FAST REACTOR MIXED-CARBIDE FUEL ELEMENT DEVELOPMENT PROGRAM

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1. INTRODUCTION

The Fast Reactor Mixed-Carbide Fuel Element Development Program is aimed at the establishment of mixed carbides as a competitive LMFBR fuel. The major objective of the program is to develop fast breeder reactor fuel rods of (UPu)C clad with stainless steel or higher strength alloys that will perform satisfactorily to a minimum of 50,000 Mwd/T and a maximum of 100,000 Mwd/T at power levels of 10 to 45 kw/ft.

The program of active tasks is organized in the following manner:

Task 1200 - High-Density Pellet Evaluation in the EBR-II (10 Rods)
Task 1300 - Annular and Low-Density Pellet Evaluation in the EBR-II (19 Rods)
Task 1500 - Unrestrained Swelling and Fission Gas Release
Task 1800 - Sodium-Bonded Fuel Evaluation (4 Rods)
Task 1900 - Fabrication Development

1950 - Low-Power Subassembly in the EBR-II (19 Rods)
1960 - Intermediate Power Subassembly in the EBR-II (19 Rods)
Task 2400 - Mixed-Oxide Fabrication
Task 4800 - Evaluation of Thermal Behavior of Uranium-Plutonium Carbide Fuels
Task 4900 - TREAT Evaluation of Mixed Carbides
Task 5100 - High-Power Subassembly in the EBR-II.
Progress during the period April 1, 1970 to June 30, 1970 on the currently active tasks of this program is described in this report.

This is the twelfth in a series of quarterly progress reports written in partial fulfillment of Contract AT(30-1)-3415 between the United States Atomic Energy Commission and the United Nuclear Corporation.

Prior reports include the following:

**Topicals**


**Progress Reports**

*Fast Reactor Mixed-Carbide Fuel Element Development Program*

- **UNC-5193** First Quarterly Progress Report, July through September 1967 (Nov. 15, 1967).
- **UNC-5224** Fourth Quarterly Progress Report, April through June 1968 (Sept. 26, 1968).


2. HIGHLIGHTS

1. UNC 86 was irradiated to 73,000 Mwd/T in the EBR-II at 20 kw/ft without failure. The fuel rod is clad with Incoloy-800 and contains helium-bonded, high-density MC + 10 v/o $M_2C_3$ fuel pellets with an initial fuel-to-clad gap of 0.010 in. (Task 1200).

2. UNC 84 and 85, the same designs as UNC 86, but clad with Type 316 SS, both failed at a burnup of 73,000 Mwd/T at 20 kw/ft (Task 1200).

3. The four sodium-bonded fuel rods irradiated to 50,000 Mwd/T at 25 kw/ft were removed from the capsules and all are intact (Task 1800).

4. The fabrication of 2.2 kg 83% dense pellets and 1.5 kg 93% dense pellets for irradiation by ANL was completed successfully (Task 2400).
3. PAST RESULTS AND CURRENT PROGRESS

FAST REACTOR IRRADIATIONS

3.1 TASK 1200 – HIGH-DENSITY PELLET EVALUATION IN THE EBR-II

Introduction and Past Results

The objective of the task is to evaluate the performance of high-density (96% of theoretical, minimum) uranium-plutonium carbide pellets with varying size helium annuli between fuel and clad to provide void space for fuel swelling. The experiments are designed to determine the fuel-clad gap and clad strength combination to give satisfactory performance.

Ten doubly encapsulated fuel rods (Rods UNC 81 through 90) were designed, and nine were inserted for irradiation in EBR-II at powers of 20 to 30 kw/ft to burnups of 50,000 to 100,000 Mwd/T. The fuel is high-density, helium-bonded (UPu)C which was prepared by carbothermic reduction of the mixed dioxides followed by cold pressing and sintering of pellets. The cladding is Type 316 stainless steel and Incoloy-800. Eight of the capsules contain full-length prototypic fuel rods, while two capsules contain four segmented experiments each. A detailed description of these experiments is given in Table 1 and UNC-5170.

Nine of the 10 rods prepared under this task have completed irradiation in the EBR-II. The peak burnups and powers achieved were 45,400 Mwd/T at 20 kw/ft
<table>
<thead>
<tr>
<th>Task No.</th>
<th>UNC Experiment No.</th>
<th>Description of Experiment</th>
<th>Pellet Diameter, in.</th>
<th>Fuel Length, in.</th>
<th>Density, % T.D.</th>
<th>Pellet (Type)</th>
<th>Shmeared a/o</th>
<th>Pu, 10^-3</th>
<th>Cladding Material</th>
<th>Linear Power (Design), kw/ft</th>
<th>Target Burnup, Mwd/T</th>
<th>Accumulated Burnup, Mwd/T</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>1200</td>
<td>81-83</td>
<td>EBR-II rods, helium-bonded (6 rods)</td>
<td>0.248 to 0.252</td>
<td>14</td>
<td>98 (solid)</td>
<td>81 to 96</td>
<td>20</td>
<td>315 SS and Inc-800</td>
<td>20 to 30</td>
<td>50 to 100</td>
<td>EBR-II 19 to 31</td>
<td>45,500</td>
<td>Irradiation complete, UNC 81-87, 89, 90 post-irradiation examination in progress.</td>
</tr>
<tr>
<td></td>
<td>84-86</td>
<td>EBR-II multiple specimen rods, helium-bonded (2 rods)</td>
<td>0.249 to 0.252</td>
<td>4</td>
<td>98 (solid)</td>
<td>82 to 90</td>
<td>20</td>
<td>315 SS and Inc-800</td>
<td>24 to 30</td>
<td>40</td>
<td>EBR-II 22 to 28</td>
<td>32,000</td>
<td></td>
</tr>
<tr>
<td>1300</td>
<td>91-114</td>
<td>EBR-II rods, helium-bonded (24 rods)</td>
<td>0.232 to 0.269</td>
<td>14</td>
<td>75 to 84 (solid), 70 to 85</td>
<td>15</td>
<td>315 SS and Inc-800</td>
<td>20</td>
<td>50 to 100</td>
<td>EBR-II 17 to 23</td>
<td>53,000</td>
<td>UNC 101 irradiation complete.</td>
<td></td>
</tr>
<tr>
<td>1500</td>
<td>115-119</td>
<td>Unrestrained swelling specimens</td>
<td>0.06</td>
<td>0.5</td>
<td>75 to 90 (solid)</td>
<td>—</td>
<td>15</td>
<td>Low</td>
<td>20</td>
<td>Hanford 14,000</td>
<td>18-rods continuing irradiation.</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>120-124</td>
<td>Fission gas release specimens</td>
<td>0.375 OD, 0.255 ID</td>
<td>0.500</td>
<td>75 to 98 (annular)</td>
<td>—</td>
<td>15</td>
<td>Low</td>
<td>20</td>
<td>Hanford 20,000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1800</td>
<td>125-128</td>
<td>Sodium-bonded rods (4 rods)</td>
<td>0.240</td>
<td>14</td>
<td>93 to 98 (solid), 83 to 87</td>
<td>15</td>
<td>304 SS and Inc-800</td>
<td>25</td>
<td>50</td>
<td>GETR 25</td>
<td>50,000</td>
<td>Irradiation complete.</td>
<td></td>
</tr>
<tr>
<td>1950</td>
<td>129-147</td>
<td>19-rod low-power subassembly, helium and sodium bonds</td>
<td>0.240 to 0.246</td>
<td>14</td>
<td>75 to 98</td>
<td>70 to 87</td>
<td>15</td>
<td>316 SS, 304 SS, and Inc-800</td>
<td>100</td>
<td>EBR-II 34,000</td>
<td>Irradiation in progress.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1990</td>
<td>185-190</td>
<td>Intermediate-power rods (6 rods)</td>
<td>0.249 to 0.253</td>
<td>14</td>
<td>84 to 96 (solid), 80 to 88</td>
<td>15</td>
<td>315 SS and Inc-800</td>
<td>30</td>
<td>50 to 100</td>
<td>EBR-II 11,000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1960</td>
<td>191-209</td>
<td>19-rod intermediate-power subassembly, helium and sodium bonds</td>
<td>0.240 to 0.246</td>
<td>14</td>
<td>84 to 96</td>
<td>—</td>
<td>15</td>
<td>315 SS, 304 SS, and Inc-800</td>
<td>30 to 100</td>
<td>EBR-II 11,000</td>
<td>UNC 185, 186 will not be irradiated, UNC 187-188 inserted with Task 1960.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>4800</td>
<td>210-221</td>
<td>&quot;Rabbit&quot; irradiations, Thermal behavior of (UPuC fuels.</td>
<td>0.300</td>
<td>3.9</td>
<td>98</td>
<td>—</td>
<td>15</td>
<td>315 SS, 304 SS</td>
<td>45</td>
<td>GETR 45</td>
<td>Design and fabrication complete.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>4900</td>
<td>222-225</td>
<td>TREAT evaluation of mixed carbides</td>
<td>0.240 to 0.246</td>
<td>14</td>
<td>97 (solid)</td>
<td>—</td>
<td>15</td>
<td>315 SS, 304 SS</td>
<td>—</td>
<td>TREAT</td>
<td>Design in progress.</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
for UNC 81 through 83, 73,000 Mwd/T at 20 kw/ft for UNC 84 through 86, and 32,100 Mwd/T at 30 kw/ft for UNC 87, 89, and 90. The irradiation conditions for these experiments are given in Table 2.

UNC 88 was never placed in the reactor because of the lack of suitable available space. A decision not to irradiate it was made recently, when post-irradiation examination revealed failures of fuel rods with a similar design.

Fuel rods UNC 87, 90, and three segments of UNC 89 were intact. One of the four segments of UNC 89, with a 3% hot void, had several brittle cracks in the Type 316 SS clad. Fuel rods UNC 81 through 83 all had brittle cracking of the Type 316 SS and Incoloy-800 clads.

The examination of the fuel microstructure of UNC 81 through 83 and 87, 89, 90 showed that it is similar to the previously examined material. The sesquicarbide phase is depleted from the high-burnup, high-temperature portion of the fuel and is distributed normally in the low-temperature, low-burnup regions. There is no central void, but porosity due to fission gas agglomeration is evident in the central region of the fuel.

Microprobe analysis of the Type 316 SS clad from UNC 89 irradiated to 40,000 Mwd/T showed that the metallographically observed dark etching zone at the clad ID contained no fuel constituents or fission products. No elements other than the constituents of stainless steel were found.

Microprobe analysis of the Incoloy-800 clad showed that some diffusion of the clad constituents into the fuel occurred at the fuel-clad interface.

Current Results

Fuel rods UNC 84, 85, and 86, containing high-density, helium-bonded $MC + 10 \text{ v/o } M_2C_3$ pellets and clad with Type 316 SS or Incoloy-800, were removed
<table>
<thead>
<tr>
<th>UNC Rod Numbers</th>
<th>81</th>
<th>82</th>
<th>83</th>
<th>84</th>
<th>85</th>
<th>86</th>
<th>87</th>
<th>A</th>
<th>B</th>
<th>C</th>
<th>D</th>
<th>A</th>
<th>B</th>
<th>C</th>
<th>D</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Material</td>
<td>316 SS</td>
<td>316 SS</td>
<td>Inc-800</td>
<td>316 SS</td>
<td>316 SS</td>
<td>Inc-800</td>
<td>316 SS</td>
<td>316 SS</td>
<td>316 SS</td>
<td>Inc-800</td>
<td>Inc-800</td>
<td>Inc-800</td>
<td>Inc-800</td>
<td>Inc-800</td>
<td></td>
</tr>
<tr>
<td>Clad Material</td>
<td>316 SS</td>
<td>316 SS</td>
<td>Inc-800</td>
<td>316 SS</td>
<td>316 SS</td>
<td>Inc-800</td>
<td>316 SS</td>
<td>316 SS</td>
<td>316 SS</td>
<td>Inc-800</td>
<td>Inc-800</td>
<td>Inc-800</td>
<td>Inc-800</td>
<td>Inc-800</td>
<td></td>
</tr>
<tr>
<td>Fuel-to-Clad Gap, diametral, m. †</td>
<td>0.008</td>
<td>0.005</td>
<td>0.006</td>
<td>0.009</td>
<td>0.010</td>
<td>0.010</td>
<td>0.006</td>
<td>0.006</td>
<td>0.002</td>
<td>0.008</td>
<td>0.009</td>
<td>0.007</td>
<td>0.003</td>
<td>0.008</td>
<td>0.009</td>
</tr>
<tr>
<td>Hot Void, % †</td>
<td>6.8</td>
<td>4.8</td>
<td>4.7</td>
<td>7.6</td>
<td>7.8</td>
<td>7.5</td>
<td>4.8</td>
<td>4.9</td>
<td>2.7</td>
<td>4.5</td>
<td>6.4</td>
<td>4.8</td>
<td>2.6</td>
<td>4.4</td>
<td>6.3</td>
</tr>
<tr>
<td>Burnup, Mwd/T (\times 10^{-2})</td>
<td>45.4*</td>
<td>42.8*</td>
<td>44.3*</td>
<td>73†</td>
<td>73†</td>
<td>73†</td>
<td>28.9*</td>
<td>32.0*</td>
<td>32.1*</td>
<td>31.1*</td>
<td>27.8*</td>
<td>31.3*</td>
<td>31.6*</td>
<td>20.7*</td>
<td>27.4*</td>
</tr>
<tr>
<td>Max Fuel Temp, °C †</td>
<td>1640</td>
<td>1495</td>
<td>1520</td>
<td>1870</td>
<td>1890</td>
<td>1700</td>
<td>1820</td>
<td>1870</td>
<td>1905</td>
<td>1800</td>
<td>1855</td>
<td>1850</td>
<td>1900</td>
<td>1815</td>
<td>1840</td>
</tr>
<tr>
<td>Max Clad OD Temp, °C</td>
<td>595</td>
<td>595</td>
<td>600</td>
<td>590</td>
<td>590</td>
<td>625</td>
<td>550</td>
<td>570</td>
<td>550</td>
<td>500</td>
<td>540</td>
<td>540</td>
<td>535</td>
<td>500</td>
<td></td>
</tr>
<tr>
<td>Max Specimen Heat Generation Rate, w/cm²</td>
<td>720</td>
<td>730</td>
<td>730</td>
<td>655</td>
<td>655</td>
<td>655</td>
<td>910</td>
<td>880</td>
<td>1010</td>
<td>930</td>
<td>730</td>
<td>910</td>
<td>1080</td>
<td>960</td>
<td>705</td>
</tr>
<tr>
<td>Max Clad Surface Heat Flux, w/cm²</td>
<td>295</td>
<td>300</td>
<td>305</td>
<td>219</td>
<td>219</td>
<td>219</td>
<td>400</td>
<td>360</td>
<td>415</td>
<td>380</td>
<td>300</td>
<td>390</td>
<td>160</td>
<td>410</td>
<td>300</td>
</tr>
<tr>
<td>(\int_{T_C}^{T_S} K_d E dT \times 10^{-2})</td>
<td>57.3</td>
<td>58.1</td>
<td>58.1</td>
<td>52.1</td>
<td>52.1</td>
<td>52.1</td>
<td>76.4</td>
<td>70.0</td>
<td>80.4</td>
<td>74.0</td>
<td>58.1</td>
<td>72.5</td>
<td>86.0</td>
<td>76.4</td>
<td>56.1</td>
</tr>
<tr>
<td>Peak Clad Fluence, nvt (\times 10^{-22})</td>
<td>3.0</td>
<td>2.9</td>
<td>3.0</td>
<td>4.9</td>
<td>4.9</td>
<td>4.9</td>
<td>0.8</td>
<td>0.9</td>
<td>0.9</td>
<td>0.9</td>
<td>0.9</td>
<td>0.9</td>
<td>0.9</td>
<td>0.9</td>
<td>0.8</td>
</tr>
</tbody>
</table>

*Burnup determined by Nd\(^{148}\) analysis.
†Calculated burnup.
‡At startup.
from the capsules at Los Alamos Scientific Laboratory. The fuel rods were ir-
radiated to a calculated peak burnup of 73,000 Mwd/T in the EBR-II at 20 kw/ft.
The irradiation conditions for the fuel rods are given in Table 2.

Fuel rods UNC 84 and 85 were clad with Type 316 SS and had a cold fuel-
to-clad diametral gap of 0.009 and 0.010 in., respectively. Both fuel rods failed
with a large number of circumferential and longitudinal clad cracks. In com-
parison, UNC 86 clad with Incoloy-800 and a cold fuel-to-clad gap of 0.010 in.
did not fail. Macro-photographs of the fuel rods are shown in Fig. 1.

The fuel structure of the three fuel rods is similar to that observed in the
previous irradiation experiments. There is depletion of the sesquicarbide in the
center of the fuel and a normal distribution toward the edge of the fuel (Fig. 2).
The amount of porosity in the center of the fuel appears to be greater than at lower
burnups. A platelet structure was observed in all the fuel, but is very similar to
a platelet structure observed in the unirradiated archive samples (Fig. 3).

The dark etching zone at the Type 316 clad ID observed in the previously
examined fuel rods at lower burnups is not nearly as pronounced in UNC 84 and
85. In many areas of the fuel clad interface this zone does not appear at all
(Fig. 4). The zone at the ID of the Incoloy-800 clad of UNC 86 was similar in ap-
pearance to the zones noted at lower burnups (Fig. 5). Previous microprobe ex-
amination identified this as an area of interdiffusion of clad components contain-
ing no fuel or fission products.

A summary of post-irradiation examination for UNC 81 through 87, 89, and
90 is shown in Table 3.
Fig. 1 — Fuel Rods UNC 84, 85, and 86
73,000 Mw.d/T  20 kw/ft  EBR-II
97% Dense Pellets  MC + 10 v/o MnC₃
500× Fuel Center Etchant - 1:1:1 H₂O, acetic, nitric acid

Fig. 2 — UNC 86 - Irradiated Fuel Microstructure
73,000 Mwd/T 20 kw/ft EBR-II
97% Dense Pellets MC + 10 v/o M₂C₃
500× Irradiated Fuel Microstructure
73,000 Mwd/T 20 kw/ft EBR-II
97% Dense Pellets MC + 10 v/o M₂C₃

Etchant:
1:1:1 H₂O, Acetic, Nitric Acid

500× Unirradiated Fuel Microstructure
97% Dense Pellets MC + 10 v/o M₂C₃

Fig. 3 — UNC 84 — Fuel Microstructure with Platelets
Fig. 4 — UNC 85 - Type 316 SS
73,000 Mwd/T  20 kw/ft  EBR-II
97% Dense Pellets  MC + 10 v/o M₂C₃
Fig. 5 — UNC 86 - Incoloy-800 Fuel Clad Interface
73,000 Mwd/T 20 kw/ft EBR-II
97% Dense Pellets MC + 10 v/o M₂C₃
### TABLE 3 — SUMMARY OF POST-IRRADIATION EXAMINATION FOR UNC 81 THROUGH 87, 89, AND 90 IRRADIATED IN THE EBR-II

<table>
<thead>
<tr>
<th>Fuel Rod No.</th>
<th>Clad Material</th>
<th>Clad Integrity</th>
<th>Fission Gas Release, % of Theoretical</th>
<th>Max Clad OD Increase, in.</th>
<th>Max Restrained Fuel %ΔV/V per 10,000 Mwd/T</th>
<th>Fuel Density Decrease, %</th>
</tr>
</thead>
<tbody>
<tr>
<td>UNC</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Based on ΔDclad</td>
<td>Based on ΔVfuel</td>
</tr>
<tr>
<td>81</td>
<td>316 SS</td>
<td>Cracked</td>
<td>*</td>
<td>0.0053</td>
<td>2.3</td>
<td>1.9</td>
</tr>
<tr>
<td>82</td>
<td>316 SS</td>
<td>Cracked</td>
<td>*</td>
<td>0.0044</td>
<td>1.8</td>
<td>2.0</td>
</tr>
<tr>
<td>83</td>
<td>Inc-800</td>
<td>Intact</td>
<td>*</td>
<td>0.0104</td>
<td>2.8</td>
<td>2.2</td>
</tr>
<tr>
<td>84</td>
<td>316 SS</td>
<td>Cracked</td>
<td>*</td>
<td>0.0122</td>
<td>3.1</td>
<td>†</td>
</tr>
<tr>
<td>85</td>
<td>316 SS</td>
<td>Cracked</td>
<td>*</td>
<td>0.0074</td>
<td>3.0</td>
<td>†</td>
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<tr>
<td>86</td>
<td>Inc-800</td>
<td>Intact</td>
<td>†</td>
<td>0.0033</td>
<td>2.3</td>
<td>†</td>
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<td>87</td>
<td>Inc-800</td>
<td>Intact</td>
<td>1.30</td>
<td>0.0032</td>
<td>2.5</td>
<td>2.1</td>
</tr>
<tr>
<td>89A</td>
<td>316 SS</td>
<td>Intact</td>
<td>†</td>
<td>0.0001</td>
<td>2.0</td>
<td>†</td>
</tr>
<tr>
<td>B</td>
<td>316 SS</td>
<td>Cracked</td>
<td>*</td>
<td>0.0035</td>
<td>2.0</td>
<td>†</td>
</tr>
<tr>
<td>C</td>
<td>316 SS</td>
<td>Intact</td>
<td>4.05</td>
<td>0.0018</td>
<td>2.5</td>
<td>3.2</td>
</tr>
<tr>
<td>D</td>
<td>316 SS</td>
<td>Intact</td>
<td>1.17</td>
<td>0.0004</td>
<td>2.3</td>
<td>†</td>
</tr>
<tr>
<td>90A</td>
<td>Inc-800</td>
<td>Intact</td>
<td>0.70</td>
<td>0.0000</td>
<td>2.5</td>
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<td>Inc-800</td>
<td>Intact</td>
<td>0.23</td>
<td>0.0030</td>
<td>2.5</td>
<td>†</td>
</tr>
<tr>
<td>C</td>
<td>Inc-800</td>
<td>Intact</td>
<td>2.47</td>
<td>0.0029</td>
<td>2.3</td>
<td>2.8</td>
</tr>
<tr>
<td>D</td>
<td>Inc-800</td>
<td>Intact</td>
<td>†</td>
<td>0.0012</td>
<td>2.3</td>
<td>†</td>
</tr>
</tbody>
</table>

*Sample lost due to clad failure.
†Analysis in progress.
¶Sample lost due to equipment failure.
§Fuel density not measured.
3.2 TASK 1300 - ANNULAR AND LOW-DENSITY PELLET EVALUATION IN THE EBR-II

Introduction and Past Results

The objective of the task is to evaluate various methods of void deployment in helium-bonded uranium-plutonium carbide pellets to provide void space for fuel swelling. Specifically, high-density annular (97% of theoretical) and low-density pellets (75 and 84% of theoretical) with various smear densities are being evaluated to determine the best fuel rod design.

The most commonly used solution to the fuel swelling problem is a controlled helium gap between the fuel and clad as in Task 1200. As an alternative approach, accommodation for swelling can be provided within the fuel. This can be in the form of a central void (cored pellets) or as uniformly distributed porosity.

Rods UNC 91 through 114 were designed for powers of 20 kw/ft to burnups of 50,000 to 100,000 Mwd/T. Design variables include power, burnup, clad material and thickness, fuel structure, and fuel smear density. Design details are shown in Table 1.

The 19-rod subassembly was discharged from the EBR-II with an average burnup of 53,000 Mwd/T. Ten fuel rods were to be removed, but approval was granted to irradiate nine of the 10 rods to 73,000 Mwd/T. The tenth rod, UNC 101, was removed for examination and shipped to LASL.

Neutron radiography of the 19 fuel rods showed no gross failures.

Post-irradiation examination of UNC 101 revealed a 3-in. crack ~3 in. above the bottom end plug. The Incoloy-800 clad fuel rod contained 84% of theoretical density pellets with a 0.001-in. fuel-to-clad gap and a hot void of 17%.
The fuel structure is the same as that observed in unirradiated 84% of theoretical density fuel. The porosity does not appear to close and accommodate the fuel swelling.

Current Results

The subassembly X-075 has accumulated a burnup of 74,000 Mwd/T and is expected to be discharged early in the next quarter.
3.3 TASK 1500 – UNRESTRAINED SWELLING AND FISSION GAS RELEASE

Introduction and Past Results

The objective of the experiments is to investigate the unrestrained swelling and fission gas release characteristics of several types of (UPu)C fuel as a function of irradiation exposure and temperature. Since temperature measurement is essential to obtain meaningful data, the experiments have to be carried out in instrumented capsules; this is feasible in thermal reactors only. The problem of flux depression and resulting ΔT's, which are not representative of fast reactors, is resolved by the use of small cross-section specimens operating at essentially isothermal conditions.

A description of the capsule trains was given in the third quarterly report, UNC-5217. The capsule trains were irradiated at design conditions. The experiments were operated in the Douglas United Nuclear Corporation special irradiation facilities with fuel temperatures between 1000 and 1600°C to 20,000 Mwd/T. A list of the experiments is given in Table 4.

Both capsule trains are at Argonne National Laboratory for post-irradiation examination.

Current Results

The post-irradiation examination of the experiments was initiated.

The fuel specimen assemblies UNC 120B, 121A, and 121B were removed intact, while the tantalum container of UNC 120A apparently oxidized and only fragments of the container remained. The contents have not been examined. Disassembly of the remaining capsules and fuel specimen assemblies is in process.
TABLE 4 — SUMMARY OF CAPSULE FUEL MATERIALS AND OPERATING DATA

Unrestrained Swelling Capsule Train

<table>
<thead>
<tr>
<th>Capsule</th>
<th>Fuel*</th>
<th>% T.D.</th>
<th>Fuel Temperature, °C</th>
<th>Burnup, $\times 10^{-3}$ Mwd/T</th>
</tr>
</thead>
<tbody>
<tr>
<td>UNC 116A†</td>
<td>MC + $M_2C_3$ ‡</td>
<td>98</td>
<td>1300</td>
<td>1289</td>
</tr>
<tr>
<td>116B</td>
<td>MC</td>
<td>84</td>
<td>1300</td>
<td>1340</td>
</tr>
<tr>
<td>117A</td>
<td>MC</td>
<td>93</td>
<td>1300</td>
<td>1325</td>
</tr>
<tr>
<td>117B</td>
<td>M(OC)</td>
<td>93</td>
<td>1300</td>
<td>1360</td>
</tr>
<tr>
<td>118A</td>
<td>$MO_2$</td>
<td>&gt;90</td>
<td>1600</td>
<td>1366</td>
</tr>
<tr>
<td>118B</td>
<td>$MO_2$</td>
<td>&gt;90</td>
<td>1600</td>
<td>1578</td>
</tr>
<tr>
<td>119A</td>
<td>MC + $M_2C_3$</td>
<td>98</td>
<td>1600</td>
<td>§</td>
</tr>
<tr>
<td>119B</td>
<td>MC</td>
<td>84</td>
<td>1600</td>
<td>1304§</td>
</tr>
</tbody>
</table>

Fission Gas Release Capsule Train

<table>
<thead>
<tr>
<th>Capsule</th>
<th>Fuel*</th>
<th>% T.D.</th>
<th>Fuel Temperature, °C</th>
<th>Burnup, $\times 10^{-3}$ Mwd/T</th>
</tr>
</thead>
<tbody>
<tr>
<td>UNC 120A</td>
<td>MC + $M_2C_3$</td>
<td>98</td>
<td>1000</td>
<td>1003</td>
</tr>
<tr>
<td>120B</td>
<td>MC</td>
<td>84</td>
<td>1000</td>
<td>978</td>
</tr>
<tr>
<td>121A</td>
<td>MC + $M_2C_3$</td>
<td>98</td>
<td>1300</td>
<td>1275</td>
</tr>
<tr>
<td>121B</td>
<td>MC</td>
<td>84</td>
<td>1300</td>
<td>1373</td>
</tr>
<tr>
<td>122A</td>
<td>MC</td>
<td>93</td>
<td>1300</td>
<td>1308</td>
</tr>
<tr>
<td>122B</td>
<td>MC</td>
<td>75</td>
<td>1300</td>
<td>1375</td>
</tr>
<tr>
<td>123A</td>
<td>$M_2C_3$</td>
<td>93</td>
<td>1300</td>
<td>1288</td>
</tr>
<tr>
<td>123B</td>
<td>M(OC)</td>
<td>93</td>
<td>1300</td>
<td>1398</td>
</tr>
<tr>
<td>124A</td>
<td>MC + $M_2C_3$</td>
<td>98</td>
<td>1600</td>
<td>1298¶</td>
</tr>
<tr>
<td>124B</td>
<td>MC</td>
<td>84</td>
<td>1600</td>
<td>1320¶</td>
</tr>
</tbody>
</table>

*M = (U$_{0.85}$Pu$_{0.15}$), U = 93% $U^{235}$.
†“A” specimen was the control specimen and operated at ±25°C of the design temperature.
‡90 v/o MC + 10 v/o $M_2C_3$.
§Thermocouple failed. Control was on “B” specimen.
¶Operating temperature was below design due to control rod shadowing.
3.4 TASK 1800 - SODIUM-BONDED FUEL EVALUATION

Introduction and Past Results

The objective of the irradiations is to evaluate the performance of sodium-bonded, uranium-plutonium carbide rods. The test parameters for the capsules are given in Table 1.

Capsules UNC 125, 126, 127, and 128 had been discharged from the GETR after achieving a peak goal burnup of 50,000 Mwd/T. All experiments operated at design power.

Neutron radiography of the four capsules indicated that approximately two-thirds of the original 0.030-in. diametral gap was utilized to accommodate fuel swelling.

Current Results

Post-irradiation examination of the capsules was initiated at Los Alamos Scientific Laboratory.

Examination of the betatron radiographs of the capsules has shown that the fuel is generally cracked and further confirms the previous measurements that approximately one-third of the original 0.030-in. diametral fuel-to-clad gap remains.

The fuel rods have been removed from the capsules and all the fuel rods are intact.
Introduction and Past Results

The objective of the experiment is to obtain performance data on uranium-plutonium carbide fuel rods at low powers. Fuel swelling data are of particular interest, because a large portion of an LMFBR will be operating at less than the 20 to 30 kw/ft peak power range at which the carbides are expected to operate. The test variables are given in Table 1.

Fuel rods and capsules for the subassembly were completed successfully and inserted in the EBR-II. This is the second all-carbide subassembly to be irradiated.

Current Results

The subassembly X-055, composed of UNC 129 through 147, operating in the EBR-II at 10 to 15 kw/ft, has accumulated an exposure of 34,000 Mwd/T.
3.6 TASK 1960 – INTERMEDIATE-POWER SUBASSEMBLY IN THE EBR-II (19 RODS)

Introduction and Past Results

The objective of the 19-rod experiments is to extend the current irradiation study to higher powers (30 to 35 kw/ft).

High linear and specific powers at high burnup are necessary in fast reactor fuels to reduce fuel cycle costs significantly. The most critical problem of high-power, high-burnup fuels is the accommodation of fuel swelling without rupturing the clad. One approach to this problem is the use of a sodium fuel-to-clad bond. The sodium provides expansion space, eliminates clad strain, and reduces fuel temperatures. Additional approaches utilizing a helium fuel-to-clad bond are finely dispersed voids in low-density pellets and high-density annular pellets.

Re-evaluation of the power output for the subassembly, based on the latest fission rates, indicated that it will operate between 26 and 29 kw/ft rather than 30 to 35 kw/ft. The reduced power will be caused by an enlarged EBR-II core which has depressed the flux near the core center.

The capsules have been fabricated, shipped, and accepted for irradiation by the EBR-II project.

Based on the results of the Task 1200 fuel rods, four high-probability-of-failure fuel rods, UNC 202 through 205, will not be irradiated but are replaced in the subassembly by UNC 187 through 190 from Task 1930.

Current Results

The subassembly X-079 has accumulated a burnup of 11,000 Mwd/T in the EBR-II.
3.7 TASK 2400 - MIXED-OXIDE PELLET FABRICATION

Past and Current Results

The objective of the task was to fabricate UO$_2$-25 w/o PuO$_2$ pellets for the Argonne National Laboratory EBR-II Irradiation Program.

Approximately 2.2 kg of 83% and 1.5 kg of 93% of theoretical density pellets were prepared and shipped. All fuel met specifications as shown in Tables 5 and 6.

The fuel pellets were prepared by the conventional cold pressing and sintering method. Fully enriched UO$_2$ and PuO$_2$ powders were pre-blended in the twin shell blender with water and binder to provide 1/4 w/o polyvinyl alcohol. The agglomerates were vacuum dried. The dried and sized agglomerates were lubricated with 0.3 w/o dry zinc stearate powder in the twin shell blender to form uniform lots of free-flowing press feed. The pellets were cold pressed in an automatic hydraulic press, and sintered in the temperature range 1560 to 1725°C in a He-6% H$_2$ atmosphere. The sintered pellets were dry centerless ground to meet the specified diametral tolerance.
TABLE 5 — CHARACTERIZATION OF UO₂-PuO₂ FOR THE ANL IRRADIATION PROGRAM

<table>
<thead>
<tr>
<th>Production Run No.</th>
<th>Pellet Quantity, g</th>
<th>Chemical Analysis</th>
<th>X-Ray Analysis</th>
<th>Alpha Autoradiography</th>
<th>Physical Analysis</th>
</tr>
</thead>
<tbody>
<tr>
<td>Batch No.</td>
<td>OX-36</td>
<td>OX-36</td>
<td>OX-37</td>
<td>OX-38</td>
<td></td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>2</td>
<td>1</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td></td>
<td>670</td>
<td>849</td>
<td>1451</td>
<td>774</td>
<td></td>
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<tr>
<td>OX-36</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>OX-37</td>
<td></td>
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<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>OX-38</td>
<td></td>
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<tr>
<td>Production Run No.</td>
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<td></td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>Pellet Quantity, g</td>
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<td></td>
<td></td>
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</tr>
<tr>
<td></td>
<td>1</td>
<td>2</td>
<td>1</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td></td>
<td>670</td>
<td>849</td>
<td>1451</td>
<td>774</td>
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<tr>
<td>Chemical Analysis</td>
<td></td>
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<td></td>
<td></td>
</tr>
<tr>
<td>O/M</td>
<td>1.96</td>
<td>1.95</td>
<td>1.96</td>
<td>1.96</td>
<td>1.97 ± 0.02</td>
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<tr>
<td>Pu, w/o</td>
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<td>21.88</td>
<td>22.00</td>
<td>21.88</td>
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<tr>
<td>U, w/o</td>
<td>66.12</td>
<td>66.18</td>
<td>66.08</td>
<td>65.96</td>
<td>None</td>
</tr>
<tr>
<td>Pu/(U +Pu)×100, w/o</td>
<td>24.96</td>
<td>24.85</td>
<td>24.98</td>
<td>24.91</td>
<td>25.00 ± 0.5</td>
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<tr>
<td>Carbon, ppm</td>
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<td>40</td>
<td>15</td>
<td>26</td>
<td>&lt;200</td>
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<tr>
<td>Fluorine, ppm</td>
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<td>3.4</td>
<td>9.5</td>
<td>8</td>
<td>&lt;50*</td>
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<tr>
<td>Chlorine, ppm</td>
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<td>6.5</td>
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<td>5</td>
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<tr>
<td>Nitrogen, ppm</td>
<td>31</td>
<td>20</td>
<td>12</td>
<td>28</td>
<td>&lt;200</td>
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<td>X-Ray Analysis</td>
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</tr>
<tr>
<td>Second phase, w/o</td>
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<td>&lt;15</td>
<td>&lt;15</td>
<td>&lt;15</td>
<td>&lt;15</td>
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<tr>
<td>Alpha Autoradiography</td>
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<td></td>
<td></td>
</tr>
<tr>
<td>Pu-containing particle size:</td>
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<tr>
<td>Maximum, μ</td>
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<td>46</td>
<td>65</td>
<td>56</td>
<td>&lt;100</td>
</tr>
<tr>
<td>Average, μ</td>
<td>35</td>
<td>38</td>
<td>40</td>
<td>38</td>
<td>None</td>
</tr>
<tr>
<td>Physical Analysis</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Density, g/cm³</td>
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<td></td>
<td></td>
<td></td>
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<tr>
<td></td>
<td></td>
<td>10.43 ± 0.02</td>
<td>10.36 ± 0.17</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>9.25 ± 0.06</td>
<td>9.20 ± 0.06</td>
<td>9.22 ± 0.07</td>
<td>9.21 ± 0.17</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Diameter, in.</td>
<td></td>
<td></td>
<td></td>
<td>All pellets met specification</td>
<td>0.214 ± 0.001</td>
</tr>
</tbody>
</table>

*Sum of F + Cl.
TABLE 6 — SPECTROCHEMICAL ANALYSIS OF UO$_2$-PuO$_2$ PELLETS FOR THE ANL IRRADIATION PROGRAM

<table>
<thead>
<tr>
<th>Production Run No.:</th>
<th>Batch No.</th>
<th>OX-36</th>
<th>OX-36</th>
<th>OX-37</th>
<th>OX-38</th>
</tr>
</thead>
<tbody>
<tr>
<td>Element</td>
<td>Specification</td>
<td>ppm</td>
<td>ppm</td>
<td>ppm</td>
<td>ppm</td>
</tr>
<tr>
<td>Al</td>
<td>500</td>
<td>41</td>
<td>53</td>
<td>19</td>
<td>13</td>
</tr>
<tr>
<td>B</td>
<td>&lt; 0.5</td>
<td>&lt; 0.5</td>
<td>&lt; 0.5</td>
<td>&lt; 0.5</td>
<td>&lt; 0.5</td>
</tr>
<tr>
<td>Ca</td>
<td>250</td>
<td>15</td>
<td>15</td>
<td>13</td>
<td>10</td>
</tr>
<tr>
<td>Cd</td>
<td>20</td>
<td>&lt; 0.5</td>
<td>&lt; 0.5</td>
<td>&lt; 0.5</td>
<td>&lt; 0.5</td>
</tr>
<tr>
<td>Cr</td>
<td>250</td>
<td>115</td>
<td>104</td>
<td>87</td>
<td>51</td>
</tr>
<tr>
<td>Fe</td>
<td>500</td>
<td>175</td>
<td>190</td>
<td>127</td>
<td>217</td>
</tr>
<tr>
<td>Mg</td>
<td>50</td>
<td>22</td>
<td>25</td>
<td>19</td>
<td>25</td>
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<tr>
<td>Na</td>
<td>500</td>
<td>4.0</td>
<td>4.7</td>
<td>3.0</td>
<td>3.3</td>
</tr>
<tr>
<td>Ni</td>
<td>500</td>
<td>91</td>
<td>100</td>
<td>47</td>
<td>61</td>
</tr>
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<td>V</td>
<td>500</td>
<td>&lt; 5</td>
<td>&lt; 5</td>
<td>&lt; 5</td>
<td>&lt; 5</td>
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<tr>
<td>Total of Cu, Zn, Si, Ti</td>
<td>800</td>
<td>132</td>
<td>133</td>
<td>93</td>
<td>79</td>
</tr>
<tr>
<td>Total of Ag, Mn, Pb, Mo, Sn</td>
<td>200</td>
<td>&lt; 10</td>
<td>&lt; 11</td>
<td>&lt; 11</td>
<td>&lt; 17</td>
</tr>
</tbody>
</table>
3.8 TASK 4800 — EVALUATION OF THERMAL BEHAVIOR OF URANIUM-PLUTONIUM CARBIDE FUELS

Introduction and Past Results

The objective of this task is to study the short-time behavior of (UPu)C fuel, with both helium and sodium thermal bonds at very high heat fluxes and linear power, to determine whether any heat transfer limitations exist with typical fuel rod designs. Both good and defective fuel-to-clad bonds will be tested.

Twelve experiments with both helium and good and defective sodium fuel-to-clad bonds have been designed to operate at 45 kw/ft with a surface heat flux of 660 w/cm².

Post-irradiation examination will include neutron radiography and capsule disassembly to examine the thermal bonds and general condition of the fuel and clad.

The metal component fabrication and the fuel fabrication were completed.

Current Results

The 12 experiments were assembled and sent to GETR for irradiation. The irradiation, however, was delayed from June 1970 until July 1970 because of additional analyses required by GE for the hazards summary report.
3.9 TASK 4900 – TREAT EVALUATION OF MIXED CARBIDES

Introduction and Past Results

The objective of the task is to evaluate mixed carbides under transient conditions.

In order to define test parameters for mixed-carbide fuels, a review of transient testing on LMFBR-type fuels was made.

Two fuel rods will be removed from the Task 1950 subassembly for the irradiated TREAT experiments. One is a sodium-bonded, 304 SS clad fuel rod with high-density, two-phase fuel with a hot void of 22%. The second is a helium-bonded 316 SS clad fuel rod with high-density, two-phase fuel with a hot void of 8%. Two identical unirradiated fuel rods will be tested for comparison.

Current Results

A parametric study was completed to define a series of transients for the experiments. The transients selected are given in Table 7. These transients yield incipient fuel melting for the sodium-bonded fuel rod and 50% fuel melting in the helium-bonded fuel rod.

The fuel rods for the unirradiated transient experiments and the replacement rods for the Task 1950 subassembly have been fabricated. Characterization of the fuel is given in Table 8 and shows that the fuel met UNC specification.
<table>
<thead>
<tr>
<th>Fuel Rod Type</th>
<th>Transient Type</th>
<th>Excess Reactivity Input, %</th>
<th>Time, sec</th>
<th>Fuel Melt, %</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sodium bonded</td>
<td>Temperature limited</td>
<td>1.1</td>
<td>3.0</td>
<td>Incipient</td>
</tr>
<tr>
<td>Helium bonded</td>
<td>Temperature limited</td>
<td>0.94</td>
<td>3.5</td>
<td>50</td>
</tr>
</tbody>
</table>
TABLE 8 — CHARACTERIZATION OF (UPu)C* PELLETS FOR TRANSIENT EXPERIMENTS AND REPLACEMENT RODS

<table>
<thead>
<tr>
<th>UNC &amp; Fuel Rods</th>
<th>Intended</th>
<th>Sintered</th>
<th>Density, % T.D.</th>
<th>Sintered Density, g/cm³</th>
<th>% T.D.</th>
</tr>
</thead>
<tbody>
<tr>
<td>146 A</td>
<td>Diameter, in.</td>
<td>0.240 ± 0.001</td>
<td>0.2396 ± 0.0002</td>
<td>13.17 ± 0.02</td>
<td>98.3</td>
</tr>
<tr>
<td>138 A</td>
<td>Diameter, in.</td>
<td>0.246 ± 0.001</td>
<td>0.2456 ± 0.0002</td>
<td>13.14 ± 0.04</td>
<td>98.1</td>
</tr>
</tbody>
</table>

**Chemical Characterization**

<table>
<thead>
<tr>
<th>UNC &amp; Fuel Rods</th>
<th>C, w/o</th>
<th>Pu, w/o</th>
<th>O₂, ppm</th>
<th>N₂, ppm</th>
</tr>
</thead>
<tbody>
<tr>
<td>146 A</td>
<td>4.71 ± 0.07</td>
<td>13.96 ± 0.06</td>
<td>3370 ± 310</td>
<td>145 ± 9</td>
</tr>
<tr>
<td>138 A</td>
<td>4.74 ± 0.05</td>
<td>14.01 ± 0.06</td>
<td>3580 ± 900</td>
<td>140 ± 5</td>
</tr>
</tbody>
</table>

**Metallographic and X-Ray Characterization**

<table>
<thead>
<tr>
<th>UNC &amp; Fuel Rods</th>
<th>Metallography†</th>
<th>Lattice Parameters, A</th>
</tr>
</thead>
<tbody>
<tr>
<td>146 A</td>
<td>% MC: 90 10</td>
<td>MC: 4.9662 ± 0.0004</td>
</tr>
<tr>
<td>138 A</td>
<td>% MC: 89 11</td>
<td>MC: 4.9652 ± 0.0001</td>
</tr>
</tbody>
</table>

*M = U₀.₈₅Pu₀.₁₅.
†Metallographic examination of the fuel pellets revealed a 10-micron layer of M₂C₃ on the pellet surface. The layer was removed by surface grinding.
3.10 TASK 5100 - HIGH-POWER SUBASSEMBLY IN THE EBR-II

Introduction and Past Results

The task objective is to conduct irradiation experiments at linear powers of up to 45 kw/ft (1500 w/cm) in the EBR-II.

A feasibility study has shown that a linear power of 45 kw/ft with a 0.030-in. diametral sodium annulus can be achieved in either an E-19 or B-19 singly clad type subassembly with a clad OD of 0.022 in. greater than the reference capsule. However, prior to 45 kw/ft operation, EBR-II prefers to operate the first singly clad carbide fuel rod at a previously demonstrated power, such as 30 kw/ft.

A peak power of 43 kw/ft could be achieved by a doubly clad experiment in Row 3.

Current Results

The design for a singly clad subassembly to operate between 30 and 35 kw/ft was started. The variables for the subassembly are shown in Table 9.

Subassembly hardware was designed by ANL based on a clad OD of 0.310 in. Analysis of the preliminary design showed that adequate coolant flow rates can be achieved for a nominal fuel rod power of 33 kw/ft.

Based on the successful irradiation of UNC 86, a helium-bonded, Incoloy-800 clad rod with a 10-mil diametral gap and several high-power, helium-bonded rods were considered with 12 to 15-mil gaps. All would have operated with center melting, a condition not considered desirable for singly clad rods at this time. The designs will be considered for doubly clad experiments on Task 1960.
### TABLE 9 — DESIGN VARIABLES FOR SINGLY CLAD SUBASSEMBLY

<table>
<thead>
<tr>
<th>UNC Rod No.</th>
<th>Fuel*</th>
<th>% T.D.</th>
<th>Fuel-Clad Bond</th>
<th>Clad</th>
<th>Burnup, Mwd/T \times 10^{-3}</th>
</tr>
</thead>
<tbody>
<tr>
<td>222</td>
<td>MC</td>
<td>93</td>
<td>Sodium</td>
<td>304 SS</td>
<td>50</td>
</tr>
<tr>
<td>223</td>
<td>MC + 10 v/o M_2C_3</td>
<td>93</td>
<td>Inc-800</td>
<td>50</td>
<td></td>
</tr>
<tr>
<td>224</td>
<td>MC</td>
<td>93</td>
<td>304 SS</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>225</td>
<td>MC</td>
<td>93</td>
<td>316 SS</td>
<td>50</td>
<td></td>
</tr>
<tr>
<td>226</td>
<td>MC</td>
<td>93</td>
<td>316 SS</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>227</td>
<td>MC + 10 v/o M_2C_3</td>
<td>93</td>
<td>Inc-800</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>228</td>
<td>MC</td>
<td>93</td>
<td>Inc-800</td>
<td>50</td>
<td></td>
</tr>
<tr>
<td>229</td>
<td>MC</td>
<td>93</td>
<td>Inc-800</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>230</td>
<td>MC + 10 v/o M_2C_3</td>
<td>93</td>
<td>Inc-800</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>231</td>
<td>MC + 10 v/o M_2C_3</td>
<td>97</td>
<td>304 SS</td>
<td>50</td>
<td></td>
</tr>
<tr>
<td>232</td>
<td>MC + 10 v/o M_2C_3</td>
<td>97</td>
<td>304 SS</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>233</td>
<td>MC + 10 v/o M_2C_3</td>
<td>97</td>
<td>316 SS</td>
<td>50</td>
<td></td>
</tr>
<tr>
<td>234</td>
<td>MC + 10 v/o M_2C_3</td>
<td>97</td>
<td>316 SS</td>
<td>50</td>
<td></td>
</tr>
<tr>
<td>235</td>
<td>MC + 10 v/o M_2C_3</td>
<td>97</td>
<td>316 SS</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>236</td>
<td>MC + 10 v/o M_2C_3</td>
<td>97</td>
<td>316 SS</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>237</td>
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<td>97</td>
<td>Inc-800</td>
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<td>50</td>
<td></td>
</tr>
<tr>
<td>239</td>
<td>MC + 10 v/o M_2C_3</td>
<td>97</td>
<td>Inc-800</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>240</td>
<td>MC + 10 v/o M_2C_3</td>
<td>97</td>
<td>Inc-800</td>
<td>100</td>
<td></td>
</tr>
</tbody>
</table>

*M = (U_{0.85} + Pu_{0.15})
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