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SODIUM-COOLED REACTORS PROGRAM

Fast Ceramic Reactor Development Program,

Edited by
F. J. Leitz

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Atomic Power Equipment Department
General Electric Company
San Jose, California
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SODIUM-COOLED REACTORS PROGRAM

FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

Fourth Quarterly Report

July-September, 1962

Prepared for the United States Atomic Energy Commission under Contract No. AT(04-3)-189, Project Agreement No. 10

ATOMIC POWER EQUIPMENT DEPARTMENT

GENERAL ELECTRIC

SAN JOSE, CALIFORNIA
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SECTION I

INTRODUCTION

The Fast Ceramic Reactor Development Program is an integrated analytical and experimental program directed toward the development of fast reactors employing ceramic fuels, with particular attention to mixed plutonium-uranium oxide. Its major objectives are:

a. Development of a reliable, high performance fast reactor having nuclear characteristics which provide stable and safe operation, and

b. Demonstration of low fuel cycle cost capability for such a reactor, primarily through achieving high burnup of ceramic fuels operating at high specific power.

Progress during the period July 1 - September 30, 1962 on the currently active tasks of this program is described in subsequent sections.

This is the fourth in a series of quarterly progress reports written in partial fulfillment of Contract AT(04-3)-189, Project Agreement No. 10 between the United States Atomic Energy Commission and the General Electric Company. Prior progress reports to the Commission under this contract include the following:


In addition, the following topical reports have been issued:


GEAP 3486  Fast Oxide Breeder Project I - Fuel Fabrication

Part I - Plutonium-Uranium Dioxide Preparation and Pelletized Fuel Fabrication by J. M. Cleveland, W. C. Cavanaugh

Part II - Fabrication of Plutonium-Uranium Dioxide Specimens by Sunging by M. E. Snyder, W. C. Cowden, August 15, 1960


GEAP 3833  The Post-Irradiation Examination of a PuO₂-UO₂ Fast Reactor Fuel by J. M. Gerhart, November, 1961

GEAP 3856  Experimental Fast Oxide Reactor by K. P. Cohen, M. J. McNelly, B. Wolfe, November 27, 1961


GEAP 4058  Analytical Studies of Transient Effects in Fast Reactor Fuels by R. B. Osborn and D. B. Sherer, August, 1962
SECTION II

SUMMARY

Task B - Vented Fuel Development
The fifth sodium logging capsule was irradiated under the severest test conditions to-date, yet cladding damage did not occur. Experimental design of the Series II capsules is underway and mixed oxide fuel specimens are being fabricated.

The FCR sodium loop has been operated successfully at design conditions for over 400 hours. Fabrication of hyperstoichiometric uranium dioxide fuel has been started.

Conceptual design of a compact dynamic sodium loopsule has been completed.

Task C - Fuel Testing in TREAT
Programmed examination of Series I large diameter UO₂ fuel specimens has been completed. Five capsules have been irradiated in TREAT and examined to identify incipient failure limits for single and repeated transients.

Capsule design for Series II has been completed and test assemblies are in the final stages of fabrication. The Hazard Analysis and Test Specifications for Series II and III were submitted to the Manager of the TREAT facility, and testing will proceed as soon as hazards approval is received.

Task E - Fuel Fabrication Development
Several batches of mixed oxide pellets satisfying density and dimensional specifications have been produced for use in the FCR fuel testing program. Equipment is being set up for the purpose of determining the oxygen to heavy metal ratio in mixed oxide fuel by means of the thermogravimetric technique.

Scouting studies have shown that the plutonium hydroxide-ammonium diuranate precipitate can be directly reduced and sintered to a high density compactible grade oxide in one firing at approximately 1500°C.
Post-irradiation examination of high exposure mixed oxide fuel (≥ 70,000 MWD/T) operated at approximately 20 kw/ft has not revealed gross plutonium migration; the migration observed was within the limits of error, ±10%.

Task G - Reactor Dynamics and Design

Earlier physics calculations on fuel cycles and reactivity coefficients for the FCR reference design were repeated using updated methods and cross sections. The newer cross sections and calculation methods produced an improvement of the fuel cycle economics and a reduction of the Doppler coefficient for the reference core size and composition.

Calculations were made to evaluate the effect of composition homogeneity in FCR on the effective absorption cross sections of uranium-238 and plutonium-239. Results indicate that the assumption of a homogeneous core is valid with insignificant error.

ZPR-III experimental data on an oxide-fueled assembly were compared with

1. multi-group diffusion calculations using the MISTRY code and a 21-group cross section set; and
2. SNG transport calculations made earlier at ANL.

The diffusion and transport results show good agreement.

Transient calculations using the FORE code indicated that rapid reactivity insertions of over a dollar in EFCR can be adequately simulated by a relatively slow addition using the main control system followed by rapid insertion of the remaining reactivity using the fast reactivity excursion device.

FORE calculations on the effects of variations in fuel-gap heat transfer coefficients and fuel thermal conductivities indicate that the net effect of increasing these quantities is to lessen the severity of an excursion for a given steady-state power and a given reactivity input.
SECTION III

TASK B - VENTED FUEL DEVELOPMENT

3.0 Sodium Logging

Four of the six Series I UO$_2$ sodium logging capsules were irradiated during the previous report period (April - June, 1962) and experimental details and results were described in the third quarterly report (GEAP 3981). During the present report period the fifth capsule containing specimen B-1-F was irradiated. As in the previous four experiments no damage to the fuel pin was noted even though more severe test conditions were imposed. The sixth specimen, B-1-E, will be irradiated at maximum power generation rates (approximately 20% greater than the other specimens) pending availability of maximum flux conditions in the Trail Cable Facility.

Experimental design of the Series II mixed oxide capsules is underway and fuel specimens are being fabricated.

3.0.1 Specimen B-1-F

3.0.1.1 Experimental Design

Specimen B-1-F contained four cored UO$_2$ fuel pellets (80 per cent of the active fuel length) with sodium wire inserted into the central axial hole (0.050 inch diameter hole in 0.220 inch diameter pellet). There was no defect in the cladding (0.006 inch thick 347 SS) and NaK was used in place of sodium as a heat transfer medium so the specimen could be taken to full power in one step instead of holding at low power until the sodium had melted. It was expected that this specimen might rupture since the sodium was placed into the portion of the fuel that would normally operate near the melting point of UO$_2$. Sodium at these temperatures has an estimated vapor pressure over twice the bursting pressure of the cladding. The one-step insertion was intended to deter sodium from filling the pellet-to-clad interface void and thereby reduce maximum fuel temperatures through improved heat transfer.

3.0.1.2 Experimental Procedure

Achievement of the design conditions of the experiment required maximum GETR power and maximum axial peak-to-average flux in the Trail Cable Facility. Under optimum
circumstances, these conditions are available for only two to three days of each GETR cycle. However, the reactor was unable to reach maximum power simultaneously with maximum peak-to-average flux during cycle 35 because of operational limitations imposed by other experiments in the reactor. An estimate of capsule performance based on available conditions in the reactor showed that power and temperatures could be realized at least as high as those in the first four experiments. Therefore, the specimen was irradiated for one thermal cycle on July 6, 1962. Indicated NaK temperatures were 850 - 900 F, corresponding to calculated clad surface temperatures of 1000 - 1055 F. Linear power generation rates of 21.3 - 22.4 kw/ft were achieved. (FCR reference conditions corrected for thermal flux effect are about 21 kw/ft maximum.)

3.0.1.3 Experimental Results
Post-irradiation examination showed no evidence of swelling or bowing in the cladding indicating that pressures across the cladding did not exceed about 1000 psi. The pre- and post-irradiation outside diameter of the cladding agreed within estimated limits of measurement (0.002 - 0.003 inches). The pre- and post-irradiation diametral measurements toward the top of the pin suggest some slight clad expansion in that region. It is thought that this is due to the use of an oversize fuel pellet (uncored) in the top of the specimen. Force was used to load this pellet. This would not be indicated on the pre-irradiation measurements since the cladding was dimensioned before it was loaded.

Figure 3-1 shows transferse cross-sections of the fuel pin. Note the sodium remaining in the central hole in the photograph. This is the only observation of sodium made from all eight surfaces exposed from this specimen. Furthermore, this one observation does not account for all of the sodium known to have been fabricated into the specimen. A longitudinal section of this specimen is being made to determine a semi-quantitative material balance on the sodium and gain insight concerning its distribution in the fuel.

3.0.2 Specimen B-1-E: (UO\textsubscript{2} fuel, 0.005 inch defect, 26 kw/ft rod power)
Specimen B-1-E was also designed to require maximum reactor power and peak-to-average flux. This capsule is completely fabricated and ready for irradiation. However, it was not irradiated during the 35th GETR cycle because of insufficient reactor power as mentioned above. The difference in the decision to operate or not operate between capsules B-1-F and B-1-E is because maximum power is a secondary variable in B-1-F while it is a primary variable in B-1-E. B-1-E will be operated pending maximum flux conditions in the Trail Cable Facility.
3.0.3 **Series II Capsules** - Preliminary design is under way on the Series II sodium logging tests based on the following concepts:

a. The experimental design and procedure will be similar to the Series I capsules except that PuO₂-UO₂ mixed oxide fuel will be used.

b. FCR reference size 347 stainless steel cladding will be used (0.250-inch OD by 0.015-inch wall).

c. The defected specimen will be plugged with a soluble or meltable plug.

d. More extensive thermal cycling will be imposed on one of the specimens.

Table III-1 shows a summary of the key variables in these experiments.

Fuel specimens for capsules B-2-A and B-2-B are being fabricated. The natural UO₂ - 20% PuO₂ fuel pellets will have a nominal diameter of 0.219 inch and a length of 0.25 inch. A fuel-clad gap of ≤ 0.003 inch is specified. Specification and fabrication of specimen B-2-C will be based on development of a soluble or meltable plug for the defect hole. This development work is scheduled to begin the end of September.

Preliminary physics and heat transfer calculations indicate that the fuel must operate at a linear power generation rate of approximately 23 kw/ft to reach an \( \dot{\theta} \) of 5000 Btu/hr-ft because of flux depression in the specimen. It is estimated that the maximum linear power generation rate available in the Trail Cable Facility for this specimen design is 24 kw/ft indicating that maximum FCR conditions can be achieved in these tests.

3.0.4 **Pressure Relief Experiments** - Room temperature burst tests using reference FCR cladding were made in an effort to study modes of cladding deformation as a means of pressure relief which could prevent sodium logging failures. A series of tests were performed using test specimens similar to that illustrated in Figure 3-2A. The specimen is comprised of twelve 1/2-inch long by 0.22-inch diameter stainless steel pellets in Type 347 stainless steel, 0.268-inch OD by 0.015-inch wall tubing. The tubing was swaged down to 0.250-inch OD with a tool steel die, then reswaged with an undersized nylon die to insure minimum pellet-to-clad gap. One end of the cladding was open when pressure was applied to the opposite end.

Figures 3-2 B and C illustrate that a similar failure occurs whether the pressure is released into the specimen instantly or gradually and that the tube tends to fail at the weakest point regardless of
**TABLE III-1**

**SERIES II EXPERIMENTAL VARIABLES**

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<th>Operating Temperature-Power</th>
<th>Defect</th>
<th>Thermal Cycles</th>
<th>Remarks</th>
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<tr>
<td>B-2-A</td>
<td>Reference design</td>
<td>None</td>
<td>8</td>
<td>Control capsule</td>
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<tr>
<td>B-2-B</td>
<td>Maximum available</td>
<td>None</td>
<td>1</td>
<td>Cored pellets filled with sodium</td>
</tr>
<tr>
<td>B-2-C</td>
<td>Reference design</td>
<td>0.005</td>
<td>100 - 1000*</td>
<td>Melttable or soluble plug</td>
</tr>
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* Approximately two days of irradiation time for 1000 cycles.
FIG. A  X-RAY PRINT OF TYPICAL TEST SPECIMEN. PRESSURE IS APPLIED FROM THE LEFT. 1X

FIG. B  X-RAY OF BURST SPECIMEN. PRESSURE (13,000 PSIG) WAS INSTANTANEOUSLY RELEASED. 1X

FIG. C  X-RAY OF BURST SPECIMEN. PRESSURE WAS SLOWLY INCREASED TO BURSTING (12,000 PSIG). 1X

FIGURE 3-2  CLADDING BURST SPECIMENS
the presence of pellets and no (small) initial pellet-to-clad gap. The pressure wave is evidently transmitted through the pellet-to-clad gap and along the tubing by clad deformation after which the tubing balloons as if no pellets were present.

The amount of clad deformation has been measured as a function of internal pressure in a shorter but similar type of specimen. Figure 3-3 illustrates the ballooning effect again and the apparent lack of effect due to the presence of pellets and no initial pellet-to-clad gap. The cladding always failed at a point about 4/5 of the specimen length from the pressure source.

The amount of elastic deformation as compared to permanent deformation was measured on one specimen. As illustrated by the data in Table III-2, elastic deformation of about 0.0005-inch on the diameter occurred at 7000 psig before permanent deformation occurred. After 12,000 psig had been reached, about 0.004-inch permanent deformation on the diameter had occurred without failure.

3.1 Na-UO₂ Compatibility

Construction of the sodium loop in Building D was completed. Start-up of the loop was also completed successfully. The completed loop is shown in Figure 3-4. Over 400 hours of loop operation indicate the design temperature and flow conditions were achieved. The loop has been operated at temperatures to 1050 F and at maximum flows to 21.9 gpm. The loop has been successfully cold-trapped to a saturation temperature near 250 F. Sodium samples have been taken and will be analyzed to establish a base for future analysis.

Fabrication of hyperstoichiometric fuel has been started. Initial results indicate that a 50 - 50 mixture of UO₂ and U₃0₈ will produce the desired pellets when pressed and then sintered in wet hydrogen at 1450 F. Varying in-furnace time is used to control the O/U ratio. The pellets are being sintered 13 at a time in a special boat. Analysis from the first pellets fabricated is not yet available. Pellet fabrication will be completed in October and initial tests with UO₂ in the loop started at that time.

3.2 Sodium Loopsule Design

Conceptual design of a loopsule for in-pile study of fuel operated in dynamic sodium systems has continued. The initial approach has been to study the feasibility, limitations, and advantages of the loopsule, and its probable configuration. The loopsule could be sufficiently small and self-contained to permit insertion in any test reactor and to be economically disposable.
FIGURE 3-3  EXPANSION OF FCR CLADDING AT DIFFERENT PRESSURES (ROOM TEMPERATURE)
### TABLE III-2

**MEASUREMENT OF OD OF CLAD AT**

**0 - 11,000 psi PRESSURE**

*(0.25 inch OD × 0.015 inch wall, 347 SS tubing, room temperature)*

<table>
<thead>
<tr>
<th>Gauge Pressure (psi)</th>
<th>OD of Clad Center of Specimen</th>
<th>OD of Clad Near End of Specimen</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>0.2538 ±0.0001</td>
<td>0.2545 ±0.0001</td>
</tr>
<tr>
<td>3000</td>
<td>0.2540</td>
<td>0.2546</td>
</tr>
<tr>
<td>0</td>
<td>0.2538</td>
<td>0.2545</td>
</tr>
<tr>
<td>5000</td>
<td>0.2542</td>
<td>0.2549</td>
</tr>
<tr>
<td>0</td>
<td>0.2538</td>
<td>0.2547</td>
</tr>
<tr>
<td>7000</td>
<td>0.2543</td>
<td>0.2550</td>
</tr>
<tr>
<td>0</td>
<td>0.2539</td>
<td>0.2547</td>
</tr>
<tr>
<td>9000</td>
<td>0.2547</td>
<td>0.2553</td>
</tr>
<tr>
<td>0</td>
<td>0.2543</td>
<td>0.2549</td>
</tr>
<tr>
<td>10000</td>
<td>0.2550</td>
<td>0.2557</td>
</tr>
<tr>
<td>0</td>
<td>0.2544</td>
<td>0.2550</td>
</tr>
<tr>
<td>11000</td>
<td>0.2557</td>
<td>0.2563</td>
</tr>
<tr>
<td>0</td>
<td>0.2550</td>
<td>0.2555</td>
</tr>
<tr>
<td>12000</td>
<td>no measurement taken</td>
<td></td>
</tr>
<tr>
<td>0</td>
<td>0.2580</td>
<td>0.2589</td>
</tr>
</tbody>
</table>
FIGURE 3–4  FCR SODIUM LOOP IN BUILDING D
The optimum loopsule size appears to be a unit designed for one gpm sodium flow as dictated by:

1. reasonably sized lines;
2. reasonable cold trap flow (0.1 gpm);
3. a heat generation capability of about 10 kw.

The preferred design is a hermetically sealed unit containing a pump, cooler, surge tank, cold trap and blower. A possible flow diagram is shown in Figure 3-5 and an isometric view in Figure 3-6. This concept offers the best opportunity for precise oxygen concentration control, eliminates the need and space requirements for a cover gas system, and appears feasible from an operating viewpoint. The design would incorporate two cold traps and a plugging indicator within the sealed unit. One cold trap would be operated prior to installation in-reactor as necessary to establish the desired oxygen level in the sodium. This trap would then be valved out of the system and cooled to freeze the sodium oxide in that location. The second cold trap would be operated to establish fission product deposition and supplement oxygen control in-reactor.

The conceptual design under consideration has been sized adequate for loop configuration estimates, preliminary shielding estimates, and cost estimates.
FIGURE 3-6  SODIUM LOOPSULE CONCEPT
SECTION IV

TASK C - FUEL TESTING IN TREAT

4.0 Summary
The purpose of the program of fuel testing in TREAT is to provide information relative to the transient behavior and performance limits of oxide fuels of interest to the Fast Ceramic Reactor program. Specific areas to be investigated are the expansion and melting characteristics of oxide fuel (including slumping effects), the effects of severe thermal stresses on the cladding, and the possibility of migration or dissociation of constituents within the fuel.

Series I testing of large diameter (EFCR type) UO₂ fuel specimens is nearly complete. Five tests have been completed and incipient failure limits have been identified for single and repeated transients. Series II irradiations are scheduled in the near future to study transient effects on small diameter (FCR type) mixed oxide fuel specimens, followed by Series III to cover the effect of pre-irradiation on the performance of Series II pins.

4.1 General
Preliminary post-irradiation examination of samples #1A through #1E was completed during the quarter. Comparison of results indicates a logical progression in the appearance of the irradiated fuel, with increased transient severity. Table IV-1 (shown on the following page) summarizes the post-irradiation findings for the samples examined to date.

Series I fuel will be kept in accessible storage pending further metallographic and RML examination. Possible areas for subsequent investigation include the oxygen-uranium ratio gradient across the radius, interaction of molten UO₂ with metallic components, thoria-urania reaction, and the presence of extra phases in the system (higheroxides, nitrides, carbides).

4.2 Series I Testing
The irradiation history and post-irradiation examination of capsules 1A, 1B, 1C, and 1D has been discussed in previous quarterly reports (GEAP 3957, January - March and GEAP 3961, April - June, 1962). Dimensioning, gamma scanning, sectioning and photographing was done on each of these
**TABLE IV-1**

**POST-IRRADIATION EXAMINATION OF SERIES I TREAT CAPSULE**

<table>
<thead>
<tr>
<th>Capsule Number</th>
<th>Irradiation History (Reactor Power)</th>
<th>Estimated Sample Temperatures °F</th>
<th>Sample Burnup MWD/T</th>
<th>Post-Irradiation Examination</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>1A</td>
<td>4 transients 19, 89, 162 and 315 MWS</td>
<td>Peak 4300 Avg 3900</td>
<td>$1.87 \times 10^{-2}$</td>
<td>Yes Slight No No Some No No No</td>
<td></td>
</tr>
<tr>
<td>1B</td>
<td>1 transient 420 MWs</td>
<td>Peak 5000+ Avg 4900</td>
<td>$1.35 \times 10^{-2}$</td>
<td>Yes Yes No Slight No No No No</td>
<td></td>
</tr>
<tr>
<td>1C</td>
<td>1 transient 585 MWs</td>
<td>Peak 5400 Avg 5000</td>
<td>$1.64 \times 10^{-2}$</td>
<td>Yes Yes No Yes Some No No No</td>
<td>4% Axial fuel expansion in uppermost section</td>
</tr>
<tr>
<td>1D</td>
<td>1 transient 765 MWs</td>
<td>Peak 7000 Avg 6500</td>
<td>$2.34 \times 10^{-2}$</td>
<td>Yes Yes Yes Yes Yes Yes Yes Yes</td>
<td>Extensive central void and vapor bubbles</td>
</tr>
<tr>
<td>1E</td>
<td>7 transients 575 MWs (typical)</td>
<td>Peak 5700 Avg 5000 (typical)</td>
<td>$9.4 \times 10^{-2}$</td>
<td>Yes Yes Yes Yes No No Yes Yes</td>
<td>Intermittent central void with some vapor bubbles</td>
</tr>
</tbody>
</table>
specimens, as well as a considerable amount of metallographic examination to study melting characteristics, expansion, and void formation.

4.2.1 Sample #1E
This fuel specimen was subjected to a series of seven repeated transients calculated to raise the peak fuel temperature to \( \sim 5700 \) F (typical). This irradiation history was described in some detail in the previous quarterly report (April - June, 1962).

Visual post-irradiation examination of this fuel specimen revealed a highly irregular cladding surface with many undulations and protuberances distributed more or less randomly over the active fuel length (see Figure 4-1, A and B). Longitudinal sectioning of the sample revealed a bubble-like central void formation with no as-fabricated axial or radial gaps remaining (see Figure 4-1, C). When the fuel pellet halves were removed from the cladding, a direct correspondence was observed between the crack pattern on the pellet surface and that on the inside surface of the cladding. (Figure 4-2, A.) This observation plus the melting of the nickel ring indicates extremely close contact (and therefore good conductance) between fuel and clad as a result of these repeated transients.

Macroscopic examination of a transverse section through pellet 13 clearly shows the distortion of both cladding and fuel pellet in this sample (see Figure 4-3). Extensive equi-axed grain growth has occurred over most of the pellet but no columnar grains are apparent in the macrostructure. The bubbles, which are found throughout the center portions of the pellet, are much like those observed in test specimen #1D (see previous quarterly report, April - June, 1962). Detailed measurements based on Figure 4-3 indicate a 4.2\% increase in the area bounded by the cladding with a 3.3\% increase in the average perimeter (cladding centerline). It appears that the outer edge of the fuel pellet remains in its "as-sintered" condition and does not become plastic during the transient. It is presumed that the cladding deformation is due to a ratcheting mechanism between the "pre-shaped" pellet segments as they expand with temperature. In some areas fusion has occurred between these segments closing that portion of the crack nearest the pellet center. The radius of curvature of the outer crust of each segment is less than the original pellet radius. This warping of the pellet perimeter is probably due to the contraction of the pellet center upon cooling.

4.3 Series II and III
Fuel fabrication techniques were developed for mixed oxice, 0.25 inch diameter TREAT specimens and essentially all Series II and III test pins were fabricated during the reporting period.

21
A) – SAMPLE #1E CAPSULE (0 DEG)

B) – SAMPLE #1E CAPSULE (180 DEG)

C) – SAMPLE #1E – LONGITUDINAL SECTION (UPPER END)

FIGURE 4-1  POST IRRADIATION EXAMINATION – SAMPLE #1E
FIGURE 4-2 POST IRRADIATION SECTIONS – SAMPLE #1E
TRANSVERSE SECTION
6X MAGNIFICATION

CLADDING
(.032" THICK
304 S.S.)

CENTRAL
VOID AREA

ENLARGED
EQUI-AXED GRAINS

"AS SINTERED"
GRAIN SIZE

IRRADIATION – SEVEN TRANSIENTS IN TREAT
MAXIMUM FUEL TEMPERATURE ~ 6000°F
(estimated melting point ~ 5000°F)

FIGURE 4-3  SAMPLE # 1E POLISHED CROSS SECTION
Pre-irradiation dimensioning of the completed pins was begun and a procedure for transmission gamma-scanning is under development. Average pellet densities range from 94.7% to 96.8% of theoretical density and diametral fuel-clad gap has been held to a maximum of 0.006 inch (average ~0.004 inch).

Capsule design for Series II has been completed and test assemblies are in the final stages of fabrication. Final assembly, checkout, and calibration of capsules awaits approval of the hazards analysis and scheduling of TREAT irradiation by ANL. Series III capsule design will be nearly identical to Series II, with additional hardware as necessary for pre-irradiation in CETR.

4.3.1 Hazards Analysis
The Hazards Analysis and Test Specifications for Series II and III testing were submitted to the Manager of the TREAT Facility on August 10, 1962. Pertinent sections of these documents are included here to provide a description of equipment design and calculations done in preparation for the planned mixed oxide testing.

4.3.2 Equipment Description and Assembly
The design requirements for the Series II and III TREAT experiments are based on simulation of certain conditions of the FCR reference design. Selection of the desired parameters for design was based on the following criteria:

1. Maintain FCR fuel composition (20% PuO₂ - 80% UO₂)
2. Simulate FCR transient temperature profiles through the fuel and clad.
3. Simulate FCR coolant temperatures.

The primary results desired will be information pertaining to the melting and redistribution of the fuel under transient conditions.

4.3.2.1 Fuel Element
The fuel will be composed of a mixture of 20% PuO₂ and 80% natural UO₂. The mixture is pressed into pellets and sintered to approximately 93% of theoretical density. These pellets are contained in a 1/4 inch OD tube, 0.015 inch wall thickness, type 347 stainless steel with a total fuel length of 6 inches (see Figure 4-4). A one inch long natural UO₂ blanket is located between the top end of the fuel and the 5-1/2 inch long gas reservoir. There is a 1/2 inch long natural UO₂ insulator pellet below the fuel, followed by a washer and the end plug.
The gas reservoir above the fuel duplicates FCR design conditions in the fuel pin. Its use as
a fission gas reservoir in Series II is negligible since very little fission gas will be generated
during the transient. However, it will simulate the FCR design by supplying space for the
fuel vapor to expand if necessary. Also, it is desirable to have Series II and III designed
similarly for comparison purposes, and a fission gas reservoir is necessary in Series III.

It is planned to attempt a measurement of fuel element pressure in Series II tests. This
pressure measurement will be made by attaching a pressure transducer to the bottom end of
the fuel element as shown in Figure 4-5. For these tests, the end plug and insulator pellet
will have a 1/16 inch diameter hole drilled through the center, axially, permitting the trans-
ducer to experience as closely as possible, the actual fuel element pressure.

A calculation of internal pressure capability of the fuel element has been made using the re-
lationships recommended for thin-walled tubing\(^1\) and closed end tubes

\[
\begin{align*}
\text{hoop stress, } \sigma_h &= \frac{P d}{2t} \\
\text{axial stress, } \sigma_a &= \frac{P d}{4t} \\
\text{radial stress, } \sigma_r &= -\frac{P}{2}
\end{align*}
\]

where \( P \) = internal pressure
\( t \) = wall thickness
\( d \) = internal diameter

Combining these stresses according to:

\[
\sigma_y = \sigma_h + \sigma_r + \sigma_a - \sigma_h \sigma_r \sigma_a - \sigma_h \sigma_a - \sigma_h
\]

where \( \sigma_y \) = combined stress,
we obtain the following:

\[
P_y = \frac{4\sigma_y (t/d)}{3 + 6 (t/d) + 4 (t/d)^2}
\]

where \( P_y \) = yield strength

when \( \sigma_y \) = yield stress
Thus, \( P_y = 0.1473 \sigma_y \), when \( t = 0.015 \) inch
\[
d = 0.220 \text{ inch}
\]

Now comparing this to a hoop stress calculation:
\[
P_y = \frac{2\sigma_y t}{d} = 0.1363 \sigma_y
\]

it can be seen that a more conservative estimate is obtained, thus the following radial stress calculations will be based on the hoop stress formula.

Since the maximum pressure will occur early in the transient, a cladding temperature only slightly above preheat (800 F) is assumed. Thus, at 900 F:
\[
\text{Yield pressure, } P_y = (0.1363) (31.5 \times 10^3 \text{ psi})
\]
\[
= 4300 \text{ psi}
\]

By the same reasoning, burst pressure, \( P_b = (0.1363) (64.0 \times 10^3) = 8730 \) psi. If it is assumed that the cladding is restrained by the heat sink before it ruptures, the ultimate strength of the end plug becomes important. In this case, the bursting pressure is derived from the axial stress formula, using the ultimate strength. Thus, at 900 F:
\[
P_b = \frac{4t \sigma_u}{d} = 0.273 (64.0 \times 10^3)
\]
\[
= 17,500 \text{ psi} \quad (2)
\]
\[
\sigma = \text{ultimate strength}
\]

4.3.2.2 Test Capsule

The test capsule design is as shown in Figure 4-5. The exterior of the capsule will be dimensionally the same as a TREAT fuel element with the exception that the tubular section will be cylindrical instead of square, and of a diameter equal to or less than the distance across flats of a square TREAT fuel element. The upper and lower ends will remain square in order to provide alignment with the surrounding fuel elements. Instrumentation and heater connections will be made in the reactor by means of an aluminum holder similar to that presently being used by TREAT for special instrumented fuel elements.
Instrumentation of the capsule will be primarily chromel-alumel thermocouples, and the pressure transducer described previously, which are located as follows:

1. Pressure transducer is connected to the bottom of the fuel element - pressure range:
   0 - 1000 psi (capable of 700% over-pressure), operating temperature limit approximately 1000 °F.

2. Two TC's in the NaK-filled annulus between the heatsink and fuel pin clad; maximum expected temperature approximately 1500 °F.

3. Two TC's in the NaK-filled annulus between the heatsink and capsule wall; maximum expected temperature - approximately 1500 °F.

4. Three TC's located at different circumferential positions at the fuel center along the capsule wall; expected temperature - approximately 1100 °F.

5. One TC located at the fuel center on the outer surface of the thermoflex insulation; maximum expected temperatures - approximately 600 °F.
   (All thermocouples have a maximum temperature limit of approximately 1650 °F.)

Prior to transient operation, the fuel element and heat sink will be electrically heated to about 800 °F to simulate the coolant temperature conditions in the FCR prior to a transient. The heater element for this purpose is wrapped around the interior surface of the thermoflex insulation. Connection to the heater element is made through the same connector that carries the thermocouple leads.

Following assembly and insertion of the fuel specimens, the inner capsule will be heated and about 30 cc of NaK will be introduced through a fill-tube. This tube is then sealed and the inner capsule pressurized with 10 atmospheres (350 psi at 800 °F) of helium to provide insurance against gross NaK boiling and void formation at elevated temperatures. The assembly will be mass-spectrometer leak tested before final connection to the upper aluminum handle and insertion into the test capsule.

A 3/4 inch layer of thermoflex insulation will be placed between the inner capsule and outer shell to minimize the loss of heat while approaching equilibrium temperature. Assuming a conservative value of K (0.05 Btu/hr-ft °F) for this insulation, the heat loss per unit temperature drop will be:
\[
\frac{Q}{\Delta T} = \frac{2 \pi KL}{\ln(D_2/D_1)} = \frac{2 \pi (0.05 \text{ BTU/hr-ft}^\circ \text{F}) (14.375 \text{ in})}{\ln (2.75/1.25) (12 \text{ in/ft}) (60 \text{ min/hr})}
\]

The capsule is expected to reach approximate thermal equilibrium within 20 sec. after a transient sufficient to reach fuel melting. During a typical operation, the average temperature of the inner capsule will be about 950 °F. If we assume the outside shell temperature to be as low as 400 °F, then the heat loss will be:

\[
Q = 0.008 \text{ Btu/min}^\circ \text{F} (950 - 400) \text{ } \text{F}
\]

\[
= 4.4 \text{ Btu/min}
\]

If the total energy input of the pulse is 60 Btu, the heat loss in the 20 sec. required to approximate equilibrium will be about 1.5 Btu or 2.5% of the total sample energy.

The heat capacity of the assembled capsule, heat sink and fuel element is estimated to be 0.203 Btu/°F between 800 °F and the expected equilibrium temperature of 1100 °F.

<table>
<thead>
<tr>
<th>Material</th>
<th>Specific Heat (Btu/°F)</th>
<th>Weight (#)</th>
<th>Btu/°F</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel</td>
<td>0.074</td>
<td>0.1072</td>
<td>0.0072</td>
</tr>
<tr>
<td>Steel</td>
<td>0.120</td>
<td>0.718</td>
<td>0.0861</td>
</tr>
<tr>
<td>NaK</td>
<td>0.210</td>
<td>0.0484</td>
<td>0.0102</td>
</tr>
<tr>
<td>Aluminum</td>
<td>0.270</td>
<td>0.368</td>
<td>0.0995</td>
</tr>
</tbody>
</table>

\[
0.203 \text{ Btu/°F}
\]

Estimates of the internal pressure capabilities of the inner capsule are again based on hoop stress as in the case of the fuel element. Using the relationships suggested by Fautself, and assuming the capsule will be at preheat temperature (800 °F) when subjected to the internal pressure, we find

\[
P_y = 0.0955 (\sigma_y)
\]

\[
= 0.0955 (21.5 \times 10^3) = 2050 \text{ psi}
\]

and

\[
P_b = 0.955 (\sigma_u)
\]

\[
= 0.955 (62.5 \times 10^3)
\]

\[
= 5960 \text{ psi}
\]
Elevated temperatures will tend to reduce internal pressure capability; thus, at the approximate equilibrium temperature (1200 °F), following a severe transient, the yield and burst pressures could become:

\[ P_y = 0.955 \times (16.5 \times 10^3) = 1575 \text{ psi} \]
\[ P_b = 0.955 \times (45.2 \times 10^3) = 4300 \text{ psi} \]

### 4.3.3 Handling of Experimental Apparatus

Since the following experiments will contain mixed oxide fuel (Series II and III) and fission products (Series III) special consideration must be given to the safe handling of the capsule. The fuel will be fabricated and sealed in its clad using glove box operations. It will then be removed from the glove box, and placed in a capsule which contains the aluminum heat sink and NaK. This operation will be performed in a helium atmosphere and will terminate by sealing the capsule under 10 atm of He pressure. The final step is to connect all T.C. leads and place the capsule in the main assembly can. This outer container will be purged with an inert gas and sealed at atmospheric pressure by a double o-ring capable of withstanding pressures up to 200 psi.

Capsule handling for Series II and III should follow the same general procedures as developed for Series I. Special care should be exercised during loading and unloading procedures to avoid dropping or jarring the capsule. It is felt, however, that the chances of such an event creating a plutonium hazard are extremely remote since the fuel has triple containment. Even assuming the outside can were split during such an accident, the inner capsule is so placed that the shock received would be reduced considerably.

Series III differs, in that it will contain fission products upon loading into TREAT. Similar handling procedures must be used here, but the hazards are essentially the same as in the case of unirradiated plutonium. The shipment cask will be modified for this series to provide additional shielding from the pre-irradiated fuel.

### 4.3.4 Mechanical Failure of Experimental Equipment

Test procedures for Series II will be similar to those in Series I as they will provide for increasing the size of the transient power generation until mechanical deformation of the pin is evident; but not for investigation of failure mechanisms beyond this point of incipient damage. It is planned to take the fuel to the melting point on the first transient and work up in transient size until cladding
deformation is achieved. Rupture of the test pin or capsule is not expected under the planned operating limits and conditions. A listing of pressures under various conditions is given in Table IV-2.

### Table IV-2

**CALCULATED MAXIMUM GAS PRESSURE - SERIES II AND III TREAT CAPSULES**

*(Estimated Under Various Conditions At Peak Temperatures)*

#### Series II

a. Typical Operation (65 Mw-sec, peak fuel temperature ~5000 F)

<table>
<thead>
<tr>
<th>Component</th>
<th>Pressure (psia)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Pin</td>
<td>120</td>
</tr>
<tr>
<td>Inner Capsule</td>
<td>300</td>
</tr>
<tr>
<td>Outer Capsule</td>
<td>15</td>
</tr>
</tbody>
</table>

b. Operation at 90 MW-sec (Peak fuel temperature ~7650 F)

<table>
<thead>
<tr>
<th>Component</th>
<th>Pressure (psia)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Pin</td>
<td>180</td>
</tr>
<tr>
<td>Inner Capsule</td>
<td>460</td>
</tr>
<tr>
<td>Outer Capsule</td>
<td>15</td>
</tr>
</tbody>
</table>

c. Transient necessary to produce inner capsule yielding assuming fuel pin failure (180 MW-sec)

<table>
<thead>
<tr>
<th>Component</th>
<th>Pressure (psia)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inner Capsule</td>
<td>2050</td>
</tr>
<tr>
<td>Outer Capsule</td>
<td>16</td>
</tr>
</tbody>
</table>

d. Maximum Accident (260 MW-sec, to burst inner capsule, assuming fuel pin failure)

<table>
<thead>
<tr>
<th>Component</th>
<th>Pressure (psia)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inner Capsule</td>
<td>5900</td>
</tr>
<tr>
<td>Outer Capsule</td>
<td>20 (if inner capsule fails)</td>
</tr>
</tbody>
</table>

#### Series III

**Assume fission gas pressure in the fuel pin after 50,000 MWD/t burnup at room temperature is 235 psia.** *(Based on 100% fission gas release)*

a. Typical Operation (65 MW-sec)

<table>
<thead>
<tr>
<th>Component</th>
<th>Pressure (psia)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Pin</td>
<td>1500</td>
</tr>
<tr>
<td>Inner Capsule</td>
<td>300</td>
</tr>
<tr>
<td>Outer Capsule</td>
<td>15</td>
</tr>
</tbody>
</table>
TABLE IV-2 (Continued)

b. Maximum Accident (200 MW-sec, bursting of inner capsule)

| Inner Capsule | 5900 psia |
| Outer Capsule | 25 psia |

Although it is not planned to approach the burst strength of the fuel element clad, an extreme case where this occurs is illustrated here for hazard purposes. Assume that an accidental transient is produced which increases the fuel element pressure above the hoop burst strength of the clad. Failure of the cladding probably would occur in the unrestrained portion of the fuel element above the heat sink. An amount of vapor equal to the volume of the fuel element at a burst pressure of ~8000 psi (average clad temperature 100 F), would expand into the capsule resulting in a pressure reduction as follows:

Assuming isothermal expansion,

\[ P_1 V_1 = P_2 V_2 \]

\[(8000) (0.502 \text{ in}^3) = P_2 (5.315 \text{ in}^3) \]

\[ P_2 = 750 \text{ psi} \]

If we assume the helium in the capsule to be at an average temperature of 800 F, the total pressure will be:

\[ P_T = P_2 + P_{He} = 750 + 350 = 1100 \text{ psi} \]

In the actual case, where mixing of the expelled vapor and the capsule NaK would occur, heat loss from the vapor to the NaK and then to the heat sink would reduce the temperature rapidly. This would simultaneously reduce the fuel vapor pressure to a negligible level and increase the He and NaK-vapor pressure. It is estimated under these conditions, that the "mixed-mean" pressure will rise to a maximum of about 1000 psi or approximately the same value as found by simple expansion. (See Maximum Accident.) Since the yield pressure of the capsule at 1200 F is nearly 1600 psi, this type of failure would not constitute a hazard.

Melting or distortion of the fuel element or heat sink will occur only through extreme heat generation in the sample. Calculations indicate that a TREAT transient of about 65 MW-sec is necessary to melt a substantial portion of the fuel. Peak cladding temperatures for such a transient would be
approximately 1800 F, and will be sustained for only a few milliseconds. At the melting point, fuel vapor pressure is negligible and a clad strength of ~1800 psi still exists. In order to reach the melting temperature of the cladding (2250 F), a transient capable of vaporizing the UO₂ will be necessary (~180 MW-sec). It should be noted that transients of this severity are nearly a factor of two greater than those required for the experiment.

4.3.5 Heat Transfer Computer Calculations

4.3.5.1 Summary of Results

The temperature distributions resulting from various TREAT transients were investigated with the TIGER V code for the Philco 2000 computer. The results for two of these transients, 64 and 91 MW-Sec, will be summarized here.

The analysis indicates that a 64-MW-sec transient should melt a substantial portion of the fuel. Estimated temperatures are shown in Figures 4-6 and 4-7. Neglecting the heat of fusion, it can be seen that the volumetric average fuel temperature peaks at 6380 F. If the 1180 F equivalent heat of fusion temperature rise is considered, the average fuel temperature peaks only slightly above the 5000 F melting point (see Figure 4-6). Maximum temperatures indicated for this transient are: fuel = 5760 F (includes heat of fusion); clad = 1830 F; NaK = 1480 F (vapor pressure = 1.85 atm); aluminum heat sink = 1100 F (estimated equilibrium temperature). The maximum heat flux through the clad is calculated to be 2.42 × (10)⁶ Btu/hr-ft².

The results of the 91 MW-sec transient are shown in Figures 4-8 and 4-9. This transient takes the fuel well above melting and would probably result in some vaporization. Maximum temperatures are: fuel = 7660 F (includes heat of fusion); clad = 2085 F; NaK = 1790 F (vapor pressure = 4.4 atm); aluminum = 1200 F (~10% through the heat of fusion).

The analysis of these two transients was based upon a gap coefficient between the fuel and clad of 1,000 Btu/hr-ft²°F. Increasing the coefficient to 5,000 Btu/hr-ft²°F results in a 17% drop in the peak volumetric average fuel temperature rise and a 38% increase in peak clad temperature rise and peak heat flux through the clad.
FIGURE 4-6 TIME VARIATION OF CAPSULE TEMPERATURES FOR 64 MW-SEC TRANSIENT
FIGURE 4-7  RADIAL FUEL TEMPERATURE PROFILES FOR 64 MW-SEC TRANSIENT (HEAT OF FUSION INCLUDED)
FIGURE 4-8  TIME VARIATION OF CAPSULE TEMPERATURES FOR 91 MW-SEC TRANSIENT
FIGURE 4-9 RADIAL FUEL TEMPERATURE PROFILES FOR 91 MW-SEC TRANSIENT – (HEAT OF FUSION INCLUDED)
4.3.5.2 Method of Analysis

A model to simulate the TREAT capsule was formed with a group of heat transfer nodes as shown in Figure 4-10. The nodes contained in the 3-inch length simulate the NaK above the active fuel region (Figure 4-5) and were added to obtain a more accurate estimate of equilibrium temperature. The temperature of the outer capsule wall was assumed constant at 600 F. A base gap coefficient of 1,000 Btu/hr-ft°F was used between the fuel and clad. The effect of raising the coefficient a factor of five was also investigated. The TIGER V code determined the temperature distribution for a given time-radial variation in power generation. Axial and angular variations were assumed negligible.

For the 91 MW-sec transient, the time variation of average power was assumed to be similar to that obtained for transient #385 (capsule 1A-2nd transient) in Series I. The 64 MW-sec transient was obtained by chopping the 91 MW-sec curve shortly after the peak. Both power curves are shown in Figure 4-11.

The material properties used for the calculations are tabulated below. The densities were taken at room temperature since the dimensions of the model assume room temperature. The specific heats and conductivities were taken at the expected average operating temperatures during the particular transient in question.

<table>
<thead>
<tr>
<th>Material</th>
<th>Specific Heat, B/lb°F</th>
<th>Density, Lb/in³</th>
<th>Thermal Conductivity, B/hr-ft°F</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel - Pu-UO₂</td>
<td>0.08</td>
<td>0.376</td>
<td>2.0</td>
</tr>
<tr>
<td>Steel</td>
<td>0.12</td>
<td>0.28</td>
<td>12.5</td>
</tr>
<tr>
<td>NaK, (78% K)</td>
<td>0.209</td>
<td>0.027</td>
<td>15.1</td>
</tr>
<tr>
<td>Thermoflex</td>
<td>0.27</td>
<td>0.00695</td>
<td>0.0542</td>
</tr>
<tr>
<td>Aluminum</td>
<td>0.27</td>
<td>0.0971</td>
<td>100</td>
</tr>
<tr>
<td>Helium</td>
<td>1.248</td>
<td>0.00133</td>
<td>0.137</td>
</tr>
</tbody>
</table>

4.3.6 Nuclear Effects on Reactor Performance

Calculations on the effect of the sample and test capsule on the reactivity of the TREAT reactor were performed using the Philco-2000 computer and the MISY program. This program involves a multi-region cylindrical approximation to TREAT and a multi-group diffusion calculation. The thermal neutron flux distribution was checked by a P₃ calculation. Results of this computation indicate that
FIGURE 4-10 TIGER V MODEL OF SERIES II TREAT CAPSULE
FIGURE 4-11 POWER CURVES FOR 64 AND 91 MW-SEC TRANSIENTS

REACTIVITY INSERTION
= 1.0% ΔK
replacing the center fuel element in TREAT with our experiment will result in a net decrease in reactivity of about - 2.3% Δk/k. The effect of the test assembly on the temperature coefficient of reactivity of TREAT was found to be negligible and the effect on neutron lifetime would likewise be small. The change in control rod worth due to insertion of the experiment was computed to be less than +5%, which is in agreement with TREAT operating experience for experiments of comparable reactivity worth.*

These calculations were performed for both series II and series III fuel and the effects were found to be similar. Thus, the above data pertains to both series.

4.3.7 Radiation Hazard

4.3.7.1 Series II

From the result of the diffusion calculation, it has been estimated that an average energy density of 240 BTU/in³, corresponding to a total of \(3.6 \times 10^{16}\) fissions in the sample would result in the gamma ray dose rates listed below:

<table>
<thead>
<tr>
<th>Activated Component</th>
<th>Time After Shutdown</th>
<th>Total Activation (Curies)</th>
<th>Distance From Center of Capsule</th>
<th>Dose Rate r/hr.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel</td>
<td>2 days</td>
<td>0.034</td>
<td>6 inches</td>
<td>0.57</td>
</tr>
<tr>
<td>NaK</td>
<td>2 days</td>
<td>0.005</td>
<td>6 inches</td>
<td>0.0005</td>
</tr>
<tr>
<td>Aluminum</td>
<td>---</td>
<td>---</td>
<td>---</td>
<td>---</td>
</tr>
<tr>
<td>Steel</td>
<td>2 days</td>
<td>0.002</td>
<td>6 inches</td>
<td>0.02</td>
</tr>
</tbody>
</table>

The indicated fuel activity is due to fission products and was calculated for a representative gamma ray energy of 2 mev. Sodium activation is due to the formation of Sodium 24. The steel activity is that of stainless steel 304 as given by the method of Bopp and Sisman. The dose rates at 6 inches include the shielding effects of the capsule.

Fission products provide the primary gamma ray source since activation of other material can be seen to be small by comparison. It is calculated that increasing the time following the transient from 2 days to 2 weeks decreases the dose rates by a factor of 6. Two hours after the transient the fission product dose rates are expected to be a factor of 6 greater than those at 2 days.

* Private communication with F. Krin, ANL - Idaho Division
In the event of a major accident, fission gases may be released from the capsule. Thus, it would be desirable to know the total activity of these gases. The following table lists the estimated activities of the krypton and xenon gases as well as those of the iodines, following a transient which results in an average of 240 Btu/in$^3$ in the sample (~70 MW-sec). The activity values are for the total fission product yields with no radioactive decay or buildup (immediately following the transient). Fission product yields and half life values were taken from data reported by S. Katcoff. (5)

**FISSION PRODUCT GAS ACTIVATION**

(For $3.6 \times 10^{16}$ Fissions)

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Half-Life</th>
<th>Yield</th>
<th>Immediate Activity - Curies</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kr$^{85}$</td>
<td>10.6 yr.</td>
<td>0.293</td>
<td>$2.9 \times 10^{-7}$</td>
</tr>
<tr>
<td>Xe$^{133}$</td>
<td>5.27 d.</td>
<td>6.62</td>
<td>$4.9 \times 10^{-3}$</td>
</tr>
<tr>
<td>Xe$^{135}$</td>
<td>9.2 hr.</td>
<td>6.3</td>
<td>$6.4 \times 10^{-2}$</td>
</tr>
<tr>
<td>I$^{129}$</td>
<td>$1.7 \times 10^7$ yr.</td>
<td>0.8</td>
<td>$5 \times 10^{-12}$</td>
</tr>
<tr>
<td>I$^{130}$</td>
<td>12.6 hr.</td>
<td>$5 \times 10^{-4}$</td>
<td>$3.7 \times 10^{-6}$</td>
</tr>
<tr>
<td>I$^{131}$</td>
<td>8.05 d.</td>
<td>3.1</td>
<td>$1.5 \times 10^{-3}$</td>
</tr>
<tr>
<td>I$^{133}$</td>
<td>20.8 hr.</td>
<td>6.9</td>
<td>$3.1 \times 10^{-2}$</td>
</tr>
<tr>
<td>I$^{134}$</td>
<td>52.5 min.</td>
<td>7.8</td>
<td>0.83</td>
</tr>
<tr>
<td>I$^{135}$</td>
<td>6.7 hr.</td>
<td>6.1</td>
<td>$8.5 \times 10^{-2}$</td>
</tr>
<tr>
<td>I$^{136}$</td>
<td>86 sec.</td>
<td>3.1</td>
<td>12.0</td>
</tr>
</tbody>
</table>

The activities of I-134, I-135, and I-136 will die out quickly with time. The Xe-135 activity will decrease more slowly, while Xe-133, I-131, and I-133 may increase in activity for a short time after shutdown.

The probability of release of fission products or radioactive gases during normal operation should be quite low due to the triple containment of the fuel and the air-tight seals at each point.
4.3.7.2 Series III

Since Series III capsules will be pre-irradiated (50,000 MWD/t) the radiation hazards will be increased over Series II. The fuel will be similar in all respects to series II except that prior to irradiation it will consist of 26% PuO$_2$ so that after irradiation the compositions are essentially identical (20% PuO$_2$ - 80% Natural UO$_2$).

The activity evolved upon irradiation will result from two main sources; fission products, and activated stainless steel. The results of calculations performed are presented in the following table IV-3. The fission products have been divided into seven energy groups (6) to give better results for the shielding calculations shown later. The activity of four different isotopes activated in the steel are also included.

**TABLE IV-3**

**CALCULATED TOTAL ACTIVITY AT VARIOUS TIMES AFTER REMOVAL FROM GETR - SERIES III TREAT CAPSULES**

(Assume equilibrium is reached at a power generation of 20 Kw/ft)

**A. Decay Time = 10 days**

**Fission Products**

<table>
<thead>
<tr>
<th>Group</th>
<th>Energy in Mev</th>
<th>Mev/Sec</th>
<th>Curies</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>0.25</td>
<td>6.6 x 10$^{12}$</td>
<td>700</td>
</tr>
<tr>
<td>II</td>
<td>0.65</td>
<td>4.5 x 10$^{13}$</td>
<td>1870</td>
</tr>
<tr>
<td>III</td>
<td>1.225</td>
<td>1.9 x 10$^{12}$</td>
<td>42</td>
</tr>
<tr>
<td>IV</td>
<td>1.65</td>
<td>2.2 x 10$^{13}$</td>
<td>360</td>
</tr>
<tr>
<td>V</td>
<td>2.0</td>
<td>8.7 x 10$^{11}$</td>
<td>12</td>
</tr>
<tr>
<td>VI</td>
<td>2.5</td>
<td>2.3 x 10$^{12}$</td>
<td>25</td>
</tr>
<tr>
<td>VII</td>
<td>2.9</td>
<td>1.08 x 10$^{10}$</td>
<td>1</td>
</tr>
</tbody>
</table>

---

3010
Induced Activity (347 SS)

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Energy in Mev</th>
<th>T-1/2</th>
<th>Curies</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cr$^{51}$</td>
<td>0.32</td>
<td>25 d</td>
<td>91</td>
</tr>
<tr>
<td>Fe$^{59}$</td>
<td>1.2</td>
<td>45 d</td>
<td>27</td>
</tr>
<tr>
<td>Ta$^{182}$</td>
<td>1.2</td>
<td>120 d</td>
<td>23</td>
</tr>
<tr>
<td>Co$^{60}$</td>
<td>1.25</td>
<td>5.3 y</td>
<td>54</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>196</td>
</tr>
</tbody>
</table>

Total = 3206

B. Decay Time = 50 days

Fission Products

<table>
<thead>
<tr>
<th>Group</th>
<th>Energy in Mev</th>
<th>Mev/Sec</th>
<th>Curies</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>0.25</td>
<td>$6.5 \times 10^{11}$</td>
<td>70.0</td>
</tr>
<tr>
<td>II</td>
<td>0.65</td>
<td>$1.25 \times 10^{13}$</td>
<td>522.0</td>
</tr>
<tr>
<td>III</td>
<td>1.225</td>
<td>$1.15 \times 10^{10}$</td>
<td>0.3</td>
</tr>
<tr>
<td>IV</td>
<td>1.65</td>
<td>$2.90 \times 10^{12}$</td>
<td>46.0</td>
</tr>
<tr>
<td>V</td>
<td>2.0</td>
<td>$1.95 \times 10^{11}$</td>
<td>2.6</td>
</tr>
<tr>
<td>VI</td>
<td>2.5</td>
<td>$5.20 \times 10^{11}$</td>
<td>5.6</td>
</tr>
<tr>
<td>VII</td>
<td>2.9</td>
<td>$9.8 \times 10^{10}$</td>
<td>0.9</td>
</tr>
</tbody>
</table>

Total = 647.4

Induced Activity (347 SS)

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Energy in Mev</th>
<th>T-1/2</th>
<th>Curies</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cr$^{51}$</td>
<td>0.32</td>
<td>25 d</td>
<td>32</td>
</tr>
<tr>
<td>Fe$^{59}$</td>
<td>1.2</td>
<td>45 d</td>
<td>15</td>
</tr>
<tr>
<td>Ta$^{189}$</td>
<td>1.2</td>
<td>120 d</td>
<td>19</td>
</tr>
<tr>
<td>Co$^{60}$</td>
<td>1.25</td>
<td>5.3 y</td>
<td>53</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>119</td>
</tr>
</tbody>
</table>

Total = 766.4
C. Decay Time = 100 days

Fission Products

<table>
<thead>
<tr>
<th>Group</th>
<th>Energy in Mev</th>
<th>Mev</th>
<th>Curies</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>0.25</td>
<td>$3.6 \times 10^{10}$</td>
<td>3.9</td>
</tr>
<tr>
<td>II</td>
<td>0.65</td>
<td>$2.55 \times 10^{12}$</td>
<td>106.0</td>
</tr>
<tr>
<td>III</td>
<td>1.225</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>IV</td>
<td>1.65</td>
<td>$2.2 \times 10^{11}$</td>
<td>3.6</td>
</tr>
<tr>
<td>V</td>
<td>2.0</td>
<td>$3.0 \times 10^{10}$</td>
<td>0.4</td>
</tr>
<tr>
<td>VI</td>
<td>2.5</td>
<td>$8.0 \times 10^{10}$</td>
<td>0.9</td>
</tr>
<tr>
<td>VII</td>
<td>2.9</td>
<td>$8.6 \times 10^{10}$</td>
<td>0.8</td>
</tr>
</tbody>
</table>

115.6

Induced Activity (347 SS)

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Energy in Mev</th>
<th>T-1/2</th>
<th>Curies</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cr$^{51}$</td>
<td>0.32</td>
<td>25 d</td>
<td>8.3</td>
</tr>
<tr>
<td>Fe$^{59}$</td>
<td>1.2</td>
<td>45 d</td>
<td>7.0</td>
</tr>
<tr>
<td>Ta$^{189}$</td>
<td>1.2</td>
<td>120 d</td>
<td>14.1</td>
</tr>
<tr>
<td>Co$^{60}$</td>
<td>1.25</td>
<td>5.3 y</td>
<td>52</td>
</tr>
</tbody>
</table>

81.4

Total = 197.0

As shown in Table IV-3 the activities are listed for three different decay times after removal from GETR. It is expected that 1-1/2 to 2 months will be required to encapsulate the irradiated fuel in a TREAT capsule after it has been removed from GETR. Therefore, the 50 day decay time should give a representative number for the activity remaining upon shipment, all work is performed in a hot-cell, providing ample protection for the higher activity involved.

Preliminary design has been performed on the shipping cask which will contain the Series III capsule enroute to TREAT. It was calculated that a lead wall thickness of 7 inches would reduce the activity on the outside of the cask to approximately 100 Mr/hr, $^{(7)}$ based on the 50 day
decay time activity, which includes seven energy groups of fission product gammas, plus the induced activity.

Consideration was also given to the activity existing outside the TREAT fuel transfer coffin, when used for transfer of the irradiated capsule to or from the reactor. Since 7 inches of lead is calculated to be sufficient for shipping, the 6 inch thickness in the loading cask, plus a considerable distance (10 feet) between it and any personnel should be sufficient to reduce the activity to below 100 Mr/hr under the same condition (~50 days decaytime).

Since there is a possibility that the capsule might be broken during shipment or loading, an estimate of the fission gas activity was made. The four elements, I, Xe, Kr, Br, are assumed completely in a gaseous state which permits them to escape if the capsule fails. (8) Xe and Kr are gases at room temperatures whereas Br and I must be at elevated temperatures, 130 F and 350 F respectively. The activities of the various isotopes are given in Table IV-4.

This shows that the gas activity is extremely low after 50 days decay (~6 curies). Thus by the time the capsule is removed from the hot cell, the fission gas activity will not present a hazard.

4.3.8 Maximum Accident

As noted previously, normal operations within test limits will not produce excessive temperature or pressure in the test capsule. In order to better evaluate capsule capabilities, however, it is advisable to analyze more extreme conditions. Hence, a transient has been determined which may produce pressures up to the burst point of the capsule (maximum accident).

It was assumed that the primary fuel containment (cladding) failed by some means, and allowed the NaK to come into contact and mix with molten fuel. The pressure necessary to produce capsule bursting (at 800 F) was found to be ~400 atms. The temperature of the NaK-fuel mixture necessary to cause this pressure is 4500 F. At this temperature the fuel vapor pressure is insignificant, and the major pressure contributions arise from the NaK vapor and He gas. Void formation in the NaK will not occur as long as the capsule remains a closed, leak-tight vessel. However, the NaK vapor pressure will begin to increase more rapidly after 3000 F, and thus become a significant source of pressure. The Helium, which is filled at a pressure of 10 atms at room temperature is assumed to rise as a function of temperature with constant volume. The total pressure at 4500 F is broken down as follows:

4-34
## TABLE IV-4

CALCULATED FISSION GAS ACTIVITY AT VARIOUS TIMES AFTER REMOVAL FROM GETR-
SERIES III TREAT CAPSULES

(Assume Equilibrium Has Been Reached)

<table>
<thead>
<tr>
<th>Isotope</th>
<th>T-1/2</th>
<th>Yield %</th>
<th>Curies</th>
<th>Mev</th>
<th>Sec</th>
<th>Curies</th>
<th>Mev</th>
<th>Sec</th>
<th>Curies</th>
<th>Mev</th>
<th>Sec</th>
<th>Curies</th>
<th>Mev</th>
<th>Sec</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kr(^{83})m</td>
<td>1.86 h</td>
<td>0.48</td>
<td>41</td>
<td>6.31 × 10(^{10})</td>
<td></td>
<td>28</td>
<td>4.32 × 10(^{10})</td>
<td></td>
<td>0</td>
<td>0</td>
<td></td>
<td>0</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>Kr(^{85})m</td>
<td>4.4 h</td>
<td>1.5</td>
<td>127</td>
<td>8.5 × 10(^{11})</td>
<td></td>
<td>108</td>
<td>7.24 × 10(^{11})</td>
<td></td>
<td>0</td>
<td>0</td>
<td></td>
<td>0</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>Kr(^{85})</td>
<td>10.4 y</td>
<td>0.3</td>
<td>2</td>
<td>~0</td>
<td></td>
<td>2</td>
<td>~0</td>
<td></td>
<td>2</td>
<td>~0</td>
<td></td>
<td>2</td>
<td>~0</td>
<td></td>
</tr>
<tr>
<td>Kr(^{87})</td>
<td>1.3 h</td>
<td>2.7</td>
<td>229</td>
<td>4.74 × 10(^{12})</td>
<td></td>
<td>134</td>
<td>2.78 × 10(^{12})</td>
<td></td>
<td>0</td>
<td>0</td>
<td></td>
<td>0</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>Kr(^{88})</td>
<td>2.8 h</td>
<td>3.7</td>
<td>312</td>
<td>2.38 × 10(^{13})</td>
<td></td>
<td>243</td>
<td>1.86 × 10(^{13})</td>
<td></td>
<td>0</td>
<td>0</td>
<td></td>
<td>0</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>Kr(^{89})</td>
<td>3.2 m</td>
<td>4.6</td>
<td>386</td>
<td>0</td>
<td></td>
<td>0</td>
<td>0</td>
<td></td>
<td>0</td>
<td>0</td>
<td></td>
<td>0</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>Xe(^{131})m</td>
<td>12.0 d</td>
<td>0.03</td>
<td>3.0</td>
<td>1.81 × 10(^{10})</td>
<td></td>
<td>3.0</td>
<td>1.81 × 10(^{10})</td>
<td></td>
<td>1.0</td>
<td>6.03 × 10(^{8})</td>
<td></td>
<td>0</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>Xe(^{133})</td>
<td>5.27 d</td>
<td>6.5</td>
<td>550</td>
<td>1.65 × 10(^{12})</td>
<td></td>
<td>547</td>
<td>1.64 × 10(^{12})</td>
<td></td>
<td>145</td>
<td>4.35 × 10(^{11})</td>
<td>1.0</td>
<td>3.00 × 10(^{9})</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Xe(^{133})m</td>
<td>2.3 d</td>
<td>0.16</td>
<td>14</td>
<td>1.21 × 10(^{12})</td>
<td></td>
<td>14</td>
<td>1.21 × 10(^{11})</td>
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<td>8.62 × 10(^{9})</td>
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<tr>
<td>Xe(^{135})</td>
<td>9.2 h</td>
<td>6.2</td>
<td>523</td>
<td>5.20 × 10(^{12})</td>
<td></td>
<td>484</td>
<td>4.80 × 10(^{12})</td>
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<td>0</td>
<td>0</td>
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<tr>
<td>Xe(^{135})m</td>
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<td>152</td>
<td>2.97 × 10(^{12})</td>
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<td>1.92 × 10(^{11})</td>
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<tr>
<td>Xe(^{138})</td>
<td>17 m</td>
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<td>465</td>
<td>~0</td>
<td></td>
<td>42</td>
<td>~0</td>
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<td>0</td>
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<td></td>
<td></td>
</tr>
<tr>
<td>Xe(^{139})</td>
<td>41 s</td>
<td>4.7</td>
<td>2030</td>
<td>~0</td>
<td>0</td>
<td>0</td>
<td>~0</td>
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<td>0</td>
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<td>0</td>
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<tr>
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<td>5.66 × 10(^{9})</td>
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</tr>
<tr>
<td>Br(^{84})</td>
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<td>93</td>
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<td>2.22 × 10(^{11})</td>
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### TABLE IV-4 (Continued)

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<th>$T_{1/2}$</th>
<th>Yield %</th>
<th>Curies</th>
<th>Mev</th>
<th>Curies</th>
<th>Mev</th>
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<th>Mev</th>
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</tr>
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<td>Br$^{88}$</td>
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<td>I$^{131}$</td>
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<td>2.9</td>
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<td>245</td>
<td>$3.54 \times 10^{12}$</td>
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<td>$4.33 \times 10^{10}$</td>
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<td>274</td>
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<td>21 h</td>
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<td>52 m</td>
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<tr>
<td>I$^{135}$</td>
<td>6.7 h</td>
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<td>498</td>
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<td>0</td>
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<td>$\sim 0$</td>
<td>0</td>
<td>0</td>
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<td>0</td>
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<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>I$^{137}$</td>
<td>22 s</td>
<td>4.9</td>
<td>414</td>
<td>$\sim 0$</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>I$^{138}$</td>
<td>5.9 s</td>
<td>3.4</td>
<td>287</td>
<td></td>
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<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>TOTAL</td>
<td></td>
<td></td>
<td>8832</td>
<td>$1.80 \times 10^{15}$</td>
<td>3458</td>
<td>$1.03 \times 10^{14}$</td>
<td>253</td>
<td>$1.94 \times 10^{12}$</td>
<td>6.0</td>
<td>$4.63 \times 10^{10}$</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
\[ \text{NaK vapor pressure} = 342 \text{ atm} \]
\[ \text{Helium pressure} = 58 \text{ atm} \]
\[ \text{Total} = 400 \text{ atm} \]

It is assumed that this pressure (~400 atm) is obtained immediately after the transient has peaked and that the heat sink and capsule clad have experienced very little temperature rise at this point. After this, the heat is dissipated throughout the capsule and equilibrium is reached.

It is estimated that these extreme conditions of temperature and pressure could be produced by providing a 260 MW·sec transient (considered the maximum accident). However, these temperatures and pressures are expected for only a few milliseconds after the peak fuel temperature is reached, and will decrease rapidly as the transient peak is passed. It should be pointed out here, that this transient is nearly a factor of 3 greater than the maximum range of expected operation.

If the capsule withstands the pressures generated in this transient, an equilibrium temperature of approximately 1290 F will be reached. This is well under the melting point of the capsule wall, and complete containment is expected. However, the aluminum heat sink will have melted, but no reactions are expected between the Al, the NaK, fuel or steel. Thus, the capsule pressure will be low (approximately 35 atms) compared to the allowable capsule internal pressure of ~100 atm at 1250 F.

It is concluded, therefore, that the mechanical integrity of the capsule is sufficient to preclude violent rupture under transient conditions of up to at least 260 MW·sec and could probably be expected to go somewhat higher since all assumptions and estimates were conservative (i.e. no heat dissipation to the sink for the first 3 seconds. This could allow a 5 - 10% increase in the maximum allowable transient level).

Since it is conceivable that a transient could accidently be produced that would cause bursting of the inner capsule, a determination of the effect of such an event on the TREAT reactor was made. The expansion of the 400 atm pressure into the outer capsule would result in an over-all pressure of about 20 psi. Since plutonium will be present (also fission products in Series III), this pressure must be completely contained in the outer capsule. Thus, the seal provided in Series I has been replaced by a double O-ring capable of withstanding 200 psi.
The expansion of fuel, first into the inner capsule and then to the outer capsule, is expected to produce a positive effect on total TREAT reactivity because of the decrease in self-shielding. It is calculated that a uniform distribution of the specimen throughout the inner capsule would increase the TREAT reactivity by only 0.16% Δk/k. If the test capsule failed and released its contents to the outer can in the central region of the core, a reactivity insertion of less than 0.2% Δk/k, above that of the core with a normal test specimen would result. If it is assumed that the entire capsule failed and the specimen fuel were distributed uniformly throughout the TREAT core, the reactivity insertion would be 0.7% Δk/k.
REFERENCES


SECTION V

TASK E - FUEL FABRICATION DEVELOPMENT

5.0 Introduction

The objective of Task E is the development of fuel with properties meeting prescribed technical requirements and the development of fabrication methods which are economically attractive. Effort during this quarter has been directed toward (1) establishing the process parameters which need to be controlled in order to produce consistent fuel samples; (2) preparing powder and pellets of acceptable standards to be used in other fuel testing phases of the program; (3) experimentally measuring the degree of plutonium migration in high exposure fuel specimens; and (4) developing a method whereby the plutonium hydroxide-ammonium diuranate co-precipitate may be compacted and sintered directly to high density compactible grade powder.

5.1 Preparation of Mixed Oxide Powder

A 270 gram batch (B-4) of mixed oxide powder was prepared for the Series II specimens of the Task C program. The feed solution contained 28 g/l Pu 72 g/l natural U and ~1M nitric acid. Precipitation was accomplished by addition of ammonium hydroxide to a final pH of 7 to 8. The coprecipitate was stirred for 15 minutes at ambient temperature and then filtered. The cake was washed three times with 0.1M NH₄OH and dried with air and acetone. The reduction cycle consisted of heating to 750 C, holding at 750 - 780 C for three hours, and cooling to room temperature in flowing 6% H₂ - 94% He. The reduced powder was ball milled for 7-1/2 hours to provide a -325 mesh powder for pressing. Pellets produced from this powder had uniform shrinkage and high density.

A 170 gram batch (B-5) was processed in which the feed solution consisted of dissolved mixed oxide pellets and miscellaneous feed sample solutions. The dissolved pellets came from batch B-1 and were outside specification limits. The purpose of processing batch B-5 was to determine if recycling of material results in acceptable powder. The pellets produced from the recycled material consisted of an unusually high number of warped pellets and pellets with blisters, cracks, and chips. It has not yet been determined what characteristic of the feed solution or powder preparation process is the controlling factor in giving the unsatisfactory results.
The feed solution for B-5 contained 20 g/l Pu, 80 g/l natural U, and ~4M nitric acid. Precipitation was accomplished by addition of ammonium hydroxide to a final pH of 7 to 8. The coprecipitate was stirred for 10 minutes at ambient temperature and filtration initiated. The vacuum pump supplying vacuum for filtration developed difficulties and finally failed; therefore, filtration and cake washing and drying were delayed and the total cycle required seven days. The reduction cycle consisted of heating to 750 C, holding at 750 - 770 C for two hours, and cooling to room temperature in flowing 6% - 94% He. The reduced powder was ball milled for eight hours to provide a -325 mesh powder for pressing. Tap density of the mixed oxide was 3.02 g/cc as compared to previous densities of 3.32 to 3.63 g/cc for previous 20% Pu - 80% normal U mixed oxide powder.

5.2 Mixed Oxide Pellet Fabrication

Several batches of powder (B-3, B-4, and B-5) were pressed and sintered into pellets. Pressing pressure was varied between 9,500 and 23,000 psig and the sintering temperature between 1480 and 1620 C. Sintering time was 4 to 5 hours. Significant data are presented in Tables V-1, V-2, and V-3, for batches B-3, B-4, and B-5 respectively.

Sintered density is found to vary inversely with green density and pelletizing pressure. One exception is batch B-4 where the sintered density does not appear to vary with pelletizing pressure.

Sintering temperatures above 1610 C have been found to result in pellet warpage. The calculated densities are also low due to the fact that the densities are determined from measurements of diameter, length and weight. Hence, hour-glass configurations result in low calculated densities.

As noted in Table V-3, the pellets produced from batch B-5 had an unusually high number of blisters, cracks, and chips and were more frequently warped. As discussed under powder preparation, batch B-5 powder was prepared from recycled feed material. Additional work is required to determine what is influencing the powder in such a manner to cause a higher rejection rate.

Additional investigations are required to further define the important process parameters which need to be controlled in order to produce pellets with consistent dimensions, density and purity.

5.3 Mixed Oxide Characterization

An analytical balance, tube furnace and auxiliary apparatus have been ordered for the determination of oxygen-to-metal ratio in mixed oxides using the method reported by Brett and Russell of Harwell.
## TABLE V-1

PRESSING AND SINTERING DATA FOR POWDER BATCH B-3

(20% Pu-80% Normal U Mixed Oxide)

<table>
<thead>
<tr>
<th>Pellet Number</th>
<th>Pelletizing Pressure (psig)</th>
<th>Green Density (% Theoretical)</th>
<th>Sintering Temp. (°C)</th>
<th>Sintered Density (% Theoretical)</th>
<th>Firing Shrinkage (%)</th>
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<td></td>
<td></td>
<td>Range</td>
<td>Ave.</td>
<td>Range</td>
<td>Ave.</td>
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<td>51.6</td>
<td>1550-1590</td>
<td>93.5-96.2</td>
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<tr>
<td>7-10</td>
<td>15,500</td>
<td>50.3-51.0</td>
<td>50.8</td>
<td>93.6-95.4</td>
<td>94.6</td>
</tr>
<tr>
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<td>19,000</td>
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<td>51.9</td>
<td>1480-1560</td>
<td>90.1-91.6</td>
</tr>
<tr>
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<td>90.5-92.3</td>
<td>91.0</td>
</tr>
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<td>51.5</td>
<td>90.6-92.9</td>
<td>91.8</td>
</tr>
<tr>
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<td>90.5</td>
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<td>89.8-92.1</td>
<td>91.2</td>
</tr>
<tr>
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<td>49.9</td>
<td>1560-1600</td>
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</tr>
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(a) Green data for eight pellets.
### TABLE V-2

PRESSING AND SINTERING DATA FOR POWDER BATCH B-4

(28% Pu - 72% U Mixed Oxide)

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<th>Pellet Numbers</th>
<th>Pelletizing Pressure (psig)</th>
<th>Green density (% theoretical)</th>
<th>Sintering</th>
<th>Sintered Density (% theoretical)</th>
<th>Firing Shrinkage (%)</th>
<th>Comments</th>
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<td>Time (hrs)</td>
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<td>Avg.</td>
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<td>4-1/2</td>
</tr>
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<td>48.5</td>
<td></td>
<td>1580-1600</td>
<td>4-1/2</td>
</tr>
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<td>51.4</td>
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<td>1580-1610</td>
<td>4-1/2</td>
</tr>
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<td>10,000</td>
<td>46.6-46.8</td>
<td>46.8</td>
<td></td>
<td>1580-1610</td>
<td>4-1/2</td>
</tr>
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<td>47.3</td>
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<td>1580-1610</td>
<td>4-1/2</td>
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<td>31-34</td>
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<td>48.5</td>
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<td>1580-1610</td>
<td>4-1/2</td>
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<tr>
<td>35-37</td>
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<td>1580-1610</td>
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<tr>
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<td></td>
<td>1560-1620</td>
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<tr>
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<td>48.8</td>
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<td>1560-1620</td>
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<td>69-74</td>
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<td>44.7-47.1</td>
<td>46.3</td>
<td></td>
<td>1560-1600</td>
<td>4</td>
</tr>
<tr>
<td>75-89(b)</td>
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<td>47.9</td>
<td></td>
<td>1560-1600</td>
<td>4</td>
</tr>
<tr>
<td>90-99(c)</td>
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<td>47.7</td>
<td></td>
<td>1560-1590</td>
<td>4</td>
</tr>
<tr>
<td>100-120(b)</td>
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<td>48.2</td>
<td></td>
<td>1560-1590</td>
<td>4</td>
</tr>
<tr>
<td>121-144(d)</td>
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<td>47.6-48.3</td>
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<td></td>
<td>1560-1590</td>
<td>4</td>
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<tr>
<td>145-168(e)</td>
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<td>47.7-48.3</td>
<td>48.0</td>
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<td>1570-1600</td>
<td>4</td>
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<tr>
<td>169-177</td>
<td>10,500</td>
<td>46.8-48.0</td>
<td>47.5</td>
<td></td>
<td>1480-1530</td>
<td>4</td>
</tr>
</tbody>
</table>

**NOTES:**
(a) hour-glass configuration  
(b) green data for three pellets  
(c) green data for two pellets  
(d) green data for seven pellets  
(e) green data for five pellets
TABLE V-3
PRESSING AND SINTERING DATA FOR POWDER BATCH B-5
(20% Pu - 80% Normal U Mixed Oxide)

<table>
<thead>
<tr>
<th>Pellet Numbers</th>
<th>Pelletizing Pressure (psig)</th>
<th>Green density (% theoretical)</th>
<th>Sintering</th>
<th>Sintered density (% theoretical)</th>
<th>Firing Shrinkage (%)</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Range Avg.</td>
<td>Temp (°C)</td>
<td>Time (hrs)</td>
<td>Range Avg. Diametric</td>
<td>Axial</td>
</tr>
<tr>
<td>1-4</td>
<td>10,500</td>
<td>46.3-46.6 46.4</td>
<td>1540-1590 4</td>
<td>92.0-95.0 93.6</td>
<td>22.3</td>
<td>19.7</td>
</tr>
<tr>
<td>5-8</td>
<td>13,000</td>
<td>47.7-48.0 47.9</td>
<td></td>
<td>92.8-93.4 93.2</td>
<td>21.1</td>
<td>19.0</td>
</tr>
<tr>
<td>9-12</td>
<td>11,500</td>
<td>46.0-46.3 46.2</td>
<td></td>
<td>91.8-93.8 93.1</td>
<td>22.1</td>
<td>20.1</td>
</tr>
<tr>
<td>13-17</td>
<td>14,000</td>
<td>47.2-47.5 47.4</td>
<td></td>
<td>91.1-93.8 92.3</td>
<td>21.3</td>
<td>19.2</td>
</tr>
<tr>
<td>18-19</td>
<td>11,000</td>
<td>46.6-46.7 46.6</td>
<td></td>
<td>93.1-93.5 93.3</td>
<td>21.7</td>
<td>20.3</td>
</tr>
<tr>
<td>20-23</td>
<td>12,000</td>
<td>47.3-47.7 47.5</td>
<td></td>
<td>93.5-94.8 94.2</td>
<td>21.7</td>
<td>19.4</td>
</tr>
<tr>
<td>24-27</td>
<td>17,000</td>
<td>49.3-49.7 49.6</td>
<td></td>
<td>89.3-94.1 92.0</td>
<td>20.0</td>
<td>17.5</td>
</tr>
<tr>
<td>28-31</td>
<td>12,500</td>
<td>46.7-47.0 46.8</td>
<td></td>
<td>94.9-95.5 95.4</td>
<td>22.3</td>
<td>20.6</td>
</tr>
<tr>
<td>32-35</td>
<td>10,000</td>
<td>45.5-45.8 45.7</td>
<td></td>
<td>94.5-95.8 95.1</td>
<td>22.9</td>
<td>21.2</td>
</tr>
<tr>
<td>36-60(a)</td>
<td>13,500</td>
<td>47.6-47.8 47.7</td>
<td></td>
<td>91.7-95.8 93.9</td>
<td>21.6</td>
<td>19.6</td>
</tr>
<tr>
<td>61-66</td>
<td>13,500</td>
<td>47.6-47.8 47.7</td>
<td>1480-1540 4-1/2</td>
<td>93.3-95.7 94.7</td>
<td></td>
<td>Blisters and chips</td>
</tr>
</tbody>
</table>
This method uses the equilibrium weight at 700 C in pure hydrogen as the reference point. An oxygen-to-metal ratio of 2.00 is obtained under these conditions.

5.4 Direct Sintering of Pu-U Coprecipitate

Scouting studies show promise that the plutonium hydroxide - ammonium diuranate coprecipitate may be compacted and sintered directly to a high density compactible grade mixed oxide in a reducing atmosphere. An aggregate density of 10.7 g/cc was obtained upon firing four compacts pressed from the washed and dried coprecipitate made from a nitrate solution containing 20 g/l Pu and 80 g/l U. The product after firing in 6% H₂ - 94% He at 1550 C for two hours consisted of severely cracked and broken pellets which appeared to make suitable feed for vibratory compacted fuel elements if further crushed to make the desired particle size blends.

Additional scouting studies were made to explore the effect of firing temperature upon final density during the direct sintering of compacts made from 20% Pu- 80% U coprecipitated with ammonia. Pellets were made from a single batch precipitated at room temperature, using a 15-minute digestion time and a final pH of 9, washed and dried; all pellets were pressed under identical conditions. Firing of groups of pellets at 1350, 1450, and 1550 C for four hours resulted in densities of 10.46, 11.13, and 10.85, respectively, as determined by the carbon tetrachloride displacement method.

A series of 15 runs in a statistically designed experiment to study the effects of precipitation variables was completed. In this series, pressing pressure was held constant and firing was completed in two heats with maximum temperature held within the range of 1450 to 1500 C. The precipitation variables were temperature (25 to 60 C), time (10 to 30 minutes) and pH (6 to 9.5). Densities ranged from 10.24 to 10.78 for these runs. An evaluation of the results of this study is in progress.

5.5 Plutonium Migration Studies on Irradiated Specimens

The objective has been to determine if plutonium migration occurs under the high temperature gradient existing in FCR fuel. Both swaged and pellet PuO₂-UO₂ fuel specimens previously irradiated to high exposure have been examined for possible plutonium migration. In the process of obtaining plutonium distribution data, fission product distribution data has also been obtained for Cs-137, Ce-144, and Zr-95.

5.5.1 Experimental Procedure

Both radial migration and longitudinal migration were studied. The radial samples investigated were obtained from a swaged fuel specimen (V-1) and a pellet fuel specimen (V-2) irradiated to
approximately $10 \times 10^4$ and $7 \times 10^4$ MWD/T respectively. Nominal pre-irradiation composition was 20% PuO$_2$ - 80% UO$_2$ and the fuel diameter was 0.150 inch. The swaged specimen material had a pre-irradiation density of 75% of theoretical and the pelletized specimen 95%. Both specimens were irradiated at an average clad surface thermal flux of $1.5 \times 10^6$ Btu/hr-ft$^2$ for 6 GETR cycles. A metal bonded diamond abrasive wheel was used to grind off concentric rings of fuel ranging in thickness of 0.002 to 0.010 inch in width. Besides these radial samples, 4 longitudinal samples were taken from fuel specimen IX-1-P (pelleted) to determine the extent of any plutonium migration from the fuel region into the insulator region under the high temperature gradient existing at the fuel-insulator interface.

The locations of the radial samples are tabulated below.

**SAMPLE LOCATION OF CONCENTRIC RING OF SAMPLE**

(mils from periphery of fuel)

**Specimen V-1 (Swaged)**

<table>
<thead>
<tr>
<th>Sample No.</th>
<th>Outer Edge</th>
<th>Inner Edge</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>B-1</td>
<td>0</td>
<td>75</td>
<td>Cross-section of fuel</td>
</tr>
<tr>
<td>B-2</td>
<td>0</td>
<td>?</td>
<td></td>
</tr>
<tr>
<td>B-3</td>
<td>2</td>
<td>7</td>
<td></td>
</tr>
<tr>
<td>B-4</td>
<td>7</td>
<td>12</td>
<td></td>
</tr>
<tr>
<td>B-5</td>
<td>12</td>
<td>17</td>
<td></td>
</tr>
<tr>
<td>B-6</td>
<td>17</td>
<td>22</td>
<td></td>
</tr>
<tr>
<td>B-7</td>
<td>22</td>
<td>32</td>
<td></td>
</tr>
<tr>
<td>B-8</td>
<td>32</td>
<td>42</td>
<td></td>
</tr>
<tr>
<td>B-9</td>
<td>42</td>
<td>52</td>
<td></td>
</tr>
<tr>
<td>B-10</td>
<td>52</td>
<td>62</td>
<td>Central void region</td>
</tr>
</tbody>
</table>
Specimen V-2 (Pelleted)

<table>
<thead>
<tr>
<th>Sample No.</th>
<th>Outer Edge</th>
<th>Inner Edge</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>C-1</td>
<td>0</td>
<td>75</td>
<td>Cross section of fuel</td>
</tr>
<tr>
<td>C-2</td>
<td>0</td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>C-3</td>
<td>2</td>
<td>5</td>
<td></td>
</tr>
<tr>
<td>C-4</td>
<td>5</td>
<td>10</td>
<td></td>
</tr>
<tr>
<td>C-5</td>
<td>10</td>
<td>15</td>
<td></td>
</tr>
<tr>
<td>C-6</td>
<td>15</td>
<td>25</td>
<td></td>
</tr>
<tr>
<td>C-7</td>
<td>25</td>
<td>35</td>
<td></td>
</tr>
<tr>
<td>C-8</td>
<td>35</td>
<td>45</td>
<td></td>
</tr>
<tr>
<td>C-9</td>
<td>45</td>
<td>55</td>
<td></td>
</tr>
<tr>
<td>C-10</td>
<td>55</td>
<td>65</td>
<td></td>
</tr>
<tr>
<td>C-11</td>
<td>65</td>
<td>75</td>
<td></td>
</tr>
</tbody>
</table>

The sample locations for the V-1 and V-2 specimens are off-center by 5 mils and 2 mils respectively.

The locations of the longitudinal samples are indicated as distance from the fuel-insulator interface taken as zero; negative distance and positive distance show the sample location in the insulator region and fuel region respectively.

Specimen LX-1-P

<table>
<thead>
<tr>
<th>Sample Number</th>
<th>Location (mils from Interface)</th>
</tr>
</thead>
<tbody>
<tr>
<td>11</td>
<td>-(450-470)</td>
</tr>
<tr>
<td>33</td>
<td>-(10-30)</td>
</tr>
<tr>
<td>34</td>
<td>- 10 + 10</td>
</tr>
<tr>
<td>57</td>
<td>20 - 40</td>
</tr>
</tbody>
</table>

The method used in the determination of the plutonium content involves a combination of mass spectrometric, alpha spectroscopic, and alpha counting data as well as the use of several nuclear constants, and is limited in accuracy to about ±5 percent. It will be seen that while this level of accuracy is certainly adequate to measure gross migration effects, it is not adequate to measure and
identify any more subtle behavior. Methods are under development to improve the accuracy for
analysis of any future samples. The fission product determinations were made by standard proce-
dures.

5.5.2 Results

The burnup determined from the fission product content and heavy element isotopic composition,
and the Pu-239 to U-238 ratios, are tabulated below for each sample.

| TABLE V-4 |
| EXPERIMENTAL DETERMINATION OF BURNUP AND Pu/U RATIO IN |
| SPECIMENS V-1, V-2, AND IX-1 |

<table>
<thead>
<tr>
<th>Sample Number</th>
<th>Cs-137</th>
<th>Ce-144</th>
<th>Zr-95</th>
<th>Heavy Element</th>
</tr>
</thead>
<tbody>
<tr>
<td>B-1</td>
<td>14.4</td>
<td>10.4</td>
<td>9.2</td>
<td>10.6 ± 1.0</td>
</tr>
<tr>
<td>B-2</td>
<td>20.2</td>
<td>10.4</td>
<td>10.1</td>
<td>11.4 ± 1.0</td>
</tr>
<tr>
<td>B-3</td>
<td>12.6</td>
<td>10.6</td>
<td>9.7</td>
<td>11.6 ± 1.0</td>
</tr>
<tr>
<td>B-4</td>
<td>12.6</td>
<td>9.5</td>
<td>9.5</td>
<td>11.7 ± 1.0</td>
</tr>
<tr>
<td>B-5</td>
<td>8.6</td>
<td>8.7</td>
<td>---</td>
<td>8.3 ± 1.0</td>
</tr>
<tr>
<td>B-6</td>
<td>4.3</td>
<td>6.5</td>
<td>---</td>
<td>8.2 ± 1.1</td>
</tr>
<tr>
<td>B-7</td>
<td>4.4</td>
<td>6.7</td>
<td>6.1</td>
<td>5.3 ± 1.2</td>
</tr>
<tr>
<td>B-8</td>
<td>4.6</td>
<td>6.9</td>
<td>5.9</td>
<td>4.2 ± 1.2</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Specimen V-2 (Pelleted)</th>
</tr>
</thead>
<tbody>
<tr>
<td>C-1</td>
</tr>
<tr>
<td>C-2</td>
</tr>
<tr>
<td>C-3</td>
</tr>
<tr>
<td>C-4</td>
</tr>
<tr>
<td>C-5</td>
</tr>
<tr>
<td>C-6</td>
</tr>
<tr>
<td>C-7</td>
</tr>
<tr>
<td>C-8</td>
</tr>
<tr>
<td>C-9</td>
</tr>
<tr>
<td>C-10</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Specimen IX-1 (Longitudinal)</th>
</tr>
</thead>
<tbody>
<tr>
<td>11</td>
</tr>
<tr>
<td>33</td>
</tr>
<tr>
<td>34</td>
</tr>
<tr>
<td>57</td>
</tr>
</tbody>
</table>

63
The estimated limits of error on the burnup measurements are as follows:

Cs-137, ±6%; Ce-144, ±7%; Zr-95, ±8%.

The uranium isotopic data showed that nearly all the samples taken from specimen V-1 were contaminated with uranium of a lower enrichment. Therefore, the values reported above have been corrected for this by making use of the U-236 to U-235 ratios with the assumption that all the U-235 and U-236 are from the original sample. This appears to be a fairly safe assumption in view of the consistency of the data.

The burnup data calculated from the heavy element isotopic data include the fission contribution from J-235, Pu-239, and Pu-241.

5.5.3 Interpretation and Conclusions

In order to determine whether or not plutonium migration has occurred in the samples investigated, the experimental ratio of Pu-239/U-238 is compared with the computed value calculated in two ways. One method involves the use of Pu-239 absorption cross-sections ranging from 1100 to 1700 barns together with the integrated flux determined from the U-235 depletion in each sample using an absorption cross-section of 667 barns. The other method makes use of the Pu-240 content (corrected for the Pu-240 capture) and a capture to fission ratio ranging from 0.44 to 0.70. The ratio of the experimental to the calculated data are shown plotted in Figure 5-1 and 5-2. The limits of error include both the experimental and calculated uncertainty. It is seen that little migration has been observed in either the pellet or swaged fuel specimen. The limits of error are such that small variations (less than 10 percent) about the ratio of 1.00 cannot be detected. It is apparent that there is a small increase in the plutonium content near the void in specimen V-1.

As shown in Table V-4, the Pu-239 to U-238 content for the longitudinal specimen is seen to be higher in sample 33 than in sample 11 although both are in the insulator region. This, at first glance, appears to indicate plutonium migration. However, this sample contained a significant increase in the U0235 content over natural uranium. If the assumption is made that the Pu-239 associated with the enriched uranium has the same Pu-239/U-238 ratio as sample 57, then a ratio of \( 7.7 \times 10^{-3} \) is calculated for the Pu-239/U-238 ratio for sample 33 compared with the observed value of \( 7.3 \times 10^{-3} \). That is, there was a slight movement of both the uranium and plutonium into this region in the same proportion as in the fuel with no apparent separation of plutonium from uranium even across the sharp temperature gradient existing at the fuel-insulator interface.
FIGURE 5-2 RADIAL VARIATION OF RATIO OF OBSERVED PLUTONIUM CONTENT TO CALCULATED PLUTONIUM CONTENT
In order to relate the radial fission product data to the problem of migration, the radial variation of integrated neutron flux must be taken into account. For the fission product isotopes, this has been done in a preliminary way by normalizing the fission product content to the U-236 content for each sample. This quantity is then multiplied by the U-236 to fission product ratio for the cross-sectional samples (i.e. B-1 and C-1) in order to normalize the quantity to 1.00. These values for the V-1 and V-2 specimen are plotted in Figures 5-3 and 5-4 respectively as a function of radial distance from the periphery. It is seen that there is essentially no migration of the Ce-144 and the Zr-95. The fact that the Ce-144 values are in general less than one in the V-1 specimen may be due to the contamination previously noted. The sample to which these were normalized (the cross-sectional sample) was one of the most highly contaminated samples. It is therefore likely that this sample contained excess Ce-144 (and Cs-137) which would depress the values. The general decrease in the Ce-144 and Zr-95 value with radial distance from the periphery, in particular for the V-2 specimen, is probably due to the assumption inherent in the normalization, i.e., that the fission product generation is proportional to the U-236 production. This is only true for the fission products produced from U-235 fission and does not really describe the fission product production from Pu-239.

The Cs-137 distribution is peaked at the periphery in both cases. It is of interest, however, that a much larger proportion of the Cs-137 is located at the periphery of the lower exposure pelleted V-2 specimen than the swaged V-1 specimen. It is also of interest that both specimens exhibit a minimum in the Cs-137 content at a mid-radius location rather than at the position closest to the central void. This behavior does not seem to be exhibited for either the plutonium or the other two fission products within experimental error.

In summary, then, it can be stated that gross migration of the plutonium, Ce-144 or Zr-95 in either the pelleted or swaged fuel specimens examined has not been observed. Fission product Cs-137 does exhibit gross migration toward the periphery and in addition, a region of depletion at mid-radius. Because of the special importance of plutonium migration as related to the Doppler effect, further investigation of the conditions under which such migration may occur is planned.
SPECIMEN V-1
SWAGED, 75% T.D.
100,000 MWD, 22 kw (MAX)
FUEL RADIUS 75 MILS

FUEL DISTANCE FROM PERIPHERY (mils)

\[
\frac{\text{Fission Product \ (U-236)}}{\text{Fission Product \ (U-236)}}
\]

\( T \) ft

Cs-137
Ce-144
Zr-95

FIGURE 5-3 RADIAL VARIATION OF FISSION PRODUCT CONTENT
WITH FLUX DEPENDENCE REMOVED
FIGURE 5-4  RADIAL VARIATION OF FISSION PRODUCT CONTENT WITH FLUX DEPENDENCE REMOVED
5.6 Plutonium Laboratory Operations

5.6.1 Dissolution of Mixed Oxide Pellets
An insoluble white residue remained when a 194 gram batch of sintered mixed oxide pellets from run B-1 was dissolved in nitric acid. This residue, when washed and dried, was 0.7 percent by weight of the original oxide. A single pellet from batch B-2 gave a similar insoluble residue amounting to 0.54% by weight. The analyses of feed plutonium and uranium did not show significantly high levels of impurities, nor did the uranium oxide previously prepared in the glove box line. Therefore, it is hypothesized that the residue may be resulting from a silica impurity which hydrolyzes upon refluxing the nitric acid. Further investigation will be made in an attempt to identify the material.

5.6.2 Recovery of Sintered Mixed-Oxide Scrap
Oxide-reduction cycling was investigated briefly as a possible means of recovering sintered mixed oxide powder from out-of-specification pellets. A 96 gram batch of 20% Pu-80% U mixed oxide pellets were subjected to prolonged firing in air at 750 to 800 C followed by heating in 6% H2 - 94% He at 750 C. Essentially no comminution of the mixed oxide pellets occurred which is contrary to the experience with UO2. In fact, measurements of a few whole pellets following the 10-hour firing in air indicated that no swelling had occurred, although an oxygen up-take amounting to 2.37% by weight was observed for the batch. (See Table V-5). A second oxidation-reduction cycle produced only spalling of some pellets; however, no fine powder was produced as is the case with UO2. It is concluded that a marked stabilizing effect against comminution of sintered pellets is rendered by the presence of plutonium oxide in the UO2 structure in the 20% composition range, and that an oxidation-reduction treatment will not be practical in this case, as contrasted to the behavior of pure UO2. Of interest in the experiment were the gravimetric changes which occurred, and which may indicate either that the original pellets were hypo-stoichiometric (oxygen-to-metal ratio less than 2.00), or that the oxidation-reduction cycling resulted in an end product which was hyperstoichiometric (oxygen-to-metal ratio greater than 2.00).

Recovery of sintered pellets by dissolution in boiling nitric acid remains as the alternative method of recycle and was utilized in the preparation of run B-5 which resulted in low grade pellets. Additional effort is required to determine how the powder from recycled fuel can be improved.
**TABLE V-5**

OXIDATION-REDUCTION CYCLING OF 20% Pu - 80% U MIXED OXIDE PELLETS (Run R-2)

<table>
<thead>
<tr>
<th>Treatment</th>
<th>Temperature (°C)</th>
<th>Time (hrs)</th>
<th>Net Weight (g)</th>
<th>Weight Change From Initial (g)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Original charge</td>
<td>---</td>
<td>---</td>
<td>95.730</td>
<td>---</td>
</tr>
<tr>
<td>Heated in air</td>
<td>750</td>
<td>2</td>
<td>97.165</td>
<td>+1.435</td>
</tr>
<tr>
<td></td>
<td>750</td>
<td>5</td>
<td>97.735</td>
<td>+2.005</td>
</tr>
<tr>
<td></td>
<td>750-800</td>
<td>10</td>
<td>97.995</td>
<td>+2.265</td>
</tr>
<tr>
<td>Heated in 6% H₂ - 94% He</td>
<td>750</td>
<td>2-1/2</td>
<td>96.749</td>
<td>+1.019</td>
</tr>
<tr>
<td>Heated in air</td>
<td>750</td>
<td>4</td>
<td>97.990</td>
<td>+2.260</td>
</tr>
<tr>
<td>Heated in 6% H₂ - 94% He</td>
<td>750</td>
<td>2-1/2</td>
<td>96.350</td>
<td>+0.62</td>
</tr>
</tbody>
</table>

5.6.3 Liquid Waste Disposal

Floccing of the plutonium- and uranium-containing liquid wastes, using a ferric hydroxide precipitation, was shown to be effective in reducing the alpha activity level of the solution phase by a factor of ~100, giving a final activity level of only 1.6 μc/l (21.2 μg Pu/l). This treatment of the plutonium facility's liquid waste, followed by concreting will permit disposal of about 80% of the volume generated via the present waste disposal contractor, who is limited by AEC license and ICC regulations to a maximum 100 μc of plutonium per standard 55 gallon dry waste container. The comparatively smaller volume of sludge, containing the bulk of the waste plutonium, will be solidified in regulation containers to await licensing approval for disposal from the site.

5.7 Analytical Chemistry

Isotopic dilution - mass spectrometry and alpha counting techniques have continued for uranium and plutonium analyses, respectively. These methods have the advantage of requiring only a dilution to be made on the sample solution within the analytical box of the plutonium facility. Subsequent operations can conveniently be performed in the hot chemistry laboratory. Titrimetric methods have not been investigated further.

Glass has been found more convenient and practical than platinum for direct alpha mounts of pellet and feed solutions. Replicate mounts agree well within 2%. Because of high salt content in waste solutions, a TTA extraction is required. Such mounts are necessarily made on platinum. Uranium analyses have been discontinued on waste solutions.
SECTION VI

TASK G - REACTOR DYNAMICS AND DESIGN

6.0 Updating Physics of FCR Reference Design

6.0.1 Introduction

The FCR reference core design (1) is based largely on fuel cycle data and reactivity coefficients calculated about two years ago. (2, 3) During this time there has been considerable effort devoted to improving cross sections and physics methods for fast reactor calculations. There has also been some experimental work on large dilute fast critical assemblies in the ZPR III facility for checking methods and cross sections. Although the latter is just a bare beginning relative to what is needed for understanding and improving fast reactor cross sections, the discrepancies observed between theory and experiment were large enough to stimulate a good deal of new work in calculation methods.

The earlier physics calculations on fuel cycles and reactivity coefficients are being repeated, using updated methods and cross sections. Modifications in calculation procedures relative to those used in the earlier work include the following:

1. The basic multigroup cross section set above 9 kev is that of Yiftah, Okrent, and Moldauer (YOM) (4) instead of the ANL-5800 set of Okrent and Lowenstein. (5) One effect of this change is that the higher plutonium isotopes (241 and 242) are included in the calculations directly instead of being treated as equivalent to 239 and 240 respectively.

2. Changes were made in the YOM set to take into account:
   a. Improved alpha values for Pu-239 measured by Hopkins and Diven. (6)
   b. More pessimistic \( \gamma \) values for U-235 based on the work of Cook and Davey, (7) and also for Pu-240 based on the work of Stevenson and Baker. (8)
   c. Higher elastic removal and transport cross sections for sodium and steel based on the ELMO code calculations by Hummel and Bhide, (9) and continuation of these calculations to lower energies at APED.
   d. Modifications in the (YOM) Pu-241 and Pu-242 cross sections as discussed in the April-June, 1962 Quarterly report under Task G.
3. A 24-group cross section set was used, instead of the 18-groups set for the earlier work, with 3 extra groups above 9 Kev in the YOM set, 1 extra group in the region of the 2 Kev sodium resonance, and 2 extra groups below 500 ev.

4. For improved calculation of Doppler and sodium coefficients.
   a. U-238, Pu-239, and Pu-240 cross sections were calculated as functions of fuel temperature and potential scattering cross section per absorber isotope. Using the RAPTURE code (10) higher angular momentum states of the compound nucleus (l, J) and an improved sampling of Porter-Thomas distributions for neutron and fission widths were used. The temperature dependent cross sections were corrected for resonance overlap.
   b. The Doppler-broadened cross sections were evaluated at specific energy points from groups 7 through 23. Three points were used for each group and 5 points per group near the 2 Kev sodium resonance, and the cross sections were then flux-averaged within each group.
   c. Heterogeneity effects were taken into account as described in Section 7.1 of this report.
   d. Reactivity changes due to the changes of fuel cross sections with temperature were made using a 24-group perturbation calculation. Reactivity changes due to changes in sodium density were also calculated by the perturbation methods.

5. The one-dimensional multigroup calculations were based on the actual cylindrical geometry with an improved basis for evaluating the axial bucklings. (11) The earlier work on reactivity coefficients was based on a slab geometry which gave a reactivity equivalent to the cylinder.

6.0.2 Results

A set of calculations of fuel cycle and reactivity coefficients was carried out using the newer methods and cross sections. These were done for a reactor size and composition very nearly that of the FCR reference design. (1) Results of these calculations are indicated in Table VI-1 and compared with earlier values.

The newer cross sections and calculation methods produced an improvement of the fuel cycle economics and a reduction of the Doppler coefficient for the FCR reference core size and composition. Both of these effects resulted primarily from (1) the harder neutron spectrum calculated with the
TABLE VI-1
UPDATED FUEL CYCLE AND REACTIVITY COEFFICIENTS*

<table>
<thead>
<tr>
<th></th>
<th>Current Calculations</th>
<th>GEAP-3721</th>
</tr>
</thead>
<tbody>
<tr>
<td>a/o Pu-(239+241)/to all (U+Pu)**</td>
<td>11.6</td>
<td>12.8</td>
</tr>
<tr>
<td>a/o Pu-239</td>
<td>10.6</td>
<td>12.8</td>
</tr>
<tr>
<td>a/o Pu-240</td>
<td>5.2</td>
<td>6.4</td>
</tr>
<tr>
<td>a/o Pu-241</td>
<td>1.0</td>
<td>---</td>
</tr>
<tr>
<td>a/o Pu-242</td>
<td>0.46</td>
<td>---</td>
</tr>
<tr>
<td>a/o U-238</td>
<td>77.0</td>
<td></td>
</tr>
<tr>
<td>a/o fission product pairs</td>
<td>5.8</td>
<td></td>
</tr>
</tbody>
</table>

Breeding Ratios for Pu-(239+241)

<table>
<thead>
<tr>
<th></th>
<th>Current Calculations</th>
<th>GEAP-3721</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core</td>
<td>0.84</td>
<td>0.75</td>
</tr>
<tr>
<td>Total</td>
<td>1.35</td>
<td>1.08</td>
</tr>
</tbody>
</table>

Isothermal Doppler Coefficient at 900°K $\Delta k/\Delta T$  

| % U-238 Contribution *** | 116 (neg)            |
| % Pu-239 Contribution    | 30 (pos)             |
| % Pu-240 Contribution    | 14 (neg)             |
| % U-238 Contribution above 9 Kev | 15          |
| % U-238 Contribution 9 to 2 Kev (Na resonance) | 10          |
| % U-238 Contribution 2 to 1 Kev (below Na resonance) | 30          |
| % U-238 Contribution 1 to 0.5 Kev | 27          |
| % U-238 Contribution below 0.5 Kev | 18****        |

Isothermal Sodium Temperature Coefficient $\Delta k/\Delta T$  

<table>
<thead>
<tr>
<th></th>
<th>Current Calculations</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core and Radial Blanket</td>
<td>-1.5x10^-6</td>
</tr>
<tr>
<td>Core only*****</td>
<td>-1.3x10^-6</td>
</tr>
<tr>
<td>Inner Radial 60% of Core</td>
<td>+0.3x10^-6</td>
</tr>
<tr>
<td>Inner Radial 40% of Core</td>
<td>+0.7x10^-6</td>
</tr>
<tr>
<td>Inner Radial 20% of Core</td>
<td>+0.7x10^-6</td>
</tr>
<tr>
<td>Core only (neglecting cross section changes of U-238, Pu-239, and Pu-240 with changes of sodium density)</td>
<td>-1.9x10^-6 +1x10^-6</td>
</tr>
</tbody>
</table>

Footnotes: See following page for footnote references for Table VI-1.
TABLE VI-1 (Continued)

Footnotes:

* Core volumetric composition for the hot operating condition is 32% PuO₂·UO₂, 16% stainless steel, and 52% sodium. Core height and diameter are 3.3 by 6.5 ft. Core fuel burnup is 100,000 MWD/t, with an assumed 15-batch loading. Burnup of the individual batches at any one time ranges from 0 to 100,000 MWD/t. The parameters listed above are for the average fuel isotopic composition in the core corresponding to a batch at about 50,000 MWD/t.

** The calculated $k_{\text{eff}}$ for this condition is 1.018.

*** The isotopic and spectral breakdown of the Doppler coefficient was not computed explicitly for the FCR reference design case; but the magnitude of the Pu-239 contribution is estimated to be about 30% of the U-238 magnitude.

**** By contrast, 41% of the Pu-239 contribution and 44% of the Pu-240 contribution are below 0.5 Kev.

***** These calculations were based on a one-dimensional radial geometry, with axial perpendicular bucklings used to give the effect of the axial leakage. A "core only" calculation run in axial geometry gave a very small positive sodium coefficient (less than $1 \times 10^{-6} \Delta k/°C$), and an axial "core and blankets" calculation gave an extremely small negative coefficient. Doppler coefficients and reactivity calculated in radial and axial geometries were in very precise agreement.
new cross section, (2) the considerable lowering of the Pu-239 alpha value at high neutron energies, and (3) the improved neutronic properties of the higher plutonium isotopes at high energies relative to the older cross sections used for Pu-240, 241, and 242. The sodium reactivity coefficient remained about the same as in the earlier calculations.

The spatial dependence of the sodium coefficient and the isotopic and spectrum breakdown of the Doppler coefficient are seen from the data listed in Table VI-1. The change in fuel absorption cross sections with sodium density, mainly reduced capture cross sections by U-238 in the strong resonance region which was not included in the earlier sodium coefficient work, is seen to produce a small positive coefficient contribution.

The variations in the sodium coefficient data listed, showing effects of spatial dependence and of reduced fuel isotope resonance absorption, have greater quantitative significance than the absolute values (see bottom footnote under Table VI-1). The calculated sodium coefficient is near zero; and the computation is not precise enough to attach much significance to that (+) or (-) of the small absolute value.

The spectral Doppler distribution indicates that further degradation of the neutron spectrum by addition of light scattering elements will substantially enhance the Doppler effect. The improved neutron economy with the newer cross sections leaves a substantial margin for the deleterious effect that a more degraded spectrum may have on the fuel cycle cost. Calculations have been started to evaluate the effects on FCR reactivity coefficients and fuel economy of inserting small amounts of moderating material into the core. (8 per cent BeO was used as an example.) Preliminary results indicate that a sizeable increase of the Doppler coefficient (∼70%) with very small penalty to the neutron economy is achieved.

The improved neutron economy also permits lowering the fuel specific power to achieve a lower average fuel temperature and, consequently, larger Doppler coefficient. The temperature dependence of the isothermal Doppler coefficient was found to be very nearly $T^{-1}$, in agreement with the earlier results. For the temperature range 300 to 3000 K, the isothermal coefficient was fitted to

$$\left(1 \ \frac{d\kappa}{dT}\right)_{Dop} = -0.00635 + \frac{0.6}{T^2}$$

yielding a reactivity change.
(2) \[
\text{k}(T) - \text{k}(T_o) = -0.00635 \log_e \frac{T-o}{T} - 0.6 \left( \frac{1}{T} - \frac{1}{T_o} \right) \text{ for an isothermal fuel temperature change}
\]
\( T_o \) to \( T \) in K. \( \frac{dk}{dT} \) and \( \text{k}(T) - \text{k}^0(1000) \) are listed below for several temperatures.

<table>
<thead>
<tr>
<th>( T ) in °K</th>
<th>( \text{k}(T) - \text{k}(1000) )</th>
<th>( \frac{dk}{dT} )</th>
</tr>
</thead>
<tbody>
<tr>
<td>300</td>
<td>+0.00624</td>
<td>-14.5 \times 10^{-6}</td>
</tr>
<tr>
<td>700</td>
<td>+0.00202</td>
<td>-7.9 \times 10^{-6}</td>
</tr>
<tr>
<td>1000</td>
<td>0.00000</td>
<td>-5.8 \times 10^{-6}</td>
</tr>
<tr>
<td>1173</td>
<td>-0.00092</td>
<td>-5.0 \times 10^{-6}</td>
</tr>
<tr>
<td>1400</td>
<td>-0.00197</td>
<td>-4.2 \times 10^{-6}</td>
</tr>
<tr>
<td>1600</td>
<td>-0.00276</td>
<td>-3.7 \times 10^{-6}</td>
</tr>
<tr>
<td>3000</td>
<td>-0.00658</td>
<td>-2.1 \times 10^{-6}</td>
</tr>
</tbody>
</table>

These values will be used to evaluate an improved balance between fuel specific power and Doppler coefficient in the current reference design, concurrent with efforts to optimize by modifications of core geometry and composition.

6.1 Heterogeneity Effects on the Physics of FCR

Calculations were made to assess the effects of composition heterogeneity on the effective absorption cross sections of fuel isotopes, particularly uranium-238 and plutonium-239. Earlier physics computations had assumed spatial homogeneity for the mixture of fuel, clad, and coolant in the core. Results of the heterogeneity calculations show that the homogeneous approximation is a very good one.

In the homogeneous approximation, the potential scattering cross section per absorber isotope, \( \sigma_p \), is the sum of the independent contributions by the fuel materials (\( \text{UO}_2 \) and \( \text{PuO}_2 \)) and the materials outside of the fuel rods (clad and coolant). The heterogeneous case may be treated as an equivalent homogeneous composition by redefining \( \sigma_p \) such that

\[
\sigma_p = (\sigma_p)_{\text{fuel}} + (\sigma_p)_{\text{escape}}
\]

* Corrections for spatial power and temperature distributions have not yet been calculated. In the earlier studies, the spatial effects resulted in about a 20 per cent increase of the above Doppler effects. Current perturbation calculations of the spatial effects indicate that this may be reduced to around a 15 per cent increase.
\( (\sigma_p)_{\text{fuel}} \) is the potential scattering cross section per absorber isotope contributed by fuel materials only and

\[
\sigma_{\text{escape}} = \sigma_t \left( \frac{P}{1-P} \right)
\]

(2)

where \( \sigma_t \) is the total cross section of fuel material per absorber isotope and \( P \) is the probability that a neutron in the fuel material (at an energy in the region of a resonance) will suffer its first collision in material outside of the fuel. Using the geometric chord length approximation for collision probabilities, \( P \) for clustered cylindrical fuel rods is given by:

\[
P = \left( \frac{P_f P_m}{N \sigma_t} \right) \left[ \frac{1}{1-(1-P_f)(1-P_m)} \right]
\]

(3)

where \( N \) is the concentration of absorber isotopes in the fuel rod, and \( d \) is the inner diameter of the fuel rod.

\[
P_f = 1 - \exp (-Nd \sigma_t)
\]

(4)

\[
P_m = 1 - \exp (-d \sum S \frac{V_m}{V_f})
\]

(5)

are respectively the probabilities \( P_f \) that a neutron entering a fuel rod will suffer a collision before emerging and \( P_m \) that a neutron emerging from a fuel rod will suffer a collision before it reaches fuel material again. \( \sum S \frac{V_m}{V_f} \) in (5) is the scattering cross section of material outside the fuel per unit volume of fuel.

Using Wigner's rational approximation for \( P_f \), Bell\(^{(12)}\) showed that \( \sigma_{\text{escape}} \) reduced simply to \( P_m \) \( d/N \). This was found to be of insufficient accuracy for the FCR over the full range of \( \sigma_t \) values which, at a given energy region, range from the potential scattering cross section of UO\(_2\) to the peak of the strongest uranium-238 resonance. It was found that for the FCR composition* an excellent fit to the right side of equation (2) was obtained by setting

\[
\sigma_{\text{escape}} \approx CP_m \frac{m}{Nd}
\]

(6)

wherefore neutron energies above 5.5 kev

\[
C = 1 + 0.54 P_m \text{ for } P_m < 0.3
\]

\[
C = 0.96 + 0.96 P_m \text{ for } 0.3 \leq P_m \leq 0.5
\]
For neutron energies below 2 kev, C = 1.12. In the energy region of the strong sodium resonance, 5.5 to 2 kev, $\sigma_{\text{escape}}$ was computed directly from equations (2), (3), (4), and (5) using $\Gamma_t = 60$ barns for uranium-238 and $\Gamma_t = 190$ barns for plutonium-239. The 60 barn value is approximately the total fuel cross section per uranium-238 atom at an average position of a uranium-238 resonance (average $\Gamma_n$ and at the position which defines the half-area under the Doppler-broadened resonance at 1400 K fuel temperature). The 190 barn value is simply $(\sigma_p)^{\text{fuel}}$ for plutonium-239 in the region of the sodium resonance.

Table VI-2 compares $\sigma_p$ values for uranium-238 calculated by the homogeneous method and the heterogeneous approximation over a range of neutron energies. It is apparent from Table VII-2 that for the fine spacings of fuel and non-fuel materials in FCR, the heterogeneity is insignificant except in the region of the strong sodium resonance peak.

Table VI-3 compares the capture cross sections for uranium-238 and fission and capture cross sections for plutonium-239 calculated by the homogeneous method and the heterogeneous approximation for two neutron energy groups which overlap the sodium resonance. (These are group 15 and 16 in the new 24-group set.) It is seen from Table VI-3 that even with the large reduction of $\sigma_p$ near the sodium resonance peak in the heterogeneous case, the differences in the group cross sections are extremely small. This is due largely to the strong depression of the neutron flux around the peak of the resonance.

It can be concluded from the very small effect of heterogeneity on the uranium-238 and plutonium-239 cross sections that the homogeneous model is adequate for FCR fuel cycle and design calculations. Further calculations are being done to evaluate the effect of the heterogeneity on the temperature dependence of the fuel absorber cross sections. In view of the extremely small effect of the heterogeneity on the cross sections, it is expected that the Doppler effect will be similarly insensitive to the small departure from homogeneity in the FCR.

6.2 Comparison of ZPR III Experimental Data with Multigroup Calculations Using the MISY Code

Calculations were made and compared with experimental results obtained from a fast uranium oxide fueled critical assembly in ZPR III. These calculations were done to provide a check on methods used in the physics design of fast reactors at APED. A 21-group cross-section set used for recent EFCR and FCR calculations was employed.
<table>
<thead>
<tr>
<th>Energy (kev)</th>
<th>$\sigma_p$ for U-238</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Homogeneous</td>
</tr>
<tr>
<td>282.2</td>
<td>34.3</td>
</tr>
<tr>
<td>203.0</td>
<td>41.3</td>
</tr>
<tr>
<td>147.1</td>
<td>43.4</td>
</tr>
<tr>
<td>103.9</td>
<td>45.1</td>
</tr>
<tr>
<td>74.7</td>
<td>45.4</td>
</tr>
<tr>
<td>45.3</td>
<td>46.4</td>
</tr>
<tr>
<td>20.0</td>
<td>46.9</td>
</tr>
<tr>
<td>10.0</td>
<td>62.2</td>
</tr>
<tr>
<td>6.0</td>
<td>87.5</td>
</tr>
<tr>
<td>5.1</td>
<td>87.8</td>
</tr>
<tr>
<td>4.3</td>
<td>104.0</td>
</tr>
<tr>
<td>3.7</td>
<td>159.0</td>
</tr>
<tr>
<td>3.2 Sodium Resonance Region</td>
<td>417.0</td>
</tr>
<tr>
<td>2.9</td>
<td>808.0</td>
</tr>
<tr>
<td>2.6</td>
<td>494.0</td>
</tr>
<tr>
<td>2.4</td>
<td>147.0</td>
</tr>
<tr>
<td>2.1</td>
<td>83.8</td>
</tr>
<tr>
<td>1.5</td>
<td>60.9</td>
</tr>
<tr>
<td>0.38</td>
<td>62.9</td>
</tr>
<tr>
<td>0.15</td>
<td>63.7</td>
</tr>
<tr>
<td>0.01</td>
<td>63.7</td>
</tr>
</tbody>
</table>
TABLE VI-3

COMPARATIVE CROSS SECTIONS OF
URANIUM-238 AND PLUTONIUM-239

I. Uranium-238

<table>
<thead>
<tr>
<th>Group</th>
<th>Homogeneous</th>
<th>Heterogeneous</th>
</tr>
</thead>
<tbody>
<tr>
<td>15</td>
<td>.730</td>
<td>.728</td>
</tr>
<tr>
<td>16</td>
<td>1.051</td>
<td>0.033</td>
</tr>
</tbody>
</table>

II. Plutonium-239

<table>
<thead>
<tr>
<th>Group</th>
<th>Homogeneous</th>
<th>Heterogeneous</th>
<th>Homogeneous</th>
<th>Heterogeneous</th>
</tr>
</thead>
<tbody>
<tr>
<td>15</td>
<td>3.079</td>
<td>3.074</td>
<td>1.650</td>
<td>1.645</td>
</tr>
<tr>
<td>16</td>
<td>3.991</td>
<td>3.988</td>
<td>2.294</td>
<td>2.291</td>
</tr>
</tbody>
</table>

* Group 15 extends from 5.5 to 3.4 kev; group 16 from 3.4 to 2.0 kev.
Experimental measurements carried out in the ZPR III assembly and checked by the MISY code calculations included:

1. Multiplication factor and critical mass.

2. Fission ratios (as neutron spectrum indicators).

3. Reactivity worth of various materials inserted in the center of the core.

ANL had carried out earlier calculations on this ZPR III assembly using a SNG transport code. Thus, the MISY calculations permitted a comparison of diffusion theory with transport theory results, in addition to comparisons between computational and experimental values. (In the FCR core a 2.5 per cent overestimate of reactivity corresponds to about a 6 per cent underestimate of required plutonium enrichment.) The multiplication factor calculated by the SNG transport code was 3.2 per cent higher than experiment, in good agreement with the MISY diffusion theory calculation.

The computed fission ratios listed in Table VI-4 indicate that the 21-group cross section set used in the MISY calculation predicted too hard a neutron spectrum. This is based on the apparently too low calculated fission rate for uranium-235 relative to plutonium-239 and uranium-238. (Calculation of too hard a spectrum results in an underestimate of enrichment, an underestimate of the magnitude of the Doppler coefficient, and probably biases the calculated sodium temperature coefficient in the negative direction.) It is not certain, however, that the discrepancies between calculated and experimental fission ratios are due to a calculated spectrum which is too hard, since small errors in the cross section values could account for the discrepancies. (Calculations being carried out now at ANL, and at Dounreay by W. Gemmell,\textsuperscript{14} indicate that the discrepancies are due to a combination of too hard a calculated spectrum, some undetermined errors in the cross sections used, and errors in the experiment.) The SNG transport and MISY diffusion calculations of the fission ratios are in good agreement as shown in Table VI-5.

Table VI-6 shows MISY calculated and experimental reactivity worths of uranium-235, uranium-238 and plutonium-239 in the center of the core. The calculated values are in good agreement with experimental results, being on the underreactive side for all three isotopes. (This tends to weaken the "hard calculated spectrum" argument used to explain the fission ratio discrepancies.) Table VI-7 shows calculated and experimental reactivity worths of non-fissile materials in the center of the core. Table VI-8 shows calculated and experimental reactivity worths for the non-fissile materials.
<table>
<thead>
<tr>
<th></th>
<th>Experimental (2)</th>
<th>Misy</th>
<th>Ratio (Calc./Exp.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$q_{f}(U-233)/q_{f}(U-235)$</td>
<td>$1.470 \pm 0.015$</td>
<td>1.55</td>
<td>1.05</td>
</tr>
<tr>
<td>$q_{f}(U-238)/q_{f}(U-235)$</td>
<td>$.0356 \pm .0004$</td>
<td>.0397</td>
<td>1.12</td>
</tr>
<tr>
<td>$q_{c}(U-238)/q_{f}(U-235)$</td>
<td>$.115 \pm .006$</td>
<td>.127</td>
<td>1.10</td>
</tr>
<tr>
<td>$q_{f}(Pu-239)/q_{f}(U-235)$</td>
<td>$1.06 \pm .02$</td>
<td>1.156</td>
<td>1.09</td>
</tr>
<tr>
<td>$q_{f}(Pu-240)/q_{f}(U-235)$</td>
<td>$.289 \pm .006$</td>
<td>.296</td>
<td>1.02</td>
</tr>
<tr>
<td>$q_{f}(U-233)/q_{f}(Pu-239)$</td>
<td>$1.39 \pm .03$</td>
<td>1.34</td>
<td>.964</td>
</tr>
<tr>
<td>$q_{f}(U-235)/q_{f}(Pu-239)$</td>
<td>$.944 \pm .02$</td>
<td>.865</td>
<td>.916</td>
</tr>
<tr>
<td>$q_{f}(U-238)/q_{f}(Pu-239)$</td>
<td>$.0336 \pm .0006$</td>
<td>.0343</td>
<td>1.02</td>
</tr>
<tr>
<td>$q_{c}(U-238)/q_{f}(Pu-239)$</td>
<td>$.108 \pm .006$</td>
<td>.1103</td>
<td>1.02</td>
</tr>
<tr>
<td>$q_{f}(Pu-240)/q_{f}(Pu-239)$</td>
<td>$.273 \pm .006$</td>
<td>.251</td>
<td>.919</td>
</tr>
</tbody>
</table>
### TABLE VI-5
COMPARISON OF CALCULATED TO EXPERIMENTAL CROSS-SECTION RATIOS FOR MISY AND SNG

<table>
<thead>
<tr>
<th>Ratio ( \frac{MISY}{Exp.} )</th>
<th>Ratio ( \frac{SNG}{Exp.} )</th>
</tr>
</thead>
<tbody>
<tr>
<td>( q_{(U-233)}/q_{(U-235)} )</td>
<td>1.05</td>
</tr>
<tr>
<td>( q_{(U-238)}/q_{(U-235)} )</td>
<td>1.12</td>
</tr>
<tr>
<td>( q_{c(U-238)}/q_{(U-235)} )</td>
<td>1.10</td>
</tr>
<tr>
<td>( q_{(Pu-239)}/q_{(U-235)} )</td>
<td>1.09</td>
</tr>
<tr>
<td>( q_{(Pu-240)}/q_{(U-235)} )</td>
<td>1.02</td>
</tr>
<tr>
<td>( q_{(U-233)}/q_{(Pu-239)} )</td>
<td>.964</td>
</tr>
<tr>
<td>( q_{(U-235)}/q_{(Pu-239)} )</td>
<td>.916</td>
</tr>
<tr>
<td>( q_{(U-238)}/q_{(Pu-239)} )</td>
<td>1.02</td>
</tr>
<tr>
<td>( q_{c(U-238)}/q_{(Pu-239)} )</td>
<td>1.02</td>
</tr>
<tr>
<td>( q_{(Pu-240)}/q_{(Pu-239)} )</td>
<td>.919</td>
</tr>
</tbody>
</table>

### TABLE VI-6
CENTRAL WORTH OF FUEL ISOTOPES

<table>
<thead>
<tr>
<th>Material</th>
<th>Weight of Sample</th>
<th>Exp. ( \Delta k/k )</th>
<th>Calc. ( \Delta k/k )</th>
</tr>
</thead>
<tbody>
<tr>
<td>U-238</td>
<td>296.5 gm</td>
<td>- .000048</td>
<td>- .0000497</td>
</tr>
<tr>
<td>U-235</td>
<td>287.8</td>
<td>.00078</td>
<td>.00076</td>
</tr>
<tr>
<td>Pu-239</td>
<td>186.0</td>
<td>.00086</td>
<td>.0008326</td>
</tr>
</tbody>
</table>
### TABLE VI-7

CENTRAL WORTH OF NON-FISSILE MATERIALS

<table>
<thead>
<tr>
<th>Material</th>
<th>Exp. Sample Weight</th>
<th>Exp. ( \Delta k/k )</th>
<th>Calc. ( \Delta k/k )</th>
</tr>
</thead>
<tbody>
<tr>
<td>Be</td>
<td>120.4 gm</td>
<td>.000286</td>
<td>.000237</td>
</tr>
<tr>
<td>( \text{B}^{10} )</td>
<td>24.0 gm</td>
<td>-.000122</td>
<td>-.000188</td>
</tr>
<tr>
<td>C</td>
<td>196.4 gm</td>
<td>.000183</td>
<td>.000112</td>
</tr>
<tr>
<td>Na</td>
<td>91.0 gm</td>
<td>.0000359</td>
<td>.0000167</td>
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<tr>
<td>Al</td>
<td>350.8 gm</td>
<td>.00001</td>
<td>.0000221</td>
</tr>
<tr>
<td>V</td>
<td>527.2 gm</td>
<td>.0000615</td>
<td>.0000102</td>
</tr>
<tr>
<td>Zr</td>
<td>846.0 gm</td>
<td>-.0000301</td>
<td>-.0000400</td>
</tr>
<tr>
<td>Ta</td>
<td>507.0 gm</td>
<td>-.000495</td>
<td>-.000342</td>
</tr>
<tr>
<td>304 SS</td>
<td>1017.0 gm</td>
<td>-.000423</td>
<td>-.000470</td>
</tr>
</tbody>
</table>

### TABLE VI-8

DISTRIBUTED MATERIAL WORTH

<table>
<thead>
<tr>
<th>Material</th>
<th>Sample Size</th>
<th>( \Delta k/k ) Calc.</th>
<th>( \Delta k/k ) Exp.</th>
</tr>
</thead>
<tbody>
<tr>
<td>C</td>
<td>196.4 gm</td>
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<tr>
<td>Na</td>
<td>91.0 gm</td>
<td>+.0000184</td>
<td>----</td>
</tr>
<tr>
<td>Al</td>
<td>527.2 gm</td>
<td>+.0000477</td>
<td>+.000049</td>
</tr>
<tr>
<td>304 SS</td>
<td>1017.0 gm</td>
<td>+.0000418</td>
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</tr>
</tbody>
</table>
distributed uniformly over the core. Generally the calculated and experimental central reactivity
worths for the non-fissile materials are not in good agreement. The discrepancy is especially great
for sodium, aluminum, and vanadium; however, both the calculated and experimental reactivity values
of these isotopes are so small that the differences may not be significant.

The new 24-group Misy cross section set has recently been compiled and is now being used to update
the physics of the FCR reference design (see Section 6.0). The new set contains more recent cross
section information than the 21-group set used for the MISY calculations reported here. A computa-
tion of the ZPR III Assembly 29 experiments is now being repeated using a 24-group cross section
set.

6.3 **FORE Code Transient Calculations**

Analysis of the results of the extensive series of FORE excursion calculations performed in June
proceeded. Included were calculations covering the effects of variations in fuel-clad gap coefficient,
fuel conductivity, fuel heat of fusion, Doppler and other reactivity coefficients, magnitude and rapid-
ity of accidental reactivity insertion, initial power and inlet coolant temperature, and scram time.

The calculations on the different fuel-clad gap coefficients and fuel conductivities have generally in-
dicated that the net effect of increasing these quantities is to lessen the severity of the excursion for
a given steady-state power and a given reactivity input, because the lower steady-state fuel tempera-
tures provide a larger Doppler coefficient, and hence decrease the energy release and transient fuel
temperature rise. However, the maximum heat fluxes and coolant temperatures are not lowered to
a very great extent, because the higher fuel conductivity and gap coefficient allows the transient heat
to flow to the coolant more rapidly.

Transient calculations were also made to help solve design problems on the EFCR fast reactivity ex-
cursion device (FRED). It was concluded that the effects of a rapid insertion of well over a dollar
of reactivity could be reasonably simulated by:

1. A relatively slow addition of slightly less than a dollar using the main reactor control sys-
tem followed by
2. rapid insertion of the remainder of the desired reactivity using FRED.
A report on the current version of FORE is being written, and a final draft is expected to be complete shortly.

6.4 Reference FCR Design

The rough draft of a report was prepared covering all of the work done in fiscal 1962 on the reference FCR design.
REFERENCES


