L

SODIUM GRAPHITE REACTOR
QUARTERLY PROGRESS REPORT
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SUMMARY

This quarter the alternate core configuration studies included the Circular Moderator Can Core (13-rod fuel element), Edge Loaded Hexagonal Moderator Cans and the Gravity Flow Thimble concept. It has been concluded that in the Circular Moderator Can Core concept no advantage in lower steam costs can be realized by decreasing the size of the core. Also, the shorter core lifetimes require higher fabrication costs of fuel and this increased cost would offset the savings in investment. The Edge Loaded Hexagonal Moderator Can concept is still in the development stage. This program includes three phases and to date only the nuclear evaluation of Phase "1" (fixed-rod size, 19 rod fuel element can size varied) has been completed. In the Gravity Flow Thimble concept a study was made for a minimum enrichment core layout. An enrichment very close to natural uranium may be possible with this approach.

Plans for the SRE start-up and low flux experiments are nearly complete. The experiments to be conducted and the order in which they will be carried out are outlined. The program for measurements of SRE construction tolerances is essentially completed. The results were used to re-evaluate the critical mass calculations and it was found that the revised hot-poisoned critical mass is 25.8 fuel clusters. This decrease from a previous value of 28.8 fuel clusters is due to there being less stainless steel inside the safety rod thimbles than was assumed in earlier calculations.

The SRE hot cell windows showed growth of crystals on the interface of the 3.3 density glass. Since the crystals increased with time, moisture was suspected and all windows were purged with dry N_2 gas. The condition is what is known in the glass industry as "weathering". It is believed that the crystals are Na_2CO_3 and K_2CO_3 . One window was removed and a chemical analysis showed the presence of Na, K, Pb and Ce in some complex compound, not identified, but probably carbonate or oxide.

The present fuel irradiation tests in the MTR are aimed at providing knowledge of alloy fuel stability at temperatures and burn-up conditions expected in the SRE. The types and numbers of samples used in the assemblies are outlined in Table I. The large hollow fuel element and all the uranium base alloys (2.78 w/o U^{235}) required for start-up tests of the SRE are being prepared for canning.

The 450 Th-U (5.4 per cent) slugs have been received. During the first inspection of finished slugs (454) at the manufacturer 100 slugs were rejected. The main cause for rejection was a combination of pitting and surface oxidation. The area of difficulty in the manufacturing process is casting good ingots prior to extrusion. In an attempt to get a good ingot surface the following steps will be taken: higher pouring temperature, a pre-heated mold, application of a magnesium-zirconate wash under-coat plus a thorium oxide wash on the graphite crucible, stopper rod and mold.

A program entitled "Advanced Fuel Materials Study" has been started this quarter. With the use of modified equipment (using the "Argonne" pressurized casting technique) numerous cycles of casting of uranium metal into rod shape have been made. A uranium rod (0.750 inch diameter and 14 inches long) was produced; the surface and density were good, however, the roundness was poor. The best cycle conditions have been a pour temperature of 1300° C, a slightly pre-heated Vycor mold, 30 inches of helium and a residence period of 60 seconds in a molten bath prior to removal for cooling. Static casting of slugs 4-3/4 inch diameter by 6 inches long has been made with normal uranium metal; results have been good.

Previous data have shown that grain growth in zirconium has an unfavorable effect on fatigue life. Substantial germinative grain growth was noted in zirconium samples (formed 0.035 inch moderator can side panel) exposed at 975° F in vacuum or at 1000° F in sodium. To determine the maximum aging temperature (not accompanied by undesirable grain growth) zirconium samples were exposed for 800 hours in sodium at 950° F. The absence of significant grain growth in any of the samples exposed suggests 950° F as a limiting useful temperature.

The pump case freeze seal has been extended 2 inches and a third anticonvection ring was added. This modification lowered the cooling requirements from 4.3 kw to 1.4 kw (1000° F sodium temperature). The seal has been satisfactory at a maximum test temperature of 1200° F. A new test section for measuring flow distribution and pressure drop across the SRE Hollow Fuel Element has been incorporated into the existing vertical hydroloop at the Tower Facility. A number of tests were run and the conditions and results are included.

A Mark II Control Rod Prototype was fabricated. In this design the ball nut screw is in the ambient temperature region above the reactor loading face as

compared to below the loading face in the Mark I design. The Mark II Control Rod Prototype has been operated for 10 days under a temperature gradient corresponding to full power operation and 8 days at room temperature.

Tests of the Mark I Safety Rod Prototype were conducted at elevated temperatures. Mechanical difficulties resulting from galling and deformation of materials in the snubber system terminated the tests. The mechanical failures did not interfere with the falling of the poison column when electrical power to the solenoid was interrupted. However, the failure did prevent the recovery of the poison column after it had dropped to the full-in position.

SECTION A

TECHNOLOGY OF THE SODIUM GRAPHITE REACTOR

The material used in this Section was contributed by the following persons:

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J. P. Carlino

R. L. Carter

J. M. Davis

R. L. Eichelberger

F. L. Fillmore

R. C. Gerber

C. Guderjahn

B. R. Hayward

R. B. Hinze

H. E. Johnson

J. E. Mahlmeister

J. J. McClure

W. Morgan

W. S. Peck

J. Stanley

J. Walter

L. Wilkinson

10

I. FULL-SCALE SGR

A. ALTERNATE CORE CONFIGURATION STUDIES (J. E. Mahlmeister, J. J. McClure, W. S. Peck and R. C. Gerber)

1. Circular Moderator Can Core - 13-Rod Fuel Element

A study of the steam costs of various core designs for the 13-rod fuel element and circular stainless-steel can has been completed. Work prior to this study indicated a series of permissible designs. The variation of these designs showed that higher specific power, with shorter core lifetimes and smaller core sizes, were possible with increased enrichment. The conclusions of the study were:

- a) No apparent advantage (in lower steam costs) can be gained by decreasing the core size since the shorter core lifetimes required higher fabrication costs of the fuel. This increased cost offsets the savings in investment cost.
- b) The optimum radius of the core with fuel grade value of plutonium credit is greater than 75 inches. The optimum with current plutonium values is between 65 and 75 inches.

2. Edge Loaded Hexagonal Moderator Cans

A program consisting of nuclear, heat transfer, and economic evaluations of the edge-loaded, scalloped moderator can has been initiated. This study program has three phases:

- 1) For a fixed rod size, 19-rod fuel element the can size is varied.
- 2) For a fixed can size, the fuel rod diameter will be varied for the 19-rod fuel element.
- 3) Investigate the use of more rods per element.

These phases will be reported on as they develop. The nuclear evaluation of Phase "1" has been completed. Some results are shown in Fig. 1, 2 and 3. The smallest core sizes are possible using 17 to 19 inches across the flats. Smaller core sizes require higher enrichment due principally to the larger fraction of resonance capture. This is further shown in Fig. 2 and 3 which indicate the increased conversion ratio associated with the lower value of the product of the resonance escape probability, p , and the thermal utilization.

k Excess = 1.03 Above Hot and Poisoned
 Variable Reflector Savings
 19 Rod Cluster -0.60 in. Slug Diameter
 Stainless Steel Moderator Can -0.018 in. Thickness
 Stainless Steel Process Tube -0.015 in. Thickness

Fig. 1. Core Radius vs Lattice Spacing

k Excess = 1.03 Above Hot and Poisoned
 19 Rod Cluster -0.60 in. Slug Diameter
 Stainless Steel Moderator Can -0.018 in. Thickness
 Stainless Steel Process Tube -0.015 in. Thickness

Fig. 2. Conversion Ratio vs Lattice Spacing

19 Rod Cluster -0.60 in. Slug Diameter
Stainless Steel Moderator Can -0.018 in. Thickness
Stainless Steel Process Tube -0.015 in. Thickness

Fig. 3. p^f vs Lattice Spacing

3. Gravity Flow Thimble Concept

Thermal expansions were calculated and detail design was studied for critical points of the concept. A reactor drawing was prepared showing a typical varying lattice spacing. This study was made in an effort to approach a minimum enrichment core layout. If zirconium proves out as a structural material, it can be used for thimbles and possibly for fuel elements without stainless to zirconium welds in the system. An enrichment very close to natural uranium may be feasible with that approach.

A fuel element was designed for this reactor which will allow the fuel section to be removed from the elaborate upper section of the fuel element. This will allow the re-use of the upper section and can result in a reduction in height of the fuel handling system.

A bottom operated control rod was proposed in this study. The thimble for cooling this control rod has been laid out and the concept included on the reactor drawing.

A glass tube model has been designed to demonstrate the hydraulic principles of the system. The operating media are water and air.

Several complications in the system could be avoided if a small amount of sodium vapor could be allowed in the atmosphere around the graphite core. A review of the available literature on the absorption of sodium vapor in graphite indicates that sodium vapor must be separated from the core atmosphere, except for short intervals of time.

B. PLANT CONTROL (W. Morgan and L. Blue)

Using the general boiler equations previously prepared for various combinations of superheaters, evaporators and economizers, the steady-state variable-load system equations for a typical SGR were prepared. This involves some 36 simultaneous equations with multiple iteration loops. Hand calculations indicated early that the objective of constant reactor temperature over the load range cannot be met using only sodium-flow control. The temperature of the sodium cold leg is influenced principally by the boiler constants, and at low loads the temperature falls well below the desired value. Several methods of alleviating this difficulty were examined, and bypassing part of the feedwater around the

feed preheating zone of the boiler was found to be the most effective. The pre-heat bypass is conveniently done, too, since the control is in the water system rather than the sodium. Hand calculations indicate that reactor temperatures can be maintained constant with this addition to the control means. The system analysis was then programmed for the IBM 704 Computer and the coding checked against the hand calculations. Computer production runs on the problem will begin soon.

An analysis of steam generator transient behavior was begun, and a preliminary set of system equations drawn up.

II. REACTOR PHYSICS

A. SRE START-UP (C. Guderjahn)

Planning and preparation for the SRE start-up and low flux experiments are nearly complete. The following experiments will be done in the order listed.

- 1) Loading of fuel with no sodium present and with the reactor at room temperature. Loading will cease when multiplication is about 35. An estimate of the critical mass under the above conditions will be obtained.
- 2) Same as 1 above but with reactor at 200° C.
- 3) Critical experiment with sodium present at 200° C.
- 4) Control rod calibration.
- 5) Flux mapping.
- 6) Measurement of change of reactivity due to change of sodium flow rate.
- 7) Addition of excess reactivity.
- 8) Control rod calibration.
- 9) Flux mapping.
- 10) Determination of worth of experimental fuel elements by danger coefficient measurements.
- 11) Measurement of steady state temperature coefficient of reactivity.

B. PARTIALLY INSERTED CONTROL ROD (F. L. Fillmore)

The program for measurement of SRE construction tolerances is essentially complete, and the results were used to re-evaluate the critical mass calculations. It was found that the revised hot-poisoned critical mass is 25.8 fuel clusters. This decrease from the value of 28.8 fuel clusters¹ is due to there being less stainless steel inside the safety rod thimbles than was assumed in the earlier calculations. Of the quantities measured, the one which contributed the greatest uncertainty to the critical mass is the thickness of the sodium layer between moderator cans. It is believed that the measurements are accurate to 20 per cent. An increase of 20 per cent in the layer would require an increase in the critical loading of 3.0 fuel clusters.

The critical mass at 180° C with no sodium present was calculated and found to be 13.4 fuel clusters.

The reactivity introduced by removing the source holder was found to be +0.10 dollars.

III. DEVELOPMENT OF HOT CELL FACILITIES AND HANDLING TECHNIQUES

(J. M. Davis)

A. DISPOSAL OF EXAMINED FUEL AND SOLID WASTES

A re-canning device has been designed and constructed so that solid wastes and fuel materials may be canned tightly for disposal. The device pressure-welds a lid on an aluminum can, and is 99 per cent complete.

These cans are approximately 3-1/8 in. OD by 0.065 wall by 7-1/2 in. long. They will hold 7 SRE fuel slugs or an equal volume of solid waste. It is planned that these cans will be packed into larger cans in the primary cell and raised to roof level by an elevator. (The elevator is in the design stage.) The cans are grappled at the end of a short hanger-rod suspending from a shield plug in the fuel-access port. The fuel handling coffin performs the transfer of the plug and can to a transfer cask. This cask is then transported to the fuel and solid waste storage facility.

B. SRE HOT CELL WINDOWS

All of the cell windows showed a growth of crystals on the interfaces between the panes of 3.3 density glass. The crystals were extensive enough to somewhat obstruct vision and photography. Through subsequent observation, the quantity of crystals appeared to increase with time. A moisture-dependent reaction was suspected and all windows were subjected to an immediate purge with dry N_2 gas in an effort to stop the reaction. The manufacturer was contacted in an effort to determine the cause of this growth. Personnel from the manufacturer inspected the windows and suggested that the condition was what is known to the glass industry as "weathering". It is believed the crystals were probably Na_2CO_3 and K_2CO_3 . It was further postulated that the weathering rate is dependent upon the moisture content of the interface environment.

One window was removed from the cell wall and disassembled. Chemical analysis of the crystals showed the presence of Na, K, Pb and Ce in some complex compound, not identified, but probably a carbonate or oxide. The crystals were removed with a damp cloth and the surface was then dried with alcohol. The glass surfaces affected (all but the one pane of 6.2 density glass) showed microscopic (at 15X) pits where the crystals had formed. It has been definitely determined, however, that the pits do not interfere with vision through the glass and the surfaces do not have to be re-polished.

The window will be cleaned and re-assembled under the direct supervision of a manufacturer's engineer. The disposition of the remaining windows will be held up until conclusive results are compiled on the first window.

IV. METALLURGY OF SGR FUELS

(B. R. Hayward, J. Walter and L. Wilkinson)

A. FUEL IRRADIATION TESTS

The present series of fuel irradiations in the MTR is aimed at providing better knowledge of alloy fuel stability at the temperature and burn-up conditions expected in the SRE. The specimens were fabricated in the same manner as

SRE fuel. The conditions of peak temperature (1200° F) and thermal stresses of the SRE fuel are the desired conditions of the specimens being irradiated.

Each assembly contains a series of capsules with a thermocouple extending from the center of each specimen to a continuous temperature recorder.

Table I outlines the type and the number of samples used in the assemblies. All samples are 3/8 in. diameter and 1-1/2 in. long.

TABLE I
TYPE AND NUMBER OF SAMPLES

Assembly	Powder Compacted U-1.2 w/o Mo	Cast U-2 w/o Zr	Alpha Rolled β H/T-U	Extruded Th-U
NAA-15-3	2	2	2	-
NAA-15-4	3	1	1	-
NAA-15-5	-	-	-	6

The irradiation has been completed on NAA-15-3 to an estimated 0.5 per cent burn-up of the total atoms. Before thermocouples were lost, due to manipulation in the reactor tank, the peak temperature attained was 1080° F. The irradiation continued despite the loss of thermocouples. Hot cell examination of these specimens is expected during January 1957.

The NAA-15-4 has been inserted in the MTR and has attained a peak temperature of 1200° F. Thus far 0.1 per cent burn-up has been accomplished. A total burn-up of 0.5 per cent is desired.

The NAA-15-5 is being fabricated and should be inserted about March 11, 1957.

B. EXPERIMENTAL FUEL SLUGS

The centrifugally cast experimental fuel slugs required for the initial tests in the SRE have been gamma-graphed. Some of the slugs exhibited small centerline voids. The gamma-graph data will be recorded and re-examined after the slugs are irradiated to establish any correlation between slug soundness and slug irradiation behavior.

Sample sections are being cut from some of the experimental fuel slugs. These sections will have surfaces prepared for metallography. Photomicrographs will be made and the samples canned with the experimental fuel slugs. It is hoped that these sections will aid in the determination of changes in slug micro-structure during irradiation.

C. CANNING OF EXPERIMENTAL FUELS

The large hollow fuel element is ready to be canned. Successful sample welds joining the element end caps to the inner and outer jacket tubes have been made. Sections taken from these welds showed good weld penetration. Micro-cracking of the jacket tubing adjacent to the weld was observed in some of the welds. Minor adjustments in the welding procedure eliminate this cracking.

Fuel rods containing thermocouples are being canned individually in a single rod loading machine at a rate of about one each week. Eleven of the twelve standard temperature monitored rods needed for the initial fuel loading have been completed. Four rods of thermocoupled experimental fuels are ready for canning.

V. METALLURGY OF BREEDER FUELS

(J. Stanley and B. R. Hayward)

The manufacturer has completed and shipped the 450 Th-U (5.4 per cent) slug commitment. The slugs are now being batched and subsequent tests will be performed prior to reactor performance. Additional development and production

of Th-U enriched slugs will be temporarily suspended pending evaluation of revised proposals by the manufacturer.

The results of our first inspection on finished slugs at the manufacturer, which include an additional 100 slugs, were as follows:

Total Inspected	-	454
Total Rejected	-	100
Slug to Good Slug	-	354
Per Cent Yield	-	78.2 per cent

On further examination of results, it was evident that a combination of pitting and surface oxidation was the main reason for slug rejection. This was attributed to the copper jacket rupturing slightly during the extrusion process. Other contributions to the rejection rate were surface seams and undersize slugs.

A condition that is causing considerable difficulty to the manufacturer is the casting of a good ingot surface prior to extrusion. Scalping of ingots is required to prepare a surface acceptable for extrusion, during which approximately 36 per cent by weight per ingot is removed. This rate is intolerable for large-scale production. It is felt that a higher pouring temperature, a pre-heated mold, and the application of a magnesium-zirconate wash under-coat plus a thorium oxide wash on the graphite crucible, stopper rod, and mold will give an ingot of much better surface.

VI. ADVANCED FUEL MATERIALS STUDY

(B. R. Hayward and J. A. Stanley)

Several months were required to modify the 15 kw M-G set and the furnace to perform melts efficiently up to 1800° C in a reasonable time with a maximum of safety. New induction coils have been wound, molds and casting equipment have been designed, and a storage tank for inert gases has been installed to insure safety during application of pressurized casting techniques.

Numerous melting and casting cycles of uranium metal into rod shape have been made with modified equipment using the "Argonne" pressurized casting technique. After several attempts, a uranium rod of 0.750 in. diameter by

14 in. long was produced (Fig. 4). The surface and density were good but the roundness was poor. A 4-in. long slug is shown (Fig. 5) after approximately 0.040 in. removal by machining. The clean-off was necessary because of the ovality of rod and not the surface condition. This rod was allowed to solidify in a Vycor tube in the cooled uranium bath in the melting crucible. Additional castings have been made with the Vycor tube mold ejected from the molten bath after a predetermined period. The best cycle conditions to date have been a pour temperature of 1300° C, a slightly preheated Vycor mold, 30 in. of helium and a residence period of 60 seconds in the molten bath prior to ejection for cooling. Regardless of the Vycor tube design, the problem is one of temperature, since Vycor readily slumps at 1300° C. It appears that this method is best applicable to casting below 1300° C temperature. Preliminary results with graphite tubes replacing the Vycor tubes were unsatisfactory. Several other mold materials are being considered for subsequent runs.

Several static castings have been made of hollow cylinders of normal uranium (Fig. 6). The slug shown is 6 in. long by 2-3/8 in. diameter OD 1-3/8 in. diameter ID. The surface was machined to remove the slight imperfections using a 0.040 in. cut (0.080 in. on diameter). About 3 inches of the cylinder was sectioned to observe shrinkage pipe but none was visually detected. Static casting of 4-3/4 in. diameter by 6 in. long slugs have been made with normal uranium metal with good results.

The development work on the melting of thorium and its alloys has been slow due to the lack of proper induction coil design. However, a coil has been recently designed and two thorium melts have been successfully made to 1800° C. Additional castings of the thorium are pending acquisition of a modified distributor head and crucibles from the graphite shop.

VII. ZIRCONIUM BEHAVIOR IN LIQUID SODIUM

(J. C. Bokros, J. P. Carlino, R. L. Carter, R. L. Eichelberger,
R. B. Hinze and H. E. Johnson)

A. HOT TRAP EXPERIMENT

An illustration showing the planned modification of the hot-trapped dynamic loop was included in the last Quarterly Progress Report². Installation of a hot

Fig. 4. Pressure Cast Slug (3/4 by 14 inches)

9304-5101A

24

9304-4706B

Fig. 5. Four-inch Machined Slug (0.040 inches removed)

25

9304-5102A

Fig. 6. Static Cast Hollow Slug (2-3/8 inch OD by 1-3/8 inch ID)

trap, containing sheet zirconium gettering material, and addition of a diffusion cold trap was completed, and loop operation was resumed. During start-up the cold trap was not operated, its temperature being maintained about 100° F above the adjacent sodium stream temperature.

After equilibration at 600° F, sodium in the loop was found to precipitate impurities in the plugging indicator at 463° F. This temperature was within the range of the charge tank filter temperatures during initial filling and subsequent level adjustments. After the hot trap had been operated a total of five days at 900° F, the plugging temperature was 480° F. This corroborates the absence of a decrease in plugging temperature previously observed during operation of the zirconium-titanium hot trap. However, analysis for oxygen in zirconium samples exposed at that time indicated an oxygen pick-up rate corresponding to an oxygen concentration in the sodium well below 10 ppm. It is clear that a plugging indication taken in a hot-trapped sodium system is not necessarily indicative of oxygen concentration.

B. GERMINATIVE GRAIN GROWTH IN ZIRCONIUM

Previously reported data have shown that grain growth in zirconium has an unfavorable effect on fatigue life. Zirconium exposed either at 975° F in vacuum, or at 1000° F in sodium showed substantial germinative grain growth in samples of formed 0.035 in. moderator can side panel. To determine the maximum aging temperature not accompanied by undesirable grain growth, zirconium samples from the following SRE components and material were exposed for 800 hours in sodium at 950° F: a formed side panel, a 0.100 in. as-rolled can head plate, a central tube with weld, and vapor blasted, chemically polished 0.035 in. flat stock. The absence of significant grain growth in any of the above materials suggests 950° F as a limiting useful temperature.

SECTION B
SODIUM REACTOR EXPERIMENT

The material used in this Section was contributed by the following persons:

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D. H. Johnson

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A. E. Miller

M. E. Nathan

E. C. Phillips

H. Schroeder

H. Strahl

J. E. Tate

VIII. LAND, UTILITIES AND BUILDINGS

(J. C. Cochran)

The installation continued and is now estimated as 99.3 per cent completed. The present status of overall construction can be seen in Fig. 7 and 8. During this period the moderator cans were loaded into the core tank (Fig. 9). Minor adjustments are being made to the installed cans. Additional coolant coils were installed in the loading face plug and in the two 40-inch and 20-inch plugs which are part of the top shield of the reactor. This modification was required by the discovery of several leaks in the cooling coils in the area where they are embedded in the lead shielding. The type of coils added are shown in Fig. 10 and 11. The loading face was filled with heavy concrete and has been installed. The installation of instrumentation wiring is in the process of completion (Fig. 12).

The fuel handling coffin was completed and turned over to the operating personnel (Fig. 13). However, certain modifications in the hoist assembly are being made to improve its performance.

The control room and the electrical distribution room, the tetralin system, the nitrogen system, the helium system and the vent system, have been turned over to the operating personnel.

The installation of underground tanks and of piping in the radioactive waste area has been completed (Fig. 14). The remaining work, completing the installation in the compressor vaults, is in progress.

The concrete closure plugs for the galleries and vaults were put in place (Fig. 15).

The zirconium foil of 4 mil thickness to be used as a gettering material in the hot traps was fabricated from scrap zirconium (Fig. 16).

IX. FUEL ELEMENTS

A. WIRE WRAP (S. Elchyshyn)

Preparations were initiated to wire wrap all of the outer fuel rods for the standard SRE fuel elements. The 10-inch pitch was adjusted to 10.3 inches to cause coincidence of the circumferential position of the starting and stopping

30

30

9693-1030

Fig. 7. Building and Site Construction Progress (December 1956)

9693-1029

Fig. 8. Building Construction Progress (December 1956)

9693-57150F

Fig. 9. Moderator Can Being Lowered into Core Tank

Fig. 10. Cooling Coil Modification, 20-inch Plug
(The camera was located approximately 3 feet from loading face surface)

9693-56246D

34

Fig. 11. Cooling Coil Modification, Outer 40-inch Plug
(The camera was located at the loading face surface)

9693-56245A

35

9693-56257

Fig. 12. Instrumentation Wiring on Loading Face

9693-58208A

Fig. 13. Fuel Handling Coffin

Fig. 14. Radioactive Waste Area

9693-58218

38

9693-56238D

Fig. 15. Concrete Closure Plugs being Lowered in Place

Fig. 16. Scrap Zirconium to be Used as Gettering Material in Hot Trap 9693-54470B

point of the ends of the wire wrap. This is desirable to prevent a wire wrap from crossing the weld region near the outside edge of a fuel rod. The resulting slight increase in overall diameter of the cluster might cause difficulty during cluster insertion into a coolant channel.

Apparatus required to wire wrap the fuel rods was fabricated and tested. This included a positioning jig to space the wire and a tension device to pull the wire taut against the tubing.

B. THERMOCOUPLED RODS (H. Strahl)

Assembly was completed of the 6 thermocoupled rods to be distributed among the standard clusters in the initial loading for SRE. In addition, 5 of another 6 thermocoupled rods requested for use as spares, were also assembled. All thermocoupled rods were assembled using the Single-Rod Loader.

C. HOLLOW FUEL ELEMENTS (C. Kuhnhofer and R. Kennedy)

Modification of the Standard Production Rod Loader was performed to adapt the apparatus for loading the SRE Hollow Fuel Elements. A new weld head was fabricated and assembled with different gear trains to allow welding of both the inside and outside tubing to the end caps. Plates for the rack were fabricated to fit the outside tube of the hollow element. Practice welds were performed to establish optimum conditions. The initial welds obtained were not leak tight. After sectioning, microscopic examination revealed intergranular cracking of the area just below the weld. Attempts to eliminate this cracking by adjustment of the electrode spacing and welding current are now in progress.

A special holder and container for electropolishing the hollow slugs were fabricated. A fixture to suspend the hollow slugs in the out-gas rig was fabricated and heated under vacuum to remove all traces of oil and other impurities. Cyclograph coils to test the tubing were fabricated and tests were performed to establish the number of turns of wire on the coil required to exhibit the most sensitive response to surface defects in the tubing. Outer and inner tubes on hand were then inspected.

X. HEAT TRANSFER

A. AUXILIARY PUMP TEST (H. Schroeder)

The pump case freeze seal has been extended 2 inches and a third anticonvection ring added. This new configuration results in lowering the cooling requirement from 4.3 kw to 1.4 kw (at 1000° F sodium temperature). Performance of this seal has been satisfactory at the maximum test temperature of 1200° F.

An SRE type sodium syphon was installed in this auxiliary secondary pump to determine its feasibility in removing sodium following loss of the freeze seal. A test of this sodium syphon system was made in conjunction with the case freeze seal test. A Calrod type immersion heater was inserted in a 1/2-inch tube and the assembly placed in the pump. Coolant was shut off at the shaft seal and sodium allowed to leak to a point 2 inches above the shaft seal. Coolant flow to the seal was restored, the case was pressurized to 5 psi and the heaters turned on. With the heaters running at half their rated voltage it required 50 minutes to initiate sodium flow through the tube and an additional 10 minutes to recover 8 pounds of sodium.

After disassembly only a trace (less than 2 ounces) of sodium remained in the pump cavity. This indicates that pump operation can be continued after syphoning without the need for disassembly.

B. HYDRAULIC EXPERIMENTS FOR SRE HOLLOW FUEL ELEMENT (C. Davidson and R. Hargrove)

Measurements of the flow distribution and pressure drop across an SRE hollow fuel element have been made. A new test section for this purpose was incorporated into the existing vertical hydroloop at the Santa Susana Tower Facility. Figure 17 is an external view of the hydroloop while Fig. 18 is a longitudinal view of the fuel element. Figure 19 is a view of the downstream or top end of the fuel element. Figure 20 shows the upstream end and the central channel orifice plate which distributes the coolant between the annular and central channel regions. At the bottom spear point, the orifice plate and the spacer are identical with those on the seven rod fuel element; as their pressure drop and flow characteristics have previously been determined³, they were not incorporated into the present experiment.

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Fig. 17. Vertical Hydroloop, Santa Susana Tower Facility

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Fig. 18. Fuel Element (Longitudinal View)

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Fig. 19. Down-Stream (or Top) End of Fuel Element

9693-52259D

Fig. 20. Up-stream End of Fuel Element

The flow distribution was determined for central channel orifice diameters of 0.450, 0.650 and 0.850 inches over a range of Reynolds Number of from 2×10^4 to 8×10^4 . The flow distribution was determined by circulating water at 180° F through the loop at a flow rate to give the desired Reynolds Number. The flow velocity through the central channel was determined by injecting a burst of brine solution into the central channel at a recorded reference time and observing the time taken for the brine to reach a conductivity cell (see Fig. 21). The output of the conductivity cell is recorded on one channel of a dual channel recorder while the other channel is used to indicate the opening of the solenoid valve to admit the brine solution.

This method of measuring flow velocities was checked against a standard orifice plate flowmeter and was found to agree within ± 2 per cent. Figure 22 shows the annular flow rate vs total flow rate for varying orifice diameters.

The pressure drop across the fuel element was determined at flow rates of 20 to 60 gpm. The maximum total pressure drop, for 60 gpm, varied from 2.94 psi with the 0.850 inch diameter central channel orifice to 6.42 psi for the 0.450 inch diameter orifice.

C. COLD TRAP EXPERIMENT (M. E. Nathan)

1. Methods

In order to ascertain the trapping characteristics of the prototype SRE mesh-packed cold trap assembly, three methods of gaseous oxide addition were tried:

- a) Bubbling argon-diluted oxygen gas directly into sodium by means of a sparger.
- b) Adding argon-diluted oxygen gas directly into sodium by means of a system of concentric nozzles. (Oxygen was passed through the inner nozzle and blanketed by argon flowing in the annulus formed by the concentric nozzles.)
- c) Adding argon-diluted oxygen gas to the cover gas above the free surface of sodium.

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Fig. 21. Conductivity Cells used to Record Flow Velocity through the Central Channel

Fig. 22. Flow Total vs Flow Annulus

Difficulties were encountered with each of the above methods. The occurrence of sodium oxide plugging in the gas inlet line could not be avoided even though high argon to oxygen volume ratios were utilized. However, using the last method, several periods of oxygen addition long enough for equilibrium runs were realized. The data obtained during these equilibrium runs are presented in Table II. Analysis of Table II indicates that although the addition of oxygen gas caused an increase in the oxygen concentration of the cold trap inlet stream, essentially no increase in oxygen concentration was found in the cold trap exit stream. This latter conclusion may be noted by the continued approximate agreement between observed plugging temperature of the cold trap exit stream and minimum cold trap internal sodium temperature before and during oxygen additions to the test loop. Table II indicates that for the range of variables investigated, the prototype cold trap was essentially 100 per cent efficient in:

- 1) Effecting complete precipitation of sodium oxide in accordance with the solubility curve for oxygen (as sodium oxide) in sodium.
- 2) Retaining all of the sodium oxide precipitated.

Following the runs reported in Table II, an attempt to obtain trapping data at still higher rates of oxygen addition resulted in complete plugging of the cold trap circuit. Elevating the temperatures above 1000° F and applying a pressure differential across the cold trap circuit were unsuccessful in clearing the circuit.

In the belief that the plug was located in the cold trap economizer, this component was cut out of the loop and the loop piping was revised to permit operation without the economizer. However, upon reloading the loop with sodium, attempts to re-establish flow through the cold trap failed. The experiment was concluded at this point since sufficient data had been accumulated to predict SRE cold trap performance.

Following the conclusion of the experiment, the stainless mesh packing was withdrawn from the cold trap for observation. Figure 23 shows three sections of annulus mesh after their withdrawal from the prototype cold trap. The section shown in the foreground of the photograph was located in the bottom third of the annulus. (The sodium flow enters the cold trap at the bottom and passes up through an annulus and returns through a central downcomer so that the sodium flow also exits at the bottom of the trap.) The photograph indicates that the bulk of the oxide precipitation occurring in the cold trap throughout the experimental program took place in the initial eight or nine inches of the cold trap annulus.

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TABLE II

TRAPPING DATA FOR EQUILIBRIUM PERIODS OF OXYGEN ADDITION
WHILE MAINTAINING COLD TRAP SODIUM FLOW = 1350 LB/HR

Oxygen Gas Addition Rate	0		1.0 scfh*		1.0 scfh*		1.4 scfh	
	Inlet	Exit	Inlet	Exit	Inlet	Exit	Inlet	Exit
1. Cold Trap Stream Sampled by Plugging Meter								
2. Cold Trap Minimum Sodium Temperature	342° F	336° F	338° F	333° F	342° F	337° F	330° F	332° F
3. Plugging Temperature Observed	337° F	329° F	411° F	323° F	445° F	346° F	415° F	328° F

Note: * A portion of the oxygen added was continually vented through a vapor trap along with the unreacting argon.

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**Fig. 23. Stainless Steel Mesh Packing, after Removal from
Prototype Cold Trap**

2. Results

a) Final Equilibrium Oxygen Concentrations

Table III indicates the final equilibrium oxygen concentrations obtained in the test loop. Resultant concentrations were obtained by operation of the prototype cold trap at various minimum sodium temperatures during cleanup of the initial loop sodium inventory, prior to oxygen additions. It may be seen that the oxide concentration is reduced to approximately the saturation concentration at the cold trap temperature.

b) Initial Oxygen Cleanup

At the conclusion of the loop initial oxygen cleanup, the sodium oxide which has been deposited in the cold trap was readily dissolved by passing 600° F sodium through the trap. This demonstrated the possibility of cleaning an "in place" cold trap by means of transferring the accumulated oxide to a disposable cold trap.

c) Pressure Drop Across Clean Cold Trap

Sodium pressure instruments located at the tube inlet and shell outlet of the cold trap economizer indicated that the pressure drop across a clean cold trap circuit at design flow (1350 lb/hr sodium) was less than 0.5 psi.

D. RECOMMENDED CHANGES RESULTING FROM COLD TRAP EXPERIMENT

1. Changes in Sodium Flows

Sodium flows in the SRE cold traps should initially be at least 3 times the design magnitude of 1350 lb/hr so that high levels of liquid toluene in the cold trap jackets can be utilized. Precipitation of oxide will then occur over a greater annular distance.

2. Heater

Electrical heat should be applied continuously to the cold trap economizers during the period required for initial cleanup of the sodium inventory in the various reactor loops.

3. Oxides

Collected oxide should be transferred to a disposable cold trap immediately after a particular trap is valved out of the reactor system.

TABLE III

CLEANUP OF INITIAL OXYGEN CONTENT

Plateau	Minimum Na Temperature in Cold Trap	Plugging Determination of Loop Circulating Sodium	Equivalent Oxygen Concentration *
1st	600° F	500° F	80 ppm
2nd	402° F	394° F	32 ppm
3rd	389° F	390° F	30 ppm
4th	370° F	368° F	24 ppm
5th	361° F	369° F	24 ppm
6th	312° F	298° F	11 ppm
7th	301° F	300° F	11 ppm **
8th	290° F	300° F	11 ppm **

* The KAPL curve for saturation concentrations of oxygen in liquid sodium is utilized. (KAPL-Memo ILG-7)

** A 20-hole plug utilizing 0.040 inch diameter holes was used in the plugging meter in this phase of the test program and was not sufficiently sensitive to give reliable plugging indications at very low oxygen concentrations. Following this phase of the test program, a 9-hole plug also utilizing 0.040 inch holes was inserted and used throughout the remainder of the experiment.

XI. INSTRUMENTATION AND CONTROL

A. CONTROL RODS (A. E. Miller and J. E. Tate)

1. Mark I Production Rods

Two Mark I Control Rods, which will be used as shim rods in the SRE, were fabricated and functionally tested. The functional test consisted of cycling the poison column, up and down, 10 times under a temperature gradient corresponding to full power operation. During this test the rods were maintained under this temperature gradient for a period of six days. These two Mark I rods are now ready for installation in the SRE.

2. Mark II Control Rod Prototype

During this period a prototype of the alternate, or Mark II, design was fabricated. In this design the ball-nut screw is in the ambient temperature region above the reactor loading face as compared to below the loading face in the original, or Mark I, design. This was accomplished by placing the screw in a separate, but easily removable, support column mounted on the loading face. The poison column is connected to the screw by means of four rods which pass through the shield section. The Mark II rods are to be used as SRE regulating rods.

The Mark II prototype has been operated for a total of 10 days under a temperature gradient corresponding to full power operation, as well as 8 days at room temperature. Some wear has been observed on the four support rods which connect the screw to the poison column. This wear does not appear to affect adversely the operation of the rod. Testing of this prototype is continuing.

B. SAFETY ROD SYSTEMS (E. C. Phillips and W. A. Davis)

1. Mark I Latch Test

The aluminum oxide flame-plated latch was tested after soaking 500 hours at 1100° F in helium. The torque required to release the latch was between 2-1/2 and 5 pound-inches. This was the same torque as measured before heating the latch indicating that self-welding did not occur during this soaking period.

2. Mark I Safety Rod Prototype

Additional tests of the Mark I safety rod prototype were conducted at elevated temperatures. All tests were terminated due to mechanical difficulties resulting from galling and deformation of materials in the snubber system. In no instance did these failures interfere with the falling of the poison column when electrical power to the solenoid was interrupted. The failures did, however, prevent the recovery of the poison column after it had dropped to the full-in position.

A snubber cylinder of Stoddy No. 6 (a high temperature cobalt alloy) and a tungsten carbide coated pull tube were then fabricated in an attempt to prevent material failures. This snubber system combination has operated satisfactorily during the small number of tests to which it has been subjected to date.

XII. REACTOR SERVICES

A. PREOPERATIONAL TESTING (D. H. Johnson)

Checkout of the reactor control room and electrical distribution room were conducted, this also included the calibration of instruments. Tests of the reactor control room have been 90 per cent completed and tests of the electrical distribution room have been 60 per cent completed. The remainder of the work in these areas is pending release of instrumentation in other areas to permit completion of tie-ins.

Tests were started on the helium, nitrogen and parts of the ventilation systems. The most apparent leaks from the insulation and thermal barrier cavities (through conduits containing thermocouples and heater leads) were sealed with epon. A detailed study of small leaks is being conducted by volume displacement tests.

Heater tests were performed on the sodium primary and secondary fill tanks.

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