DEC 26 1961

GEAP 3835

INFORMAL AEC RESEARCH AND DEVELOPMENT REPORT

MASTER

FUEL CYCLE PROGRAM
A BOILING WATER REACTOR
RESEARCH AND DEVELOPMENT PROGRAM
FIFTH QUARTERLY REPORT

October 10, 1961

Prepared for

The U.S. Atomic Energy Commission
under
Contract AT(04-3)-189, Project Agreement No. 11

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.

GENERAL

ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT

SAN JOSE, CALIFORNIA

47-1

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.

FUEL CYCLE PROGRAM
A BOILING WATER REACTOR
RESEARCH AND DEVELOPMENT PROGRAM

FIFTH QUARTERLY REPORT
JULY 1961 - SEPTEMBER 1961

OCTOBER 10, 1961

Prepared for
The U. S. Atomic Energy Commission
under
Contract No. AT(04-3)-189, Project Agreement No. 11

EDITED BY:

J. A. Hodde

J. A. Hodde, Engineer
Engineering Development

APPROVED BY:

C. L. Howard

C. L. Howard, Project Engineer
Fuel Cycle Research & Development Program

APPROVED BY:

D. H. Imhoff

D. H. Imhoff, Manager
Engineering Development

GENERAL ELECTRIC COMPANY
ATOMIC POWER EQUIPMENT DEPARTMENT
SAN JOSE, CALIFORNIA

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.

Do not distribute

DISTRIBUTION

	<u>No. of Copies</u>		<u>No. of Copies</u>		<u>No. of Copies</u>
William D. Manly Oak Ridge National Laboratory Oak Ridge, Tennessee	(1)	U. S. Atomic Energy Commission Washington 25, D. C. Attention: Chief, Water Systems Project Branch Army Reactors Division of Reactor Development	(1)	Advanced Technology Laboratory A Division of American-Standard 360 Mission Road Mountain View, California Attention: C. J. Barnes	(1)
Assistant Director for Naval Reactors Division of Reactor Development USAREC Washington 25, D. C. Attention: I. E. Mandil	(1)	U. S. Atomic Energy Commission Chicago Operations Office 9700 South Cass Avenue Argonne, Illinois Attention: Director, Reactor Program Division	(5)	Consuevers Power Company 212 West Michigan Avenue Jackson, Michigan	(1)
Westinghouse Electric Corporation Atomic Power Division Bethis Plant, Clairton Site P.O. Box 1526 Pittsburgh, Pennsylvania Attention: A. Squire	(1)	General Nuclear Engineering Corporation P.O. Box 245 Dunedin, Florida Attention: J. M. West	(1)	Dr. F. Charles Moschel Office of Technical Information U. S. Atomic Energy Commission Washington 25, D. C.	1
Knoolls Atomic Power Laboratory P.O. Box 1072 Tehenectacy, New York Attention: W. Coshin	(1)	Combustion Engineering Incorporated Nuclear Division Prospect Hill Road Windsor, Connecticut Attention: W. H. Zinn	(1)	Dr. David B. Langmuir, Director Electronic Research Laboratory Ramo-Wooldrige Corporation 6433 Fallbrook Avenue Canoga Park, California	1
Robert Macnerry Argonne National Laboratory Argonne, Illinois	(1)	Westinghouse Electric Corporation Atomic Power Department P.O. Box 355 Pittsburgh 30, Pennsylvania Attention: Dr. W. E. Shoupp	(1)	Dr. Stanley A. Sawlewicz, Chief Reactor Safety Branch Division of Reactor Development U. S. Atomic Energy Commission Washington 25, D. C.	1
F. W. Albaugh General Electric Company Hanford Works Hanford, Washington	(1)	Allis-Chalmers Manufacturing Company Atomic Energy Division Milwaukee 1, Wisconsin Attention: C. B. Graham	(1)	Dr. Warren E. Myer SPEPT Project Manager Phillips Petroleum Company P.O. Box 1259 Idaho Falls, Idaho	1
U. S. Atomic Energy Commission Washington 25, D. C. Attention: Chief, Water Reactors Branch Civilian Reactors Division of Reactor Development	(2)	Allis-Chalmers Manufacturing Company Nuclear Power Department P.O. Box 8697 Washington 11, D. C. Attention: E. Etherington	(1)	J. F. O'Brien Mail No. W-719 Martin Company, Nuclear Division Middle Fox, Maryland	(1)
Northern States Power Company Minneapolis 2, Minnesota Attention: D. F. McElroy	(1)				

Do not

47-3

TABLE OF CONTENTS

	<u>Page No.</u>
I Introduction	1
II Summary	2
III Task A - Advanced Fuel Power-Limit Test	3
A. Task A-II - Fabrication of Fuel	3
B. Task A-III - Instrument and Calibrate Three Assemblies	3
C. Task A-IV - Stability Test	3
1. Test Plan	4
2. Instrumentation	5
3. Data Reduction and Analysis	5
4. Analytical Program	6
D. Task A-VI - Irradiation in VBWR	6
1. Irradiation Status	6
2. Operating and Loading Schedule	9
E. Task A-VII - Special Fuel Program	11
IV Task B - Heat Transfer & Fluid Dynamics	15
A. Task B-I - Burnout Heat Transfer	15
1. Single Rod Burnout Data	15
2. High Pressure Observational Experiment	15
B. Task B-IV - Two-Phase Pressure Drop	18
V. Task C - Experimental Physics	20

LIST OF FIGURES

	<u>Page No.</u>
Figure 1 - Schematic Diagram Instrumentation System	22
Figure 2 - Fuel Bundle 23-J	23
Figure 3 - Single-Rod Test Section with Rough Liner	24
Figure 4 - Increase in Burnout Heat Flux Due to Roughened Liner	25

LIST OF TABLES

Table I	Irradiation Status of Fuel Cycle Assemblies	7
Table II	VBWR Operating Schedule	9
Table III	VBWR Loading Schedule	10
Table IV	Special Fuel Program	12
Table V	Selected High Speed Data Films	16

I. INTRODUCTION

The Fuel Cycle Program is an integrated program of investigation in the Vallecitos Boiling Water Reactor (VBWR) and other facilities to improve the technological limits of boiling water reactors in the following areas:

Task A

1. Extend fuel life information on oxide fuel at high specific power operation and raise the performance limits of oxide fuels.
2. Study power stability and performance characteristics of an oxide-fueled core under natural and forced circulation to improve design limits.

Task B

Conduct out-of-pile experiments in heat transfer and fluid dynamics in the areas of burnout heat transfer, hydraulic instability, steam void observational studies, and two-phase pressure drop to support in-core work.

Task C

1. Study long-term reactivity characteristics of fuels having lattice characteristics of large power reactors.
2. Measure initial conversion ratio of a number of lattice configurations characteristic of large, oxide-fueled power reactors.

This report is written in partial fulfillment of contract AT(04-3)-189, Project Agreement No. 11, Fuel Cycle Program, between the United States Atomic Energy Commission and the General Electric Company. Prior reports to the Commission under this contract have included the following:

1. GEAP 3516, First Summary Progress Report, March 1959 - July 1960.
2. GEAP 3558, First Quarterly Progress Report, August 1960 - September 1960.
3. GEAP 3627, Second Quarterly Progress Report, October 1960 - December 1960.
4. GEAP 3628, Prediction of Two-Phase Flow From Mixing Length Theory, S. Levy, December 27, 1960.
5. GEAP 3709, Third Quarterly Progress Report, January 1961 - March 1961.
6. GEAP 3655, Pressure Drop Along a Fuel Cycle Fuel Assembly, Various Orifice Configurations, E. Janssen and J. A. Kervinen, May 22, 1961
7. GEAP 3781, Fourth Quarterly Progress Report, April 1961 - June 1961.
8. GEAP 3794, Plan for VBWR Stability Experiment, W. H. Cook, et al, Aug. 30, 1961.
9. GEAP 3628, Revision I, Prediction of Two-Phase Pressure Drop and Density Distribution from Mixing Length Theory, S. Levy (To be issued).

II. SUMMARY

Task A. - Advanced Fuel Power-Limit Test

1. Fabrication of the 50 basic fuel assemblies has been completed.
2. The three instrumented fuel assemblies were assembled and calibrated.
3. Equipment and procedures have been prepared for the VBWR stability tests during the next quarter.
4. Forty-two assemblies were irradiated during the last VBWR run. Average exposure for all assemblies which have been irradiated are:

<u>Fuel</u>	<u>Number of Assemblies</u>	<u>Ave. Exposure MWD/T</u>	<u>Maximum Exposure MWD/T</u>
Fuel Followers (H)	10	1680	2010
Stainless Clad (I)	15	1820	2150
Zircaloy Clad (J)	18	760	1400
Special Fuel	3	590	1280

5. Detailed examination of the centermelt calibration assembly (I-L) is still in progress. Preliminary metallography indicates that centermelting did not occur. The conductivity of UO_2 is appreciably higher than the design basis of $1.1 \text{ Btu/hr-ft}^2 (^\circ\text{F/ft})$. This result is supported by experiments at Hanford and Chalk River, and indicates that the fuel performance limit is higher than the burnout heat flux limit.

Task B - Heat Transfer and Fluid Dynamics

1. Use of a "rough" liner in the single rod test section resulted in a 10 to 43% increase in burnout heat flux at various operating conditions.
2. The high speed movies of the high pressure observational boiling experiment have been edited and assembled. Analytic work is being continued.
3. A theoretical analysis of two-phase pressure drop and density distribution has been made.

Task C - Experimental Physics

Isotopic analyses of un-irradiated fuel pellets show good uniformity. Preliminary calculations of expected isotopic compositions during irradiation are being made.

III. TASK A - ADVANCED FUEL POWER-LIMIT TESTS

A. Task A-II - Fabrication of Fuel

The objective of this subtask was fabrication of 50 "basic" fuel assemblies for irradiation in VBWR. The fabrication was completed in the previous quarter except for minor modification of three assemblies:

- 17J - Task C, Isotopic Composition Experiment
- 18J - Task C, Isotopic Composition Experiment
- 23J - Plutonium Recycle Assembly

The attachment of guide tubes for flux wire irradiation was changed slightly to facilitate replacement of fuel rods. A description of the assemblies was given in the previous reports. Figure 1 shows the assembly drawing of assembly 23J. Assemblies 17J and 18J are identical to assembly 23J except that they contain unsegmented fuel rods of 2.7% enriched UO_2 . Assembly 23J is segmented with the upper fuel section of UO_2 , and the lower nine inches of each fuel rod containing natural UO_2 enriched 1.5% in PuO_2 . (Assembly 23J is shared by this program and the Plutonium Recycle Program, Project Agreement No. 21, between the Commission and General Electric Company).

A topical report on fabrication of the basic fuel is in preparation.

B. Task A-III - Instrument and Calibrate Three Assemblies

Three fuel assemblies were fabricated with temperature, flow, pressure drop and neutron flux instrumentation. This equipment will be installed in VBWR during October for the stability tests under Task A-IV. The instrumentation system consists of the three fuel assemblies, instrument leads which will penetrate the reactor enclosure and carry the signals to the readout panels located in the VBWR control room. The readout panels will be shared by the Fuel Cycle and the High Power Density Instrumented Assemblies.

The fabrication and calibration of the three assemblies was completed in September 1961. Reduction of the calibration data will continue into the next quarter.

C. Task A-IV - Stability Test

Early development work on boiling water reactors concentrated on the problem of the stability limits because of reports of power oscillations of large magnitude. The result was a set of empirical conditions limiting stable reactor operation, which have been used in the design of subsequent reactors. However, achievement of favorable economics in improved systems may require operation beyond the original empirical limits. As a result further development efforts in boiling water reactor stability have become necessary. This development is being conducted concurrently with the heat transfer and fuel performance work to raise all the limits on reactor operation.

The present schedule provides for shutdown of the VBWR on October 5 to make preparations for the stability test. The tests will begin during the latter part of October and continue through December 1961. A topical report, GEAP 3794, Plan for VBWR Stability Experiment, by W. H. Cook, J. A. Hodde, C. L. Howard, R. O. Niemi, was published August 30, 1961.

1. Test Plan

The stability experiment will be performed at approximately 30 reactor operating points (i.e., power levels and recirculation flows). An average of about 15 test runs will be performed at each point. During each run 20 variables will be recorded on the fast readout instruments for periods of time varying from 40 seconds to 5 minutes. The significant variables which will be explored are:

- (a) Natural versus forced circulation modes of operation.
- (b) Power level variations (15, 22, 30 MW and maximum attainable - maybe 36 MW).
- (c) Forced circulation flow rate variations (2,500 to 20,000 gpm).
- (d) Minimum versus maximum flow resistance in recirculation loop (6 to 300 ft. head).
- (e) Reactor water level variations
- (f) Effect of orificing on individual fuel channel stability (one instrumented assembly is orificed).
- (g) Extent of parallel-channel hydraulic oscillation.
- (h) Extent of recirculation loop hydraulic oscillation.

The tests to be performed include the following:

- (a) Transfer Function Test - A control rod will be oscillated at number of frequencies between 0.03 and 3.0 cycles per second. The readout system will be connected to measure ion chamber responses, fuel flow rates, the reactor steam, feedwater and recirculation flows, the reactor pressure and the control rod motion. The transfer function from the control rod to each measured variable can be calculated.
- (b) "Noise" Tests - With the reactor system at steady-state conditions, recordings of the small scale oscillations (called "noise") in neutron fluxes, flow rates and pressures will be made. The recordings will be made by both direct-writing oscillograph and by tape recorder. The oscillographs will give a record which can be analyzed visually while the tape recorder will give a record which can be analyzed on analog or digital computers. An indication of important frequencies and phase relationships between various parameters will be obtained. A band pass filter will be used for direct measurement of the frequency spectrum.

- (c) Control Rod Calibration - To determine the control rod reactivity worth as a function of position, the control rod oscillation technique will be used. The control rod will be oscillated at several rod positions. This process will be repeated both for the hot zero-power case and for the reactor at power operation.
- (d) Transient Tests - After the reactor has reached a steady operating condition, ramp changes in steam flow, feedwater flow or recirculation flow will be made. A recording of in-core and out-of-core system response will be made.
- (e) Water Level Variation Test - Recordings of the "noise" (i.e., small scale oscillations) in neutron fluxes, flow rates and pressures will be made for various levels of water in the reactor vessel.

2. Instrumentation

The instrumentation and read-out systems for the stability test were designed and major items of equipment were procured during the past quarter. The system is designed to record any 20 out of 42 signals from in-core and out-of-core transducers.

The major items in the instrumentation are three instrumented fuel assemblies from the Fuel Cycle Program (Task A-III) and two instrumented assemblies already in the VBWR core from the High Power Density Program. The readout system consists of two 6-channel and one 8-channel direct writing oscillograph, a 14-channel FM tape recorder and a readout system switching and "signal conditioning" center. A general schematic of the system is shown in Figure 1. Twelve of the tape recorder channels and twelve of the oscillograph channels will be connected in parallel and will be used to measure the most significant variables simultaneously (usually neutron fluxes and in-core channel flow rates).

The rod oscillator mechanism was installed and tested on a spare control rod drive. The control system for the mechanism was synthesized and necessary modifications were made to the mechanism. A demonstration of the mechanism in operation was made before the Vallecitos Atomic Laboratory Safeguards Group on August 31, 1961, and subsequent approval for use of the system was obtained.

3. Data Reduction and Analysis

The reduction and analysis of the data has been carefully considered. Three major schemes will be used:

- (a) Visual examinations of the data recorded on the direct-writing oscillographs will be made. An estimate of frequency and phase relationships will be determined.
- (b) The tape recorded signals will be fed into analog computers. Three such computers presently planned are (1) the frequency analyzer (band-pass filter); (2) the cross-correlating

analyzer; and (3) an analog computer programmed for various portions of the boiling water reactor mathematical model. From this data-reduction scheme, quantitative estimates of important frequencies in one signal or of the relationships between several variables can be made.

- (c) More accurate estimates of frequency and phase relationships between parameters can be obtained from analysis of the data on digital computers. First the signals are converted to digital form. Then auto-correlation, cross-correlation or statistical analyses are performed.

4. Analytical Program

In parallel with the stability test preparations which have occurred during the past three months, an analytical program has been conducted. The progress to date has included:

- (a) The development of an analytical model.
- (b) Plans for predicting the outcome of the stability tests in VBWR prior to performance of the tests.

As the test data become available, they will be compared with the analytical predictions. The range of applicability of the model will be explored and modifications of the model will be made as required.

D. Task A-VI - Irradiation in VBWR

This subtask provides for irradiation of the Fuel Cycle basic and special fuel assemblies in VBWR. Thirty-eight basic fuel assemblies (10 H-type, 15 I-type, and 13 J-type) were irradiated in VBWR run #134, July 15 to August 27, 1961. The previous run #133 had all of the above assemblies except for 4 J-type Zircaloy clad assemblies. Assembly 25-J was added in September for run #135. The VBWR core composition scheduled for start-up in October will contain 51 assemblies including all 24 of the Zircaloy clad, 23 of the stainless clad basic core assemblies (2-I, 7-I, 13-I will be taken out) plus four Special assemblies.

The irradiation status of both the basic fuel and the special fuel assemblies are combined in Table I.

Safeguards approval was given by the Vallecitos Safeguards Group for irradiation of the Zircaloy-4 defected and non-defected fuel rods in Assembly 25-J in VBWR. This assembly contains three Zircaloy-4 and thirteen Zircaloy-2 clad rods. Initial operation of 25-J was in VBWR run #135 for a scheduled heat flux of 203,000 Btu/hr-ft² and 187 kw power.

TABLE I

IRRADIATION STATUS OF FUEL CYCLE ASSEMBLIES

<u>VBWR Run</u>	<u>Dates</u>	<u>Assembly Designation</u>	<u>Assembly Power (KW)*</u>	<u>Peak Heat Flux (Btu/hr-ft²)*</u>	<u>Total Burnup MWD/T</u>
133 and 133A	6-27-61 through 7-16-61	1I	306	360,000	
		2I	247	285,000	
		3I	281	312,000	
		4I	286	321,000	
		5I	268	312,000	
		6I	188	206,000	
		7I	245	264,000	
		8I	285	320,000	
		9I	321	378,000	
		10I	241	294,000	
		11I	323	386,000	
		12I	255	294,000	
		13I	253	306,000	
		14I	353	400,000	
		16I	208	242,000	
		1J	197	218,000	
		2J	132	146,000	
		3J	139	153,000	
		4J	182	193,000	
		5J	204	208,000	
		6J	166	174,000	
		7J	135	144,000	
		8J	186	218,000	
		9J	210	242,000	
		1L	272	509,000	
134	7-16-61 through 7-31-61	All fuel assemblies listed above run at same conditions. Four new Fuel Cycle assemblies inserted in core operated as follows:			
		11J	340	360,000	
		12J	360	381,000	
		13J	370	392,000	
		14J	330	350,000	
135 and 135A	8-28-61 through 10-5-61	1H	194	208,000	1353
		2H	194	226,000	1341
		3H	242	260,000	1631
		4H	298	321,000	1994
		5H	263	288,000	1770
		6H	241	249,000	1668
		7H	224	236,000	1736
		8H	254	269,000	1720

*At a reactor power of 30 MW

4/7-12

Table I - Irradiation Status of Fuel Cycle Assemblies (Cont.)

<u>VBWR Run</u>	<u>Dates</u>	<u>Assembly Designation</u>	<u>Assembly Power (KW)*</u>	<u>Peak Heat Flux (Btu/hr-ft²)*</u>	<u>Total Burnup MWD/T</u>
135 and 135A	8-28-61 through 10-5-61	9H	316	338,000	2015
		10H	238	265,000	1575
		1I	322	353,000	2039
		2I		0	1494
		3I	288	311,000	1999
		4I	310	346,000	1991
		5I	272	298,000	1876
		6I	188	202,000	1504
		7I		0	1501
		8I	284	309,000	1936
		9I	209	244,000	1971
		10I	239	264,000	1712
		11I	180	198,000	2002
		12I	272	301,000	1855
		13I		0	1550
		14I	249	206,000	2147
		16I	219	244,000	1665
		1J	188	198,000	897
		2J	130	139,000	613
		3J	135	146,000	645
		4J	178	186,000	842
		5J	197	200,000	941
		6J	159	170,000	767
		7J	132	145,000	627
		8J	187	207,000	871
		9J	215	239,000	976
		10J	304	328,000	313
		11J	384	405,000	1290
		12J	357	383,000	1315
		13J	411	442,000	1397
		14J	345	396,000	1224
		15J	248	309,000	255
		16J	227	252,000	234
		17J		0	0
		18J		0	0
		19J		0	0
		20J	237	242,000	244
		21J		0	0
		22J		0	0
		23J		0	0
		24J		0	0
		25J	187	203,000	193
		1L		0	1283
		2L		0	0
		3L		0	0
		4L		0	0
		5L	215	268,000	268
		6L		0	0
		7L	175	186,000	209
		8L		0	0

VBWR Operating and Loading Schedules

Tables II and III present the VBWR operating and loading schedules for the remainder of 1961.

TABLE II

VBWR OPERATING SCHEDULE

October 1-October 5	Complete Run 135A
October 6-October 29	Shut down to prepare for stability tests and start ESADE installation.
October 30,31 & Nov.1	Startup, check out stability test instrumentation and run zero-power tests.
November 2-November 6	Shut down for VESR and VBWR containment vessel pressure tests.
November 7-December 7	Run Stability Tests
December 8-December 11	Shut down - prepare for Natural Circulation portion of stability tests.
December 12-December 18	Complete stability tests.
December 19-December 28	Shut down, close baffles, and rearrange fuel.
December 29	Run routine, maximum power operation.

TABLE III
VBWR LOADING SCHEDULE

	<u>Sept. 30</u>	<u>Oct. 31</u>	<u>Nov. 30</u>	<u>Dec. 31</u>
Consumers HPD ^(a)	24	24	24	30
AEC Fuel Cycle ^(b)	42	51 ^(c)	51	48 ^(d)
Du Pont	2	2	2	2
SADE	0	1	1	--
SAVANNAH I (or III)	3	3	3	3
SAVANNAH II	6	6	6	6
Euratom Pressure Build-up	0	1	1	1
HAPO	1	1	1	1
Dresden	15	10	10	10
Cellular Plates	2	0	0	0
Fuel Rod Cyclers	--	--	--	1
	<hr/>	<hr/>	<hr/>	<hr/>
Total Test Assemblies	95	99	99	102
APED Drivers	15	9	9	10
	<hr/>	<hr/>	<hr/>	<hr/>
TOTAL	110	108	108	112

- a. The number of consumers elements is currently limited to 24 because of reactivity considerations. This number must be increased as soon as possible to meet program commitments.
- b. Whenever possible, APED driver elements will be replaced by Fuel Cycle assemblies. Total Fuel Cycle elements are not limited to 48.
- c. Add 8L, 1L, 15I, 17J, 18J, 19J, 21J, 22J, and 23J. Make up "stability test" and "isotopic variation test" core areas.
- d. Add 2L, 3L, 4L (one defected rod), 6L. Remove 15I, 21J, and 22J. Remove 5I, 11I, 12I and 14I if necessary

E. Task A-VII - Special Fuel Program

The objective of this program is to determine performance capabilities of 12 assemblies consisting of fuel concepts which show potential increased performance and fuel cycle economy.

The special assemblies will be tested to determine their performance characteristics relative to the type 304 stainless steel clad "H" and "I" assemblies, and the Zircaloy-2 clad "J" assemblies which represent current fuel design and fabrication processes. Some of the special fuel assemblies comprise reactor performance tests of fuel concepts which have been developed by other Commission contractors. The special assemblies and their status are summarized in Table IV. The irradiation status of both the basic and special fuel is given in Table I.

1-L, Centermelt Calibration Assembly

After running at a peak flux of 417,000 Btu/hr-ft² (240 kw) in VBWR during run #132 in June 1961, assembly 1-L was moved into a higher power level position. During runs #133 and #133A, the assembly reached a nominal heat flux of 509,000 Btu/hr-ft² (272 kw). In this run, rod 4.3-2 reached the calculated target peak heat flux of 530,000 Btu/hr-ft². Rods 4.3-1 and 4.3-3 reached a peak of approximately 515,000 Btu/hr-ft² at 30 MW VBWR operation.

After fuel rod 4.3-2 in assembly 1-L reached the calculated target peak heat flux of 530,000 Btu/hr-ft², it was removed from the assembly gamma-scanned, and sectioned. Metallographic determination of the temperature and position of several points across the rod is in process. Chemical burnup analyses are being made and will be checked with the peak burnup physics calculation of approximately 1895 MWD/T.

Metallographic indications are that centermelting did not take place. This indicates that the UO₂ conductivity is appreciably higher than the value of 1.1 Btu/hr-ft² (°F/ft) which was used as a design basis.

This result is in agreement with independent tests at other sites. For example, work at Hanford* indicates an average thermal conductivity of UO₂ of about 2.3 Btu/hr-ft² (°F/ft) between 1000 and 5000 °F. The conductivity is actually a function of temperature which increases at temperatures above 2000 °F.

This higher value of conductivity results in a considerable improvement in the (calculated) fuel performance. Some of the implications are as follows:

1. The economic incentive for the use of conductivity promoters is greatly reduced.
2. Performance of UO₂ fuel of small diameter ($\frac{1}{2}$ " or less) is limited by burnout heat flux long before it is limited by central temperature.

*J. L. Bates, "Thermal Conductivity of UO₂ Improves at High Temperature," Nucleonics, Vol. 19, No. 6, June 1961, p. 83-87.

TABLE IV
SPECIAL FUEL PROGRAM

<u>Assembly Number</u>	<u>Type of Fuel Test</u>	<u>UO₂ Fabrication</u>	<u>Rod Fabrication</u>	<u>Status</u>
1-L	UO ₂ centermelt cali- bration	GE	GE	Calibration rod irradiated, removed from assembly, metallography and burnup analysis are in progress.
2-L 3-L	Centermelting assemblies	GE	GE	
4-L	Hot gas isostatic pressing of UO ₂ inside of stainless steel clad.	BMI	BMI	Four Battelle rods received 9/5/61.
5-L	Co-extrusion of UO ₂ inside of stainless steel clad.	NMI	NMI	Fabricated and in VBWR .
6-L	Zircaloy-4 clad	GE	GE	Tubing being processed for rod fabrication.
7-L	Zircaloy-2 clad co-extruded with type 304 stainless steel lining.	NMI	GE	Fabricated and in VBWR.
8-L	Thin stainless steel clad of .005" thickness	GE	GE	Fabricated.
9-L	UO ₂ with molybdenum filters	--	--	To be designed.
10-L	Metallic-coated UO ₂ particles	--	--	To be designed.
11-L	Extruded UO ₂	Allis Chalmers/ Electric-Autolite	GE	Pre-design complete.
12-L	Burnable poisons	--	--	To be designed.

3. Special assemblies 2-L and 3-L are approximately $\frac{1}{2}$ " diameter and cannot be center melted in VBWR because of burnout heat flux limits. Operation of these assemblies at the maximum heat flux permitted (now 540 Btu/hr-ft²) is however a valuable test of the life of such fuel and its gas release at higher temperature than the basic fuel. (Assemblies 2-L and 3-L have higher enrichment and larger diameter rods than the basic fuel.)
4. If UO₂ center melting is to be achieved in VBWR, it will be possible only with a large diameter rod. Such rods are not of economic interest because of the burnout heat flux limitation.

These implications will be explored further as more detailed results from assembly 1-L become available.

2-L and 3-L Centermelt Assemblies

Irradiation of assemblies 2-L and 3-L will be started after the 1-L calibration and complete analysis has been made.

4-L Isostatic Pressed

Fabrication of assembly 4-L (BMI hot gas isostatic pressed UO₂ in stainless clad) will be processed when assembly 6-L (Zircaloy-4 clad) is fabricated. The four Battelle fabricated rods were received September 5, 1961. Three of these rods will be placed in the original assembly. One will be held out for adding to the assembly later as a defected rod. The remaining rods will be conventional UO₂ pellets in stainless steel clad. (Because of the small number of rods involved, it may be desirable to combine the test rods from assemblies 4, 9, and 10-L into one 16-rod assembly.)

5-L and 7-L

Assembly 5-L and 7-L and 25-J were loaded in the VBWR run #135 in September.

The regular rods of assembly 5-L (UO₂ co-extruded in stainless steel clad) and 7-L (Zircaloy-2 co-extruded with thin stainless lining) were fabricated by Nuclear Metals, Inc. The majority of the work on these assemblies was completed in the previous quarter. During this quarter, the following work was done to complete these assemblies:

Fabrication of the spacer, top tieplate and extension handle modification were completed on 5-L. A reweld of the end plugs on 7-L eliminated porosity which existed in the first welding. Spacers using a constant spring pressure against the clad were fabricated during July. One of these spacers replaced the top tieplate in assembly 7-L. These assemblies were given end plug weld X-ray, leak test, and dimensional inspection before shipment to the VBWR in July.

6-L, Zircaloy-4 Clad

The Zircaloy-4 tubing for assembly 6-L has been received from Harvey Aluminum Company and the wall and length have been dimensionally inspected.

A Comptor gage will be used to record the maximum and minimum I.D. after which ultrasonic inspection and fabrication processing will be performed. Visual examination and ultrasonic inspection of several preliminary advanced samples looked good (somewhat better than an earlier lot of smaller diameter tubing processed on this program). A sample of tubing is being given a routine burst test, and chemical analysis including calcium. The 2.98% UO_2 pellets have been removed from the old Zr-Nb clad with about an 85% yield.

8-L, Thin Stainless Clad

Assembly 8-L (stainless steel clad of 5 mil thickness) was completed. Leak testing, final assembly, and pressure testing of two spare fuel rods to 1300 psi air at room temperature was completed satisfactorily. A plenum length of 7.0 inches is used to accommodate fission gas release since the thin clad prohibits high internal pressure.

Six fuel rods have an external plenum support with a wall thickness of .017" and a .413" O.D. which was welded to the .005" clad at the lower end and to the end plug at the upper end. After swaging the .017" thick external plenum support tube increased in thickness to .018". Six other rods of .005" thick clad do not have the external support but have a conventional internal plenum support of .016" thickness. The 5 mil clad was originally a 0.375" I.D. before swaging over 0.370" UO_2 pellet of 4.58% enrichment.

Four corner rods have .020" thick clad type 304 stainless steel with a nominal diametral gap from clad to pellet of .005". The UO_2 pellets in the corner rods are 0.370" and of 3.9% enrichment. Fuel lengths are 37" $\pm$ 1/8".

Other Special Fuel Assemblies

The four remaining special assemblies are presently planned as:

1. 9-L, UO_2 fuel with molybdenum fibers. Two mil molybdenum fibers have been procured. Test pellet pressing will be done and evaluated in the next month. (Test rods may be combined with 4-L).
2. 10-L, metallic-coated UO_2 particles. Quotations are out to four vendors for cost and delivery of coated UO_2 particles (coated with either niobium or molybdenum). (Test rods may be combined with 4-L).
3. 11-L, extruded UO_2 pellets. 0.485 $\pm$.002" O.D. UO_2 extrusions of approximately 6" lengths will be supplied by Allis-Chalmers/Electric Autolite. This fuel will be loaded into stainless steel type 304 tubing of 0.495 $\pm$.002" I.D. x 0.022 $\pm$.002" wall.
4. 12-L, burnable poison. Planning and design of this assembly have been started. Under consideration are: (a) Use of boron stainless cladding, and (b) particles of zirconium diboride coated with BeO incorporated into UO_2 .

IV. TASK B - HEAT TRANSFER AND FLUID DYNAMICS

A. Task B-I - Burnout Heat Transfer

The objective of this subtask is to develop precise burnout correlations in support of in-pile experiments. The work is aimed at extending the present range of burnout data by investigating the effects of flow rate, fluid properties and geometry. A parallel effort is being made to understand the phenomena affecting the burnout heat flux, prepare theoretical predictions of burnout, and investigate means of raising the burnout heat flux.

1. Single Rod Burnout Work

The previous experimental work consisted of varying mass flow rate, velocity profile, heat flux axial profile, hydraulic diameter, heated length and pressure. The work was done in the single-rod test section described in GEAP 3558, First Quarterly Progress Report, with an electrically heated stainless steel rod.

This work has been extended with efforts to raise the burnout heat flux. Figure 2 shows a sketch of a portion of the single rod test section fitted with a rough liner. The liner was roughened by fitting it with .709 inch I.D. x .870 inch O.D. x .080 inch thick washers in its interior. The distance between the washers was typically one inch and the annular flow path was .167 inch between the O.D. of the heated rod and the I.D. of the washer.

Figures 3a and b show the increase in burnout heat flux that occurred as a result of the roughened liner for two mass flow rates (1.12×10^6 and 1.68×10^6 lb/hr ft²). The effects of washer spacing and annular clearance will be investigated next.

2. High Pressure Observational Boiling Experiment

In addition to the single rod burnout tests, a complementary effort is being carried out to enlarge fundamental understanding of burnout in forced convection with bulk boiling.

The experimental phase of this investigation was described in GEAP 3709, Third Quarterly Progress Report. It was comprised of critical heat flux experiments at 1000 psi in the observational test section in conjunction with high speed motion pictures of the two-phase flow patterns at the inception of burnout as well as at lesser fluid enthalpies and heat fluxes. The critical heat flux data obtained is described in GEAP 3781, Fourth Quarterly Progress Report.

Work during this quarter was on the theoretical phase of the investigation. In addition, final titling and editing of the high speed data films were completed, and the film negatives are being prepared for reproduction of copies to be released later to the AEC as a supplement to the final report of the project. The final selection of data films to be released are described in Table V. Discard of the other films taken during the experiment is on the basis of photographic quality and duplication (for

TABLE V
HIGH PRESSURE OBSERVATIONAL BOILING EXPERIMENT

SELECTED HIGH SPEED DATA FILMS**

Reel No.	Film No.	Heat Flux Btu/hr-ft ²	Steam Quality	Enthalpy of Sub- Cooling Btu/lb.	Fraction of "Critical" Heat Flux	Heater Element Temp. *	Mass Velocity lbs/sec-ft ²	Pressure psia
I	2-1	465,000	0.258	--	--	Steady	50.1	1000
	1-1	343,000	0.318	--	--	Steady	50.1	1000
	2-2	616,000	0.450	--	--	Steady	50.1	1000
	2-3	698,000	0.556	--	100%	Pulse (trip)	50.1	1000
	1-2	588,000	0.656	--	100%	Osc. (4mm)	50.1	1035
II	3S-1	245,000	--	91.4	--	Steady	102	940
	3S-2	492,000	0.056	--	--	Steady	99.4	1015
	3S-3	491,000	0.097	--	--	Steady	99.4	1020
	3S-4	491,000	0.166	--	--	Steady	99.4	1005
	3R-1	465,000	0.223	--	--	Steady	99.4	1000
	4-5-a	876,000	0.303	--	82%	Osc. (Temp. rise in lower sect.)	101	1005
	3R-2	837,000	0.454	--	92%	Pulsing (1mm)	99.4	1000
	3R-3	857,000	0.465	--	96%	Osc. (2mm)	99.4	1000
III	6S-1	298,000	--	2.8	--	Steady	199	1000
	6R-1	612,000	0.167	--	--	Steady	199	1000
	6R-5	903,000	0.252	--	100%	Osc. (1mm) & Pulse (trip)	196	1000
	7-1-b	796,000	0.104	--	--	Steady	202	995
	7-2	911,000	0.151	--	91%	Pulsing (1mm)	199	1000
	7-5	957,000	0.160	--	100%	Osc. (4mm)	199	1000
	8-1	989,000	0.084	--	--	Steady	199	1035
	8-3	1,080,000	0.105	--	100%	Pulsing (1mm)	199	1000

*(mm) refers to peak amplitude of safety monitor recorder.

** 9/16 in. high view of critical heat flux zone. Fastex camera at 4500 frames per second.

Table V - High Pressure Observational Boiling Experiment - Selected High Speed Data Films** (Cont.)

Reel No.	Film No.	Heat Flux Btu/hr-ft ²	Steam Quality	Enthalpy of Subcooling Btu/lb	Fraction of "Critical" Heat Flux	Heater Element Temp.*	Mass Velocity lbs/sec-ft ²	Pressure psia
IV	9S-1	245,000	--	61.3	--	Steady	398	995
	9S-2	493,000	--	17.6	--	Steady	398	995
	9R-1	857,000	0.096	--	98%	Pulsing (1mm)	398	1005
	9R-2	884,000	0.093	--	100%	Osc. (4 mm)	395	1000
	9A-1	987,000	0.074	--	100%	Osc. (2mm) & Pulse (trip)	398	1000
	9B-2	1,141,000	0.037	--	100%	Osc. (1/2 mm) & Pulse (trip)	398	1005
V	11S-1	489,000	--	112.3	--	Steady	99.4	985
	11S-2	612,000	--	72.4	--	Steady	99.4	1000
	12S-1	489,000	--	170.0	--	Steady	199	1000
	12S-2	612,000	--	151.2	--	Steady	199	1000
	12-2	844,000	0.113	--	80%	Pulsing (1mm)	199	1013
	18S-1	367,000	0.234	--	--	Steady	99.2	1005

photographic purposes) of test conditions.

The theoretical analysis has proceeded to relate analytically the critical heat flux to: local steam quality, fluid properties, channel geometry, and flow mass velocity. This work is based on the qualitative evidence of the two-phase flow patterns shown in the high speed data films. Initial results have provided a tentative theoretical relationship between these several variables, involving empirical determination of three constants, which appears to correlate the critical heat flux in a compact manner rather well.

The analysis will be refined further during the next quarter and will be applied to correlating the single rod test section data (involving a different geometry and a pressure range from 600 to 1400 psia).

B. Task B-IV - Two-Phase Pressure Drop

An analysis of two-phase flow was made using single-phase turbulent mixing length methods. Two-phase density, velocity distributions, and pressure drops were derived by treating the two-phase system as a continuous medium where the turbulent exchanges of momentum and density are equal. The results parallel those for single-phase flow and are being reported in a topical report GEAP 3628, Revision I, "Prediction of Two-Phase Pressure Drop and Density Distribution From Mixing Length Theory," by S. Levy, May 31, 1961. The following is a summary of the results:

1. The density and velocity distributions are approximately of the power type and their shape varies with total liquid Reynolds number and mean fluid density. The calculated profiles are much more curved than in single-phase flow and agree with the density distributions measured by Petrick.
2. The two-phase pressure drop in horizontal flow is a function of total liquid Reynolds number and mean fluid density. In vertical flow, it also depends upon the equivalent total liquid velocity head.
3. The ratio of two-phase pressure drop to total liquid pressure drop decreases as the total liquid Reynolds number increases. Also, it varies only slightly as the channel geometry changes from horizontal to vertical or from pipe to parallel plates. Available low and high pressure data in horizontal and vertical channels confirm the predicted trends. Good agreement is obtained between test and analytical two-phase pressure losses especially at high mean fluid density and Reynolds number.
4. The mean fluid density is a function of total liquid Reynolds number and homogeneous fluid density. It also depends slightly upon channel geometry and upon the equivalent total liquid velocity head in vertical flow. The mean fluid density decreases as the total liquid Reynolds number increases and falls below experimental data unless local slip between liquid and gas is permitted at every position in the flow channel. An analytical prediction based upon local slip and experimental data at very low gas content is presented. It gives good agreement with

available test results.

5. The proposed model reduces a two-phase problem to an equivalent one in single-phase flow. As such, it can be extended to predict the effects of film boiling, entry length, and heat transfer in two-phase flow. Typical values are given for film boiling conditions.

V. TASK C - EXPERIMENTAL PHYSICS

This task provides for the study of long-term reactivity characteristics of fuels irradiated in the VBWR. Isotopic analyses of uranium depletion and plutonium production will be experimentally performed and compared to theoretical predictions.

Two fuel elements of 2.7% enrichment, designated J-17 and J-18, will be used for this experiment. Both elements contain five 0.313" diameter stainless steel guide tubes which will allow flux wire measurements to be made in five locations within the element. Wires for the measurements are inserted in 0.25" diameter stainless steel thimbles, which are positioned in the stationary guide tubes. This arrangement allows insertion and withdrawal of the flux wires during reactor operation.

Fabrication of these guide tubes and thimbles is complete. These two elements will be inserted in the VBWR in positions F-5 and G-5 during the scheduled shutdown in early October. All adjacent fuel locations will contain similar Fuel Cycle Program fuel elements to provide as homogeneous a region as possible.

After a few hundred MWD/T exposure, two rods will be removed and a conversion ratio determined. The plan for rod replacement is as follows:

- A. Remove elements from positions F-5 and G-5.
- B. Remove rods for analysis from element irradiated in position G-5.
- C. Replace with rods irradiated in similar flux, from element irradiated in position F-5.
- D. Replace these rods with clean rods fabricated from same fuel lot as that in the two test elements.
- E. Replace elements in reactor in the same orientation as during previous irradiation.

This plan allows, as nearly as possible, each rod to be irradiated in a constant flux with a constant gradient, and eliminates many uncertainties in data analysis. Assembly J-17 in position F-6 is used to supply rods of uniform irradiation to replace rods removed from assembly J-18.

It is planned to remove the second pair of rods at approximately 2000 MWD/T exposure at the expected power density of 25 - 30 KW/KgU; this exposure will be obtained in about three months.

After the fuel rods are removed from the fuel assemblies, in the VBWR pool, they will be transferred to RML where examination will include:

- A. Visual examination for identification.
- B. Gamma activity survey.
- C. Sectioning rods at several axial positions for pellet sampling.
- D. Dissolution of a sample from each of above locations.

E. Obtaining several samples along one pellet radius and preparation for chemical analysis.

The chemical analysis of each sample will then include measurement of:

A. Pu/U ratio.

B. U-234, U-235, U-236, U-238, Pu-239, Pu-240, Pu-241 and Pu-242 and Pu-238 by alpha spectrometry.

C. Burnup by Cs-137/U, plus one alternate FP/U as developed or available.

Ten sample pellets from the fuel lot used to fabricate the two test elements have been isotopically analyzed. Results of this analysis are:

Pellet #	Atom Percent		
	U-234	U-235	U-236
1	.0175	2.802	.0209
2.	.0165	2.805	.0212
3	.0168	2.803	.0203
4	.0180	2.796	.0220
5	.0160	2.802	.0201
6	.0170	2.802	.0210
7	.0170	2.803	.0208
8	.0159	2.802	.0205
9	.0162	2.802	.0209
10	.0161	2.795	.0208
Average	.0167 $\pm$.0005	2.801 $\pm$.002	.0209 $\pm$.0004

It is encouraging that there is essentially no difference in these ten pellets and that the precision of measurement is very high. This analysis also indicates that the U-235 alone gives a rather sensitive MWD/T indicator for several thousand MWD/T.

The isotopic atom percent obtained from the above analysis has been used to obtain a set of three-group cross sections for the test elements. These cross sections will be used to compute axial and radial power distribution. The axial power distribution will also be obtained from the wire measurements at as many as five locations within the element. Power densities obtained will then be used to compute isotopic concentrations as a function of exposure time.

Considerable effort has been spent in trying to obtain a reliable means of predicting theoretically these isotopic concentrations. Of primary concern is an accurate prediction of the isotopic concentration for short exposure time, i.e., a few hundred MWD/T, since pellets will be removed from the element for analysis at a few hundred MWD/T. Several available burnup codes are being investigated. Once a code has proven to be acceptable, variations in cross sections will be studied to determine changes in concentration to be expected as a function of the cross sections. This is a first step toward using the experimentally determined concentrations to calculate the cross sections.

FIGURE 1
 SCHEMATIC DIAGRAM INSTRUMENTATION SYSTEM FUEL CYCLE STABILITY EXPERIMENT

10-1-1

762D226

GENERAL ELECTRIC

762D226

145A5481	✓	2	762D226	FUEL BUNDLE
APPROVED FOR ISSUE		DATE	DESIGNED BY	CHECKED BY
145A5481		10/1/54	W. J. BROWN	J. E. BROWN
145A5481		10/1/54	W. J. BROWN	J. E. BROWN

SECTION AA

± 0.010
76.0 TYPICAL SPACING

SECTION BB

SECTION CC

NOTE 1 BEND TUBES (PTS 7 & 9 10 & 11) TO POSITIONS AS SHOWN WITH MINIMUM BEND

2 TIE SPACERS (PTS 14 & 15) IN PLACE BY TWISTING WIRE (PT 18) APPROX CENTER ALL FOUR SIDES

3 PLACE PINS (PT 25) IN TUBES (PTS 7 & 9 10 & 11) SO THAT THEY DO NOT PROTRUDE BEYOND O.D. OF TUBES AND STAKE IN

PL ISSUED

1/16 DIA. DRILL (4 HOLES)
AND STAKE IN PINS P25

SEE NOTE 2

SEE NOTE 3

DESCRIPTION OF GROUPS	REVISIONS	REMARKS
	1. 100% AS BUILT	
	2. 100% AS BUILT	
	3. 100% AS BUILT	

762D226

FIGURE 3
SINGLE-ROD TEST SECTION
WITH ROUGH LINER

FIGURE 4A & B
INCREASE IN BURNOUT HEAT FLUX DUE TO ROUGHENED LINES