NAA-SR-3456
COPY MASTER

SRE EXPERIMENTAL FUEL PROGRAM

(INTERIM REPORT)

AEC Research and Development Report

ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.

LEGAL NOTICE

Price $1.25.
Available from the Office of Technical Services
Department of Commerce
Washington 25, D. C.

NAA-SR-3456
METALLURGY AND CERAMICS
45 PAGES

SRE EXPERIMENTAL FUEL PROGRAM

(INTERIM REPORT)

By
B. R. HAYWARD
J. H. WALTER

ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.
P.O. BOX 309 CANOGA PARK, CALIFORNIA

CONTRACT: AT(11-1)-GEN-8
ISSUED:

OCT 15 1959

DISTRIBUTION

This report has been distributed according to the category "Metallurgy and Ceramics" as given in "Standard Distribution Lists for Unclassified Scientific and Technical Reports" TID-4500 (14th Ed.), October 1, 1958. A total of 605 copies was printed.

CONTENTS

TABLES

FIGURES

ABSTRACT

A program has been set up to develop and evaluate suitable fuels for Sodium Graphite Reactors. The method being used is to irradiate various fuel materials in full-size element designs, in the SRE, under measured temperatures and known reactor conditions. To date, uranium, dilute uranium alloys, thorium-uranium alloys, and UO_2 have been fabricated, assembled into fuel elements, and loaded into the SRE. A schedule has been established for these fuels to be evaluated in the SRE hot cell after irradiation. New fuel materials are being investigated and will be tested in a similar manner.

I. INTRODUCTION

Sodium Graphite Reactors (SGR) offer the potential advantages of high coolant temperatures, high heat-transfer rates, reasonable neutron economy, and low cost electrical power. With the building of the Sodium Graphite Reactor Experiment (SRE),[1] a program was established to provide a suitable fuel for this reactor. The requirements of fuel for this system are principally: a) to be capable of long burnup at high surface and center temperatures, b) to be low in cost, c) to be amenable to economical reprocessing. The SRE experimental fuel program is set up to develop both metallic and ceramic fuels for Sodium Graphite Reactors. The operating limitations of burnup and temperature of various fuels in the SRE will be established. Included in the development program is the evaluation of the overall fuel element design. The end result will also include improved specifications for fuels with lower procurement cost, and longer life. The information developed in the SRE will be useful to the physical metallurgist, ceramist, and solid state physicist, in developing and substantiating theories of irradiated fuel behavior.

The primary problem with metallic fuels operating at high surface temperatures, is swelling. This dimensional change results in a decreased density. The phenomenon has not yet been fully explored, but the principal factors believed to affect swelling in metal fuels are:

a) Center temperature.

b) Surface temperature.

c) Temperature difference (surface to center).

d) Total burnup.

e) Burnup rate.

f) Properties and structure of the fuel material.

g) Thermal cycling.

h) Phase change kinetics.

It is only recently that an attempt has been made to irradiate fuels at surface temperatures over 700°F. The problem of swelling increases as the surface

temperature increases. To date, only a few widely scattered experimental data are available, on the problem of swelling.[2,3,4] From this information, it appears that each fuel material has a threshold temperature, or temperature range, beyond which swelling increases at a rapid rate. It is not known whether the surface or central temperature (or both) is the controlling factor. This temperature range varies beyond 660 and 1200°F and depends on factors such as burnup rate, fuel composition and structure, and geometry. It does not presently appear feasible to extrapolate irradiation data, from plate to cylindrical geomety. Similarly, extrapolation from small cylinders to large cylinders, is not recommended. Each fuel design, and each fuel material, must be evaluated under the intended operating conditions in order to predict their expected performance.

The primary difficulties experienced with ceramic base fuels are low thermal conductivity and low uranium atom density. These limitations increase the size of the core with proportionately higher fuel costs. Additional ceramic fuel problems that require investigation are: a) the amount of fission gas release, and b) the extent of thermal ratcheting between the fuel and the cladding. As with metal base fuels, the only feasible method of establishing limits of burnup, heat transfer rates, and temperature, is to irradiate full-scale fuel elements under design conditions.

The method being followed in accomplishing the goals of the SRE experimental fuel program is as follows:

a) Various alloys of metallic fuels are selected for test purposes on the basis of thermal cycling, physical and mechanical property evaluation.

b) Samples of promising fuels are irradiated in the MTR under temperature-monitored conditions.

c) Fuel element designs, suitable for testing experimental fuels in the SRE, are developed.

d) Fabrication techniques are developed for promising fuels, and all processing and quality variations are carefully noted.

e) Fuel elements are assembled, and irradiated in the SRE under measured temperature conditions.

f) Fuel materials, and fuel element designs, are evaluated in the SRE hot cell after irradiation. Examination includes gross effects such as dimensional changes, density changes, fission gas release, metallurgical changes in structure, and physical properties.

g) The optimum fuel (or fuels) for use in SGR reactors, are selected. The selection includes the establishment of temperature and burnup operating limits and the material specifications required to produce the selected fuel.

h) The economics of the entire fuel cycle are evaluated.

This report is restricted to the fabrication, irradiation conditions, and irradiation damage of fuel materials. Studies of the nuclear, heat transfer, and flow characteristics of fuel elements, have been carried out under other projects.[1]

The first round of experimental fuels has been fabricated and inserted in the SRE. Irradiation schedules and examination procedures have been established. The flexibility of the program permits the selection of new fuels, and the development of improved fabrication techniques. This report covers the results achieved to date and outlines plans for future work.

II. OBJECTIVES OF THE SRE EXPERIMENTAL FUEL PROGRAM

The objective of this program is to develop an economical fuel for SGR reactors. To be economically feasible, the fuel must be capable of: a) long burnup, b) a low fabrication cost, and c) low cost reprocessing. In order to utilize the Na coolant in SGRs properly, the fuel must operate at high surface and center temperatures, and at high surface heat fluxes. The fuel should have a high uranium density and a low neutron absorption cross section. The operating limitations of fuel burnup and fuel temperature can be established only by irradiating full-sized fuel elements under actual design conditions.

To achieve the objective of an economical fuel, all fuel variables starting with fabrication, and continuing through the entire irradiation, must be investigated. Close attention to fabrication details, and material quality, is necessary if future specifications and predictions of irradiation behavior are to be realistic. This program encourages the development of improved low cost fabrication procedures and handling techniques.

The final objective of this program is the selection of the most suitable fuel materials and fuel element designs, for Sodium Graphite Reactors. The limiting values of fuel burnup, fuel temperature, and rate of fuel power output, will be established under actual SGR conditions. Realistic specifications for fuel procurement will be established. These specifications will reflect the effect of fabrication variables, chemical compositions, manufacturing tolerances, and defects on irradiation behavior, to allow fabrication of minimum cost fuel elements.

III. SRE IRRADIATION CONDITIONS[1]

The SRE loading presently consists of 43 fuel elements. With the reactor operating at 21 Mw thermal, the average neutron flux in the fuel is approximately 1.3×10^{13} thermal and 1×10^{13} fast. The maximum power per channel is approximately 530 kw. The Na inlet temperature is 525°F and the mixed mean Na outlet temperature is 900°F. The individual fuel channel outlet temperatures vary due to differences in Na flow and channel power output. The peak sodium temperature is limited to about 960°F by germinative grain growth and associated loss of structural properties in the zirconium moderator cladding.[5]

Temperature profiles for SRE metal fuel rods with a 43 element loading, and no allowance for hot channel factors, have been calculated on the basis of a 1.3 to 1 peak-to-average vertical flux distribution. The calculated temperature profile for a fuel rod operating with a 525°F Na inlet and a 840°F Na outlet in the central fuel channel of the SRE, is given in Figure 1. The maximum temperature in Figure 1 is about 1010°F, which is appreciably less than the fuel temperature of 1200°F originally calculated for a 37 element loading with allowance for hot channel factors.

Control rods, burnup of the fuel, and the reactor fuel loading, will cause deviations from the calculated flux distribution. Temperature profiles based on actual flux distribution and measured fuel temperatures will be established for each experimental fuel element. The actual flux distribution will be determined from burnup profiles. It should be noted that both surface and center temperatures cover a wide range. Irradiation results, representing this spread in temperature, will be useful in evaluating important parameters to design a metal fuel element for operation at high temperatures.

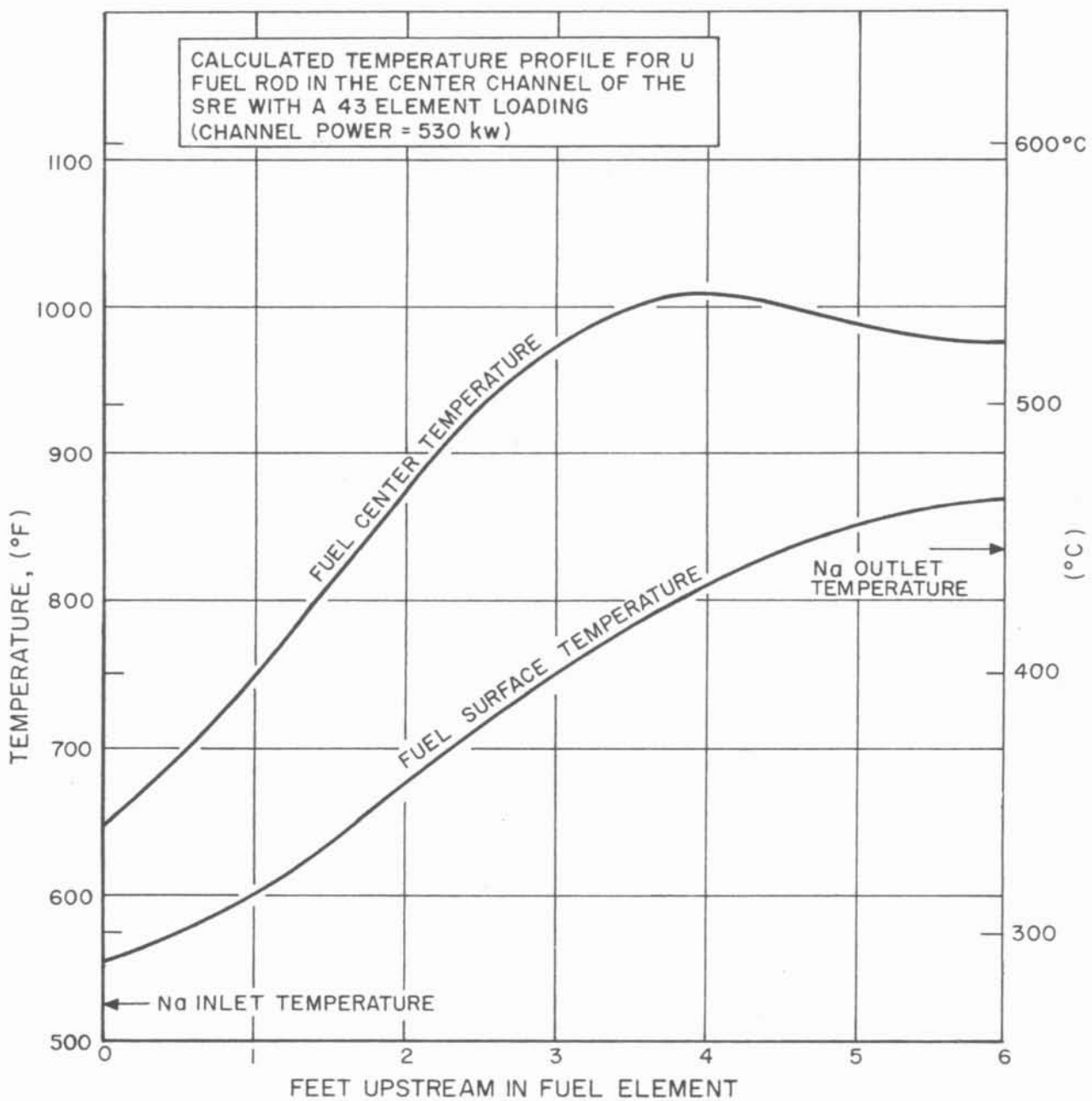


Figure 1. Temperature Profile of a Standard SRE Fuel Rod

IV. SELECTION OF FUELS

Due to the high operating temperatures (surface and central) employed in the SRE, any fuel, including the original core loading of unalloyed uranium, must be classified as experimental. Irradiation growth and swelling are the greatest limiting factors on the life of metal fuels. Irradiation growth, which is due to the anisotropic properties of alpha uranium, causes a change in shape with only small density changes. Irradiation swelling, which is thought to be caused by the nucleation and growth of small bubbles of fission gas, causes gross density decreases in the fuel.

A. SRE FIRST CORE LOADING

The choice of alpha-rolled, beta heat-treated uranium, as fuel for the original core loading, was made in 1954. An alternate fuel, U - 2 wt % Zr alloys was discarded at that time in favor of the unalloyed uranium, due to the sensitivity of the U - 2 wt % Zr alloys to fabrication variables.[6,7] It was originally expected that the high fuel-slug temperatures (500 to 1200°F) would decrease the phenomena commonly known as irradiation growth. The target irradiation exposure was 2500 Mwd/tonne.*

B. SRE SECOND CORE LOADING

As an alternate to uranium and dilute uranium alloys, a Th-U alloy was chosen for test as a potential SGR breeder fuel. The physical metallurgy of these alloys was studied under a separate project.[8,9] It was hoped that both the lack of anisotropy, and of phase change below 2500°F in the face-centered cubic thorium lattice, would result in a fuel that could be irradiated to high burnup at high temperatures with minimum distortion.[10] A Th - 7.6 wt % U alloy will be evaluated as the main fuel in the second SRE fuel loading.

C. EXPERIMENTAL METALLIC FUELS

After selecting the first core loading, an investigation of the means of extending the maximum radiation life beyond 2500 Mwd/tonne and the maximum operating

* Mwd/tonne will be used throughout this report as the unit of burnup. (Mwd/tonne is defined as the total energy produced, expressed in megawatt days per 1000 kilograms of fuel.)

temperatures above 1200°F was initiated. Thermal cycling tests and metallographic studies were conducted on dilute uranium alloys. Alloying compositions investigated originally were limited to about 5 at. % maximum to minimize the effect of alloying on neutron economy.

On the basis of resistance to thermal cycling, fine grain size, feasibility of fabrication, and irradiation data from other sites, cast 2 wt % Zr, cast U - 1.5 wt % Mo, and powder compacted U - 1.2 wt % Mo, were initially chosen for test in MTR capsules and SRE.

Recently the experimental program has been expanded to include the U - 3 wt % Mo alloy, in an effort to provide an alloy with greater radiation stability.[12]

A potential breeder fuel alloy, with a composition of 5.4 wt % highly enriched uranium in thorium, was chosen for test in the SRE. This composition approximates the reactivity equivalent of the 2.78 wt % enriched uranium first SRE core loading. This alloy has a microstructure consisting of a very fine uranium precipitate in a matrix of Th-U solid solution. The solubility of uranium in thorium is about 1% at 1100°F.[8] To test Th-U alloys in the MTR, a Th - 11 wt % U alloy was required to provide SRE temperature conditions in a 3/8-in. diameter pin.

There are presently no clear-cut criteria to use in selecting a metal fuel to operate at elevated temperature. Three different approaches now being investigated as possible means of improving the irradiation damage resistance of metal fuel elements at elevated temperatures are:

1) High creep strength fuels.

2) Metastable gamma phase uranium alloys.

3) High strength fuel claddings.

Small amounts of silicon or aluminum appreciably increased the creep strength of the dilute U-Mo alloys.[13] Of these alloys, U - 3 wt % Mo - 0.5 wt % Si, and U - 3 wt % Mo - 0.1 wt % Al, are now being prepared for test. The finely divided second phase in these alloys may also act as nucleating sites for fission gas

atoms and minimize the fission gas atom mobility. A series of alloys consisting of U - 3 wt % Mo, U - 5 wt % Mo, U - 7.5 wt % Mo, and U - 10 wt % Mo, are being prepared for test. These irradiations should establish the amount of gamma stabilizer required to produce irradiation stability under SRE conditions. The effect of irradiation on phase transformation kinetics in U-Mo alloys will be investigated. Other metastable gamma alloys, such as U - 10 wt % Nb - 4 wt % Zr, are under consideration. An attempt will be made to restrain the swelling of uranium by use of high strength cladding materials such as Nb, Mo, or high strength Zr alloys.[14]

D. EXPERIMENTAL CERAMIC FUELS

After a review of properties, such as thermal conductivity and compatibility with Na, UO_2 was chosen as a ceramic fuel for test in the SRE. The existing irradiation data of UO_2 indicates no gross problems with swelling or distortion at high burnups and at central fuel temperatures up to the melting point of UO_2.[15] However, a new series of problems arise with the use of UO_2 that require investigation, both in-pile, and out-of-pile. These problems include, 1) thermal ratcheting, 2) the effect of irradiation on thermal conductivity, and 3) the amount of fission gas release.

The physical properties, particularly the thermal conductivity, are very important when ceramic and cermet fuels are used. Uranium atom density is another important consideration. Uranium carbide presently looks very promising from these two aspects and will be included in the SRE program, as soon as a suitable fuel element can be fabricated. Other uranium compounds and cermets are being investigated.

V. FUEL ELEMENT DESIGN

The standard SRE fuel elements are seven-rod clusters containing NaK-bonded uranium slugs, in type 304 stainless steel jackets (Figure 2). The active fuel length of the element is 72 in. The 6.0-in. long by 0.750-in.-diameter slugs of unalloyed uranium are bonded by 0.010 in. of NaK to the 0.010-in.-wall stainless steel jecket, for good heat transfer. The fuel element channels through which the sodium coolant flows, are 2.80-in. nominal diameter. The spacing between rods is maintained by spirally wrapped spacer wires. The end fittings position the rods and the element in the channel, and regulate the coolant flow.

The experimental fuel elements for the SRE are designed to equal the reactivity and the power generation of standard fuel elements. Experimental alloy fuels of the proper enrichment are loaded into elements of the standard design. In addition to testing different fuel materials the SRE irradiation tests will be used to evaluate variation in fuel element design. Alternate designs contain the proper quantity of fuel at the desired enrichment and provide adequate heat transfer surfaces.

The cross sections of the alternate fuel elements to be tested in the SRE are illustrated in Figure 3. In each case the fuel length is 72 in. and proper positioning orificing fittings are provided.

A. SEVEN-ROD CLUSTER

The SRE core was designed for a fuel loading of 37 seven-rod fuel elements. The seven-rod element is designed to provide adequate heat transfer surface to limit the center fuel temperature to 1200°F, when the SRE is operating at 20 Mw thermal. There are two possible disadvantages to this design:

1) If isotropic swelling of the fuel occurs, the fuel will have filled the 0.010-in. NaK bond annulus after 8% volume expansion. This expansion will exert pressure on the fuel jacket with the possibility of eventual element rupture.

2) The nonsymmetrical flux distribution in the seven-rod fuel element will produce nonsymmetrical thermal stress and radiation damage which could cause slug warp.

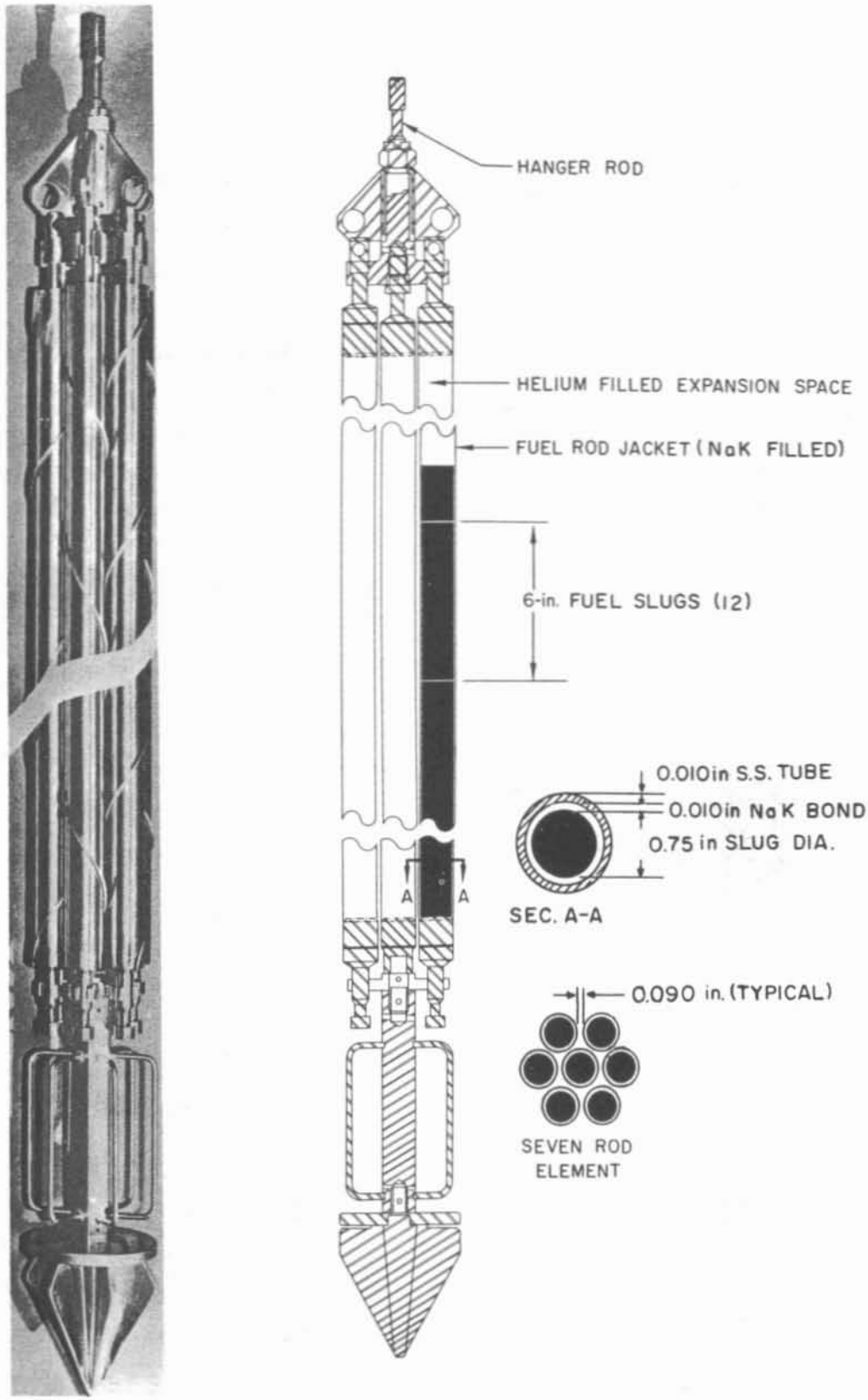
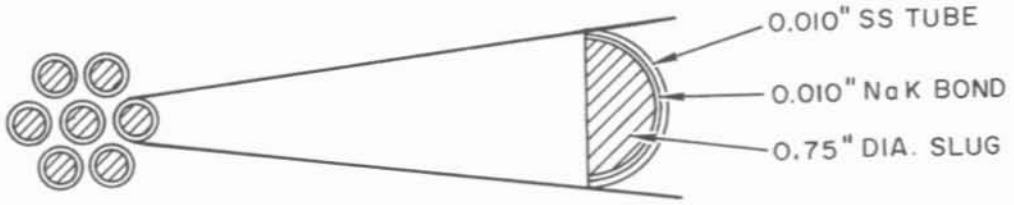


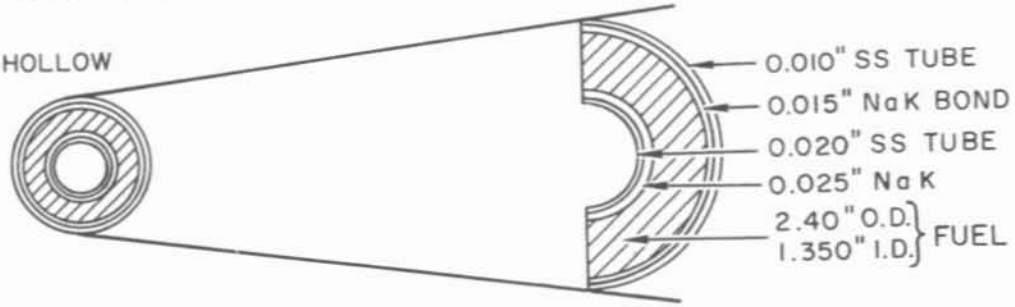
Figure 2. The Standard SRE Fuel Element

1. STANDARD 7–ROD



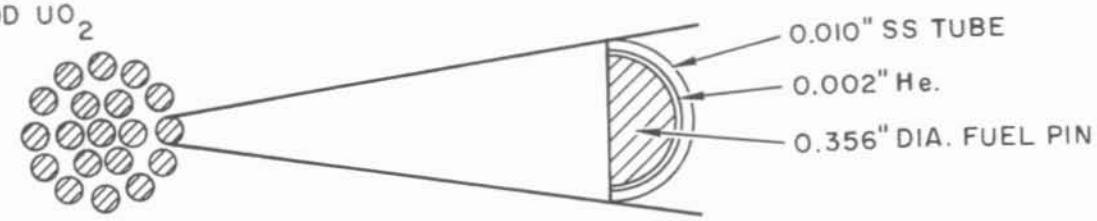
A VARIATION OF THE ABOVE IS 0.720" DIA. SLUGS AND 0.025" NaK BOND

2. LARGE HOLLOW



A VARIATION OF THE ABOVE IS 2.35" O.D. FUEL AND 0.040" NaK BOND

3. 19 ROD UO_2



4. TUBULAR UO_2

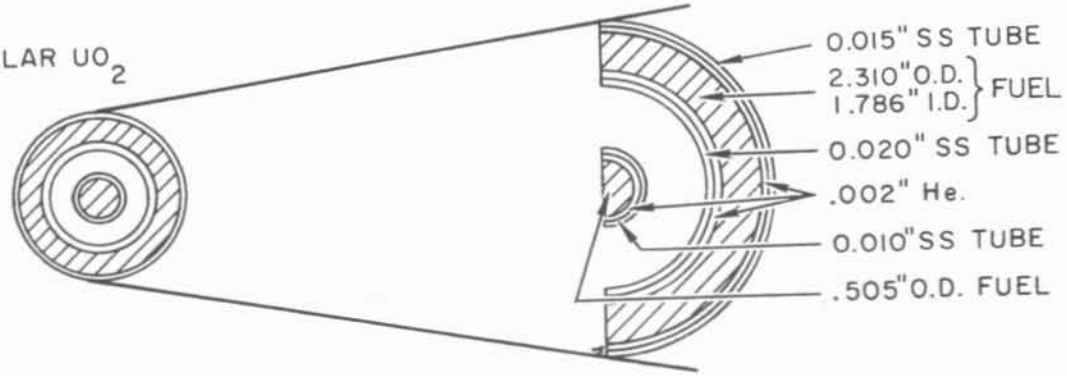


Figure 3. Cross Sections of SRE Experimental Fuel Elements

NOTE: Individual lengths are indicated in the detailed specifications.

The present metal fuel element design provides space in the NaK annulus to accommodate about 3% diameter increase. Therefore, two variations of metal fuel slug designs which allow more space for fuel expansion will be tested in the SRE. The first variation will be to increase the NaK bond to 0.025 in. with a corresponding smaller diameter slug (0.720-in.) which will allow up to 20% volume expansion. The other variation will be to fabricate a center core in the fuel slug to provide increased volume for fuel swelling. In addition, two techniques are to be attempted to reduce slug warp. One technique is to use short slugs about 1 in. long. The other technique is the large hollow fuel element design described below.

B. LARGE HOLLOW ELEMENT

The large hollow fuel element (Figure 3), which contains sixteen four-in. long uranium cylinders, is an alternate design to the seven-rod cluster. It will reduce the possibility of fuel warp and take advantage of possible fuel fabrication cost reductions by using large fuel pieces. This design has the same cross sectional fuel area (3.1 $in.^2$) as the standard seven-rod fuel element. However, in order to provide an equivalent heat transfer surface, the diameter must be increased. The original design allows only 3.75% isotropic volume expansion of the fuel before the outer annulus is filled. If the outer annulus is increased from 0.015 to 0.040-in. by reducing the outside diameter of the slug from 2.400 to 2.350-in. a 10% volume expansion will be possible without interference. Isotropic swelling would not decrease the inside diameter of the fuel; thus the present 0.025 in. inside NaK annulus should be adequate. The 0.020-in. wall of the inside jacket tube is required to prevent buckling of the tube due to pressure developed inside the fuel element during irradiation.

C. 19-ROD UO_2

In an effort to provide an SGR fuel which could be used for long burnups, UO_2 fuel elements were added to the SRE fuel program. Small fuel slug diameters were required to avoid center melting of UO_2 in the SRE, if the experimental element is to match a standard SRE fuel element in power generation. The 19-rod cluster (Figure 3) meets this requirement.

D. TUBULAR UO_2

As an alternate to the 19-rod UO_2 fuel element, a tubular UO_2 element was designed (Figure 3). It has the advantage of requiring fewer pieces of both UO_2 and jacket tubing. The amount of stainless steel in the fuel element is also reduced.

VI. FABRICATION OF FUEL MATERIALS

In the development of fuels for a power reactor, considerable attention should be paid to potential fabrication costs and eventual reprocessing of the fuel. To compare the various known methods of fuel fabrication, and to initiate the development of improved methods, the SRE experimental fuels were fabricated by several different methods. The goal is a fabrication technique that is low in cost, has a high yield, and is adaptable to remote techniques for reprocessing. A list of the fuels fabricated for test in the SRE is given in Table I.

Extensive fabrication records were required to evaluate the irradiation behavior of the various fuels, to establish realistic inspection and quality control procedures, and to evaluate the economics of the various fabrication techniques. All metal slugs in the SRE program are numbered. The slug number relates the slug to individual process variables such as ingot heat, extrusion rod, or sinter batch. By use of the slug number, all of the material can be traced back to the original containers of UF_6. Although UO_2 pellets are not numbered, the batches are isolated so that variations in processing procedures can be evaluated. Similarly, all fuel rods are labelled and the assembly sequence records carefully maintained.

A. METAL FUEL FABRICATION

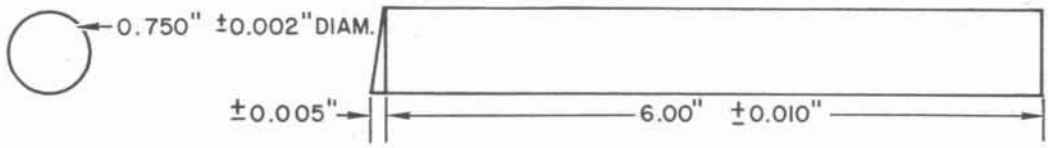
In the initial standard fuel loading for the SRE, 5230 unalloyed alpha-rolled, beta heat-treated 2.78 wt % enriched uranium slugs, 3/4 by 6 in. were fabricated. No development work was necessary prior to the enriched uranium processing. Figure 4 shows the complete specification used for these slugs. Subsequent dilute uranium alloy slugs, and Th-U alloy slugs, were fabricated to the same specifications. Minor modifications were adopted for the powder-compacted short slugs, and the large hollow slugs.

Each slug was visually inspected and dimensionally checked against the specification. Most rejects were caused by surface seams and end cracks. Fifty reject slugs with light seams, heavy seams, and end cracks, were accepted for evaluation in the SRE. One of the purposes of the irradiation examinations is to correlate performance with specifications. It is anticipated that by more liberal

TABLE I

EXPERIMENTAL FUELS FOR THE SRE

Material	Fabrication Technique	Number Pieces Fabricated	Dimensional Specifications (inches)	Uranium Enrichment (wt % U^{235} in U)
Uranium	Alpha-rolled, beta heat-treated	5,230	Figure 4	2.778
	Powder compacted			
U - 1.2 wt % Mo	Cold pressed and sintered	515	0.750 ± 0.002 dia. x 1.000 ± 0.002 length	2.778
U-large hollow	Hot pressed	36	1.35 x 0.002 ID x 2.40 x 0.002, 4.000 ± 0.010 length	
U - 2 wt % Zr	Centrifugally cast and centerless ground	215		
U - 1.5 wt % Mo		120	Figure 4	2.778
U-beta heat-treated		25		
U - 3.0 wt % Mo	Statically cast to final diameter	60	0.747 ± 0.005 dia., 6.00 ± 0.010 length	3.1
U-3.0 wt % Mo large hollow		14	1.35 ± 0.005 ID, 2.35 ± 0.005 OD, 4.000 ± 0.010 length	3.25
Th - 5.4 wt % U	Extruded and swaged to final diameter	450		
Th - 7.6 wt % U (SRE second core)		3,800	Figure 4	Above 90
UO_2 pellets	pressed and sintered	11,000		
UO_2 rings		49	Figure 5	8



DIMENSIONAL SPECIFICATIONS

End radius deburred with no sharp corners.
All surfaces, 250 root mean square.
Maximum single-throw warp 0.003 in. per 6.00-in. length.
Ends to be square with centerline to ± 0.005-in.

MATERIAL QUALITY SPECIFICATIONS

Isotopic Content: ± 0.005 at. % U^{235}
Purity: See Table 2.
Density (min.): Unalloyed U 18.86 g/cc; U - 1.5 wt % Mo 18.45 g/cc; U - 3.0 wt % Mo 18.20 g/cc; U - 2.0 wt % Zr 18.20 g/cc; Th - 5.4 wt % U 11.8 g/cc; Th - 7.6 wt % U 11.9 g/cc.
Grain Size: Maximum of 0.21 mm.
Alloy Content: ± 0.1 % of alloy element; from slug to slug and within one slug. Exception of + 0.2 wt % U in Th - 7.6 wt % U alloy.
Metallurgical Quality: Uniform microstructure, preferred orientation must be less than reactor grade alpha-rolled, beta heat-treated uranium.
Surface, Subsurface, and End Imperfections: Seams or cold shuts longer than 1/2-in., wider than 1/32-in., or deeper than 1/32-in. shall be rejected. Only three defects within a 1/2-in. area on one piece are acceptable. Striations are cause for rejection only when in large quantity. Surface or subsurface porosity, or shrinkage cavities, greater than 1/32-in. deep or 1/32-in. wide with two or more within a 1/8-in. area on one slug are cause for rejection. Cracks greater than 1/32-in. in length are cause for rejection. Machining defects are cause for rejection. Any macroscopic nonmetallic inclusions must be removed and the slug must pass all specifications after removal.
Destructive Tests: Representative samples will be destructively examined for specification verification.

Figure 4. SRE Metal Fuel Slug Specifications

specifications the process yield will be substantially increased and the fuel cost reduced. Samples were taken throughout the fabrication cycle for isotopic and spectrochemical analysis. Table II lists the typical chemical impurity analysis of the fuel. Samples from each rolled rod were also retained for metallographic analysis.

The various fabrication techniques listed in Table I were first used to produce unenriched process development slugs, which were evaluated for surface specifications, internal soundness, chemical homogeneity, and metallurgical structure. The uranium feed material for the enriched uranium alloy slugs, was rejected slugs, and end crops from the fabrication of the enriched unalloyed

TABLE II

CHEMICAL ANALYSIS OF SRE METAL FUEL

Uranium Metal

Element	Amount (ppm)	Element	Amount (ppm)
Carbon	50 - 250	Chromium	Trace
Iron	20 - 60	Boron	Trace
Nitrogen	10 - 20	Silver	Trace
Nickel	40 - 90	Cadmium	Trace
Manganese	10	Cobalt	Trace
Silicon Dioxide	20	Lead	Trace
Copper	20	Phosphorus	Trace
Magnesium	5 - 15	Zinc	Trace

Thorium Metal

Element	Amount (ppm)	Element	Amount (ppm)
Carbon	600 - 700	Chromium	50
Iron	300	Boron	0.8
Manganese	5	Nickel	180
ThO_2	1%	Cobalt	10
Copper	50 - 100	Lead	10
Magnesium	10	Beryllium	10 - 150*
Silicon	110	Molybdenum	20
Aluminum	30	Zirconium	50 - 100*

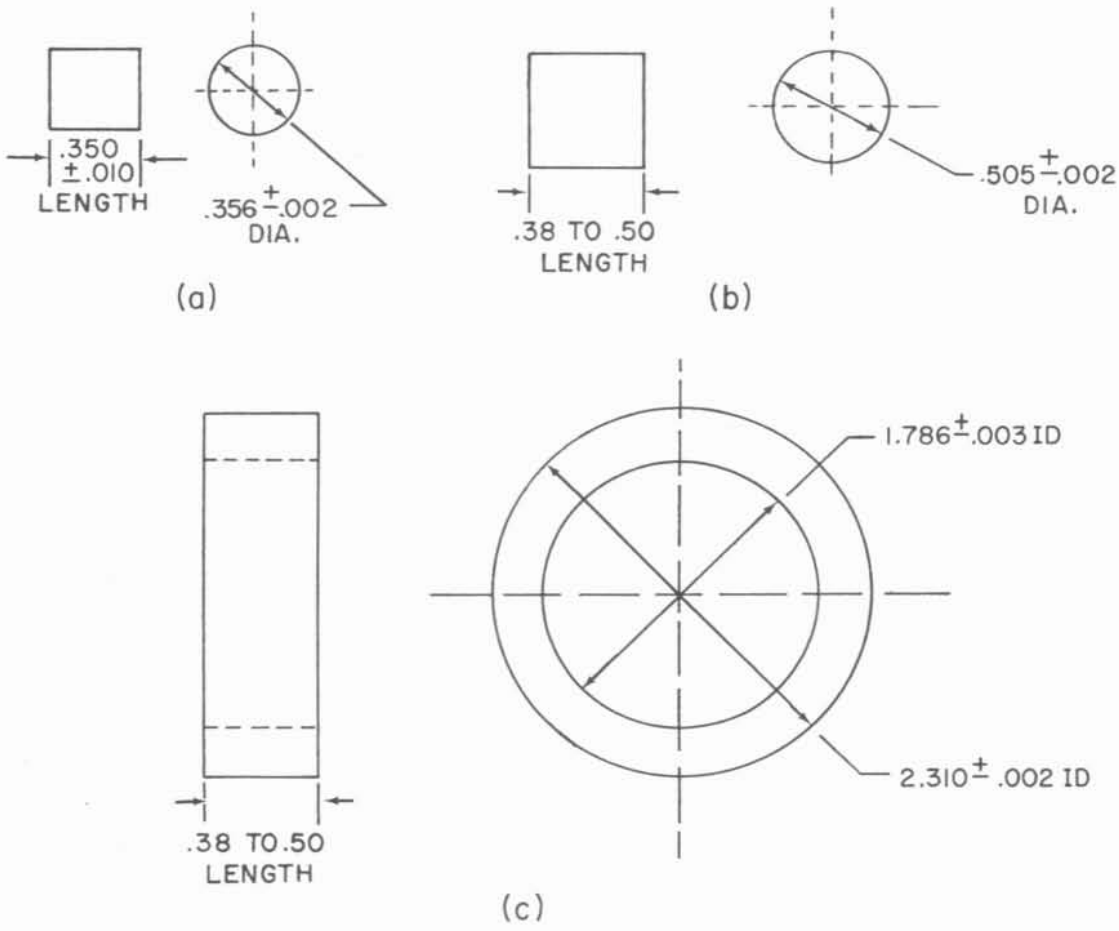
* Due to crucible and mold wash.

uranium for the first core loading. Double arc-melted thorium and highly enriched uranium were used to produce the Th - 5.4 wt % U, and Th - 7.6 wt % U alloy fuel slugs.[10] All slugs were inspected to the specifications in Figure 4, with the exception of the dimensions noted in Table I. Extensive samples, from each processing step, were taken for analysis and to provide samples for comparison with irradiated material.

Fuel specimens for irradiation in the MTR were fabricated in the form of 0.375 ± 0.005-in. diameter x 1.500 ± 0.010-in. long pins. Whenever possible the SRE fuel fabrication techniques, and quality specifications were utilized. Uranium enriched to 10 wt % U^{235} was required to duplicate SRE fuel temperatures in these small diameter uranium and uranium alloy pins. The Th-U alloy fabricated for MTR testing contained 11 wt % of highly enriched uranium.

B. UO_2 FUEL FABRICATION

For an initial test of UO_2 in the SRE, 11,000 pellets and 49 rings were fabricated cated. The pellets were 0.356-in. diameter for use in 19-rod clusters and 0.505-in. diameter for use as center rods in tubular elements. The rings for the tubular elements were 2.310-in. OD and 1.786-in. ID. The UO_2 pellets and rings were fabricated to the specification shown in Figure 5. All UO_2 powder was produced from the same withdrawal of eight wt % enriched UF_6. The UO_2 powder was prepared by an ammonia precipitation technique. Since numbering of the small pellets is not feasible, the pellets were kept segregated by process lot so that any effect of processing on the UO_2 irradiation behavior in the reactor can be isolated. All 0.356-in. diameter pellets were produced by cold pressing and sintering which produced pellets to the dimensional specification without grinding. The resultant pellets had an oxygen to uranium ratio of 2.02. The fabricator also produced pellets with two 0.032-in. axial holes, one at the center axis, and one 0.030 in. from the edge; to be used for thermocouples for temperature monitoring. The UO_2 rings were fabricated by two suppliers to evaluate processing techniques.



DIMENSIONAL SPECIFICATIONS

ALL SURFACES 250 MICROINCHES, R.M.S.
(a) & (b) ENDS FLAT AND PARALLEL WITHIN 0.0015 INCHES WITH THE PLANES 90 ± ½° TO THE CYLINDER AXIS
(c) ENDS FLAT AND PARALLEL WITHIN 0.005 INCHES WITH THE PLANES 90 ± ½° TO THE CYLINDER AXIS

MATERIAL QUALITY SPECIFICATIONS

ISOTOPIC CONTENT ± 0.1 ATOM % U235
HEXAVALENT U-2 % AS RECEIVED
DENSITY 10.1 gr/cc MINIMUM
PURITY: MAXIMUM IMPURITY CONTENTS IN PPM:

Fe-200	C-100	Ti-500
Si-100	Ni-150	B-1
F-200	Pb-5	Cd-50
Mo-5	Sn-5	Cu-50
Ag-0.2		

Figure 5. UO_2 Fuel Specifications

VII. IRRADIATION OF FUELS

A. PRE-IRRADIATION EVALUATION

As metallic fuels were received from the vendor, samples of the fuel were evaluated for chemical homogeneity, microstructure, microhardness, density, and physical integrity. All cast slugs were gamma-graphed to determine the amount of center line shrinkage.

Oxide pellets and rings were measured for density and dimensions. Since the pellet diameter is expected to affect the center temperature of the fuel, by varying the size of the helium gap, the pellets were segregated into groups by diameter.

B. ASSEMBLY OF EXPERIMENTAL FUEL ELEMENTS

After pre-irradiation evaluation, the fuel materials were divided into batches, with each batch containing the amount of fuel required for one fuel rod. Each metal fuel batch contained only one fuel composition produced by a single fabrication technique. Variations in slug surface conditions and slug processing lots were included in each batch. One-half in. wafers were cut from two slugs in each experimental metal fuel batch, prepared for metallographic examination, and weighed, to determine density. After photomicrographs were taken, each wafer was returned to its own batch. This technique provides for pre- and post-irradiation examination of the same area. Oxide fuels were batched by selecting pellets from the diameter groups so that each batch contained pellets with uniform dimensional tolerances, and similar processing and fabrication histories.

After batching, each group of fuel units was assembled into one fuel rod. A canning procedure was established for standard SRE metallic fuel which consisted of: 1) electropolishing each slug, 2) vacuum outgassing the slugs at 1000°F, to remove hydrogen, 3) loading 12 slugs (one batch) into a fuel jacket with a welded bottom end cap, 4) vacuum out-gassing 21 rods at 750°F in a multi-rod loading machine 5) loading NaK to a specific level above the fuel slugs, and 6) welding top-end caps under a helium-argon atmosphere.[1] Quality control procedures on the assembled rod included an eddy current inspection for NaK bond continuity, and a helium-leak inspection of the welds.

Oxide fuel rods are fabricated by a similar procedure which consisted of welding caps on loaded rods in the multirod loading machine under a helium atmosphere. The helium acts as the heat transfer medium between the UO_2 and the cladding. A NaK bond is not used since the center temperature of the UO_2 fuel is above the boiling point of NaK. There would be a possibility of high NaK vapor pressures being generated in a NaK-bonded UO_2 element when the UO_2 pellets fractured.

Thermocouples were installed in selected metal and oxide fuel rods to record the fuel temperatures during irradiation. Thermocoupled metal fuel rods contain 1/8-in. diameter stainless steel clad chromel-alumel thermocouples extending down an axial 3/16-in. diameter hole drilled in the slugs. These axial holes provide a cored fuel design, for evaluation, as well as providing the path for the thermocouples. The sheath of the thermocouple is welded leak tight through the fuel rod end cap. A coil wound in the thermocouple, and extra depth in the axial hole, provide for fuel expansion. The depths that the thermocouples extend into the experimental metal fuels and the composition of the thermocoupled rods, in the SRE, are shown in Figure 6. Each thermocoupled UO_2 rod contains two Pt:Pt+Rh thermocouples; one centrally located and the other 0.030 in. from the edge of the UO_2 pellets. The distribution of thermocouples in UO_2 fuel elements is shown in Figure 7. The thermocouples used are 0.025-in. diameter Pt sheathed, with a single Pt+Rh conductor insulated with MgO_2. These bayonet thermocouples are inserted into the 0.030-in. holes in the UO_2 pellets. A splice is made in the helium gap above the UO_2 between the Pt-sheathed thermocouple and stainless steel clad compensating lead wires. The stainless clad of the lead wires is welded leak tight through the fuel rod end cap. Similar techniques were used to install thermocouples in metallic large hollow fuel elements and tubular UO_2 fuel elements. In addition to the thermocouples in each fuel material, every fuel element contains a thermocouple located in the hanger rod to monitor the exit sodium temperatures.

Finished experimental fuel rods are assembled into fuel elements as shown in Table III. The fuel elements listed are those which have been completed or are in assembly at the present time.

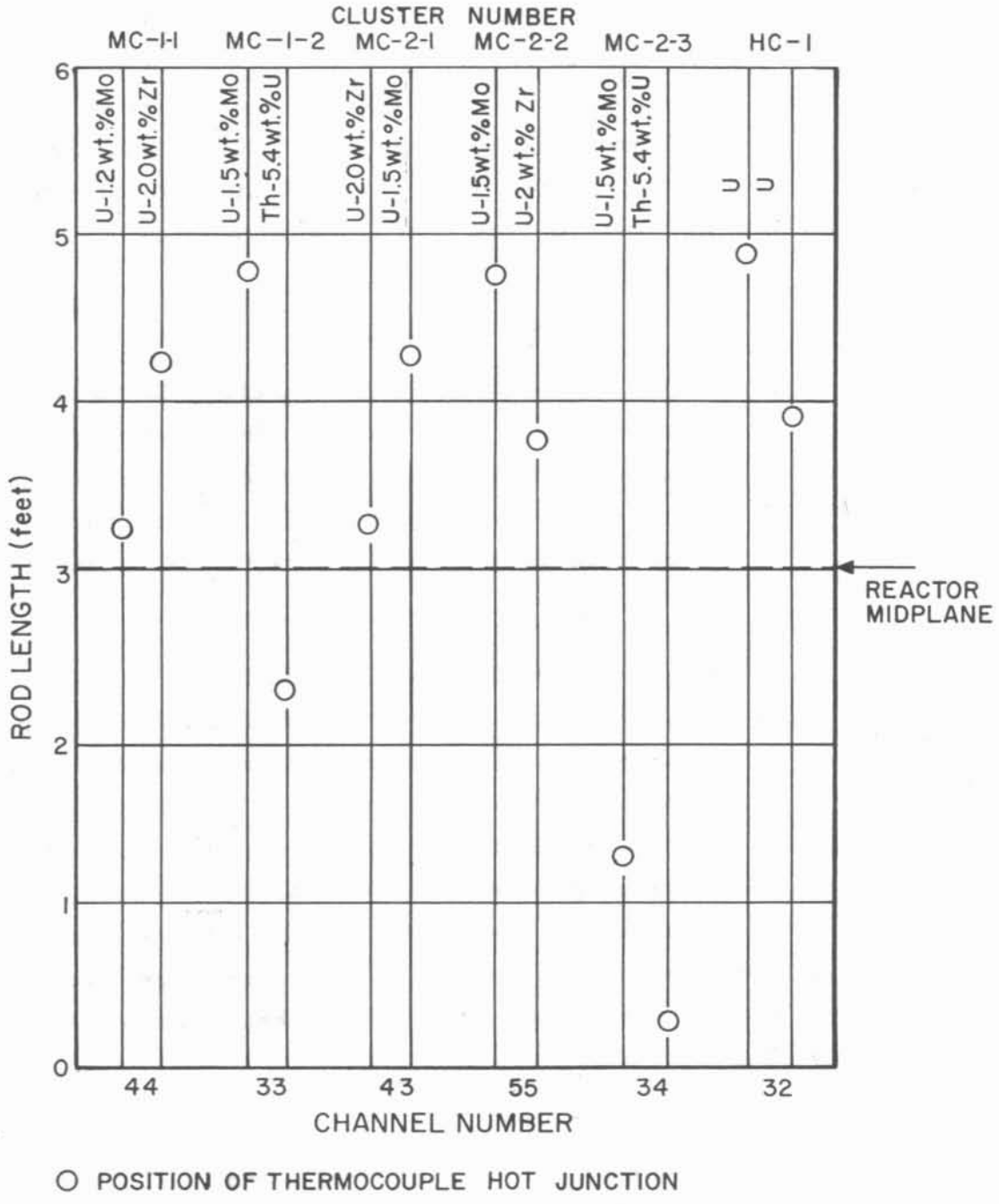


Figure 6. Thermocouple Distribution in the SRE Experimental Metal Fuels

1. MC-1 Fuel Elements

These two 7-rod cluster fuel elements were assembled to provide a test of each experimental fuel alloy in an element which contained unalloyed alpha-rolled, beta heat-treated fuel, as a control. These elements are to be irradiated to burnups equal to, or less than, the allowable burnup on the standard fuel. Each element contains two thermocoupled fuel rods positioned diametrically opposite each other.

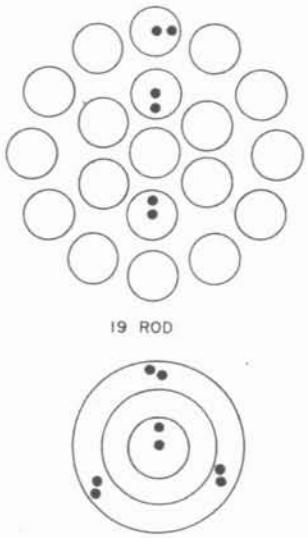


Figure 7. Thermocouple Location in Oxide Fuels

2. MC-2 Fuel Elements

These three 7-rod fuel elements contain only alloyed fuels. The elements are intended to be exposed to higher burnups than the unalloyed, alpha-rolled, beta heat-treated standard fuel. Each element is thermocoupled in the same manner as the MC-1 clusters.

3. Reduced Diameter Slug Element

Since the standard SRE design will permit only 8% fuel volume change, it is doubtful whether any of the unalloyed uranium, or diluted uranium alloy slugs, can be irradiated much beyond 1000 Mwd/tonne. To provide data at higher burnup, and to test the hypothesis that increasing the volume available for expansion will increase the useful life of a fuel element, one seven-rod fuel element is to be fabricated with 0.720 in. diameter slugs. This element will contain five-rods of unalloyed, alpha-rolled, beta heat-treated uranium and two-rods of U - 2 wt % Zr.

4. Large Hollow Fuel Elements (HC)

The two large hollow metal fuel elements (HC-1 and HC-2) containing unalloyed uranium are intended for burnup equal to, or less than, the primary fuel loading. This is due to the small amount of available space for volume expansion of the fuel. These elements will establish the validity of the hollow fuel element design. The HC-3 element, containing U-3 wt % Mo, was provided

TABLE III

EXPERIMENTAL FUEL ELEMENTS FOR THE SRE

Fuel Element Designation	Fuel							Outside Diameter Fuel (in.)
	Unalloyed U β-Heat-Treated		U - 2 wt % Zr	U - 1.2 wt % Mo	U - 1.5 wt % Mo	U - 3 wt % Mo	Th - 5.4 wt % U	
	α-rolled	cast	Cast	Powder compacted	Cast	Cast	Extruded	
7-rod metal fuel mixed clusters								
MC-1-1	2*	1	1	1	1		1	0.750
MC-1-2	2	1	1	1	1		1	0.750
MC-2-1			2	1	1	1	2	0.750
MC-2-2			1		2	2	2	0.750
MC-2-3			2	1	1	1	2	0.750
Reduced diameter	5		2					0.720
Metal large hollow element								
HC-1	Hot pressed U cylinders							2.40
HC-2	Hot pressed U cylinders							2.40
HC-3	Cast U - 3 wt % Mo cylinders							2.35
UO_2 19-rod								
CO-1	Pressed and sintered UO_2							0.356
UO_2 tubular								
HO-1	Pressed and sintered UO_2							2.310

* Indicates number of rods of given alloy in the experimental fuel element. These fuel elements are presently in the SRE or in the final stages of fabrication.

with 10% volume expansion of the fuel and is expected to go to greater burnup than the primary fuel loading. This is a result of the increased expansion volume and of the alloy fuel material. Each hollow fuel element contains two thermocouples positioned to read maximum fuel temperatures.

5. 19-Rod UO_2 Fuel Element

The 19-rod UO_2 fuel element will be used to establish the validity of its design. The effect of variations in UO_2 pellet diameters and densities on radiation behavior will be evaluated. Thermocouples are located near the vertical midplane of the element to record maximum fuel temperatures. The location of two thermocouples within each rod, one at the center, and one just below the surface, is expected to provide some data on the thermal conductivity of irradiated UO_2.

6. Tubular UO_2 Fuel Element

Irradiation behavior of the tubular UO_2 fuel elements will be used to evaluate UO_2 rings and pellets with different dimensional and density tolerances. Thermocouples are located near the vertical midplane of the element to record maximum fuel temperatures.

C. SCHEDULE OF IRRADIATIONS

The initial loading of experimental fuels is shown in Figure 8. In order to achieve the highest rate of burnup, experimental metal fuels have been loaded into the central region of the SRE. Due to uncertainties in the thermal conductivity of the first UO_2 fuel elements, a UO_2 19-rod fuel element was loaded into an outer channel where flux and temperature are relatively low. Localized overheating of the fuel channel during reactor scrams is a potential problem with UO_2 fuel elements under present SRE sodium flow conditions. Overheating might result from the large amount of heat stored in a UO_2 fuel element, as compared to the metal fuel element. The initial SRE tests on the 19-rod UO_2 fuel element indicate that the element is compatible with present SRE conditions in an outer fuel channel.

The present loading of experimental fuels provides a distribution of fuel thermocouples as shown in Figure 6. Originally five standard unalloyed uranium

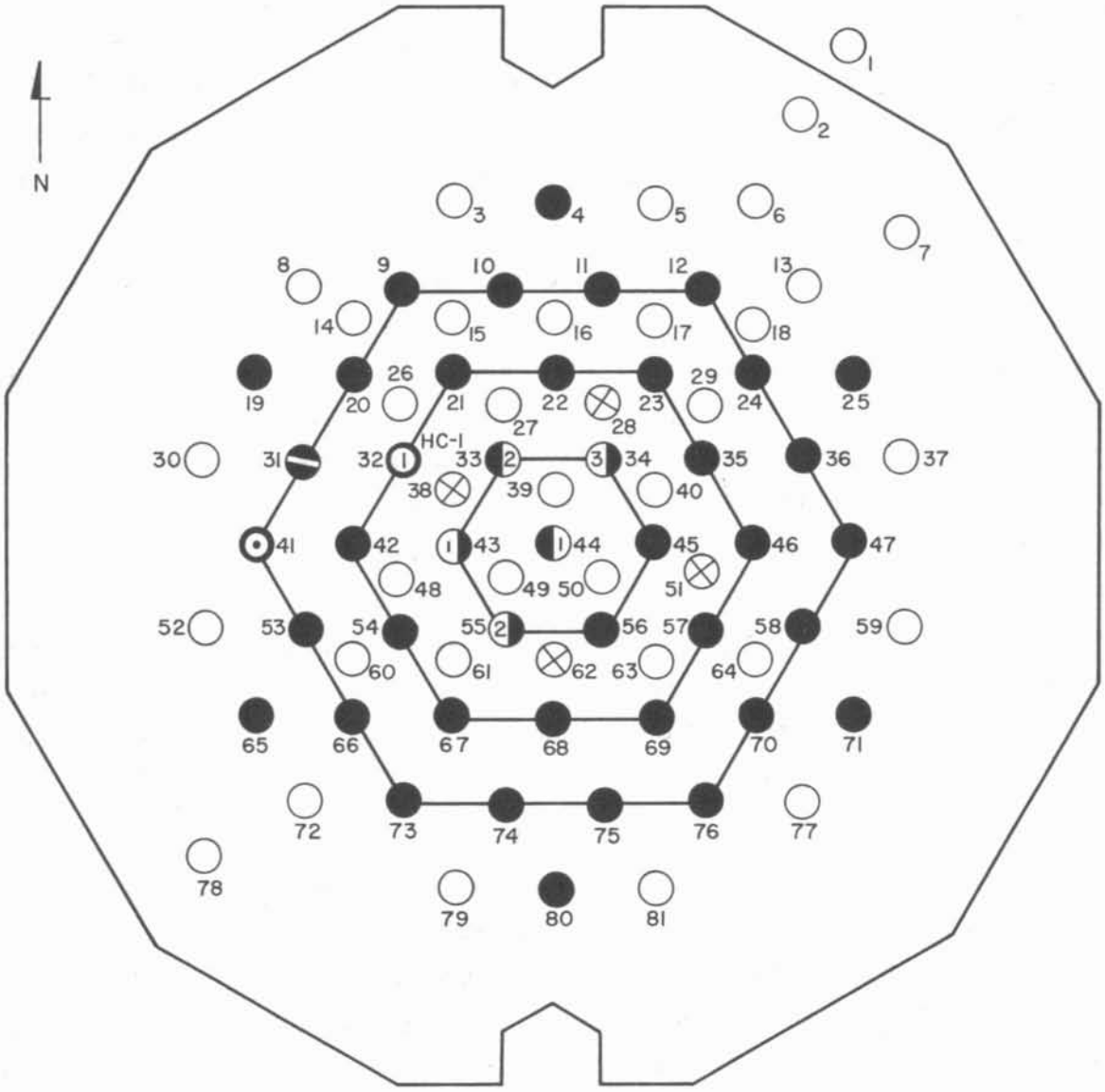


Figure 8. Experimental Fuel Loading

fuel elements containing thermocouples, were loaded into outer fuel channels. These thermocouples failed during the initial hours of full-power operation, due to failure of either the insulated lead wires, or the splices in the shield plugs. The experimental fuel elements were modified so that stainless steel clad thermocouples extend all the way to the reactor face. These modified thermocouples have operated satisfactorily.

The scheduling of fuel element removals has been established to satisfy two requirements: 1) the maximum limit of burnup for the present unalloyed core loading must be determined so that the present fuel can be utilized fully and the Th - 7.6 wt % U fuel loading can be scheduled into the SRE, with a minimum of reactor down-time, 2) the unalloyed uranium fuel and the experimental fuels are to be examined at roughly equal fractions of expected maximum burnup, based on fuel distortion. By this schedule, the mechanisms and variables of radiation damage can be properly evaluated. In addition to removing fuel elements for destructive examination to evaluate fuel slug distortion, periodic nondestructive examinations are made to insure gross fuel element integrity. Fuel elements are examined visually in the hot cell, and fuel rod dimensions are measured without removal from the element.

The schedule which has been followed through the first 1100 Mw days of reactor operation and the tentative schedule for the remainder of the estimated life of the first core loading is shown in Figure 9. The examinations carried out during the first 1100 Mwd of SRE operation were: 1) after about 50 Mwd/tonne at low temperature, 2) after about 80 Mwd/tonne which included the first full-power and design temperature run, and 3) after about 300 Mwd/tonne consisting primarily of steady full-power operation. These destructive examinations of standard seven-rod fuel elements established that the fuel element design was functioning properly and established a zero point on which to base further full-power operation results.

D. MTR TESTS

To obtain advance information on radiation stability, at SRE temperatures, and to eliminate unpromising fuels from further consideration, irradiation tests are conducted in the MTR. The irradiations are conducted in capsules designed

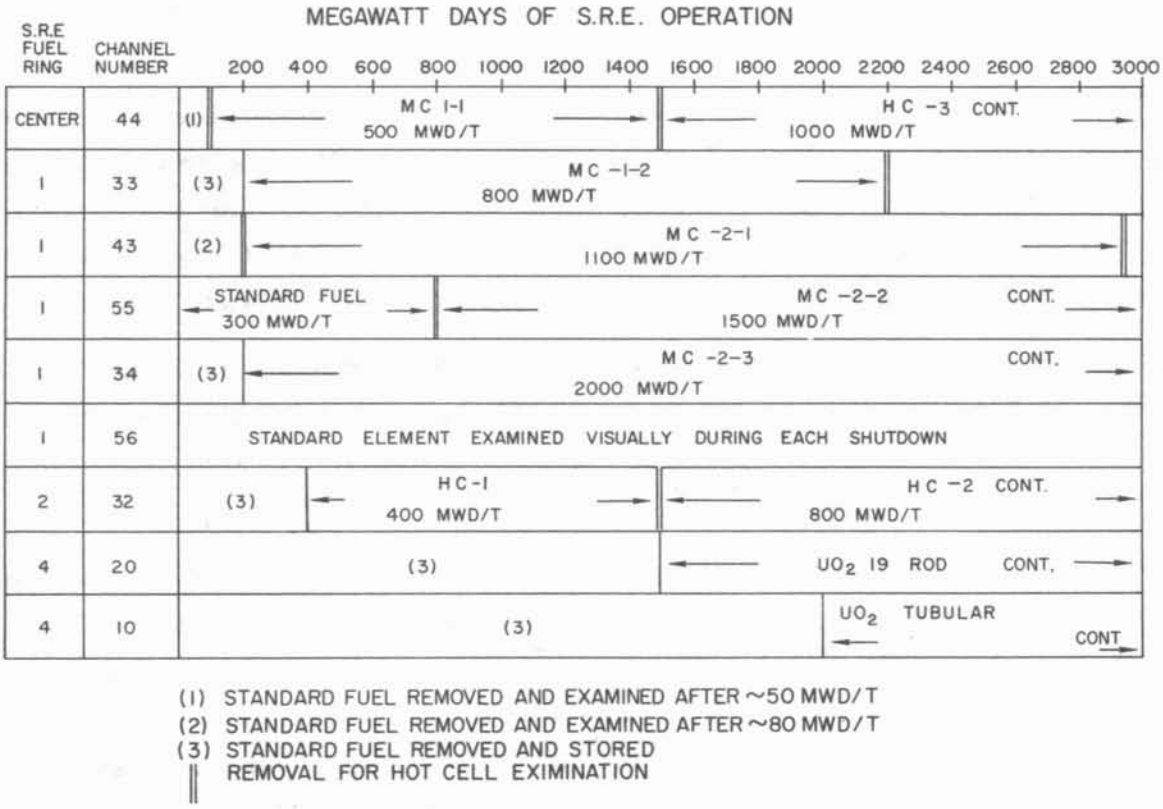


Figure 9. Irradiation Schedule for SRE Experimental Fuels

to produce maximum SRE center and surface fuel temperature conditions in 3/8-in. diameter fuel pins. The central temperature of the fuels is measured with a thermocouple. Six assemblies, each containing five or six specimens, have been irradiated. Unalloyed alpha-rolled beta heat-treated U, U - 1.2 wt % Mo, U - 2 wt % Zr, and Th - 11 wt % U were tested. Details of these irradiations have been reported separately.[16]

VIII. HOT CELL EXAMINATION

After an SRE fuel element has reached its prescribed burnup, the reactor is shut down. The element is cooled until the afterglow temperature does not exceed 1200°F, during subsequent manipulation. The fuel element is removed from the reactor with a shielded cask, washed free of Na in a wash cell, and lowered into the SRE hot cell. During these operations the fuel temperature of the experimental fuel elements is monitored to guard against overheating. The afterglow heating data will contribute to improved cask and wash cell design.

In the hot cell, the general appearance of the element, fuel jacket diameters, and fuel rod straightness are determined. Photographs are taken at each examination. Elements removed for visual examination only, are then returned to the reactor.

When the element is to be destructively examined, the metallic fuel rod, or rods to be examined, are removed from the cluster and decanned. Fission gas release will be determined on both experimental metal and UO_2 fuel rods. Decanned metal slugs and clad UO_2 rods will be given a complete dimensional inspection. Dimensional inspection for diameter, length and warp has been made on the fuel elements examined to date. In future experimental fuel examinations, density measurements also will be taken on full-size slugs and on the wafers that were canned as part of the fuel rods.

Selected metal slugs and UO_2 rods will be sectioned to provide metallographic specimens and burnup samples. Gamma-counting techniques and burnup analysis will establish the burnup profile along the length of the fuel rod. The metallographic specimens will be available for post-irradiation metallography, microhardness, and annealing studies.

IX. IRRADIATION RESULTS

A. SRE RESULTS

Nondestructive examinations of the seven-rod elements and the hollow elements, after 1100 Mw days of SRE operation, have revealed no distortion of either the fuel elements as a whole or the individual jacket tubing. Three fuel rods containing unalloyed alpha-rolled, beta heat-treated slugs have been destructively examined in detail. A tabulation of the results of these examinations is given in Table IV. Slug warp apparently occurs at very low burnups, but the effect either saturates with burnup, or is restrained by the cladding. This initial warp may be caused by the relief of residual fabrication stresses. Slug length increased progressively with burnup. It is interesting to note that the amount of length change is roughly proportional to the relative flux over the entire rod, with no evident effect of temperature. Diameter increases were noted only on the highest burnup slugs. Since bumping on the slug surfaces and the preirradiation tolerances practically invalidate any calculation of density changes, any conclusions regarding swelling will have to await more precise density measurements. A more detailed evaluation of the results will be made, when an accurate burnup profile is available, which will also allow a more complete temperature profile to be calculated.

B. MTR RESULTS

Five MTR irradiation assemblies have been opened for postirradiation examination. Due to difficulties in reproducing SRE conditions with 3/8-in. fuel pins in the MTR, only fragmentary information was obtained on unalloyed U and U - 2 wt % Zr. However, neither of these fuels appeared promising on the basis of qualitative observations.

Four U - 1.2 wt % Mo fuel pins were examined quantitatively for dimensional and density changes. The results of these examinations is given in Table V. These tests indicate that this alloy will swell appreciably after 2000 to 3000 Mwd/tonne irradiation at fuel temperatures between 800 to 1200°F.

Th - 11 wt % U did not swell appreciably during MTR tests under SRE conditions. A total of 17 specimens were evaluated for postirradiation density and

TABLE IV

RESULTS OF DIMENSIONAL EXAMINATION OF THREE STANDARD SRE FUEL RODS

Burnup Fuel Temp.	50 Mwd/tonne Less than 750°F			80 Mwd/tonne Less than 1000°F			300 Mwd/tonne Less than 1000°F		
Slug No.	Average % Diameter Change	Average % Length Change	Single Throw Warp, Mills	Average % Diameter Change	Average % Length Change	Single Throw Warp, Mills	Average % Diameter Change	Average % Length Change	Single Throw Warp, Mills
12 (top)	0.0	0.1	4	0.0	0.3	16	0.0	1.0	21
11	0.0	N.R.	5	0.3	0.3	7	0.3	1.2	27
10	0.0	N.R.	10	0.1	0.3	3	0.3	1.2	22
9	0.0	0.1	8	0.0	0.5	25	0.5	1.4	20
8	0.0	N.R.	26	0.0	0.4	12	0.3	1.5	26
7	0.0	0.2	7	0.0	0.6	23	0.4	1.5	19
6	0.0	N.R.	4	0.1	0.7	25	0.4	1.6	23
5	0.0	0.2	8	0.0	0.7	26	0.7	1.8	26
4	0.0	N.R.	11	0.1	0.6	22	0.1	1.5	18
3	0.0	0.3	18	0.0	0.6	21	0.1	1.3	22
2	0.0	N.R.	17	0.0	0.6	24	-0.1	1.2	14
1 (bottom)	0.0	0.1	6	0.1	0.3	10	0.0	1.0	16
(Average)	0.0	0.2	10	0.0	0.5	17	0.3	1.4	21

NOTE: 1) Changes are positive unless labeled negative.
2) Due to preirradiation tolerances diameter changes are ± 0.3% and length changes are ± 0.2%.
3) All fuel was alpha-rolled, beta-heat-treated.
4) Temperatures represent maximum central temperature.

dimensional changes. Included were fuel pins irradiated to burnup greater than 10,000 Mwd/tonne, and fuel pins with measured center temperatures up to 1495°F. Details of the irradiation conditions and the postirradiation evaluation are given in Table V. The capsule designs, the details of the irradiation, and the detailed results for the MTR irradiations have been reported separately.[16]

TABLE V

DIMENSIONAL CHANGES OF MTR IRRADIATED SPECIMENS

Specimen Composition (wt %)	Measured Central Temperature (°F)		Calculated Surface Temperature (°F)		Burnup (Mwd/tonne*)	Fission per cc (x 10^{20})	Maximum Increase in Diameter (%)	Increase in Volume (%)†	Sample NAA-15-
	Range	Average	Range	Average					
U - 1.2 Mo	830 to 700	756	655 to 555	598	2,400	1.2	9.9	22	3-5
U - 1.2 Mo	1050 to 920	972	825 to 740	763	3,100	1.6	13	31	3-1
U - 1.2 Mo	1245 to 905	1017	980 to 715	799	2,100	1.1	7.7	6.9	4-1
U - 1.2 Mo	1130 to 990	1052	890 to 775	827	2,100	1.1	9.6	7.3	4-5
Th - 11 U	1100 to 1000	1059	899 to 820	866	2,900	1.4	1.6	0.47	5-5
Th - 11 U	1300 to 1110	1172	1060 to 905	957	2,900	1.4	0.8	0.78	5-1
Th - 11 U	1225 to 1195	1214	1000 to 975	991	2,900	1.4	1.1	0.86	5-2
Th - 11 U	1230 to 1215	1219	1005 to 992	995	2,900	1.4	1.5	1.9	5-3
Th - 11 U	-	-	-	-	2,900	1.4	2.4	1.6	5-4
Th - 11 U	1135 to 1010	1071	925 to 825	876	5,400	1.7	0.7	2.9	7-6
Th - 11 U	1230 to 1160	1198	1005 to 950	979	5,900	1.9	1.4	4.7	7-5
Th - 11 U	1255 to 1115	1187	1025 to 910	970	6,100	2.0	1.3	4.1	7-4
Th - 11 U	1300 to 1090	1176	1060 to 890	961	6,300	2.0	**	**	7-3
Th - 11 U	1415 to 1163	1252	1150 to 950	1022	6,400	2.1	2.3	5.1	7-2
Th - 11 U	1495 to 930	1130	1220 to 760	924	6,600	2.1	4.4	7.7	7-1
Th - 11 U	-	970§	-	800§	9,100	2.9	1.1	3.2	6-6
Th - 11 U	-	1030§	-	850§	9,700	3.1	1.5	2.7	6-1
Th - 11 U	-	1100§	-	900§	10,700	3.4	1.5	4.4	6-2
Th - 11 U	-	1100§	-	900§	10,800	3.5	1.9	4.8	6-5
Th - 11 U	-	1100§	-	900§	11,100	3.6	1.7	4.1	6-3
Th - 11 U	-	1100§	-	900§	11,000	3.5	1.5	4.9	6-4

* Based on chemical analyses of some specimens and heat balance on all specimens.
† Based on density of measurements of central sections.
§ Brief thermocoupled life required that these temperatures be calculated.
** Specimen could not be removed from capsule.

X. CORRELATION OF RESULTS

The results of hot cell examination for physical changes in the various fuels will be compared with the complete irradiation history, the temperature profile during irradiation, and the burnup profile in the fuel rod, to establish limits of temperature and burnup under SRE operating conditions. Variations in fuel element design, fabrication techniques, and material quality will be evaluated. Oxide fuels will be evaluated for the percentage of fission gas release and thermal ratcheting effects. Pre- and postirradiation microstructure, and microhardness of the various fuels, will be evaluated to provide a basis for the selection of new fuel materials. Fuel materials, which appear promising under SRE conditions, will be further evaluated for physical and mechanical properties, to establish criteria for selecting future materials and designs worthy of reactor testing.

The effect of dimensional tolerances, surface condition, and chemical composition on fuel performance, will be evaluated to establish realistic specifications for the procurement of future fuel materials.

XI. FUTURE TESTS

An important phase of this program is the continuing evaluation of promising new fuel materials and fabrication techniques. Fuel materials that show potential as promising SGR fuels, after physical property tests, mechanical property tests, and nuclear analysis, will be tested under SGR operating conditions in the SRE. These future tests will follow the general plan of action established for the present tests.

A. METAL FUELS

The second SRE Core Loading of Th - 7.6 wt % U, in the seven-rod element design, will lower the thermal neutron flux in the SRE by approximately 30% due to its higher U^{235} density. It is hoped to establish a suitable flux for testing fuels of the present core enrichment by loading the center seven fuel channels of the reactor with low-enrichment, uranium base fuels; and by loading the 30 outer

fuel channels with Th - 7.6 wt % highly enriched uranium fuel. Thermocoupled Th - 7.6 wt % U elements are being fabricated to provide temperature mapping of the Th-U core. It is planned to include cast Th - 7.6 wt % U in this loading, in both the seven-rod, and hollow element design, as fabrication techniques are developed.

Uranium fuels that are to be irradiated in the SRE in the near future are U - 10 wt % Mo, U - 7.5 wt % Mo, U - 5 wt % Mo, U - 3 wt % Mo, U - 3 wt % Mo - 0.5 wt % Si, U - 3 wt % Mo - 0.1 wt % Al. The uranium alloy fuels will be assembled into two seven-rod elements. One element will be irradiated to 3000 Mwd/tonne and the other to 6000 Mwd/tonne. Other metal fuels under consideration for test are restrained uranium, and uranium alloys, Th-Pu alloys, and additional gamma-stabilized uranium alloys. An effort will be made to increase the center fuel temperature of metal fuels to 1300°F in some of the future tests.

A new series of MTR irradiations on metal fuels will investigate the variables of compostion, restraint, coring, geometry, and fabrication techniques. Two assemblies, each containing twelve Th - 13 wt % U specimens, will be irradiated at 1200°F maximum surface temperature, and 1500°F maximum center temperature. The test will contain as-cast, and cast and swagged specimens. The effect of varying carbon content and the amount of cold work will be studied. Tentative burnups desired are 7,000 and 14,000 Mwd/tonne.

Two additional assemblies, containing twelve samples each, will be irradiated to test cast U-Mo alloys containing 3, 5, 7.5, and 10 wt % Mo. Maximum temperature conditions will be 1000°F surface and 1350°F center. Desired burnups are 3,000 and 7,000 Mwd/tonne.

Five assemblies will be used to evaluate the effect of restraint, coring, and ternary additions to U-Mo alloys, on fuel behavior. These assemblies will also contain a number of unalloyed uranium specimens to be used as a basis for comparison of behavior. Molybdenum, stainless steel, and a high-strength zirconium alloy, will be used in different wall thicknesses as sleeves to restrain the swelling of unalloyed uranium. Some specimens will be cored and some will be solid slugs. The alloys to be studied are U - 3 wt % Mo, U - 3 wt % Mo - 0.5 wt % Si, U - 3 wt % Mo - 0.1 wt % Al, and U - 10 wt % Mo.

B. CERAMIC FUELS

The 19-rod UO_2 element will be irradiated to extended burnup in the present reactor position. Rods will be removed at intervals for destructive hot cell examination. Since the 19-rod UO_2 element operated successfully, the tubular oxide element will be irradiated. The present plan is to include several rods of uranium carbide in either the 19-rod UO_2 element, or a seven-rod metal element, as an initial test of uranium carbide in the SRE.

MTR irradiations of uranium carbide will investigate the effect of carbon content, irradiation temperature, and burnup on irradiation behavior. Additional compounds, and cermets have been scheduled for MTR irradiations.

XII. SUMMARY

A program has been established and is in operation to develop and evaluate potential fuels for high temperature and long burnup in the SRE. The limitations on operating temperature and burnup for various fuels will be determined after irradiation under SRE conditions in full-size fuel elements. The feasibility of the basic seven-rod element design will be established and alternate fuel element designs will be evaluated.

Promising fuel materials are selected through the evaluation of physical and mechanical properties, thermal cycling, fabrication feasibility, MTR irradiations, and irradiation data from other sites. The fuels are fabricated under controlled and recorded conditions to isolate variations in irradiation behavior initiated by fabrication. The fuels are assembled in a manner which provides known positions of defects and dimensional tolerances in a fuel element. The fuel elements are either full-size seven-rod SRE fuel elements, or alternate designs which produce the same power output as the seven-rod element. These experimental fuel elements contain thermocouples to measure central fuel temperatures. The first round of experimental fuels, which are now operating in the SRE, consists of unalloyed uranium, dilute molybdenum and zirconium alloys of uranium, thorium-uranium alloys and UO_2. The removal of fuel elements for examination in the SRE hot cell is scheduled for each fuel material to be evaluated, at regular intervals. In the hot cell, the fuel materials are evaluated for dimensional stability, density changes, and microstructural changes. New fuel materials are being preirradiation tested and promising ones will be tested in the SRE, following the same general procedure.

This program will result in the selection of the most suitable fuel materials and fuel element designs for Sodium Graphite reactors. The limiting values of burnup, temperature, and rate of power output, will be established under actual SGR conditions. Realistic specifications for fuel procurement will be established after the evaluation of the effect of fabrication variables, chemical composition, and dimensional tolerances and defects, on irradiation behavior. Due to the carefully controlled irradiation conditions, the results should provide valuable data for the studies of the fundamental effects of irradiation behavior that will allow new improved minimum cost fuel materials to be designed.

REFERENCES

1. C. Starr and R. W. Dickinson, Sodium Graphite Reactors (Addison-Wesley Publishing Co., Inc., Reading, Mass., 1958)

2. W. H. Wilson and T. O. Ziebold, "Performances of Fuel Elements at Elevated Temperatures," WASH-741, (AEC Classified Report) September 4, 1957

3. a. J. H. Kittel, "Effects of Irradiation on Power Compact of Uranium and Some Uranium-Base Alloys," ANL-5664, June 1957

   b. S. H. Paine and J. H. Kittel, "Irradiation Effects in Uranium and its Alloys." Selected Papers from the First Nuclear Engineering and Science Congress, Cleveland, 1955. (New York, Pergamon Press, 1957) 1, p. 107-117

4. I. F. Barwood et al., "The Effects of Irradiation on Some Highly Enriched Fuels," AERE-M/R-2004, (AEC Classified Report) August 28, 1956

5. J. Bokros, "Germinative Grain Growth Characteristics of Zirconium," NAA-SR-1867, June 15, 1957

6. "Metallurgy Division Quarterly Report for July, August, and September, 1953," ANL-5153, September 30, 1953

7. "Metallurgy Division Quarterly Report for January-March 1954," ANL-5257, March 31, 1954

8. a. G. Bentle, "A Physical Metallurgical Study of Thorium-Rich, Thorium-Uranium Alloys," NAA-SR-2069, January 15, 1958

   b. G. Bentle, "Study of Thorium-Uranium Alloy System," Second International Conference on the Peaceful Uses of Atomic Energy (Geneva, 1958) Paper 706

9. G. Bentle, "Annealing Effects in Th-U Alloys," NAA-SR-2969, to be published

10. B. R. Hayward and P. Corzine, "Thorium-Uranium Fuel Elements of SRE," Second International Conference on the Peaceful Uses of Atomic Energy, (Geneva, 1958) Paper 785

11. B. R. Hayward, "Dimensional Changes Resulting from Alpha-Beta Thermal Cycling of Uranium and Uranium Alloys," NAA-SR-1434, March 15, 1956

12. a. K. Smith, "Irradiation of Uranium-Fission Alloys and Related Compositions," ANL-5736, September 1957

    b. J. H. Kittle and S. H. Paine, "Effect of Irradiation on Fuel Materials," Second International Conference on the Peaceful Uses of Atomic Energy, (Geneva, 1958) Paper 1890

c. K. Smith, "Irradiation of Uranium-Fissium Alloys and Related Compounds." In Symposium on Radiation Effects on Materials (Philadelphia, American Society for Testing Materials, (1958). Vol. 3, p. 136-144

13. W. Friske, M. Binstock, and H. Kline, "Uranium Alloy Development for OMR," NAA-SR-3169, to be published

14. R. Wagner and H. Kline, "High Strength Zirconium Alloys," NAA-SR-3481, to be published

15. J.D. Eichenberg, P.W. Frank, T.J. Kisiel, B. Lustman, and K.H. Vogel, "Effects of Irradiation on Bulk UO_2," WAPD-183, October 1957. Also in Fuel Elements Conference, Paris, November 1957, TID-7546, p. 616-716

16. a. L.E. Wilkinson and B.R. Hayward, "Radiation Behavior for Fuel Materials for Sodium Graphite Reactors," NAA-SR-3411, to be published.

b. B.R. Hayward, L.E. Wilkinson and C.C. Woolsey, "Radiation Behavior of Fuel Materials for Sodium Graphite Reactors." Symposium on Radiation Effects on Materials (Philadelphia, American Society for Testing Materials, 1958). Vol. 3, p. 127-135.