdot \text{cm}^{-3}$ (277 pounds/ft³). Previous to this, all experimental work with heavy concrete has had as its goal achieving a concrete with a minimum density of $3.70 \text{ gm} \cdot \text{cm}^{-3}$ (231 pounds/ft³). This latter concrete was to be used, for the most part, in the portions of the SRE foundations where heavy concrete was required.

B. CONTROL ROD (A. R. Vernon)

Haynes 25 alloy, a 53 w/o cobalt alloy, is being considered for the ball nut lead screw of the control rod assembly. The radiological hazards to be expected have been investigated. The specific problem analyzed was that of the maximum radiation levels expected at the surface of the fuel handling coffin, which will be used to remove or replace the control rod assembly. With the screw located above the reactor core some of us were concerned over the possibility of the induced Co^{60} activity in the screw becoming so large that the shielding of the coffin, which is reduced in thickness in the region to be occupied by the ball screw (above the region normally occupied by the active fuel), would be inadequate. The results of the calculations indicated that, since mechanical life of these screws is not expected

to exceed one year, the maximum radiation level at the surface of the fuel handling coffin would be 0.3 r/hr.

C. THIMBLE SHIELD PLUGS (A. R. Vernon)

The design of the shield plugs for the fission and ion chamber thimbles was analyzed. The effect of neutron streaming was evaluated utilizing recent experimental data on the transmission of neutrons through bent ducts.⁵ The results of the analysis indicated that a two foot thick ordinary concrete plug would provide adequate protection. It was observed, however, that some redesign was in order since the hole for the instrument positioning cables markedly reduced the effectiveness of the shield plug. This was accomplished and a satisfactory design agreed upon.

D. FUEL HANDLING COFFIN (A. R. Vernon, R. L. Ashley)

1. Radiation Levels within Coffin - Radiation levels at some of the more sensitive regions within the fuel handling coffin have been calculated. The calculations were based on a fuel element exposed for 300 days, generating 0.725 mw. Cooling times of both 200 minutes and 24 hours were considered.

Because of the uncertainty in the vertical power distribution in the reactor (control rod shadowing, etc.), the peak radiation levels from the fuel are somewhat in doubt. It was estimated, however, that at 10 cm from the center line of the fuel element, the peak radiation level would be in the range from 1.6×10^6 to 2.4×10^6 r/hr for 200 minute cooling and 0.9×10^6 to 1.4×10^6 r/hr for 24 hour cooling. The integrated dose received during one raising or lowering operation was calculated with somewhat less uncertainty. With the hoist traveling at 6 ft/min., the dose received 10 cm from the center line during one raising or lowering operation would be 2.5×10^4 r and 1.5×10^4 r, for 200 minute and 24 hour cooling, respectively. At a distance of 20 cm, which is more appropriate for the pickup device when traveling opposite the active fuel, the radiation exposure will be approximately one-half those quoted for 10 cm.

After the coffin is loaded, the lower pickup device is raised slightly to plug the opening in the shield at the bottom of the coffin. The upper-most O-ring will then be located approximately 50 cm below the lower end of the active fuel. In this position, the dosage rate will be 7.5×10^4 r/hr after 200 minute cooling or 4.4×10^4 r/hr after 24 hour cooling. The remaining O-rings in this pickup device will be at least partially shielded by the 4-inch lead shield under the fuel element. The

radiation levels will thus be as much as a factor of 100 less than at the top unshielded position. The radiation level at the pickup device located above the active fuel element will be approximately 2.2×10^3 r/hr for 200 minute cooling and 1.3×10^3 r/hr for 24 hour cooling.

2. Fuel Coffin Shielding - The basic shielding dimensions recommended for the fuel handling coffin have been used in its design. However, additional shielding is required in many places to compensate for voids caused by various design details. This additional shielding is needed because of steel structural members which penetrate the bulk lead shielding, passages for helium and vacuum lines, openings for ventilation inlet and of outlet pipes, and the region where lead shot has replaced the solid lead shield. The thickness of lead patches required for several of these regions has been determined.

The bulk shield of the fuel handling coffin is to be built up from lead blocks. If this type of construction is used, it will be difficult to predict the extent to which gaps will disturb the radiation levels expected outside the shield. Since the bulk shielding has been designed to radiation levels which are already quite high (40 times tolerance at a distance of 10 feet), the effect of such gaps cannot be ignored. Thorough testing of the coffin shielding using slightly irradiated fuel or some other equivalent source of radioactivity, is a way of determining the magnitude of this effect. Such a testing procedure is recommended before the coffin is used for handling highly irradiated fuel.

3. Radiation Level of Surface of Coffin - The radiation level at the surface of the fuel handling coffin has been re-evaluated after considering the most recent available information. Making use of a more complete compilation of fission product gamma rays, and taking into account the fact that the central rod in the fuel element will contain only about 10 per cent of the fission product activity, as well as the effect of the peaked power distribution in the vertical direction, it was found that the maximum radiation level (which will occur at the surface of the coffin opposite the lower portion of the fuel element) will range from 1.5 to 2.0 r/hr. At 10 feet, the maximum radiation level will be about 40 times AEC tolerance, rather than 10 times tolerance as had been previously calculated. The maximum radiation level at the fuel coffin operator was calculated to be 20 mr/hr.

E. HOT CELL (R. L. Ashley, A. R. Vernon)

1. Auxiliary Coffin Shielding - Shielding requirements have been determined for the proposed auxiliary coffin which will be used to raise fuel element assemblies 8 feet above their normal position while in the hot cell. The raising operation is required to enable the manipulators to reach the orifice plate at the bottom of the fuel element. The following requirements have been established:

1. The lower 3 feet of the coffin should be shielded by 5 inches of lead.
2. Four inches of lead is required up to a height of 5.25 feet.
3. Above 5.25 feet, 3 inches of lead are required.
4. If only emergency access to the top of the coffin is anticipated, a 5-inch lead shield on the top will be adequate.

More shielding at the base of the coffin will be necessary to protect personnel from radiation which has attenuated through short paths of heavy concrete in the floor. This shielding should be in the form of a lead collar, 5 inches thick, poured around the access plug casings to a depth of 9 inches (in the roof block of the hot cell). An alternative, a lead collar or lead bricks could be placed around the base of the auxiliary coffin, extending 13 inches beyond the outer surface. The height of the collar could vary from one inch at its outer extremity, to 5 inches where it joins the 5 inch thick coffin shield. This shielding is adequate for elevating the fuel assembly no more than 8 feet above its normal position in the hot cell, and for a maximum clearance of 1/4 inch between the OD of the shield plug and the ID of the coffin.

2. Shielding for Hot Cell Doors - An analysis of the hot cell rolling door design indicates that more shielding is required to compensate for the voids created by the wheel cutouts. The following shielding arrangement eliminates excess radiation levels caused by the slant penetration of gamma rays through the wheel cutouts in the hot cell door.

- a. Directly behind the wheel cutouts, either 9.5 inches of steel or 6.3 inches of steel and 2 inches of lead are required.
- b. This shielding should extend 3 inches above the wheel cutouts in the door.
- c. Lead slabs, 2 inches thick, should be placed on the inside of the wheel cutouts and in the portion of the hot cell wall next to the wheel cutouts. The dimensions of this lead should be approximately 2 by 8 by 12 inches.

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Excessive radiation levels might result from the fact that the hot cell door contains only one step and that a gap of from 1/4 to 3/8 inches is expected between the door and the hot cell foundation. It was recommended that the design change of these doors be made to incorporate two steps, dividing the shield into thirds. An alternative would be to place a thick steel slab in the hot cell foundation directly in front of the rear portion of the gap and to place a second slab in the door behind the forward gap. This should be done on both sides, as well as on top of the door.

3. Hot Cell Shielding - The design of the hot cells has been reviewed in detail and a number of recommendations for eliminating weak spots are as follows:

- a. The outlet vent pipe should be moved toward the outside wall of the hot cell.
- b. Two equal steps should be made in the shield block in the ceiling, so as to divide the shield into three equal sections. The gap between this block and the foundation should be reduced to 1/4 inch.
- c. The lower 7-1/2 inches of the fuel access port liner should be replaced by a steel cylinder (with the same ID) with one inch thick walls. The total length of this piece should be equivalent to 12 inches of steel.
- d. The outlet vent for the metallurgical cell should come straight up through the shield. Even with this change, it will be necessary to add shielding to compensate for the 8 inch void of the vent pipe and also for the 12 inch void of the vent pipe plus the vent outlet pans. Four inches of lead placed internally behind the vent pipe, or 8 inches of concrete added to the outside wall will be required in the case of the former, while 6 inches of lead or 12 inches of concrete will be required for the latter. This shielding should be designed so as to shadow the void regions and should extend from floor level to a height of at least 8 feet, to protect operations personnel working at floor level.
- e. The present design requires the fuel element to be located almost directly in front of one of the manipulator ports during fuel loading operations and removal of the orifice plate. Since the manipulator port contains essentially no shielding material, very high radiation levels can be expected in the hot cell operations area. The radiation level at the surface of the manipulator port was calculated to be $< 3 \times 10^3$ r/hr. Scatter from the hot cell operations area wall and "sky shine" will

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F. PRIMARY GALLERY SHIELDING (R. L. Ashley, A. R. Vernon)

The results of work performed during this quarter have shown that the specific activity of the sodium in the primary coolant loop will be $0.32 \text{ curies} \cdot \text{cm}^{-3}$, rather than $0.10 \text{ curies} \cdot \text{cm}^{-3}$, previously calculated. As a result, a re-evaluation of the shielding requirements for the primary sodium system and the primary fill tank was begun. This analysis indicated that the shielding is inadequate with radiation levels of about 4 to 6 mr/hr prevailing over the heat exchanger wing of the main primary gallery. Maintaining the current shield thickness over both the main and auxiliary primary galleries (4.25 feet), the radiation levels as a function of concrete density, for various portions of the primary sodium system, are shown in Fig. 37. In order to reduce the radiation levels to the recommended design value of 0.75 mr/hr (10 per cent of AEC tolerance) it will be necessary either to increase the density of the concrete above that which has been achieved thus far in the test pours, or to add lead or steel below the present shielding. To meet the recommended design tolerance, the density of the concrete over the heat exchanger wing of the main primary gallery should be a minimum of $3.97 \text{ gm} \cdot \text{cm}^{-3}$ (after long aging). The galleries, as now designed, have a uniform shield thickness of 4.25 feet, which corresponds to the original recommendation for the thickness of shielding required over the main primary heat exchanger only.

92

03-10-2008

CONFIDENTIAL

Fig. 37. Radiation Level vs Shield Density

result, the shielding requirements for the narrow wing of the main gallery and the entire auxiliary primary gallery need not change.

With regard to the shielding required over the heat exchanger wing of the main primary gallery, an alternative to the use of a minimum concrete density of $3.97 \text{ gm} \cdot \text{cm}^{-3}$ would be to add steel or lead below the proposed concrete shielding. Using a density of $3.60 \text{ gm} \cdot \text{cm}^{-3}$, it was found that a steel plate 2.5 inches thick, or a lead plate 1.5 inches thick would be required in this portion of the gallery.

Further analysis of the design of the shielding for the primary galleries revealed that streaming of gamma rays through the 1/2 inch gaps between the heavy concrete blocks over the main primary gallery would cause radiation levels of about 150 mr/hr. These blocks contain, in the present design, a single step. It was recommended that two equal steps be made, so as to divide the shielding into thirds. Reduction of the radiation level to 10 per cent of AEC tolerance is not possible in this instance, since the bulk shield itself does not accomplish this objective for any concrete density less than $3.97 \text{ gm} \cdot \text{cm}^{-3}$. By using two steps, the radiation levels will be reduced to a value slightly less than four times tolerance in the worst case, and proper placement of the additional steel or lead shielding which is required in this portion of the main gallery should reduce the maximum level to less than tolerance.

G. SPECIFIC ACTIVITY OF PRIMARY SODIUM (A. R. Vernon)

A detailed re-appraisal of the effect of the increased gap between the moderator cans on the specific activity of the primary sodium coolant was made. A nominal gap width of 0.172 inches, increased from 0.094 inches, was used in this analysis increasing the volume of sodium between the cans from 14.9 ft^3 to 27.3 ft^3 . The results showed that the specific activity had increased to $0.32 \text{ curies} \cdot \text{cm}^{-3}$ (see Table VI), the previous result being $0.10 \text{ curies} \cdot \text{cm}^{-3}$. The major reasons for the increase were the increased gap between the moderator cans and an increase in the estimate of the neutron flux in the reflector region.

H. DISPOSABLE COLD TRAP SYSTEM (R. L. Ashley, A. R. Vernon)

A design analysis of the disposable cold trap system indicates that radiation will pass from the main primary gallery through the wall into the cold trap vault. To reduce this radiation to a tolerable level within the vault, 7 inches of lead shielding or equivalent will be required. Radiation within the vault is also produced

95

TABLE VI
SPECIFIC ACTIVITY OF PRIMARY SODIUM

	Volume of Sodium (ft. ³)	Mass of Sodium (lb.)	Total Cross Section 10 ³ cm ²	Neutron Flux (n/cm ² -sec)	Captures/sec (x 10 ¹⁷)
Moderator can gaps					
Core	5.9	310	1.22	6.2×10^{13}	0.76
Reflector	21.4	1,150	4.42	3.0×10^{13}	1.32
Fuel Channels	7.3	390	1.50	1.16×10^{13}	0.17
Dummy Fuel Channels	2.1	110	.425	1.5×10^{13}	0.06
Above reflector	538.	27,690	-	-	0.32*
Below reflector	174.	9,670	-	-	0.32*
Annulus around reflector	143.	7,650	29.5	5.0×10^{11}	0.15
Piping & heat Exchangers	47.	2,630	-	-	-

TOTALS 939 3.10

$$= 2.66 \times 10^7 \text{ cm}^3$$

$$\text{Specific Activity} = \frac{3.10 \times 10^{17}}{(3.7 \times 10^{10}) (2.66 \times 10^7)}$$

$$= 0.32 \text{ curies} \cdot \text{cm}^{-3}$$

*Calculated using an average leakage thermal neutron current of 3.6×10^{11} n/cm²-sec over an area of 9.0×10^4 cm², assuming 100 per cent capture of thermal neutrons. This does not account for fast leakage neutrons which are thermalized and captured in the sodium. This might increase these values by a maximum of about 10 per cent.

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by sodium oxides collected in the cold trap itself and sometimes by sodium and helium in the primary fill tank vault which is not completely shielded from the cold trap vault. If access to the cold trap vault is required while the reactor is operating, first the contaminated cold trap must be removed and the radiation level within the vault be measured to be sure it is not excessive. In any case, access to the cold trap vault cannot be achieved for at least two weeks after flushing of the gallery cold traps. This waiting period is required to give the sodium activity in the portion of the flushing system located within the cold trap vault time to decay to toleration range. A disposable cold trap system, designed for completely remote operation is recommended so that access to the vault would not normally be required. Shielding the passages between the main primary gallery and the cold trap vault will permit limited access to the vault two weeks after flushing the primary cold traps (while the reactor is operating at full power) and after removal of the contaminated cold trap. More shielding may be needed between the primary fill tank and the cold tank vault.

I. LIQUID RADIOACTIVE WASTE DISPOSAL SYSTEM (R. L. Ashley)

A proposed redesign of the radioactive liquid waste disposal system holdup area has been analyzed. Shielding requirements and radiological safety requirements were determined. The earth fill above the holdup tanks should accordingly be 7 feet, with a similar thickness required between the access tunnel and the bottom of the holdup tanks. The overall utility of this access tunnel was questioned, however, especially since a leak in any of the piping which carries the liquid waste to and from the holdup tanks would cause a serious contamination problem in the tunnel and would, in all probability, prevent access for sampling the contents of the holdup tanks. Some other method of sampling these tanks should be considered.

J. GASEOUS RADIOACTIVE WASTE DISPOSAL SYSTEM (R. L. Ashley)

A review of the layout of the gaseous radioactive waste disposal system has been made. The analysis indicates that the gas decay tanks and the compressor suction tank should be moved about 10 feet east, to prevent excessive radiation levels in the access tunnel located beneath the radioactive liquid holdup tanks. Also, the compressor area should be situated so that a minimum of 5 feet of earth shield-protects personnel on the operations platform and the area should have a minimum of 5 feet of earth shielding above it.

K. SHIELD CONCRETE TESTS (W. A. Hay)

Tests have been conducted to develop concrete suitable for the SRE shield. The test results indicate that a concrete bulk density of $3.075 \text{ gm} \cdot \text{cm}^{-3}$ is obtainable with the magnetite ore acquired for the SRE shielding. One cubic foot forms were filled with machine mixed and vibrated concrete to attain this density. Aggregates and cement were proportioned to yield the densest possible mix. Although the concrete contained only enough water to show a slump of one inch, it was easily placed and compacted using vibratory equipment. Subsequent breakage of two of the concrete blocks showed an excellent internal structure free from voids and segregation.

Results of moisture weight loss tests showed that the use of a cement dispersing agent would result in about 1 per cent lower loss of moisture at elevated temperatures. Compression test cylinders were used for this testing. Samples specifically made for weight loss determination should be fabricated and heated for accurate data.

Tests are presently being conducted at Santa Susana on the intrusion grouting of heavy aggregates to prove concrete proportions and grouting techniques to be used in the SRE Top Shield. Moderate success has been realized so far in the program, and it is expected that the present test program will be successfully completed during the next few weeks.

L. GALLERY SEALS (M. Tarpinian)

Development work has been concentrated on two types of gas seals for gallery shield plugs: (1) a low melting point alloy metal seal, and (2) a neoprene rubber gasket seal.

The basic features of the metal seal are shown in Fig. 38. The cover plate was fabricated from 1/32-inch thick sheet steel with two corrugations down the center and a standing lip 1/4 inch high. The corrugations were rolled into the plate to take up any movement between the components. When in place the plate was 1/4 inch from the walls; thus forming a cross section of 1/4 by 1/4 inch into which Cerrobend alloy was poured. This alloy contains 50 per cent bismuth, 26.7 per cent lead, 13.3 per cent tin, and 10 per cent cadmium. Its low melting point, 158 F, and the fact that it expands upon freezing make it a good sealer. However, to insure good wetting it was found necessary to tin all appropriate surfaces with 50-50 solder. After 4

days at 6 psig and 3 days under vacuum, no leaks were detected. Quick removal was accomplished by chiseling out at the cutaways, as shown in Fig. 38.

The second type of seal, shown in Fig. 39, is simpler in handling and installation. A 3/8-inch thick neoprene gasket 2 inches wide is laid over the gap and clamped by a 1-1/2 by 3/4-inch channel iron. Force is applied to this channel by cross-pieces also made of channel iron notched so as to fit perpendicular to the clamping piece. The force is adjusted by means of studs and nuts.

To date tests have been completed on a straight section with satisfactory results. Under pressures of 12 psig no leaks were detected. Experiments are in progress on corner and cross sections to more closely simulate the situations that will be encountered on the reactor itself.

XVI. REACTOR SERVICES

A. INERT GAS SYSTEMS

1. Design (A. M. Stelle) - Piping and instrumentation diagrams, piping specifications, and descriptions of the inert gas systems have been released.

All helium and nitrogen lines have been relocated to parallel the toluene coolant system lines. The trenches around the reactor have been increased to accomodate both the toluene and the inert gas systems.

The volume of the constant low pressure tank which maintains a pressure of 1/4 psig in the pipe galleries has been set at 900 cubic feet. This will provide expansion space for normal temperature changes within the galleries but will not provide expansion space for temperature changes within the fill tank or disposable cold trap vaults.

The constant pressure compressor suction tank in the vent system has been eliminated in favor of a constant volume variable pressure tank. The pressure will vary from 1/2 psi below atmospheric to 3 psi above atmospheric. This change facilitates shielding and simplifies the manifolding system.

The vent gas decay tank system has been changed to include four 350-cu. ft. tanks in place of the two original 700 cu. ft. tanks. The new tanks are to withstand a pressure of 250 psig and satisfy the requirements for lethal gas containers as prescribed in the ASME Unfired Pressure Vessel Code. In normal operation, the

Fig. 38. Cerrobend Metal Alloy Gallery Seal

Fig. 39. Neoprene Gasket Gallery Seal

pressure in the tanks will not exceed 100 psig. Increasing the number of tanks provides longer holdup times between exhausting periods.

The cleaning cell is vented into the inert gas system through a rotary ballast vacuum pump which provides a pressure of 100 microns of mercury inside the cell. By purging the cell with helium and then evacuating it, the oxygen concentration within the cell can be reduced to less than three parts per million.

The decay tanks, compressor station, and compressor suction tank have been moved to the plateau north of the reactor building. This area is now considered a radioactive waste storage area.

2. Purification of Inert Gas (Bradshaw) - The final phases of the purification experiment emphasizing life tests and entrainment have been completed. As reported in the previous quarterly progress report,⁵ the oxygen content of tank helium was reduced from 6 to 2 parts per million to 0.2 parts per million after passing through the purification train.

The purification tests were run originally with a 1-1/2-inch diameter NaK bubbler. However, due to NaK entrainment, the tests were completed with 4-inch bubblers. Figure 40 represents two runs made on the 1-1/2 inch bubbler, one without check valves and the other with check valves to prevent blow-back of NaK. Using the larger bubblers with flow rates up to about 13 cu.ft/min, the contamination can be kept to 0.2 parts per million. This is true even when using nitrogen containing 2000 to 4000 parts per million of oxygen.

During life tests made with the 4-inch bubblers, about 65 liters of oxygen were absorbed. This corresponds to 475 tanks of helium containing 20 parts per million of oxygen. The limiting factor in the life tests was not the exhaustion of the NaK, but the problem of NaK entrainment which depleted the supply in the bubblers.

Some approximate calculations resulted in the curve of Fig. 41. With the 1-1/2-inch bubbler, no appreciable entrainment takes place at flow rates up to about 1.3 cu.ft/min. With the 4-inch bubbler the corresponding flow rate is about 10 cu.ft/min. It can be assumed, therefore, that the oxygen absorption result mentioned above is a minimal amount and can be increased by at least a factor of two by keeping flow rates below 1.3 cu.ft/min with the 1-1/2-inch bubbler or 10 cu.ft/min with the 4-inch bubbler.

101

Fig. 40. Flow Rate vs Oxygen Concentration, Inert Gas Purification Tests

Fig. 41. Rate of NaK Entrainment in Helium Flow Through 1 1/2 inch ID Bubbler

B. FUEL HANDLING EQUIPMENT

The sodium system piping for the fuel process cylinder has recently been completed. This includes a plugging loop for oxygen determination, a diffusion-type cold trap to minimize oxide concentration in the sodium, an electromagnetic pump for sodium circulation, and a sodium dump tank. Components of this system are shown in Fig. 42, 43, and 44.

The fuel handling coffin has been checked out for proper operation with particular regard to sequence functions and sequence interlock. The interlock region of the coffin with the pickup device extended is shown in Fig. 45. Some difficulty has been experienced with limit switch positioning. These switches have been reworked by replacing the roller arms with phosphor bronze spring leaves and now seem to be satisfactory. The parts for the dummy fuel element have been received and assembled and the completed unit has been handled (dry and cold) in a satisfactory manner by the coffin.

The dry storage cell and the fuel washing cell have been received and installed in the Tower (see Figure 46). Preliminary tests of the water spray nozzles originally selected for the washing showed that these were unsatisfactory. The spray pattern was too coarse and would not drift down the washing cell as originally planned. New spray nozzles were selected by actual comparison of the spray. Since spray nozzles require from 20 to 150 psig water pressure, it was necessary to install a booster pump on the 46-ft deck of the Tower. (Normal water pressure at ground level is about 50 psig.)

Pump down tests were made on the washing cell in the dry clean condition and in the wet condition. A gas ballast pump designed to handle water vapor was used. The dry washing cell requires 50 minutes to attain an absolute pressure of 50 microns. When 250 cm³ of water were added to the clean dry cell, the gas ballast pump required 6-1/2 hours to attain an absolute pressure of 50 microns. Since this is probably less water than would remain after fuel washing, a higher pumping rate for water vapor is needed. For this purpose, a Nash-Hytor pump has been ordered. This pump is capable of handling a much larger volume of water vapor to a higher ultimate pressure. The Nash-Hytor pump will be used for roughing, and the gas ballast pump for finishing the washing cell vacuum.

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Fig. 42. Upper End of Process Cylinder with Coffin Locating Ring in Place

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Fig. 43. Lower End of Process Cylinder,
Sodium Loading Connection, Diffusion Cold Trap, and Sodium Return Line

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Fig. 44. Plugging Loop, Electromagnetic Pump, and Flow Meters

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Fig. 45. Interlock Region of Coffin with Pickup Device Extended

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(9693-5850)

Fig. 46. Dry Storage and Fuel Washing Cells

C. RADIOACTIVE LIQUID WASTE SYSTEM

The hold-up tanks and storage tanks have been moved to the radioactive waste storage area north of the reactor building. This new location is 20 feet higher than the radioactive sump outlet which prevents gravity drainage from the sump outlet to the hold-up tanks. The double walled pipe and other safety features which previously drained leakage to the hold-up tank vault have now been removed because leakage back to the sump is not tolerable.

The cleaning cell feed water has been isolated from the domestic water supply by a metering tank to measure the amount of water used in the fuel cleaning operation. This eliminates the necessity of automatically controlling the water level in the cleaning cell.

XVII. REACTOR OPERATION

A. SYSTEMS ANALYSIS (D. J. Cockeram, C. A. Trilling)

Studies of various transients in the reactor system have been started on an analog computer. To date the effects of a reactor scram, the loss of air flow in the main airblast heat exchanger, the loss of the primary pump, and the loss of the secondary pump on the temperatures in the main intermediate heat exchanger have been investigated. In no case did the measured maximum time rate of change of temperature within the heat exchanger exceed 17°F per second.

A more detailed analysis than that reported previously⁵ has been carried out to determine the cooling characteristics of irradiated fuel elements after removal from the reactor. The more recent results are presented in Table VII.

B. COORDINATION OF EXPERIMENTAL PROGRAM (F. E. Faris)

The review of the experiments to be performed in the reactor has continued. The following special elements are being designed and built in connection with the experimental program.

1. Fuel elements (uranium, uranium alloys, and thorium-uranium alloy) in the form of seven rod clusters with thermocouples at the center of some rods--to be used for determining central fuel element temperatures in connection with materials and physics experiments.

110

TABLE VII

DECAY TIMES REQUIRED TO STAY BELOW GIVEN FUEL ROD TEMPERATURES

	T _{uranium} 1337° F			T _{uranium} 1200° F		
	30	90	360	30	90	360
Irradiation time days						
Location of fuel element after irradiation						
Coffin, hours	1-1/2	2	2-1/2	3-1/2	5-1/2	8
Cleaning cell	1-1/2	2-1/2	3	4	6	8-1/2
Cleaning cell evacuated	8-1/2	12-1/2	19	21	35	60
Hot Cell	2	3	4	5-1/2	8	11-1/2

2. Fuel element in the form of a hollow cylinder - to be used for materials and physics investigations.
3. Fuel element containing less than seven rods--to be used for critical experiments
4. Fuel element with fuel rods easily removable and space for foils--to be used for flux measurements at the time of the dry critical experiments.
5. Thimbles containing nuclear instruments and foils--to be used for critical experiments and for later flux measurements.
6. Dummy control rod segment with thermocouples--to be used at start-up for determining control rod temperatures.
7. Sample plugs for main gallery--to be used for studying mass transfer in the sodium system.
8. Corner channel stainless steel clad element with gas tube--to be used for studying outgassing of graphite.
9. Corner channel zirconium clad elements containing special graphites--to be used for studying the effects of operation on various moderator materials.
10. Special moderator cans with gas tubes and thermocouples--to be used for studying gas evolution, effects of operation on the moderator, and (in conjunction with fuel elements containing thermocouples) temperature coefficient of reactivity.

C. REACTOR PERSONNEL TRAINING (K. H. Campbell, F. E. Faris)

Plans have been completed for a training program to prepare the reactor operators and the maintenance men for their jobs on the Sodium Reactor Experiment. The training program will include the following items:

1. A brief introduction to reactor technology.
2. Operation of the Water Boiler Neutron Source with a discussion of the differences between this reactor and the Sodium Reactor Experiment.
3. A general description of the Sodium Reactor Experiment.
4. Specific information (instrumentation, radiation safety, preoperational testing, operating characteristics, and standard operating procedures) relating to the Sodium Reactor Experiment.
5. Systems and equipment operation under the direction of the shift supervisors during the pre-operational tests.

5. Systems and equipment operation under the direction of the shift supervisors during the pre-operational tests.
6. Training followup by the shift supervisors with the aid of a check list.

Several of the salaried personnel in the Reactor Operations Group have received operations training at the Materials Testing Reactor and the Experimental Breeder Reactor during the past quarter.

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