U - 10 WT % MO FUEL ELEMENT

IRRADIATION IN SRE

AEC Research and Development Report
LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Price $2.00
Available from the Office of Technical Services
Department of Commerce
Washington 25, D. C.
U - 10 wt % MO FUEL ELEMENT

IRRADIATION IN SRE

By

J. L. ARNOLD
K. J. MILLER
R. M. PETERSON

ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.

CONTRACT: AT(11-1)-GEN-8
ISSUED: AUG 14, 1965
DISTRIBUTION

This report has been distributed according to the category "Metals, Ceramics, and Materials" as given in "Standard Distribution Lists for Unclassified Scientific and Technical Reports" TID-4500 (41st Ed.), May 1, 1965. A total of 590 copies was printed.

ACKNOWLEDGMENT

The authors are deeply indebted to the many colleagues at Atomics International who have contributed significantly to the conception, design, irradiation, and examination of this fuel element. Special thanks are due to A. R. Schmitt and B. R. Hayward for their direction and advice; to R. M. Willard, R. D. Hahn, and A. D. Schwartz for their advice and consultation; to personnel at the Atomics International Hot Laboratory for their advice and examination efforts, and to personnel at the Sodium Reactor Experiment for their efforts in the irradiation and data reduction for this element.
CONTENTS

Abstract......................................................... 5
I. Introduction .................................................. 7
II. Experiment Description ....................................... 9
III. Irradiation History .......................................... 11
IV. Post-Irradiation Examination, Procedures and Results ....... 13
   A. Variable Orifice ........................................... 13
   B. Gamma Scanning ........................................... 13
   C. Fission Gas Release ....................................... 14
   D. Disassembly ............................................... 15
   E. Visual Examination of the Fuel ................................
   F. Dimensional Measurements ................................... 17
   G. Density Measurements ....................................... 19
   H. Metallography ............................................... 19
      1. Fuel .................................................. 19
      2. Cladding .............................................. 21
I. Carburization and Bend Test of Cladding ......................... 21
V. Discussion and Evaluation ..................................... 23
   A. Irradiation Stability of the U-10 Wt % Mo Alloy ............. 23
      1. General Background Information .......................... 23
      2. SU-9 Results ........................................... 25
      3. Fuel Slug Cracking and Warping .......................... 28
   B. Variable Orifice ........................................... 31
   C. Fuel Rod Cladding ......................................... 31
   D. HNPF Core I U-10 Wt % Mo Fuel Elements ...................... 32
VI. Conclusions ................................................... 34
References ....................................................... 35

TABLES
1. Summary of SRE Operating History While SU-9 Fuel Element was
   Irradiated .................................................. 12
2. Physical Changes in Rods No. 4 and No. 5 .......................... 18
3. Density Changes of Fuel in Rod No. 5 ........................ 19
4. Operating Parameters of Fuel Metallographic Specimens .......... 21

NAA-SR-11121
3
# FIGURES

<table>
<thead>
<tr>
<th>FIGURE</th>
<th>DESCRIPTION</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>SU-9 Fuel Element</td>
<td>10</td>
</tr>
<tr>
<td>2.</td>
<td>SU-9 Fuel Element Axial Temperature Profile</td>
<td>12</td>
</tr>
<tr>
<td>3.</td>
<td>Schematic Arrangement of the Fission Gas Collection Apparatus</td>
<td>14</td>
</tr>
<tr>
<td>4.</td>
<td>Pinhole Gamma-Ray Camera Photograph of the Interface of the Middle Two Slugs in Rod No. 5</td>
<td>15</td>
</tr>
<tr>
<td>5.</td>
<td>View of the Fuel and the Cladding from Rod No. 4 of the SU-9 Experiment</td>
<td>17</td>
</tr>
<tr>
<td>6.</td>
<td>Physical Changes of Fuel in Rods No. 4 and No. 5</td>
<td>20</td>
</tr>
<tr>
<td>7.</td>
<td>Uranium-Molybdenum Phase Diagram</td>
<td>24</td>
</tr>
<tr>
<td>8.</td>
<td>Time-Temperature-Transformation Curves for Uranium-Molybdenum Alloys</td>
<td>24</td>
</tr>
<tr>
<td>9.</td>
<td>Theoretical Critical Fission Rate as a Function of Temperature</td>
<td>25</td>
</tr>
<tr>
<td>10.</td>
<td>Typical Pre-Irradiation Microstructure of U-10 Wt % Mo Alloy (Dark Areas Indicate Coring)</td>
<td>26</td>
</tr>
<tr>
<td>11.</td>
<td>Post-Irradiation Microstructure of the U-10 Wt % Mo Alloy</td>
<td>29</td>
</tr>
<tr>
<td>12.</td>
<td>Typical Microstructures of Fuel Rod Cladding</td>
<td>32</td>
</tr>
<tr>
<td>13.</td>
<td>Dimensional Change Data for U-10 Wt % Fuel Slugs from SU-9 Experiment</td>
<td>33</td>
</tr>
</tbody>
</table>
ABSTRACT

The fuel element assembly was successfully irradiated in the SRE to a maximum burnup of 5300 Mwd/MTU, at a peak fission rate of approximately $1.5 \times 10^{13}$ fissions/cm$^3$-sec and a maximum central temperature near 1200°F. The post-irradiation examination of the fuel rods showed a low fission gas release of 0.1% and fuel dimensional changes ranging from -0.2% to +2.2% with corresponding density decreases from 0.2% to 6.4%. The fuel slugs which had burnups less than 4,000 Mwd/MTU were in excellent condition with maximum dimensional changes of less than +0.6%. Fuel slugs with burnups greater than 4,000 Mwd/MTU had extensive surface cracking and greater fuel dimensional changes.

Metallographic examination of the fuel showed that the slugs which had experienced the extensive surface cracking, had most of their structure transformed to the $\alpha + \gamma'$ phase. The structure of the fuel which had no surface cracking and small dimensional changes was essentially in the $\gamma$ phase. There was evidence that the fuel which operated at lower temperatures had been homogenized as a result of fission events, the mechanism being the radiation induced phase reversion. Metallographic examination of the cladding showed that there were no compatibility problems associated with this system, although there was some slight carburization as shown by chemical analysis.

In predicting the limiting life of the HNPF Core I U-10 wt % Mo fuel elements, it appears that a lifetime limit below a maximum burnup of 4,000 Mwd/MTU is highly unlikely, and a maximum burnup in excess of 6,000 Mwd/MTU may be achieved. Because of uncertainties in application of the SU-9 data to Hallam, a more quantitative estimate is believed unwarranted. Periodic examination of the Core I U-10 wt % Mo fuel elements will be required to form the basis for a more exact estimate of allowable burnup.
I. INTRODUCTION

As part of a development program for the Hallam Nuclear Power Facility (HNPF), a partial prototype of a Hallam Core I fuel element (SU-9) was fabricated and irradiated in the Sodium Reactor Experiment (SRE). The purpose of this irradiation experiment was to test the mechanical aspects of the fuel element design and to define the behavior of the U-10 wt % Mo fuel alloy used in the Hallam Core I fuel elements at temperatures and fission rates representative of those expected in the first core loading.

The first core for HNPF contains U-10 wt % Mo fuel elements and utilizes a variable orifice for controlling the temperature in each of these elements. Hydraulic tests had been conducted out-of-pile on this variable orifice, but the feasibility of using it under sodium reactor conditions had never been demonstrated. One of the purposes of inserting the SU-9 fuel element into the SRE was to test this variable orifice concept under sodium reactor conditions to determine:

a) the response and sensitivity of the orifice to changes in orifice position during steady state reactor operation;

b) the response and sensitivity of the orifice to transient reactor conditions;

c) the possibility of binding of the orifice drive tube due to sodium vapor condensation and/or fouling.

Although the variable orifice and drive mechanism used in this experiment were not identical in design to those used in Hallam Core I, they were similar in terms of sensitivity, drive tube connection, etc.

Following completion of the testing of the variable orifice in 1962, the irradiation of the SU-9 fuel element was continued to evaluate the irradiation performance of the U-10 wt % Mo fuel slugs which were equivalent in composition and geometry to the fuel used in the first core loading of the HNPF.

Early studies on the irradiation swelling behavior of uranium metal showed that by adding small amounts of alloying elements, the swelling resistance could be markedly increased.\(^1\) Screening tests were performed on many of
these alloys by various companies\(^2, 3, 4\) and on this basis Atomics International selected the U-10 wt \% Mo alloy for Hallam Core I fuel. Since that time, a comprehensive irradiation program was conducted to determine the effects of the variables that would be encountered during reactor operations.\(^5, 6\) The basic objective of this program was to determine the burnup that could be achieved without excessive fuel swelling. Due to the uncertainties in the data at the lower burnups, the predicted life of the fuel elements in Hallam Core I was given as 4000 ± 2000 Mwd/MTU. The SU-9 experiment was the first test of this fuel under sodium reactor conditions and as such should best represent the irradiation performance of this fuel in the HNPF.
II. EXPERIMENT DESCRIPTION

The SU-9 fuel element, shown schematically in Figure 1, consisted of two basic parts, the fuel bundle assembly and a support assembly.

The support assembly consisted of a shield plug, hanger tube, and orifice housing assembly. Except for the inclusion of the orifice control drive unit, this shield plug was identical to the regular SRE shield plugs. Control of the orifice was accomplished by a worm gear drive assembly operated from the top shield by means of a special wrench. This worm gear drive actuated a drive rod attached to a flexible cable which in turn was fastened to the drive rod for raising or lowering the orifice bulb. The orifice housing assembly was located at the top of the fuel element and the sodium exit temperature was controlled by controlling the sodium flow through the element. This control was achieved by varying the elevation of the tapered orifice bulb located in the tapered section of the orifice housing assembly.

The fuel bundle assembly consisted of six fuel rods numbered 2 through 7 and equally spaced around a 5/8 in. diameter support tube. Eight fuel rods were initially fabricated for this experiment, but only rods No. 2 through No. 7 were used. Fuel rod spacing was maintained by wrapping each rod with 1/8 in. diameter tubing. The center tube was designed to provide the standard connection necessary for the modified SRE Lower Guide.

Each of the fuel rods contained a column of twelve 0.590 in. diameter by 6 in. long cast U-10 wt % Mo slugs which were sodium bonded to 0.010 in. thick cladding. The cladding was 0.660 in. outer diameter, welded and drawn, Type 304 stainless steel tubing. This provided approximately a 0.025 in. sodium annulus which was used both as a heat transfer medium and expansion space for fuel growth. A helium filled space approximately 17 in. long was left above the fuel column to provide space for collection of fission gases and for accommodation of sodium displacement due to fuel growth. Temperatures in the element were recorded with thermocouples positioned both in the fuel and on the cladding surface at depths varying from 18 to 54 in. from the top of the fuel column.

The in-fuel thermocouples which were protected from the fuel by a tantalum foil sheath were located in specially prepared slugs which had a 0.205 in. diameter axial hole through them. Four of the six rods of this element were equipped with in-fuel thermocouples.
Figure 1. SU-9 Fuel Element
III. IRRADIATION HISTORY

The SU-9 fuel element was irradiated in the SRE over a period of three years. The element was originally inserted in November, 1961 and was ultimately removed in January, 1964 because of malfunction of the orifice control drive. During the intervening time the SU-9 fuel element was removed three times and examined in the SRE maintenance cell. Visual examination of the fuel rod cladding revealed no observable surface defects or signs of operational damage. No diameter changes were noted on any of the fuel rod claddings.

The variable orifice control drive malfunctioned three times during the course of the irradiation. These malfunctions were mainly the result of hardware deficiencies in that the star adapter, which actuated the position counter in the orifice control box, would become loose and result in erroneous orifice position indication. The first malfunction occurred in April, 1963 when the drive rod was forced down until it became disengaged from the drive gear. This was repaired by removing the drive unit and re-engaging it with the drive rod. Due to a false setting on the position indicator, the brass retaining sleeve on the adjusting side of the drive was stripped when the orifice was adjusted beyond the full open position in June, 1963. The fuel element was removed from the reactor in January, 1964, after the third failure. (The examination of the variable orifice is discussed in section IV-A.)

The average burnup of the SU-9 fuel element upon removal from the SRE in January, 1964 was calculated to be 3350 Mwd/MTU. Using a peak-to-average flux ratio of 1.65 (as determined from post-irradiation gamma scan profiles) a peak burnup of 5300 Mwd/MTU was calculated. Prior to February, 1963 the SRE operated intermittently and at relatively low power levels (10 megawatts thermal or less) such that an average burnup of only 425 Mwd/MTU was accumulated to this time. However, during the last 11 months of irradiation, the SRE operated consistently at higher power levels (15 to 20 megawatts thermal) during which time the major portion of the fuel burnup was accumulated. Table I shows the percentage of time that the SRE operated at the various power levels during the period that the SU-9 fuel element was in operation.
TABLE 1
SUMMARY OF SRE OPERATING HISTORY

<table>
<thead>
<tr>
<th>Power Level</th>
<th>5 Mwt</th>
<th>10 Mwt</th>
<th>15 Mwt</th>
<th>20 Mwt</th>
</tr>
</thead>
<tbody>
<tr>
<td>Percentage of Time at Power Level</td>
<td>26%</td>
<td>43%</td>
<td>21%</td>
<td>10%</td>
</tr>
</tbody>
</table>

Figure 2 shows the axial temperature profiles for the fuel centerline and surface, cladding surface, and the bulk sodium coolant. These temperatures were based on 550°F inlet coolant temperature, 870°F outlet coolant temperature, and a reactor power level of 20 megawatts.

![Temperature Profile Diagram](image)

Figure 2. SU-9 Fuel Element Axial Temperature Profile

Upon final removal from the SRE, the element was washed and stored in the SRE storage cell until disassembly of the element could be initiated at the Atomics International Hot Laboratory (AIHL) and the SRE maintenance cell.
IV. POST-IRRADIATION EXAMINATION

A. VARIABLE ORIFICE

The post-irradiation examination was initiated in April, 1964 in the SRE maintenance cell. The fuel bundle assembly was removed from the hanger rod and was shipped to AIHL for destructive examination.

The orifice housing and hanger rod assemblies were cut into ten sections, each about one ft long. After visually examining each section it was concluded that the malfunction of the orifice control drive was located in the shield plug. Subsequent examination of the shield plug was initiated by removing the drive gear assembly. It was found that the drive rod, which had been swaged onto the flexible cable, had pulled loose.

B. GAMMA SCANNING

After the fueled portion of the element was transferred to the Hot Lab three fuel rods were removed from the element for detailed examination. Fuel rods No. 4 and No. 5 were thoroughly examined and No. 7 was used for obtaining a fission gas analysis.

Fuel rod No. 5 was subjected to a detailed gamma scan analysis. The fuel rod was mounted in a holder which in turn was mounted on a motor driven stage which moved the rod back and forth in front of a 1/8 x 1/8 x 42 in. orifice through the cell wall. The stage was adjusted so that the centerline of the fuel rod intersected the centerline of the orifice at all times. Outside the cell wall was located a 2 x 2 in. sodium iodide crystal and photomultiplier tube which were used to measure the relative gamma activity of the fuel rod. A single channel pulse height analyzer was used to discriminate and record the gamma activity. The analyzer was set on the counting peak of Zr$^{95}$, a gamma emitter with a long half life, to obtain the gamma activity profile for the fuel rod. This relative gamma activity profile was used to determine the peak-to-average burnup ratio for the rod and to construct a sampling plan for chemistry and metallographic specimens of the fuel and the cladding.
C. FISSION GAS RELEASE

Each of the three fuel rods was individually placed in the fission gas collection chamber (Figure 3). When the chamber had been evacuated, the rod was punctured and the fission gas was collected in glass vials which contained activated charcoal. These traps had previously been out-gassed and were chilled in liquid nitrogen during the gas collection process. An acetone-dry ice cold trap in the line ahead of the charcoal traps effectively removed any condensable vapors present in the system. Helium carrier gas was used to flush the system to assure collection of all the released fission gas. When gamma radiation emanating from charcoal traps reached a constant level and the pressure within the chamber returned to a point near the level noted prior to puncturing the rod, the glass vials were sealed off. The contents of these vials were then analyzed for Krypton-85. Based on an average burnup per rod of 3350 Mwd/MTU, the following results were obtained for the percent of Krypton-85 gas released:

a) Rod No. 4 - 0.053%
b) Rod No. 5 - 5.5 x 10^{-5} %
c) Rod No. 7 - 0.112%

Figure 3. Schematic Arrangement of the Fission Gas Collection Apparatus
Subsequent investigation revealed that the fission gas from rod No. 5 had been inadvertently released when the rod was removed from the fuel bundle. Thus the data obtained by fission gas collection and analysis for this rod were therefore disregarded.

D. DISASSEMBLY

The fuel slugs were removed from fuel rods No. 4 and No. 5 by slitting the cladding longitudinally at two orientations 180° apart, using an Elox* cutting machine. The fuel rods were too long (94-7/8 in.) to make one cut over the entire length. Therefore, a pinhole gamma ray camera was utilized to determine the position of the interface between the middle two slugs in each fuel rod. A camera located immediately outside the cell wall was used to photograph the fuel rod with both gamma ray sensitive film and light sensitive film. By superimposing the two negatives, the exact position of the slug interface could be determined. Figure 4 shows the pinhole camera gamma ray photograph taken of the middle two slugs in rod No. 5. The cladding was cut at this position with a tubing cutter and the longitudinal slitting of the cladding was then accomplished without difficulty.

Figure 4. Pinhole Gamma-Ray Camera Photograph of the Interface of the Middle Two Slugs in Rod No. 5
(Note thermocouple hole in slug on bottom)

*Trade name for electrical discharge method of machining conducting materials.
E. VISUAL EXAMINATION OF THE FUEL

Upon completion of the clad slitting operation, the fuel slugs and the cladding from rods No. 4 and No. 5 were rinsed in n-Butyl alcohol to remove the remaining sodium from the surfaces. The fuel slugs were then placed in a vacuum distillation furnace where they were heated under a vacuum (5 x 10^-3 mm Hg) at 500°F for four hours to remove any sodium which had impregnated the fuel slugs. After this treatment, much of the fuel still had a thin white film over most of the surface, which was believed to be a sodium butylate deposit. Most of the remaining film was removed after subsequent rinsing in ethyl alcohol and scrubbing with a fine wire brush.

The top six slugs in both fuel rods appeared to be in very good condition with no signs of surface cracking (Figure 5A). The top six slugs in rod No. 5 had an in-fuel thermocouple hole along the axial centerline while the top six slugs in rod No. 4 did not. The appearance of the fuel slugs from the two rods was quite similar. The fuel in the bottom half of the fuel rods, especially the seventh through the tenth slugs from the top of the column, had experienced extensive cracking, in both longitudinal and transverse directions (Figure 5B). These cracks appeared to be the result of thermal stress since they were observed to have started at the fuel surface and propagated into the central portions of the slug. The cracks also appeared to be the most severe in slugs which had operated at the greatest ΔT, across the fuel radius and at the highest fission rate. The bottom two slugs in both rods were free of this general gross cracking, but did have some fine hairline cracks which were oriented in both longitudinal and transverse directions. All of the fuel had experienced warping (or bowing) to varying degrees, but there were no signs of deformation, ridges, or bulges in the 0.010 in. thick, stainless steel cladding. This warping did not visibly strain the cladding, and apparently was restrained by the cladding.

Several of the fuel slugs were broken into two, and in some cases three, separate pieces, apparently because of a propagation of the transverse cracks completely through the slug. The seventh slug from the top of rod No. 5 and the sixth slug from the top of rod No. 4, which did not have the extensive, gross cracking typical of the other broken slugs, were broken during disassembly.
F. DIMENSIONAL MEASUREMENTS

Post-irradiation dimensional measurements were taken on all of the fuel slugs in rods No. 4 and No. 5 to determine the dimensional changes. A profilometer was used to obtain the diameters of the fuel slugs which were not broken. This device consists of a modified lathe bed with two end stocks to support the object being measured, and two highly sensitive linear displacement transducers which are translated axially along the fuel slug. These transducers transmit a signal to a dual channel recorder which prints a continuous profile of the fuel surface outside the cell. Maximum bowing of the fuel slugs was determined from the profiles obtained in this manner. These values were given as maximum deviations from a theoretically straight slug. Vernier calipers were used to measure the diameters of the fuel slugs which were not completely intact. The profilometer was considered to have an accuracy of ± 0.0003 in, while the vernier calipers were accurate to ±0.0015 inches.

Vernier calipers were also used for obtaining rough length measurements. Broken fuel slugs were

Figure 5. View of the Fuel and the Cladding from Rod No. 4 of the SU-9 Experiment
fitted together and then the length measurements were made. Hence the accuracy of the length changes determined for the broken slugs is questionable.

Table 2 gives the diameter changes, length changes and the magnitude of warping for the fuel from rods No. 4 and No. 5. Also listed in this table are the average dimensional changes, defined as:

$$\Delta L_{\text{average}} = \frac{2 \times \% \text{ change in diameter} + \% \text{ change in length}}{3} \ldots (1)$$

<table>
<thead>
<tr>
<th>Slugs No.</th>
<th>Avg. Dim. Change (2 ΔDia. + ΔL₃)</th>
<th>Maximum Bowing (In.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rod No. 4</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1 (Top)</td>
<td>-0.05</td>
<td>-0.27</td>
</tr>
<tr>
<td>2</td>
<td>0.25</td>
<td>-0.39</td>
</tr>
<tr>
<td>3</td>
<td>0.40</td>
<td>-0.25</td>
</tr>
<tr>
<td>4</td>
<td>0.79</td>
<td>+0.30</td>
</tr>
<tr>
<td>5</td>
<td>1.14</td>
<td>+0.32</td>
</tr>
<tr>
<td>6*</td>
<td>2.50</td>
<td>+1.33</td>
</tr>
<tr>
<td>7</td>
<td>1.72</td>
<td>+1.37</td>
</tr>
<tr>
<td>8</td>
<td>1.95</td>
<td>+1.47</td>
</tr>
<tr>
<td>9*</td>
<td>1.94</td>
<td>+1.35</td>
</tr>
<tr>
<td>10</td>
<td>0.72</td>
<td>+0.59</td>
</tr>
<tr>
<td>11</td>
<td>0.23</td>
<td>-0.08</td>
</tr>
<tr>
<td>12 (Bottom)</td>
<td>0.08</td>
<td>-0.15</td>
</tr>
<tr>
<td>Rod No. 5</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1 (Top)</td>
<td>-0.17</td>
<td>-0.20</td>
</tr>
<tr>
<td>2</td>
<td>-0.06</td>
<td>-0.03</td>
</tr>
<tr>
<td>3</td>
<td>0.20</td>
<td>-0.08</td>
</tr>
<tr>
<td>4</td>
<td>0.56</td>
<td>0.32</td>
</tr>
<tr>
<td>5</td>
<td>0.53</td>
<td>0.45</td>
</tr>
<tr>
<td>6</td>
<td>0.82</td>
<td>0.74</td>
</tr>
<tr>
<td>7*</td>
<td>3.06</td>
<td>1.50</td>
</tr>
<tr>
<td>8</td>
<td>2.44</td>
<td>2.02</td>
</tr>
<tr>
<td>9*</td>
<td>2.65</td>
<td>1.28</td>
</tr>
<tr>
<td>10*</td>
<td>1.60</td>
<td>0.79</td>
</tr>
<tr>
<td>11</td>
<td>0.35</td>
<td>0.34</td>
</tr>
<tr>
<td>12 (Bottom)</td>
<td>0.28</td>
<td>-0.07</td>
</tr>
</tbody>
</table>

*Broken slugs.

**ERRATA AT END OF ITEM**

NAA-SR-11121

18
This term is approximately equal in magnitude to $1/3$ of the density change assuming small density changes and isotropic fuel growth. Figure 6 gives a schematic representation of these data.

G. DENSITY MEASUREMENTS

The density determinations were performed by Archimede's method and using n-octyl alcohol as the liquid medium. The density measurements were made on the first, sixth, seventh, eighth, ninth, and twelfth slugs from the top of the fuel column on rod No. 5. The pre- and post-irradiation densities of these fuel slugs (normalized to $20^\circ$C) and their respective percentage changes are given in Table 3. Figure 6 also shows a comparison of the values for the dimensional changes as calculated from the density changes and a comparison of these values with the measured dimensional changes.

<table>
<thead>
<tr>
<th>Slug Number (from top)</th>
<th>Pre-Irradiation Density (gm/cc)</th>
<th>Post-Irradiation Density (gm/cc)</th>
<th>Percent Density Change (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 (Top)</td>
<td>17.138</td>
<td>17.114</td>
<td>-0.14</td>
</tr>
<tr>
<td>6</td>
<td>17.130</td>
<td>16.669</td>
<td>-2.69</td>
</tr>
<tr>
<td>7</td>
<td>17.195</td>
<td>16.115</td>
<td>-6.28</td>
</tr>
<tr>
<td>8</td>
<td>17.201</td>
<td>16.113</td>
<td>-6.33</td>
</tr>
<tr>
<td>9</td>
<td>17.153</td>
<td>16.242</td>
<td>-5.31</td>
</tr>
<tr>
<td>12 (Bottom)</td>
<td>17.196</td>
<td>17.070</td>
<td>-0.73</td>
</tr>
</tbody>
</table>

H. METALLOGRAPHY

1. Fuel

Three fuel specimens from rod No. 5 were selected for metallographic examination. These specimens were 1/2 in. long disks cut from the center of the first, seventh, and twelfth slugs from the top of this rod. For convenience, these specimens are referred to as M-1, M-7, and M-12, respectively. The operating conditions for these three specimens are given in Table 4. Specimen M-1 contained an axial 0.205 in. diameter in-fuel thermocouple hole while M-7
Figure 6. Physical Changes of Fuel in Rods No. 4 and No. 5
TABLE 4
OPERATING PARAMETERS OF FUEL METALLOGRAPHIC SPECIMENS

<table>
<thead>
<tr>
<th>Specimen I.D.</th>
<th>Distance From Top of Fuel Column (In.)</th>
<th>Peak Fuel Centerline Temperature (%)(°F)</th>
<th>Peak Fuel Surface Temperature (%)(°F)</th>
<th>Burnup (Mwd/MTU)</th>
<th>Peak Fission Rate (Fissions/cm³·sec)</th>
</tr>
</thead>
<tbody>
<tr>
<td>M-1</td>
<td>3</td>
<td>1020</td>
<td>900</td>
<td>1200</td>
<td>0.4 x 10¹³</td>
</tr>
<tr>
<td>M-7</td>
<td>39</td>
<td>1180</td>
<td>750</td>
<td>4800</td>
<td>1.4 x 10¹³</td>
</tr>
<tr>
<td>M-12</td>
<td>69</td>
<td>840</td>
<td>620</td>
<td>2800</td>
<td>0.8 x 10¹³</td>
</tr>
</tbody>
</table>

and M-12 were solid slugs. The specimens were mounted in bakelite, ground, polished and subsequently were etched with a solution of one part chromic acid, one part ammonium bifluoride, four parts nitric acid, and thirty-two parts water. (Results of the examination of the fuel are discussed in Section V-A-2.)

2. Cladding

Three cladding specimens from rod No. 5 were selected for metallographic examination. These specimens were 1/2 in. long sections taken at positions adjacent to the first, sixth, and twelfth slugs from the top of this rod. For convenience, these specimens are referred to as C-1, C-2, and C-3, respectively. When this element was operating at full power, these cladding specimens were at temperatures of approximately 880°, 750°, and 600°F, respectively. The specimens were mounted in bakelite, ground, polished, and etched electrolytically using a 10% oxalic acid solution. (Results of the examination of the cladding are discussed in detail in Section V-C.)

I. CARBURIZATION AND BEND TEST

Three cladding specimens for chemical analysis and three cladding specimens for bend tests were also cut from rod No. 5. These specimens were taken from positions immediately adjacent to the cladding metallographic specimens.

The cladding specimens were analyzed for total carbon content by using a conductometric carbon analyzer. The total carbon content of specimens taken 3, 36, and 70 in. from the top of the fuel column was 0.24, 0.13, and 0.18 wt %, respectively.
respectively. The pre-irradiation carbon content for this material was 0.08 wt % maximum. This shows that there was a substantial increase in the total carbon content of this cladding.

A crude bend test was performed on the three cladding specimens in which a specimen was placed in a bench vise and bent back and forth (at a 90° angle) repeatedly until broken. An unirradiated specimen of equivalent material required approximately 40 bends before breaking. The irradiated specimens required 20, 9 and 7 bends to break for the top, middle, and bottom portions, respectively, indicating some loss of ductility.
V. DISCUSSION AND EVALUATION

A. IRRADIATION STABILITY OF THE U-10 WT % Mo ALLOY

1. General Background Information

   The irradiation performance of a fuel depends upon a complex interrelation of several variables, each of which plays a significant role. A complicating factor with the U-10 wt % Mo alloy is a phase change near the temperature range of interest. In addition an effort was made to include multiple objectives in the experiment because of the cost of performing in-pile tests. As a result of these considerations, some difficulty was encountered in attempts to isolate the variables as separate entities.

   The U-Mo system (Figure 7)\(^{(7)}\) has an eutectoid transformation at 1065°F and 10.5 wt % molybdenum. The U-10 wt % Mo alloy possesses a disordered, single-phase (\(\gamma\)) structure at temperatures above 1100°F, but transforms, under equilibrium conditions, to a two-phase (\(\alpha + \gamma'\)) structure at temperatures below 1065°F. The gamma prime phase is an ordered form of the gamma phase. This transformation, upon cooling, is sluggish, requiring approximately 20 hours at 900°F to reach the nose of the time-temperature-transformation (T-T-T) curve shown in Figure 8.\(^{(8)}\) Therefore, the high temperature \(\gamma\) phase of this alloy may be retained at room temperature by quenching or even by comparatively slow cooling, such as in casting the fuel slugs.

   The radiation stability of the U-10 wt % Mo alloy depends to a considerable extent upon its ability to retain the gamma phase during irradiation. When irradiations are conducted at temperatures below ~1065°F, this alloy tends to transform to the alpha and gamma-prime phases which are thermodynamically stable at these lower temperatures. The effect of fission-induced-displacements and thermal spikes is to oppose this tendency, the spikes tending to disorder the gamma-prime phase and to produce a homogeneous composition of uranium and molybdenum in the gamma phase.\(^{(9,10)}\) The critical fission rate is that needed to produce the minimum number of displacement and thermal spikes required to maintain the gamma phase in opposition to the thermodynamic tendency of the alloy to transform to the alpha and gamma-prime phase. This critical fission rate is temperature dependent and decreases with decreasing temperature below the nose of the T-T-T diagram shown in Figure 9.\(^{(5)}\)
Figure 7. Uranium-Molybdenum Phase Diagram

Figure 8. Time-Temperature-Transformation Curves for Uranium-Molybdenum Alloys
2. SU-9 Results

Microscopic examination of the fuel, prior to irradiation, indicated that the cooling rate of the as-cast material was sufficient to maintain the metastable gamma phase at room temperature, with the resultant structure showing some evidence of coring. Figure 10 shows typical pre-irradiation microstructures of the fuel used in this experiment. Because of their faster cooling rate the hollow fuel slugs had a more cored or inhomogeneous structure than the solid fuel slugs. The reasons for this faster cooling rate were: (a) less material in the fuel slug, and (b) an additional heat sink present in the graphite rod used in forming the thermocouple hole.

Metallographic fuel specimen M-1 was a transverse section of a hollow fuel slug. This specimen operated at low fission rates and relatively high temperatures, such that the observed change in microstructure was thought to be due primarily to the thermodynamic tendency of the alloy to transform to the lower temperature stable phases. The entire specimen had a cored structure. Some type of transformation had occurred at the grain boundaries, but the transformed material did not have the typical lamellar structure of the \((\alpha + \gamma')\) phase (see [Further details or references].)
A. Solid Fuel Slug

B. Hollow Fuel Slug

Figure 10. Typical Pre-irradiation Microstructure of U-10 wt % Mo Alloy (Dark areas indicate coring)
Figure 11). The inhomogeneous nature of the pre-irradiation structure of the hollow fuel slugs probably contributed to this type of transformation. The molybdenum rich center of a grain would indeed transform less readily than the peripheral regions since increasing the molybdenum content moves the nose of the T-T-T curve to the right. The six-point stars seen at 100X in the center of this specimen are thought to be caused by this transformation initiating at the grain boundaries in an orderly manner, leaving the center in the untransformed γ phase.

Metallographic specimen M-7 was a transverse section of a solid fuel slug. This specimen operated at relatively high fission rates and temperatures. While operating at full reactor power, the microstructure of the central portion of this specimen should have been a single phase γ structure since the temperature of this portion of the fuel specimen was above 1065°F. There also should have been a well defined, outer ring of transformed α + γ' since, at the fuel surface temperature (~750°F), the original γ phase material would start to transform in approximately 50 hr, assuming no irradiation induced phase reversion. During the last 700 hr of irradiation, the reactor was operated at only three-fourths of full power and the resulting temperature of the central portion of the fuel was ≤1050°F. From the U-Mo TTT diagram (Figure 8), it may be seen that the transformation of a U-10 wt % Mo composition alloy from an all γ to an α + γ' structure requires ~20 to ~70 hr, when operating over the temperature range of 1065 to 750°F. The same transformation takes from ~70 to ~700 hr if the temperature range is 750 to 650°F. Therefore, the microstructure of this specimen should have been entirely that of a lamellar α + γ' structure considering only the thermodynamic properties of this alloy. The resultant structure of this specimen, however, was lamellar α + γ' only in the central regions with a well defined peripheral ring of single phase γ structure. The presence of this single phase structure was attributed to homogenization of the alloy in this region due to the higher fission rates and sluggish transformation at the lower temperatures nearer the surface of the fuel slug. This phenomenon is in agreement with that predicted by the theoretical critical fission rate curve(5) shown in Figure 9. The fact that this fuel slug had experienced substantially larger dimensional changes than the other two metallographic specimens was attributed primarily to the presence of the α + γ' phase.
Metallographic specimen M-12 was a transverse section of a solid fuel slug. This specimen operated at moderate fission rates and relatively low temperatures. The center of the specimen had transformed only slightly and still possessed a cored structure. This transformed structure did not have the typical lamellar structure, but rather had a cellular type of transformed structure at the grain boundaries. The peripheral portion of this specimen was almost entirely a single phase $\gamma$ structure. The fission rate in this region was apparently sufficient to homogenize the fuel operating at these lower temperatures. The resultant microstructure of the fuel in the peripheral region was more homogeneous than the pre-irradiation microstructure.

3. Fuel Slug Cracking and Warping

The fuel which had experienced the gross cracking was limited primarily to that which operated at the highest power ratings. The cracking was both transverse and longitudinal in nature and appeared to be the result of a thermal stress relief type of phenomenon. The cracks appeared to have initiated at the surface, propagated toward the center, and terminated in the central region. Thermal stress calculations were performed and the magnitude of the outer fiber stresses created was consistent with the degree of cracking observed. Although the absolute values calculated ($\sim 25,000$ psi maximum) were less than required to cause this cracking on the unirradiated fuel, it has been shown that the mechanical properties of this alloy deteriorate significantly at modest burnups.\(^{(11)}\)

All of the fuel slugs in both of the rods examined had experienced varying degrees of warping. The magnitude of this warping, as given in Table 1, did not seem to follow any function of burnup or temperature, and in all cases appeared to be contained within the gap provided for expansion of the fuel. At operating temperatures, the strength of this fuel alloy is very low and little restraint would be required to inhibit the warping. It is believed that warping of the fuel progressed until a point contact was made with the cladding and then the restraint offered by the cladding prevented further warping. This type of behavior was observed in SRE Core I irradiations of unalloyed uranium$^{(12)}$ where the warping progressed only until the fuel contacted the cladding. No signs of cladding deformation or strain were observed on any of the rods of this element.
Figure 11. Post-Irradiation Microstructure of the U-10 Wt % Mo Alloy

NAA-SR-11121

29
B. VARIABLE ORIFICE

Tests to evaluate the performance of the variable orifice showed that the response and sensitivity of the orifice to changes in position during steady state reactor operation were within 20% of that expected over most of the operating range. The channel outlet temperature varied from 18% higher than the SRE mixed mean outlet temperature with the orifice closed, to 7% lower with the orifice fully open. Another objective of this part of the experiment was to assess the possibility of the orifice drive binding or freezing due to condensation of sodium vapors and/or other types of fouling. The SU-9 orifice drive did not bind or freeze. However, because of the frequent washings which the SU-9 received, this lack of binding cannot necessarily be considered indicative of normal power reactor performance.

C. FUEL ROD CLADDING

The SU-9 fuel element handling at the SRE provided an excellent test of sensitized Type 304 stainless steel fuel rod cladding. The fuel rod cladding was known to be sensitized prior to its irradiation and thus special precautions were exercised during storage. These consisted of evacuating and then back-filling the storage cell with helium bubbled through NaK to insure a dry inert atmosphere. This procedure was repeated four times. Whenever necessary, the SU-9 was washed and examined in the SRE maintenance cell which does not have a controllable atmosphere. In spite of these examinations, the cladding did not incur any visible corrosion as evidenced by post-irradiation metallographic examination.

The fuel rod cladding in the element performed successfully as a containment for fission gases. Metallographic examination of the cladding material revealed a highly sensitized structure with the carbides precipitated along grain boundaries and slip lines (Figure 12). There were no signs of fuel-cladding reactions in any of the specimens examined. Conductometric carbon analyses of the cladding showed that there had been some carburization of the stainless steel but this was not observed metallographically as a case type of carburization. Due to the lack of testing equipment available when the element was examined, the change in the mechanical properties was not determined quantitatively. Qualitatively it was found that the cladding was still ductile as indicated by the previously described bend tests.
Figure 12. Typical Microstructures of Fuel Rod Cladding

D. HNPF CORE I U-10 Wt % Mo FUEL ELEMENTS

Since the SU-9 element utilized the same fuel geometry as HNPF Core I and was operated under sodium reactor conditions, it should represent the irradiation performance of the U-10 wt % Mo fuel in HNPF Core I. Previous irradiations of this fuel\(^6\) were conducted in a water cooled test reactor, and therefore are not truly representative of conditions expected in a sodium cooled power reactor. The present design conditions for the HNPF Core I fuel rods allow for a dimensional change of +2.5% averaged over the entire length of the fuel column. This dimensional change corresponds to fuel swelling which would increase the internal pressure sufficiently to stress the cladding to its yield point. A fuel diameter change of +8.5% at any position along the length of the fuel rod also was considered to be a limiting condition where firm fuel-cladding contact would be made. Due to uncertainties in the swelling behavior of the fuel at lower burnups, a limiting burnup of 4000 ± 2000 Mwd/MTU was specified, based on the previous series of irradiation tests.\(^6\) A plot of the average dimensional change versus burnup for the fuel slugs in the SU-9 element is given in Figure 13. It can be seen from this plot that the limiting peak burnup would
probably not be reached until at least 6000 Mwd/MTU. This result is not inconsistent with the earlier work done on this fuel, but rather indicates that successful attainment of the upper burnup limit is likely.

Figure 13. Dimensional Change Data for U-10 Mo Fuel Slugs from SU-9 Experiment
VI. CONCLUSIONS

The major conclusions to be drawn from the irradiation and examination of the SU-9 fuel element are:

1. The variable orifice provided the control required to maintain the fuel element outlet temperature within plus or minus 1% of the upper plenum sodium temperature without excessive sensitivity.

2. The variable orifice drive tube did not bind due to sodium vapor condensation and/or fouling, provided the element was washed frequently.

3. The U-10 wt % Mo fuel could be operated under Hallam Core I conditions to peak burnups in excess of 6000 Mwd/MTU without excessive fuel swelling and fission gas release.

4. Type 304 stainless steel could be used in a sensitized condition under sodium cooled reactor conditions without appreciable corrosion or intergranular attack on the cladding. There was some carburization of the cladding as shown by chemical analyses, but this had little apparent effect on its mechanical properties.

5. No fuel melting or compatibility problems were encountered in the Type 304 stainless steel-sodium-U-10 wt % Mo system under the conditions expected in the fuel elements in Hallam Core I.

6. Metallographic examination of the U-10 wt % Mo fuel reconfirmed the fact that $\alpha + \gamma'$ phase material was significantly less stable in an irradiation environment than the material remaining in the $\gamma$ phase. Also there was evidence that the fuel which operated at lower temperatures had been homogenized to the $\gamma$ phase as a result of the radiation induced phase transformation.

NAA-SR-11121
34
REFERENCES


ERRATA
U-10 Wt % MO FUEL ELEMENT
IRRADIATION IN SRE
NAA-SR-1121

Page No. Correction
√ 18 Table 2, fourth column heading from left reading:

<table>
<thead>
<tr>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>(2 Dia. + L)</td>
<td>3</td>
</tr>
</tbody>
</table>

√ 21 Table 4, second and third column headings from left reading:

<table>
<thead>
<tr>
<th>Peak Fuel Centerline Temperature (%)</th>
<th>Peak Fuel Surface Temperature (%)</th>
</tr>
</thead>
</table>

should read:

<table>
<thead>
<tr>
<th>Peak Fuel Centerline Temperature (^F)</th>
<th>Peak Fuel Surface Temperature (^F)</th>
</tr>
</thead>
</table>

ATOMICS INTERNATIONAL
A Division of North American Aviation, Inc.

Corrected by M. Binder July 8, 1965