CONFIDENTIAL

UNCLASSIFIED

MASTER

ORNL-4461

C-92b - Systems for Nuclear Auxiliary
Power (SNAP)-Reactor SNAP Program

AEC RESEARCH AND DEVELOPMENT REPORT

EFFECTS OF IRRADIATION ON
URANIUM NITRIDE UNDER
SPACE-REACTOR CONDITIONS

(Title Unclassified)

S. C. Weaver

J. L. Scott

R. L. Senn

B. H. Montgomery

69 179

OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U.S. ATOMIC ENERGY COMMISSION

UNCLASSIFIED

CONFIDENTIAL

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

RESTRICTED DATA This document contains restricted data as defined in the Atomic Energy Act of 1954. Its transmission or the disclosure of its contents in any manner to an unauthorized person is prohibited.

GROUP 1 Excluded from automatic downgrading and declassification.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

~~CONFIDENTIAL~~

LEGAL NOTICE

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

- A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or
- B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

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

~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

UNCLASSIFIED

Classification cancelled (or changed to)

UNCLASSIFIED

by authority of

letter April 10, 1973
Bram Feldman

ORNL-4461

by

GB

DTIE, date

6/20/73
plv of class. Wash DC

Exempt from CCRP Re-review Requirements
(per 7/22/82 Duff/Caudle memorandum)

Contract No. W-7405-eng-26

METALS AND CERAMICS DIVISION

REACTOR DIVISION

EFFECTS OF IRRADIATION ON URANIUM NITRIDE
UNDER SPACE-REACTOR CONDITIONS

S. C. Weaver R. L. Senn
J. L. Scott B. H. Montgomery

OCTOBER 1969

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

~~RESTRICTED DATA~~

~~This document contains Restricted Data as defined in the Atomic Energy Act of 1954. Its transmittal or the disclosure of its contents in any manner to any unauthorized person is prohibited.~~

~~GROUP 1~~

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

~~Excluded from automatic downgrading and declassification~~

UNCLASSIFIED

~~CONFIDENTIAL~~

NOTICE
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Blank Page

CONFIDENTIAL

iii

CONTENTS

	Page
Abstract	1
Introduction	2
Description of Experiments	4
Instrumentation and Controls	7
Irradiation History	8
Results of Postirradiation Examination	13
Capsule Disassembly	13
Macroscopic Examination of Fuel Pins	15
Dimensional Measurements of Fuel Pins	16
Release of Fission Gas	16
Macroscopic Examination of Fuel Pellets and Cladding	17
Measurements of Fuel Density	19
Metallography of Cladding	19
Metallography of Fuel	21
Discussion and Conclusions	29
Acknowledgments	35

CONFIDENTIAL

CONFIDENTIAL

EFFECTS OF IRRADIATION ON URANIUM NITRIDE
UNDER SPACE-REACTOR CONDITIONS

S. C. Weaver, R. L. Senn,¹ J. L. Scott,
and B. H. Montgomery¹

ABSTRACT

Fifteen UN capsules containing a total of 41 fuel pins were irradiated to burnups up to 4.58 at. % U. The pins contained pressed-and-sintered UN pellets clad in 0.025-in.-thick PWC-11 (Nb-1% Zr-0.1% C) with a 0.005-in. tungsten reaction barrier. Most of the pins operated under SNAP-50 Reference Design No. 3 conditions of 1200°C cladding temperature at linear heat ratings of about 2 kw/ft. To assess the effects of higher burnups and heat ratings, however, we modified four of the capsules to allow the pins to operate at linear heat ratings up to 8 kw/ft but at lower cladding temperatures of 980°C.

The fuel pins performed well with no failures despite burnups of up to three times the design burnup and irradiation times up to 12,000 hr. Twenty-two of the 41 fuel pins were successfully tested to the design test time of 10,000 hr; those pins tested less than 10,000 hr did not fail but were removed because they had completed their irradiation testing. All of the fuel pins performed better than the performance criteria of 2.0% increase in pin diameter and 20% release of fission gas as outlined in SNAP-50 Reference Design No. 3. Fuel volume increases exceeded design only in one capsule, 57-003, which had a burnup of 4.2% and a central temperature of about 1325°C. The PWC-11 cladding showed no effects of corrosion by lithium after the tests.

Metallography of the fuel showed that bubbles of fission gas migrated up the temperature gradient and became trapped at grain boundaries. The fuel that was operated to higher burnups (> 4 at. % U) showed bubbles of fission gas within the grains as well as at grain boundaries.

¹Reactor Division.

CONFIDENTIAL

~~CONFIDENTIAL~~

2

INTRODUCTION

Uranium mononitride is of significant interest for both high-temperature space reactors and commercial fast breeder reactors. For both applications, one wishes to operate the fuel at the maximum temperature permissible at the center of the fuel. Since space reactors reject heat by radiation, the temperature of the fuel cladding must be at least 1000°C. The size of space radiators can be significantly reduced by increasing the cladding temperatures, because the radiation of heat increases as the ratio of the fourth power of temperature. For the lower cladding temperatures of the fast breeder reactors, an increase in operating temperature at the center of the fuel permits an increased heat rating for the fuel pins.

Before the tests described in this report, uranium mononitride had been successfully irradiated in capsules designed and fabricated by Pratt and Whitney Aircraft-Connecticut Advanced Nuclear Engineering Laboratory (CANEL) at nominal fuel-center temperatures of 1500, 1300, and 1100°C and cladding temperatures of 1200 and 1000°C (ref. 2). Those experiments indicated that fuel swelling and release of fission gas were small at fuel-center temperatures of 1300°C and below. At 1500°C both fuel swelling and release of fission gas were significant but not excessive. The Pratt and Whitney experiments were clad with either PWC-11 (Nb-1% Zr-0.1% C) or Nb-1% Zr, 0.025 in. thick with a 0.005-in. tungsten barrier between the fuel and cladding. In addition results from a series of UN experiments with T-111 and D-43 cladding irradiated in the Low Intensity Test Reactor at Oak Ridge National Laboratory suggested that stronger cladding could significantly reduce the increases in diameter caused by swelling of the fuel.

When the Pratt and Whitney-CANEL ceased operation³ in July 1965, ORNL assumed responsibility for continuation of the CANEL program on

²S. C. Weaver and J. L. Scott, Comparison of Reactor Fuels for High Temperature Applications, ORNL-TM-1360 (December 1965). Classified.

³The Pratt and Whitney Aircraft-Connecticut Advanced Nuclear Engineering Laboratory was organized in the mid-1950's to work on the Aircraft Nuclear Propulsion project. When this project was cancelled,

~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

3

irradiation of UN at the Materials Testing Reactor (MTR). At that time, six capsules were undergoing irradiation testing, four had been removed from testing, and five were completely constructed and ready for insertion into the reactor. All 15 capsules were designed to test UN at SNAP-50 Reference Design No. 3 conditions.⁴ These conditions, together with design limitations, are shown below.

Thermal power level, Mw	2
Maximum fuel temperature, °C	1260
Maximum power density, kw/cm ³	0.4
Maximum fuel burnup, at. % U	1.5
Maximum fission-gas release, %	20
Maximum diametral pin swelling, %	2
Maximum fuel density decrease, %	8

The SNAP-50 reactor was designed as a conventional, liquid-metal-cooled, multiloop system that would use lithium as the primary coolant.

Previous irradiation tests by CANEL were designed for two sets of conditions.⁵ The first irradiation tests operated at a nominal temperature of 1500°C at the fuel center at power densities of 1.5 to 2.1 kw/cm³. The highest burnup attained was 2 at. % U, and six fuel pins exceeded the burnup limit for Reference Design No. 3. Of these, three pins exceeded the design limit for release of fission gas, and three pins exceeded the design limit for decrease in fuel density. However, the increase in the diameter of the cladding, the most significant effect, was within the design limits to 2 at. % U burnup. The second set of irradiation tests at CANEL contained fuel pins that operated at a low fuel-center temperature

the work shifted to research on the Lithium-Cooled Reactor Experiment. When this project was also discontinued, work began there on the SNAP-50 project, which for 3 to 4 years included irradiation testing of UN fuel such as that described in this report. In July 1965, all of the AEC-sponsored work was discontinued, and the facility was closed.

⁴SNAP-50/SPUR Program Engineering Progress Report April 1, 1964-June 30, 1964, PWAC-642 (July 30, 1964). Classified.

⁵M. A. DeCrescente, M. S. Freed, and S. D. Caplow, Uranium Nitride Fuel Development SNAP-50, PWAC-488 (Oct. 12, 1965). Classified.

~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

4

of 1100°C. The power densities of the fuel were in the range 0.4 to 0.8 kw/cm³ with burnups to 1.0 at. % U. The release of fission gas was negligible for all pins, and the increases in volume and diameter were well within the design limits.

The discontinuation of the CANEL program for irradiation of UN left many areas to be investigated. Some of these were studied in the series of tests discussed in this report. Principal interest was in the performance of UN fuel at central temperatures of 1100 to 1300°C; the effects of higher burnup (up to 4.6 at. % U) and longer times (up to 12,000 hr) at these temperatures were investigated.

DESCRIPTION OF EXPERIMENTS

Figure 1 is a schematic diagram of the UN fuel pins, all of which contained UN pellets that were cold pressed and sintered and clad with PWC-11. The cladding was 0.250 in. in outside diameter and 0.025 in. thick and had a 0.005-in.-thick reaction barrier of chemically vapor-deposited tungsten on the inside surface. Capsules 57-642, -643, and -649 each contained a single fuel pin 4.74 in. long with a fuel column 3.30 in. long. The other 12 capsules contained short fuel pins 1.58 in. long, each with a fuel column 0.65 in. long. Capsules 57-002 and -003 contained four pins, and the remaining ten capsules each contained three

~~CONFIDENTIAL~~
ORNL-DWG 66-10391B

Fig. 1. Schematic Diagram of Typical UN Fuel Pin.

~~CONFIDENTIAL~~

CONFIDENTIAL

5

pins. The PWC-11 end plugs in the pins were separated from the fuel by hollow cylinders of tungsten, which also provided a gas plenum at both ends of the fuel column. All fuel pins were backfilled and sealed in a helium atmosphere. The UN fuel pellets were 0.188-in.-diam right-circular cylinders. The unirradiated fuel contained nominal impurity contents of 500 ppm C and 1000 ppm O, and the preirradiation microstructures (Fig. 2) showed very little second-phase UO_2 . The uranium was enriched to 10% ^{235}U .

Fig. 2. Typical Microstructure of As-Sintered Uranium Mononitride.

Figure 3 is a schematic of an irradiation capsule. The inner and intermediate container tubes were Nb-1% Zr, and the outer container was type 316 stainless steel. The fuel pin was immersed in ^7Li , covered with helium, and sealed within the inner container. Spacers made of Nb-1% Zr centered the fuel pins in the static lithium bath. Four thermocouples were fitted into grooves in the outer surface of the inner container. There were no thermocouples within the region of the fuel specimen itself.

CONFIDENTIAL

~~CONFIDENTIAL~~

6

~~CONFIDENTIAL~~
ORNL-DWG 67-6367R

Fig. 3. Schematic Diagram of Specimen Region of Capsule Showing One Long Fuel Pin.

The desire to maintain accurate temperature control independent of neutron flux perturbations and fuel burnup dictated the capsule design. The capsule was designed so that fission heat alone would result in a fuel temperature somewhat below that desired during the test. The actual operating temperature and fine temperature control were then achieved by electric resistance heaters after the reactor had been brought to power. The fission-heat power and the temperature of the cladding were calculated from the measured irradiation temperature, the power consumption by the resistance heaters, the design calculations for heat transfer, and the preirradiation calibration of capsule temperature versus the electrical consumption by the resistance heaters. The power density reported for these capsules was an average calculated from the burnup data and the operating time above 980°C.

A 2000-w electric heater was wound spirally over the inner Nb-1% Zr tube and the thermocouples in the annulus between the inner and intermediate containers. The heater and thermocouples, both sheathed with Nb-1% Zr alloy, were brazed through the top bulkhead of the intermediate tube. A final closure weld was made after filling this annulus with NaK and helium cover gas. An argon-filled gas gap was provided between the intermediate and outer tubes. The outer tube was joined to the 3/4-in. lead tube through a transition bulkhead to which both were welded. The

~~CONFIDENTIAL~~

CONFIDENTIAL

7

thermocouple and heater leads passed through and were seal brazed to this transition bulkhead. The outer wall was designed to fit within an MTR X-basket. The measuring ring on the lead tube (shown in Fig. 3) provided a means for measuring the vertical position of the capsule relative to the reactor core during installation. The outer wall of the capsule was a 1 1/8-in.-OD $\times$ 12 15/16-in.-long stainless steel tube. A 25-ft-long $\times$ 3/4-in.-OD stainless steel lead tube attached to the top of the capsule served as a conduit for thermocouple leads and as a positioning device to fix the capsule in the reactor.

The four thermocouples located in grooves in the outer surface of the inner capsule wall specially made by Pratt and Whitney, were 1/16 in. in outside diameter, clad with Nb-1% Zr, and insulated with alumina. In the capsules that contained the long fuel pins, two thermocouples were located at the midplane and one 1/2 in. from either end of the fuel pin; in the capsules that contained short fuel pins, two thermocouples were placed at the midplane of the center pin and one at the midplanes of both the top and bottom pins. In capsule 57-642, one of the midplane thermocouples (TC-4) was Chromel-P vs Alumel. The other three were W-5% Re vs W-26% Re. In all other capsules, all of the thermocouples were Chromel-P vs Alumel.

INSTRUMENTATION AND CONTROLS

Irradiation test data were recorded on instrument panels located on the main floor of the MTR building. Each panel contained a multipoint Brown recorder with a millivolt readout and a single-pen recorder. A rotary selector switch permitted continuous monitoring of any desired signal from the multipoint recorder by the single-pen recorder. Power was supplied to the electric heater for the capsule from a manually or automatically controlled powerstat. On automatic control, the capsule temperature was controlled by the single-pen recorder-controller, which regulated the temperature through the motor-operated powerstat.

Capsule temperatures, heater power, and reactor power were recorded by the multipoint. The heater voltage and amperage were converted to a power signal through a transducer and recorded as a calibrated voltage

CONFIDENTIAL

CONFIDENTIAL

signal. The heater power was then computed from this voltage signal and corrected for line losses by the use of preirradiation resistance measurements.

The capsule operation was not tied into the safety circuitry of the MTR. If any temperature exceeded a preset value, the heaters automatically cut off and an alarm was sounded; the heater was then reset manually.

IRRADIATION HISTORY

Irradiation histories of the 15 capsules in this test series are summarized in Table 1. Note that eight capsules containing a total of 22 fuel pins operated successfully for design irradiation times of about 10,000 hr. Most of the capsules operated at conditions that approximated SNAP-50 Reference Design No. 3. However, four of the five untested capsules ORNL obtained from Pratt and Whitney-CANEL were modified at ORNL so as to achieve higher power densities and, therefore, higher burnups than initially intended. In two of these capsules (57-612 and -665) the fuel-center temperatures were maintained at about 1175°C, and the cladding temperature was reduced to about 980°C. In the third capsule (57-669), the central temperature was reduced to 1075°C and the cladding temperature to 980°C. In the fourth capsule (57-003), the central temperature was increased to about 1325°C, and the cladding temperature was maintained at 1200°C.

Cladding temperatures were calculated from thermocouple readings and the calculated power generation of the fuel pins. Temperature drops from the cladding surface to the thermocouples were in the range of 10 to 15°C so that even large errors in calculating the rate of heat generation still result in fairly small errors in estimating the temperature of the cladding. We believe the thermocouples were accurate to $\pm 10^\circ\text{C}$. It is unlikely that the total temperature error was greater than $\pm 15^\circ\text{C}$. Plots of typical axial and radial temperature distribution of the fuel pin in capsule 57-642 are shown in Figs. 4 and 5. These curves show both significant axial and radial temperature gradients.

CONFIDENTIAL

CONFIDENTIAL

Table 1. Results of Postirradiation Examination of CANEL-UN Capsules

Capsule Number ^a	Irradiation Test Time (hr)	Average Cladding Temperature (°C)	Calculated Temperature at Fuel Center (°C)	Burnup ^b (at. % U)	Calculated Power Density (w/cm ²)	Surface Heat Flux (w/cm ²)	Fission Gas Release (%)	Fuel Density Decrease (%)	Fuel Volume Increase (%)	Change in Pin Diameter	
										(in.)	(%)
57-612	5205			2.54	1.41	127					
TTS		870	1065				0.05	4-6 ^c	4-6 ^c	0.0021	0.84
MTS		975	1180				0.37	4-6 ^c	4-6 ^c	0.0031	1.24
BTS		915	1110				0.09	4-6 ^c	4-6 ^c	0.0022	0.92
57-642	9966	1180	1225	1.18	0.34	31	0.34	4.0	4.1	0.0020	0.80
57-643	4292	1180	1260	0.85	0.57	51	d	3.2	3.3	0.0007	0.36
52-649	10,512	1160	1215	1.43	0.40	36	0.038	2.66	2.74	0.0015	0.59
57-652	3180	1095 (1000 hr)	1185	0.7 ^e	0.64 ^e						
TTS		1205 (2180 hr)	1295			58	f	g	g	nil	nil
MTS							0.09	g	g	0.0004	0.16
BTS							f	g	g	0.0005	0.20
57-656	3180	1095 (1000 hr)	1180	0.6 ^e	0.53 ^e	48					
TTS		1205 (2180 hr)	1290				0.03	g	g	0.0006	0.24
MTS							0.11	g	g	0.0003	0.12
BTS							0.13	g	g	0.0016	0.48
57-658	9779			1.08	0.32	29					
TTS		1115	1155				d	3.7	3.8	nil	nil
MTS		1185	1230				0.14	3.8	4.0	0.00045	0.18
BTS		1165	1210				0.09	3.5	3.6	0.0004	0.16
57-660	2635	1095 (455 hr)	1200	0.7 ^e	0.76 ^e	68					
TTS		1205 (2180 hr)	1310				f	g	g	0.0007	0.28
MTS							0.05	g	g	0.0007	0.28
BTS								g	g	0.0005	0.20
57-662	10,270			1.16	0.37	33					
TTS		1065	1115				0.01	4.1	4.3	0.0012	0.47
MTS		1200	1250				0.08	4.3	4.5	0.0011	0.43
BTS		1190	1240				0.04	4.5	4.7	0.0010	0.40
57-664	6209			0.58	0.27	24					
TTS		1110	1145				<0.01	1.8	1.8	0.0003	0.12
MTS		1150	1190				<0.01	1.4	1.5	-0.0002	-0.08
BTS		1165	1205				<0.01	1.2	1.2	0.0002	0.08
57-665	9583			4.58							
TTS		860	1050		1.39	125	1.36	7.1	7.7	0.0025	1.00
MTS		965	1153				5.96	7.9	8.5	0.0038	1.52
BTS		905	1100				1.78	7.7	8.3	0.0026	1.07
57-667	6209			0.78	0.36	32					
TTS		1115	1165				<0.01	3.0	3.1	nil	nil
MTS		1175	1225				0.02	1.7	1.8	0.0003	0.12
BTS		1180	1230				0.03	1.8	1.9	0.0004	0.16
57-669	10,352			2.72	0.76	69					
TTS		835	940				0.05	6.1	6.5	0.0013	0.52
MTS		980	1075				0.13	6.9	7.4	0.0009	0.36
BTS		930	1035				0.02	5.6	6.0	0.0009	0.36
57-002	10,173			1.12	0.32	29					
TTS		1125	1170				<0.02	2.4	2.5	0.0010	0.39 ^h
2TS		1185	1230				<0.02	2.2	2.3	0.0014	0.55 ^h
3TS		1203	1245				0.06	1.9	2.0	0.0018	0.71 ^h
BTS		1165	1205				d	i		0.0019	0.75 ^h
57-003	11,985			4.17	1.01	91					
TTS		1100	1240				6.0	9.7	10.7	0.0036	1.42
2TS		1160	1300				10.3	9.4	10.4	0.0048	1.90
3TS		1185	1325				11.5	9.3	10.2	0.0051	2.02
BTS		1160	1300				12.6	10.8	12.1	0.0048	1.90

^aTTS = top test specimen; MTS = middle test specimen; BTS = bottom test specimen.^bBased on isotopic analyses unless otherwise stated.^cEstimated values; preirradiation inspection data not available.^dSample lost.^eBased on calculations of heat balance. These values are probably high.^fNo sample taken.^gNot measured.^hThese numbers are based on conservative calculations and are probably high.ⁱAll pellets were split during handling of this pin, and the resulting densities are thought to be erroneous.

CONFIDENTIAL

CONFIDENTIAL

~~CONFIDENTIAL~~
ORNL-DWG 68-3226R

Fig. 4. Axial Temperature Distribution in Capsule 57-642.

Fig. 5. Radial Temperature Distribution in Capsule 57-642 at Axial Center Plane Under Normal Operating Conditions.

CONFIDENTIAL

Detailed calculations of the temperature distribution across the fuel in capsules 57-642 and -643 were made by using the PETP06 (ref. 6) and ANISN (ref. 7) computer codes. The ANISN code was used to calculate the depression of thermal flux as a function of the fuel radius; the results are shown in Fig. 6. The PETP06 code was used to calculate the temperature distribution as a function of the fuel radius using input from the ANISN code, thermal conductivity for the UN fuel of $0.25 \text{ w cm}^{-1} \text{ }^{\circ}\text{C}^{-1}$, a gamma heating rate of 0.5 w/g , and energy per fission of 177 Mev. Because of the effects of flux depression, the rate of heat generation in the center of the fuel was calculated to be only 63.4% of that on the surface.

⁶R. B. Fitts, personal communication.

⁷W. W. Engle, Jr., A User's Manual for ANISN, K-1693 (Mar. 30, 1967).

Fig. 6. Thermal Flux Depression in Capsules.

As input to the PETPO/6 computer code and to calculate the temperature distribution in the fuel pins, we used average linear heat ratings calculated from the burnup data. From the linear heat ratings, we calculated radial temperature drops across the fuel pins as follows:

$$\Delta T_{\text{total}} = \Delta T_{\text{UN}} + \Delta T_{\text{He}} + \Delta T_{\text{W}} + \Delta T_{\text{Nb-1\% Zr}} ,$$

where

ΔT_{total} = temperature drop from fuel centerline to cladding surface,

ΔT_{UN} = temperature drop across the UN fuel,

ΔT_{He} = temperature drop across a 0.001 in. helium gap,

ΔT_{W} = temperature drop across tungsten barrier, and

$\Delta T_{\text{Nb-1\% Zr}}$ = temperature drop across cladding.

For calculating the temperature drops across the helium gap, tungsten barrier, and PWC-11 cladding, we used the following equation:

$$\Delta T_i = \frac{H \ln \left(\frac{d_a}{d_b} \right)}{2 \pi K_i} ,$$

where

ΔT_i = temperature drop across material,

d_a = outside diameter of material,

d_b = inside diameter of material,

K_i = thermal conductivity of material, and

H = average linear heat rating.

In these calculations we assumed the equivalent of a 0.001-in. helium gap between the cladding and fuel. It is significant to note that Pratt and Whitney-CANEL personnel assumed the equivalent of a 0.002-in. helium gap for all of their calculations. If a 0.002-in. gap is assumed, calculated temperatures at the fuel center would be higher by 20 and 75°C at 0.4 and 1.5 kw/cm³, respectively.

RESULTS OF POSTIRRADIATION EXAMINATION

Table 1 summarizes the results of the postirradiation examination of these capsules. Capsules 57-652, -656, and -660 were given only cursory examinations because they had undergone mixed irradiation conditions and because the burnups (about 0.4 at. % U) were too low to add significantly to the existing data. The rest of the capsules were examined extensively.

Capsule Disassembly

Figure 7 shows a typical capsule after disassembly with the NaK removed. Measurements of the outer Nb-1% Zr container showed an average increase of 0.030 in. in diameter after the irradiation test, which reduced the gas gap an average of 57%. This explains why the heater required increasing amounts of power with increasing time during irradiation to maintain the same cladding temperature.

We cut and removed the inner Nb-1% Zr container and examined the lithium for voids and general distribution. In the lithium surrounding

Fig. 7. Capsule 57-667 After Initial Disassembly and Removal of NaK.

the pins we found several voids that we believe resulted from the change in the volume of the lithium during solidification when the capsule was cooled to room temperature. There was no evidence to suggest that these voids, which would have caused hot spots, were present during irradiation testing. All fuel pins were axially gamma scanned to determine the relative burnups within each pin. Figures 8 and 9 show two such scans for a capsule that contained long pins and one for a capsule that contained three short pins. The burnup was relatively uniform in all capsules, with peaking at the ends of the fuel column because more neutrons have access to the ends of a fuel column.

Actual burnup values were calculated from isotopic analyses for U, ^{137}Cs , and ^{144}Ce and from the advertised MTR neutron flux. The values reported are those based on isotopic analyses used to calculate burnups with the RCR/SS computer code.⁸ The code provided the following data of

⁸A. R. Olsen and J. W. Prados, Estimation of Resonance Effects in Fuel Burnup Analysis, ORNL-TM-1760 (March 1967).

Fig. 8. Capsule 57-642. Integrated counts between 0.72 and 0.80 Mev.

~~CONFIDENTIAL~~
ORNL-DWG 68-3227

Fig. 9. Capsule 57-660. Integrated Counts Between 0.72 and 0.80 Mev.

of interest: the resonance ratio, total neutron flux, average flux, and percentage of heavy metal fissioned.

Macroscopic Examination of Fuel Pins

After removing the lithium, we examined each of the pins closely under the stereomicroscope for possible defects; none were observed in any of the 41 pins in this series of tests. A typical example of the appearance of the fuel pins after test is shown in Fig. 10. The only change appeared to be a brightening of the surface of the cladding.

Fig. 10. A Typical Example of the Appearance of a Fuel Pin After Test.

Dimensional Measurements of Fuel Pins

We used opposing dial gages to measure diameters to ± 0.0002 in. and a micrometer to measure length to ± 0.001 in. For the long fuel pins, we measured the diameter every 0.25 in. starting 0.25 in. from the top of the fuel pin. For the short pins, we measured the diameter at the center plane, $1/8$ in. on either side of the center plane, and $1/4$ in. from either end. A set of measurements was made, the pins were rotated 90° , and another set of measurements was made. The results, expressed as increases in pin diameter, are summarized in Table 1.

None of the pins exceeded the SNAP-50 Reference Design No. 3 limit of 2% increase in pin diameter, though three of the pins in capsule 57-003 approached this limit with increases of 1.9, 1.9, and 2.0%, respectively. But these increases occurred only after the fuel pins went to 4.17 at. % U burnup at a power density (and therefore linear heat generation rate) of four times the design. Only five other pins increased 1.0% or greater, all at burnups that exceeded the limits of SNAP-50 Reference Design No. 3. In general, the increases were small, and it appears that the SNAP-50 design criteria were conservative.

Release of Fission Gas

To sample the specimens for gas release, we punctured the pins in an end plenum and pumped the gas into a sampling rig. A stainless steel

system with a mercury toepler pump provided a technique for sampling to $\pm 0.03 \text{ cm}^3$. The gas sample was pumped into a known volume, and the pressure was measured. From the known pressure and volume, the volume of gas was calculated and converted to a volume at standard temperature and pressure. Gas samples were submitted to both mass spectrometry and gamma counting. Using these analyses and the calculated volume of gas sampled, we determined values for release of fission gas (Table 1).

All values for gas release were lower than required by SNAP-50 Reference Design No. 3, even though burnup values were more than three times the reference design. The maximum gas release (10 to 12%) was observed in those fuel pins that operated at the highest fuel-center temperature (1325°C).

Macroscopic Examination of Fuel Pellets and Cladding

The fuel pellets from all capsules appeared to be in excellent condition. Typical examples of the appearance of the fuel after irradiation are shown in Figs. 11 and 12. No cracking of the fuel occurred due to the irradiation exposure, though a few pellets were cracked or chipped during the subsequent examination in the hot cells. Some pellet interfaces sintered, especially at the higher burnups. We often observed small lines in the pellet surfaces that corresponded to cracks in the reaction barrier (Fig. 13). We postulated that the lines correspond to a thermal etching caused by localized hot spots resulting from a reduction in gap conductivity in the vicinity of the crack in the barrier.

Fig. 11. Irradiated Fuel Pellets of Capsule 57-642 After Pin Was Cut Open. The lines on the pellet probably resulted from thermal etching caused by the cracked tungsten barrier.

CONFIDENTIAL

Fig. 12. Middle Test Pin From Capsule 57-612 After the Pin Was Cut Open. Note the good appearance of the fuel after a burnup of 2.54 at. % U.

Fig. 13. Extensive Cracking Observed in the Tungsten Barrier of Capsule 57-642 After Irradiation. The inside surface is shown.

CONFIDENTIAL

These cracks in the tungsten did not appear to affect the performance of the fuel.

Measurements of Fuel Density

The densities of the fuel pellets after irradiation were calculated from weight measurements and from volume measurements made with a mercury pycnometer especially designed and built for this program. This equipment is described in detail by Dismuke and Stone.⁹ The weighing accuracies were ± 0.0003 g, and the volumetric accuracies were ± 0.0003 cm³. With the exception of that for capsule 57-003, the changes in fuel volume tabulated in Table 1 were all less than the minimum performance limitation of 8.7% (8.0% decrease in fuel density).

Metallography of Cladding

Two metallographic samples were made from the cladding from each of the short fuel pins: a transverse section at the center plane of each pin and a longitudinal section from the center plane to the end of the pin. Three or four samples from each long fuel pin were examined.

The PWC-11 cladding performed very well under the service conditions of 980 and 1200°C. No failures or evidences of corrosion by lithium were observed (Fig. 14). The only significant effect was a 0.001-in. reaction zone at the interface between the tungsten barrier and cladding (Fig. 15) that consisted of a region of interdiffusion of niobium and tungsten and a region containing a uranium-niobium intermetallic. At 980°C, this reaction zone was less than 0.0001 in. thick. The reaction did not appear to be detrimental to the performance of the fuel pins for the times and temperatures in these experiments.

⁹S. E. Dismuke and R. L. Stone, A Vacuum Pycnometer Using Mercury with Plunger Displacement for Hot Cell Use, ORNL-TM-1743 (February 1967).

CONFIDENTIAL

Fig. 14. Appearance of the Irradiated Cladding After Etching. The outer surface was in contact with lithium at 1180°C for 9966 hr.

Fig. 15. Interface Between Tungsten Barrier and PWC-11 Cladding.
 (a) As polished. (b) Etched.

CONFIDENTIAL

CONFIDENTIAL

Metallography of Fuel

Metallographic examination revealed bubbles of fission gas in the microstructure of the fuel pellets. In the pins that operated at a cladding temperature of 1200°C, these bubbles migrated radially up the temperature gradient until they reached a grain boundary, where they became trapped (Fig. 16). This effect was particularly evident in

Fig. 16. Transverse Section of Fuel from Bottom Test Pin of Capsule 57-667 (1200°C Cladding Temperature 0.78 at. % U Burnup). Fission gas bubbles have apparently migrated to grain boundaries parallel to the outside diameter of the pellets. As polished. 200x.

regions near the surface of the pellets where the temperature gradient was greatest. The result of this oriented bubble migration is that there was a preponderance of bubbles on grain boundaries parallel to the fuel surface. Bubbles were also trapped at grain boundaries parallel to the ends of the fuel column; this suggests the presence of an axial temperature gradient (Fig. 17). Few bubbles were observed at grain boundaries near the centers of the pellets where (1) the relatively flat temperature profile apparently did not provide a significant driving force for bubble migration, and (2) fewer gas atoms were present due to the depression of the thermal neutron flux.

CONFIDENTIAL

~~CONFIDENTIAL~~

Fig. 17. Photomicrograph Showing the End of the Fuel Column in the Bottom Test Pin of Capsule 57-667. Bubbles apparently became trapped at grain boundaries parallel to the end of the column suggesting the presence of an axial temperature gradient. 500x.

By reducing the cladding temperature from 1200 to 980°C while increasing the rate of linear heat generation from 2 to 8 kw/ft, we could assess the effects of higher linear heat ratings. In these tests, the mobility of the bubbles of fission gas near the periphery was significantly reduced (Fig. 18). Under these conditions, very few bubbles reached grain boundaries by radial migration in spite of the fact that the temperature gradient was increased to four times that for specimens that operated at cladding temperatures of 1200°C. Because of the axial gradient, however, gas bubbles did line up at grain boundaries parallel to the end of the fuel column. Apparently the temperature of the pellets at the end of the fuel stacks but near the radial center was high enough so that the bubbles were mobile.

Electron microscopy of fractured surfaces (Figs. 19-23) of the fuel from capsules 57-612 and -665 illustrate the distribution of fission gases.

~~CONFIDENTIAL~~

CONFIDENTIAL

23

Fig. 18. Fuel From Middle Test Pin of Capsule 57-612 Showing Absence of Bubbles of Fission Gas Near the Periphery of Fuel at Burnup of 2.54 at. % U. (a) 100x. (b) 500x.

CONFIDENTIAL

~~CONFIDENTIAL~~

24

Fig. 19. Fractograph of Capsule 57-612, Pellet 3, Middle Test Specimen. Mode of fracture was intergranular. 12,000X. Reduced 47.5%.

Fig. 20. Fractograph of Capsule 57-612, Pellet 3, Middle Test Specimen. Transition of fracture mode from intergranular to transgranular. 12,000X.

~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

25

Fig. 21. Fractograph of Capsule 57-612, Pellet 3, Middle Test Specimen. Mode of fracture was ductile. View shows plastic slip along slip planes. 12,000x.

Fig. 22. Typical Electron Photomicrograph from Fractured Surface of UN Pellet from Capsule 57-665. 7000x. Reduced 52.5%.

~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

Fig. 23. Typical Electron Photomicrograph From Fractured Surface of UN Pellet From Capsule 57-665. 7000X. Reduced 41.5%.

Near the periphery, the fracture was intergranular, and large bubbles had coalesced at the grain boundaries. At about midradius, the fracture was intergranular, and bubbles, about 800 A in diameter, were observed within the grains. Near the center, the fuel appeared to have flowed plastically, indicating that restraint by the cladding and the outer regions had caused the fuel in the center to extrude axially.

The fuel (Figs. 24 and 25) that went to greater than 4 at. % U burnup (capsules 57-665 and -003) behaved unusually in that most of the fission gases were retained as bubbles within the grains instead of at grain boundaries. Several grain boundaries contained many bubbles while others contained almost none. We suspect that the high-angle grain boundaries probably trapped bubbles while low-angle grain boundaries did not.

~~CONFIDENTIAL~~

Y-95496

Fig. 24. Fuel From the Middle Test Specimen of Capsule 57-665 at About Midradius. Most of the bubbles of fission gas are retained within the grains. The center of the fuel pellet was toward the right. 500X. As polished.

CONFIDENTIAL

CONFIDENTIAL

Fig. 25. Fuel From the Bottom Test Pin of Capsule 57-003 at About Midradius. Most of the bubbles of fission gas are retained within the grains. The center of the fuel pellet was toward the right. 500x. As polished.

CONFIDENTIAL

DISCUSSION AND CONCLUSIONS

The irradiation performance of the 41 fuel pins in this test series suggests that the design criteria for SNAP-50 Reference Design No. 3 were conservative. The following list of conclusions illustrate the good performance of the fuel:

1. None of the 41 fuel pins failed despite burnups of up to three times the design burnup and irradiation times of up to 12,000 hr.
2. Twenty-two of the 41 fuel pins were successfully tested to the design test time of 10,000 hr; those fuel pins tested less than 10,000 hr did not fail but were removed because they had completed their scheduled irradiation testing.
3. None of the fuel pins exceeded the performance criteria of 2.0% increase in pin diameter and 20% release of fission gas as outlined in SNAP-50 Reference Design No. 3. Fuel swelling exceeded design only in capsule 57-003, which had a burnup of 4.2% and a central temperature of about 1325°C.
4. No corrosion of the PWC-11 in contact with the lithium was observed in any of the fuel pins.

The release of fission gas from these tests is compared graphically with data generated by Pratt and Whitney-CANEL at fuel-center temperatures of 1100 and 1500°C in Fig. 26. All of the values for gas release were low except that for capsule 57-003, which operated at 1325°C to 4.2 at. % U burnup. It should be noted that the release of fission gas from the middle test pin in capsule 57-665 (4.6 at. % U burnup) was higher (5.96%) than that from the top and bottom pins (1.36 and 1.78%, respectively). This higher release in the middle test pin resulted because we inadvertently punctured the fuel pin in the region of the fuel instead of the end plenum, thereby fracturing the fuel and causing an unusually high release of the fission gases. The actual gas released during the test was probably less than 2.0%.

It is important to note that except for capsule 57-003 the pins tested to high burnup operated at cladding temperatures of only 980°C, or 220°C less than the SNAP-50 design temperature. Consequently some caution is necessary before concluding that fuel pins operating at 1200°C

CONFIDENTIAL

~~CONFIDENTIAL~~
ORNL-DWG 69-6264

Fig. 26. The Release of Fission Gas Data as a Function of Burnup.

cladding temperature to 4.6 at. % U burnup would release only 2% of the gas. The maximum burnup achieved at close to SNAP-50 temperatures, power levels, and test time showed a gas release of 0.34% after 1.18 at. % U burnup. The only high (but not excessive) gas release was from capsule 57-003, in which the center of the fuel operated at about 1325°C to a burnup of 4.2 at. % U. Comparing capsule 57-003, which operated at a cladding temperature of 1200°C to 4.2 at. % U burnup but at a higher power density than the reference design, with capsule 57-665, which operated at a cladding temperature of 980°C to 4.6 at. % U burnup, we can conclude that under conditions of SNAP-50 Reference Design No. 3 but with a burnup of 4.2 at. % U, the fuel should release from 2 to 10% of the fission gases generated.

Examination of the data for release of fission gas as a function of increase in fuel volume shows that one must take into account the operating conditions of the pins to predict release of fission gas. As the cladding temperature is lowered and the volume-average temperature of the fuel is consequently lowered, less fission gas will be released per unit volume increase of the fuel. Figure 27 illustrates this by showing release of fission gas as a function of fuel density decrease for the

CONFIDENTIAL

Fig. 27. The Release of Fission Gas Shown as a Function of Volume Increase of Fuel.

data from these tests with data from tests at a fuel-center temperature of 1500°C by Pratt and Whitney-CANEL. The results show that the gas generated at lower temperatures has less chance of being released for a given amount of swelling than do gases generated at the higher temperatures.

The microstructures of the fuel support these observations. At high cladding temperatures but low burnup, a significant portion of the fission gases migrated to grain boundaries and became trapped. Because the bubbles migrated preferentially up a temperature gradient, a preponderance of bubbles appeared at grain boundaries parallel to the outside surface of the fuel. This enhanced the coalescence of the fission gases at some grain boundaries. When the cladding temperature was dropped to 980°C, however, most of the bubbles remained within the grains and proportionately fewer were trapped at grain boundaries. In this case, some grain boundaries lying perpendicular to the temperature gradient appeared to have trapped gas bubbles while others in the same orientation relative to the temperature gradient did not. The high-angle grain boundaries probably trapped the bubbles, whereas, low-angle grain boundaries probably did not.

The retention of fission gases within the grains at low cladding temperatures but at high linear heat ratings suggests that release of fission gas will probably be low for LMFBR fuel pins that contain (U,Pu)N. The data also suggest that at high cladding temperatures when

CONFIDENTIAL

the fuel operates significantly above 1500°C, most of the fission gases should be released.

Fuel swelling is plotted as a function of burnup in Fig. 28 and is compared with data from Pratt and Whitney-CANEL for fuel-center temperatures of 1500 and 1100°C. All of the fuel swelling was less than the minimum performance criteria of 8.0% decrease in fuel density (8.7% increase in fuel volume) as outlined in SNAP-50 Reference Design No. 3 except that capsule 57-003 increased in fuel volume as high as 12.1%. The volume increases correlated with the postirradiation metallography. Most of the spread in the data can be accounted for by the different operating conditions of the fuel pins.

The small increases in pin diameter (Fig. 29) were particularly encouraging since diametral increases are the principal design limitation, provided the pins do not fail as was the case in these tests. None

Fig. 28. Fuel Volume Increase as Function of Burnup. The two lines represent data obtained by Pratt and Whitney-CANEL at fuel-center temperatures of 1100 and 1500°C. The points represent data from the series of tests described in this report.

~~CONFIDENTIAL~~

33

~~CONFIDENTIAL~~
ORNL-DWG 69-6263R

Fig. 29. Increase in Specimen Diameter as Function of Burnup. The box in the lower left-hand corner represents a scatter band of the data generated by Pratt and Whitney-CANEL at 1100°C fuel-central temperature. The line represents data obtained by Pratt and Whitney-CANEL at 1500°C fuel-central temperature. The points shown represent data from the series of tests described in this report.

of the pins exceeded the limit of 2.0%, although one pin in capsule 57-003 swelled at the border line. This swelling, however, was observed after the pin operated at a fuel-center temperature of 1325°C for 11,985 hr to a burnup of 4.2 at. % U.

The reaction zones at the interface between the tungsten barrier and PWC-11 cladding did not appear to be detrimental to the performance of the pins. In the cladding operated at 1200°C, the reaction zones were about 0.001 in. thick. When the cladding temperature was dropped to 980°C, this reacted region was barely perceptible - only about 0.00005 to 0.0001 in. thick. The reaction zones appeared to be uniform over the fueled region and in most cases were not worsened by the cracks in the tungsten barrier. The reaction appeared to be self-limiting in that it did not increase significantly in thickness with burnup. The tungsten

~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

34

barrier apparently does not entirely prevent the reaction between UN and PWC-11. We suspect that the uranium migrates down the grain boundaries of the CVD tungsten barrier. Since the tungsten was deposited by a CVD technique, the resulting grains were columnar and the grain boundaries formed straight-line paths between the UN fuel and the PWC-11. The reaction zone does not appear to be detrimental and will probably be of importance only for significantly longer times or higher temperatures.

These tests show that UN may be an attractive fuel for space reactors from the standpoint of fuel performance. We observed no cracking of the fuel pellets except that which occurred during handling in the postirradiation examination. Some sintering occurred at the pellet interfaces; it was enhanced by irradiation and appeared to increase with burnup. Microstructures of the sintered interfaces showed some of the interfaces to be so well bonded that the original interface was discernable only at the outer surface of the fuel.

A point of interest in the performance of the fuel pins was the apparent lack of migration and coalescence of the bubbles of fission gas at high linear heat ratings. The apparently even distribution of the bubbles of fission gas at high temperatures suggests that they are soluble in the UN matrix and might be construed as support for the argument that these bubbles precipitate and grow locally much as ordinary second-phase precipitates do. However, the bubbles in the fuel pins operated at low heat ratings but at higher average fuel temperatures appeared to migrate to grain boundaries, at least near the surface of the fuel. Furthermore, this migration appeared to be preferentially up the temperature gradient and resulted in a preponderance of bubbles at grain boundaries perpendicular to the temperature gradient. An explanation which might resolve the apparently different behavior is the presence of strain fields around the bubbles of fission gas that might have prevented their coalescence. Weaver noted¹⁰ a similar behavior in measurements of the migration of helium bubbles in UN out of the reactor.

¹⁰S. C. Weaver, Helium Gas Bubble Migration in Uranium Mononitride in a Temperature Gradient, ORNL-TM-2016 (December 1967).

~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

35

ACKNOWLEDGMENTS

We wish to acknowledge the help of the many people at the High Radiation Level Examination Laboratory who helped us examine these capsules, particularly Ray Wallace and Earl Sims who performed the photography and metallography on the samples.

We also wish to thank Wayne Proaps of the Ceramics Laboratory and W. C. Colwell for the preparation of the drawings, D. J. Rucker for the editing of the manuscript, and Thelma Reedy for the typing and preparation of the report for reproduction.

~~CONFIDENTIAL~~

Blank Page

~~CONFIDENTIAL~~

37

ORNL-4461

C-92b - Systems for Nuclear Auxiliary
Power (SNAP)-Reactor SNAP Program
M-3679 (61st ed.)

INTERNAL DISTRIBUTION

1-3.	Central Research Library	36.	E. J. Breeding
4-5.	ORNL - Y-12 Technical Library	37.	C. M. Cox
	Document Reference Section	38.	F. L. Culler
6-30.	Laboratory Records	39.	J. H. DeVan
31.	Laboratory Records, ORNL RC	40.	W. S. Ernst, Jr.
32.	ORNL Patent Office	41.	A. P. Fraas
33.	File Point 27 for:	42.	J. M. Googin (Y-12)
	G. M. Adamson, Jr.	43.	W. R. Grimes
	J. E. Cunningham	44.	A. G. Grindell
	R. B. Fitts	45.	W. O. Harms
	J. H. Frye, Jr.	46.	M. R. Hill
	W. Fulkerson	47.	H. W. Hoffman
	A. L. Lotts	48.	H. G. MacPherson
	R. W. McClung	49.	R. E. MacPherson
	D. L. McElroy	50.	A. J. Miller
	C. J. McHargue	51.	B. A. Montgomery
	A. R. Olsen	52.	P. Patriarca
	T. N. Washburn	53.	J. L. Scott
	J. R. Weir, Jr.	54.	R. L. Senn
34.	S. E. Beall	55.	K. R. Thoms
35.	G. E. Boyd	56.	S. C. Weaver

EXTERNAL DISTRIBUTION

- 57. Dan Albreds, University of California, Berkeley, Calif.
- 58. Benjamin C. Allen, Battelle Memorial Institute, Columbus, Ohio
- 59. G. M. Anderson, AEC, Washington
- 60. R. L. Anderson, Space Nuclear Systems Division, Space Electric Power Office, AEC, Washington
- 61. C. H. Armbruster, Wright-Patterson AFB
- 62. V. P. Calkins, General Electric, NMPO
- 63. S. S. Christopher, AEC, Washington
- 64. D. F. Cope, RDT, SSR, AEC, Oak Ridge National Laboratory
- 65. J.W.R. Creagh, NASA, Lewis Research Center
- 66. G. W. Cunningham, AEC, Washington
- 67. J. Davis, Jet Propulsion Laboratory
- 68. R. W. Dayton, Battelle Memorial Institute, Columbus, Ohio
- 69. G. K. Dicker, AEC, Washington
- 70. E. M. Douthett, AEC, Washington
- 71. D. DeHalas, Pacific Northwest Laboratory, Richland
- 72. J. K. Elbaum, Code MAMP, Wright-Patterson AFB

~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

73. R. Gluyas, NASA, Lewis Research Center
74. D. C. Goldberg, Westinghouse Astronuclear Laboratory
75. D. H. Gurinsky, Brookhaven National Laboratory
76. R. Hall, NASA, Lewis Research Center
77. R. C. Hamilton, Institute for Defense Analyses,
1666 Connecticut Avenue, N.W., Washington
78. E. E. Hoffman, General Electric, Cincinnati
79. W. R. Holman, Lawrence Radiation Laboratory
80. F. J. Huegel, University of California, Berkeley, Calif.
81. C. Johnson, Space Nuclear Systems Division, Space Electric
Power Office, AEC, Washington
82. J. S. Kane, University of California, Berkeley, Calif.
83. Don Keller, Battelle Memorial Institute, Columbus, Ohio
84. P. J. Levine, Westinghouse Advanced Reactors Division,
Waltz Mill Site, Box 158, Madison, Pa., 15663
85. J. J. Lombardo, NASA, Lewis Research Center
86. J. J. Lynch, NASA, Washington
87. T. P. Moffitt, NASA, Lewis Research Center
88. T. A. Moss, NASA, Lewis Research Center
89. R. A. Noland, Argonne National Laboratory
90. W. E. Ray, Westinghouse Advanced Reactors Division,
Waltz Mill Site, Box 158, Madison, Pa., 15663
91. Louis Rosenblum, NASA, Lewis Research Center
92. A. J. Rothman, Lawrence Radiation Laboratory
93. A. L. Rothman, University of California, Berkeley, Calif.
94. B. Rubin, Lawrence Radiation Laboratory
95. N. D. Saunders, NASA, Lewis Research Center
96. H. Schwartz, NASA, Lewis Research Center
97. F. C. Schwenk, AEC, Washington
98. J. M. Simmons, AEC, Washington
99. D. K. Stevens, AEC, Washington
100. A. Strasser, United Nuclear Corporation
101. A. Van Echo, AEC, Washington
102. G. W. Wensch, AEC, Washington
103. G. A. Whitlow, Westinghouse Advanced Reactors Division,
Waltz Mill Site, Box 158, Madison, Pa., 15663
104. M. J. Whitman, AEC, Washington
105. Laboratory and University Division, AEC, Oak Ridge Operations
- 106-226. Given distribution as shown in M-3679 (61st ed.) under Systems
for Nuclear Auxiliary Power (SNAP)-Reactor SNAP Program category

~~CONFIDENTIAL~~