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INFORMAL AEC RESEARCH
AND DEVELOPMENT REPORT
JULY 10, 1962

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MASTER

FUEL CYCLE PROGRAM

**A BOILING WATER REACTOR
RESEARCH AND DEVELOPMENT PROGRAM**

EIGHTH QUARTERLY PROGRESS REPORT

APRIL 1962 – JUNE 1962

COMPILED By
J.A. HODDE

U.S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-189
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ATOMIC POWER EQUIPMENT DEPARTMENT

GENERAL ELECTRIC

SAN JOSE, CALIFORNIA

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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, steam void observational studies, and two-phase pressure drop to support in-core work.

Task C

1. Study long-term reactivity and isotopic composition 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 - September 1960.
3. GEAP-3627, Second Quarterly Progress Report, October-December 1960.
4. GEAP-3628, Prediction of Two-Phase Flow From Mixing Length Theory, S. Levy, December 27, 1960. Revision I, May 31, 1961.
5. GEAP-3709, Third Quarterly Progress Report, January - 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 - June 1961.
8. GEAP-3794, Plan for VBWR Stability Experiment, W. H. Cook, et.al., August 30, 1961

I. INTRODUCTION (continued)

9. GEAP-3835, Fifth Quarterly Progress Report, July - September 1961.
10. GEAP-3898, Sixth Quarterly Progress Report, October - December 1961.
11. GEAP-3953, Seventh Quarterly Progress Report, January - March 1961.
12. GEAP-3766, Critical Heat Flux & Flow Pattern Characteristics of High Pressure Boiling Water in Forced Convection, F. E. Tippetts, April 1962.
13. GEAP-3961, Prediction of The Critical Heat Flux in Forced Convection Flow, S. Levy, June 20, 1962.

II. SUMMARY

Task A - Advanced Fuel Power-Limit Tests

1. Frequency analysis of the VBWR stability test data has resulted in the following observations:
 - (a) All flux noise readings examined have a strong peak at a frequency of 0.85 (± 0.15 / ± 0.10) cycles per second. The frequency of this peak did not appear to change when flow rate was changed by a factor of 7.
 - (b) The flow noise in an individual fuel channel varied from 0.45 / 0.10 cycles per second at low flow to 0.8 / 0.1 cycles per second at high flow. Intermediate flows have not been analyzed.
 - (c) Recirculation loop flow rates vary at a frequency of 0.7 to 1.0 cycles per second.
 - (d) All frequency spectra examined for neutron flux, core flow and loop flow have similarities in shape that suggest that they are related.
2. Interpretation of the steady-state noise observed during the VBWR stability tests has involved consideration of several possible sources of excitation. One of these is a theory of water surface waves which can be summarized as follows:
 - (a) The upward flow of water and steam in the center of the reactor vessel excites standing waves on the water surface.
 - (b) The waves constitute a frequency-regulating mechanism that depends only on the vessel geometry and the modes of oscillation that are being excited. Several modes and frequencies can occur simultaneously.
 - (c) The waves cause variations in reactor flow and pressure and the flow and pressure variations drive the waves. This is a regenerative feedback loop that could produce the flux noise.
 - (d) Calculated frequencies for the fundamental wave modes are in the range measured experimentally for flux noise and flow oscillations.
 - (e) Experiments in an air-water loop at atmospheric pressure indicate that waves occur at amplitudes that can be greater than vessel diameter. Amplitude is an increasing function of air and water flow rates. The waves have the frequencies and types of pressure variation predicted theoretically.
3. Visual and destructive examination of selected fuel rods resulted in the following major observations:
 - (a) Fretting corrosion of the zircaloy clad fuel rods against the stainless steel spacers has been observed. This appears to be associated

with opening up of rod-to-spacer clearance by deformation of the spacer wires during fuel handling, thus permitting vibration of the rod against the spacer. Spring clips to provide more firm rod support are to be tested. The stainless-clad rods did not exhibit fretting.

- (b) Further analysis of the centermelt calibration rod provides evidence that UO_2 thermal conductivity increases with time. As irradiation proceeds, growth of larger, columnar grains improves thermal conductivity, thus gradually decreasing the diameter of the molten region and solidifying a ring of high density, porosity-free UO_2 . This ring is distinguishable from the center region that solidifies quickly at reactor shutdown and has large spherical and tear-drop voids.
- (c) Hot gas isostatic pressed fuel rods failed after 316 MWD/T irradiation and five months storage under water. All three rods broke in normal handling, apparently the result of intergranular corrosion of the stainless steel clad. Analysis is continuing.

4. Irradiation status on June 10, 1962 was as follows:

<u>Fuel Type</u>	<u>Number Irradiated</u>	<u>Average Exposure of Lead Assembly, MWD/T</u>
Stainless clad, H	10	4436
Stainless clad, I	16	4050
Zircaloy clad, J	24	4222
Special, L	5	3842

- 5. The following fuel assemblies have been fabricated and are ready for VBWR irradiation:
 - (a) Fuel rods (assembly 9L) having molybdenum fibers or UO_2 particle coating as a conductivity promoter.
 - (b) Fuel rods (assembly 11L) made from UO_2 extrusions from CICA and Allis Chalmers.
 - (c) Fuel rods (assembly 4L) loaded with low temperature sintered pellets from United Nuclear Corporation.

Task B - Heat Transfer & Fluid Dynamics

- 1. A superposition model for predicting critical heat flux was developed and reported in a topical report.
- 2. Shakedown operation of a seven-rod test section for measuring burnout heat flux was conducted. Modifications to the heater tube internal support and spacers were made.

Task C - Experimental Physics

1. Comparison of measured and calculated isotopic composition for uranium and plutonium isotopes after 400 MWD/T irradiation indicates that the calculations underestimate capture in uranium-238 and overestimate capture in plutonium-239. Chemical analyses for 900 and 1800 MWD/T are in progress.
2. Flux wire irradiations have provided axial and radial flux profiles and the ratio of thermal to fast flux as a function of axial position.

III. TASK A - ADVANCED FUEL POWER-LIMIT TESTS

A. Stability Tests

The work of this task includes the development of a mathematical model to predict the stability of boiling water reactors and the performance of stability tests to provide an experimental basis for the model. A hydraulic stability loop has been constructed and is undergoing trial operation prior to initiation of the experimental program. Stability tests were performed in VBWR during November and December 1961. The scope of these tests was reported in GEAP-3794, Plan For VBWR Stability Experiment.

One aspect of the test is the analysis of the reactor "noise" during steady state operation. The "noise" is the spontaneous variation in neutron flux, flow rates, and pressures. The noise analysis was conducted by passing tape recorded reactor data through a band pass filter noise analyzer, which has been described previously. The result is a graph of amplitude versus frequency. A total of 34 different steady-state frequency analyses have been made. Some observations and possible interpretations of this investigation are presented.

Observations which have been made are as follows::

1. The peak frequency of the flux noise is approximately independent of the reactor operating conditions. This is shown by the following tabulation of results from the band pass filter analysis of signals from an external ion chamber (all at 15 Mwt).

Run	Flow, gpm			Pump Head, ft	Peak Noise Frequency, cps
	Pump 1	Pump 2	Total		
15-1	10,000	10,000	20,000	200	0.85
15-9	4,000	0	4,000	290	.85
15-19	0	4,400	4,400	7½	.95
15-16	4,000	0	4,000	290 Baffles open	.75
15-3	10,000	0	10,000	200	1.00
15-11	2,600	0	2,600	300	0.80
15-18	Natural Circulation		4,300	--	<u>0.75</u>

Average

0.85 ± .10
/ .15

Considering the fact that the accuracy of determination of the peak frequency is in the order of ± 0.10 , it can be inferred that the peak frequency is practically independent of the parameters listed in the above table. This suggests that some factor not previously considered is having a major effect in determining the peak frequency.

2. Loop flow rates vary at a peak frequency of 0.7 to 1.0 cps for all operating conditions (except 2,600 gpm which was 5 cps).

3. In-core flow rates vary at a peak frequency of 0.4 to 0.8 cps for all operating conditions.
4. The amplitude versus frequency curves for neutron flux, core flow and loop flow have certain similarities in shape.
5. The transfer function for the VBWR is observed to have a peak at a frequency of approximately 1.2 to 1.5 cycles per second, as determined from the rod oscillator tests. The system noise is a function of both the excitation and the reactor system transfer function. Since the system response has a peak frequency different from the peak of the transfer function, then a strong peak frequency must exist in the excitation. Moreover, the peak frequency in the excitation is relatively independent of the operating point of the reactor.

An effort was made to find a mechanism which could cause the loop flow, core flow, and flux to vary at a frequency of approximately 0.8 cps. This must be a mechanism that gives the same frequency even though the flow rate is varied by a factor of 7. The standing surface wave hypothesis was formulated as a possible explanation of the observed behavior.

1. Statement of Surface Wave Hypothesis for Reactor Flux Noise

This surface wave hypothesis is stated specifically for the VBWR but could be applied to other reactors having free-surface steam separation. It is thought that it could be modified to fit reactors having different water surface geometries.

- (1) The upward flow of water and steam in the center of the vessel excites standing waves on the surface.
- (2) The possible modes of oscillation of the surface are governed by mathematical solutions to the wave equation for the particular geometry in question. The amplitude is governed only by the strength of the driving force.
- (3) These waves are superimposed on the swell, bubble breaking, foaming, splashing, and flow phenomena which may also be present. These effects may be related by feedback mechanisms.
- (4) Any continuous surface configuration may be considered as a summation of an infinite series of solutions to the wave equation. This is a generalized Fourier series and involves Bessel functions for a cylindrical vessel.
- (5) The wave modes constitute a timing or frequency — regulating mechanism that depends on only vessel geometry and the modes of oscillation that are being excited.
- (6) Combinations of different modes of oscillation may give the appearance of continuous frequency distributions when observed with methods having low resolving power for frequency differences.

(7) The waves have a controlling effect on reactor outlet flow if they result in periodic uncovering of the reactor outlet nozzles. (Severe low slugging has been observed from this cause in model tests of the Dresden and Consumers Big Rock Point reactors when operated without baffles.)

(8) Standing surface waves cause periodic pressure variations in the fluid. These are of two types:

(a) First order effect - pressure variations are largest at surface and decrease exponentially with depth in the fluid.

(b) Second order effect - pressure variations occur at an amplitude which is not attenuated with depth. This effect predominates at depths (below surface) greater than vessel radius. The frequency of this type of pressure variation is twice the wave frequency.

(9) The pressure variations can be transmitted in two directions:

(a) Out through the recirculation loop where the pressure variation is felt as a change in saturation temperature and a change in cavitation in pump, valves, and elbows. This causes flow variations. The flow and saturation temperature changes are transmitted to the core.

(b) Down through the core where the pressure change is felt primarily as a change in saturation temperature. The core responds as though the inlet subcooling were variable.

(10) The standing waves influence the two-phase flow in and above the core in a way such that the wave is excited by the resulting flow perturbation. That is, the waves and the two-phase flow constitute a regenerative feedback loop.

2. Application to VBWR

This wave hypothesis can be applied to VBWR as follows:

(1) Since the vessel is cylindrical the wave modes expected are:

An annular ring
Side to side sloshing
Higher harmonic modes of each of these

(2) The calculated frequencies of these waves (for the VBWR 7 foot diameter vessel) are as follows:

n	Annular Rings		Side-to-Side	
	Wave	Second Order Pressure Change	Wave	Second Order Pressure Change
1 (fundamental)	0.945 cps	1.89 cps	0.625 cps	1.25 cps
2	1.27	2.54	1.11	2.22
3	1.54	3.08	1.41	2.82

(3) The modes expected are the fundamental ($n = 1$) mode for the side-to-side type with smaller amplitudes for the other modes. These modes are thought to exist concurrently and the relative amplitudes of each vary in a random manner.

(4) The actual frequencies of the waves, and the pressure variations, are reduced below the calculated values by two effects:

- (a) The presence of a gas phase
- (b) Obstructions in the vessel such as control rod shafts, and plenum support members.

It is concluded that the peak frequency of the noise observed in the reactor of 0.85 cps is comparable to the wave hypothesis prediction of 1.25 cps for the second order pressure change resulting from side-to-side sloshing. The calculated frequency would be reduced by taking into account the two-phase nature of the fluid. This does not constitute proof that waves cause or control the flux noise. Further work is needed to do this.

The next three sections give further theoretical and experimental background on waves.

3. Wave Equations for a Cylindrical Vessel

The wave equations have been solved for vessels of several shapes by H. Lamb in the book "Hydrodynamics", and the following analysis is based on this source. The fluid is assumed to be non-viscous, vertical acceleration is taken into account, and the vessel is assumed to have no obstructions. Obstructions below the water level by over one-half wave length (no more than vessel radius) have no effect.

The wave equation for cylindrical coordinates is

$$\frac{\partial^2 \phi}{\partial r^2} + \frac{1}{r} \frac{\partial \phi}{\partial r} + \frac{1}{r^2} \frac{\partial^2 \phi}{\partial \theta^2} + k^2 \phi = 0$$

The boundary conditions are (1) zero vertical motion of fluid at the bottom of the tank (at depth $z = -h$) and (2) zero radial movement of fluid at the tank wall. i.e. $\frac{\partial \phi}{\partial r} = 0$ at wall.

The frequency of the oscillations is given as

$$\sigma = \sqrt{gk \tanh(kh)}$$

or if kh is "small"

$$\sigma = k\sqrt{gh}$$

If kh is "large"

$$\sigma = \sqrt{gk}$$

Where symbols have meaning listed below.

Solutions to the wave equation for a cylindrical vessel have the form

$$\phi = A_s J_s(kr) \cos(s\theta) \cos(\sigma t + \epsilon)$$

- where ϕ = height of surface above its rest position.
 A_s = arbitrary constant which depends on driving force
 s = 0, 1, 2, 3,
 r = distance along vessel radius
 θ = angular position (arbitrary reference)
 σ = wave frequency, radians per second
 t = time, seconds
 k = function determined by boundary condition

$$\frac{\partial \phi}{\partial r} = 0 \quad @ \quad r = R$$

(1) For $s = 0$ the waves are symmetrical about the origin and are annular rings. A cross section through the waves has the shape of Bessel function $J_0(x)$. Application of the boundary condition gives:

$$\phi = A_0 J_0(kr) \cos(\sigma t + \epsilon)$$

$$kR = \alpha_n \quad \text{OR} \quad k = \alpha_n / R \quad \text{AND} \quad \alpha_n = \text{ROOTS OF } J'_0(kR) = 0$$

The frequency is given as

$$f_n = \frac{1}{2\pi} \sqrt{\frac{g\alpha_n}{R} \tanh \frac{h\alpha_n}{R}}$$

- where h = height of water, ft.
 R = radius of tank, ft.
 g = 32.2 = gravitational acceleration
 α_n = zeros of $J_1(x)$ = 3.832, 7.016, 10.173, etc.
 f_n = frequency in cycles/sec for n^{th} harmonic

This is simplified for two special cases

(a) "Shallow" vessel

If $\frac{h\alpha_n}{R}$ is small (< 0.35)

then $\tanh\left(\frac{h\alpha_n}{R}\right) \approx \frac{h\alpha_n}{R}$ and

formula becomes for "shallow" vessel

$$f_n = \frac{\alpha_n \sqrt{gh}}{2\pi R}$$

(b) "Deep" vessel

If $\frac{h\alpha_n}{R}$ is large (> 1.8)

then $\tanh\left(\frac{h\alpha_n}{R}\right) \approx 1.0$ and $f_n = \frac{1}{2\pi} \sqrt{\frac{g\alpha_n}{R}}$

(2) For $S = 1$, the fundamental wave is a side-to-side sloshing with the surface remaining relatively plane. Mathematically this is composed of two waves rotating around the vessel in opposite directions.

$$\phi = A_1 J_1(kr) \cos(\theta) \cos(\sigma t + \epsilon)$$

$$kR = \beta_n \quad \text{OR} \quad k = \beta_n/R$$

The frequency of the waves becomes

$$f_n = \frac{1}{2\pi} \sqrt{\frac{g\beta_n}{R} \tanh\left(\frac{h\beta_n}{R}\right)}$$

where $\beta_n =$ zeros of $J_1'(x)$

OR $\beta_n = 1.841, 5.332, 8.536, \text{ etc.}$

The special cases become

(a) "Shallow" vessel

$$f_n = \frac{\beta_n \sqrt{gh}}{2\pi R}$$

(b) "Deep" vessel

$$f_n = \frac{1}{2\pi} \sqrt{\frac{g\beta_n}{R}}$$

where "shallow" and "deep" have some meaning as above.

(3) For $S = 2, 3, 4$ etc. The frequencies can be obtained in a similar manner. The surface has S nodal diameters. These modes are considered of secondary importance to reactors.

4. Pressure Variations Resulting from Standing Waves

The importance of standing waves in a reactor vessel is thought to stem from the resulting pressure variations in the fluid. The equations for the pressure variations of the two types are as follows* (for one-dimensional waves in x-direction):

*R.I.B. Cooper and M. S. Longuet-Higgins, "An Experimental Study in Standing Water Waves", Royal Society Proceedings, Series A, May 7, 1961.

(1) First-order effect

$$P = 2\rho g a e^{-kz} \cos(kx) \cos(\sigma t)$$

(2) Second-order effect

$$(P_2)_\infty = 2\rho a^2 \sigma^2 \cos(2\sigma t)$$

where p = pressure variation in fluid

ρ = fluid density

g = acceleration of gravity

a = amplitude of surface displacement, zero to peak

σ = wave frequency, radians/sec.

x = horizontal distance

z = depth below undisturbed surface

$k = \frac{\sigma^2}{g}$ for "deep" vessel

It should be noted that the first order effect decreases exponentially with depth and becomes less than the second order effect at depths greater than one-half wave length (vessel radius or less). This first order effect might influence the pressure and flow in the reactor outlet lines, if they are close to the free surface. This is in addition to the severe flow slugging that occurs if the liquid level periodically drops below (or near to) the top of the outlet nozzles.

The second order effect predominates at the greater depths. If the wave amplitude is large the amplitude of the pressure variation is very significant since it depends on wave amplitude squared. The second order pressure variation is independent of both x and z and represents a pressure applied uniformly to the entire fluid. It is of twice the frequency of the surface wave.

The second order effect makes itself felt as a pressure variation on the entire core. This is thought to act primarily through the resulting change in saturation temperature and hence, boiling boundary and average core void fraction. Unfortunately, the wave frequencies encountered (about 1 cycle/second) are at the frequency for which the system has an appreciable response to saturation temperature changes. That is, the peak of the transfer function for saturation temperature change to reactor power occurs at about 1 cycle per second, as shown by pre-test analysis.

The presence of pressure variations of the two types was verified experimentally by Cooper and Longnet-Higgins in the paper referenced above.

5. Experimental Verification

Several experiments have been conducted to give at least a qualitative confirmation of the feasibility of surface waves and to gain a feel for their behavior.

First, the side-to-side mode of oscillation was verified by imparting a lateral motion to a cup of coffee. It oscillated with a plane surface, this being the fundamental mode. This mode has the lowest frequency of the various standing wave modes.

Next, a 10" pot of water was boiled on an electric burner. A 5" diameter riser was used. The presence of the fundamental annular wave was verified. The amplitude of this wave increased as the heat input was increased. At the maximum heat input the amplitude was approximately 1/8" at the edge and 3/8" (including swell) in the center. (Bessel function $J_0(x)$ has an amplitude at the first annular wave crest of 40% of the peak value at the center). Even in this crude test, the wave effect was clearly evident.

A 13" diameter circular dishpan was the vessel for the next experiment. The waves were excited by bubbling air (from plant air supply) in at the bottom center of the pan. Many modes of oscillation were excited by this method. As the air flow was increased the following changes were observed:

- (1) At very low air flows (and a small diameter column of bubbles rising in the center) the mode of annular waves excited was high ($n = 8$). Amplitudes were very small and were hardly discernable at the outer edge of vessel.
- (2) At intermediate air flow (and middle sized bubble column) a clear $n = 4$ annular mode was evident. Amplitude was approximately 1/2 inch at outside edge of vessel.
- (3) At high air flow (and large bubble column) the oscillations changed to a shifting combination of the following modes of oscillation:
 - (a) Fundamental annular mode ($n = 1$).
 - (b) Higher harmonics of annular mode ($n = 2, 3$).
 - (c) A side-to-side sloshing which gradually rotated in axis of symmetry.
 - (d) A second side-to-side mode with an axis of symmetry rotated 90° from the first one. This could be considered the $J_2(x)$ mode with $n = 1$.

The relative amplitudes of the different modes changed in an apparently random manner, so that at one moment one mode was dominant and later another mode (or combination of modes) was dominant. The combined amplitude was approximately 2" at edge of vessel. Bubble column was less than 3 inches in diameter.

The above experiment suggested that the mode of the oscillation depended strongly on the ratio of the diameter of the bubble column to the vessel diameter. It would seem that the modes excited would be ones having a diameter of the center crest about equal to (or greater than) the diameter of the bubble column. The waves tend to shift to lower frequencies as the excitation is increased. Also, the amplitude

increased with increasing air flow. The rate of air bubble release at the surface appeared to oscillate at a frequency controlled by the waves, i.e., the most bubbles came to the surface when the center wave was at its peak.

A brief wave test in an existing air-water loop having a 36" diameter vessel is being conducted as a test of the wave hypothesis. Observations and photographs of wave modes and amplitudes are being made to determine the effects of water flow rate and air flow rate. Pressure transducers are used in an attempt to verify the two types of pressure variations. Results will be compared to the theory described above. A simple baffle will be tested to determine its effectiveness in reducing waves.

The loop is a simulation of a reactor vessel having a forced circulation system and air injection to simulate steam formation in the core. The vessel is open at the top to permit direct observation of the water surface. Use of grid plates to simulate a core did not appear to affect the results.

Although the results are incomplete, the following preliminary conclusions have been reached:

- (1) Very large waves can be excited. Wave amplitudes (zero to peak) equal to vessel radius and splashing amplitudes of over vessel diameter are easily achieved.
- (2) The wave modes specifically observed, listed in order of decreasing amplitude are:
 - (a) Side-to-side sloshing ($n = 1$).
 - (b) Annular ring ($n = 1$)
 - (c) Side-to-side sloshing ($n = 2$).
 - (d) Higher harmonics of indistinguishable types.

These all occur concurrently and at amplitudes which vary randomly.

- (3) The wave amplitude at the vessel wall, as determined visually, appeared to depend on air and water flow rates approximately as follows:

$$a = C_1 (Q)W^2 / C_2Q / C_3$$

where a = wave amplitude

W = water flow rate

Q = air flow rate

$C_1(Q)$ = constant which is dependent on air flow

C_2 & C_3 = constants

Small waves are generated with no air flow. Larger waves are formed when the air flow is maintained with no water flow. Very large waves are generated by flow of the air and water together.

(4) Pressure transducers connected to taps on the sides of the vessel indicate that:

- (a) The primary mode of surface wave is side-to-side.
- (b) The pressure variations of the first and second order types are clearly evident and occur at approximately the frequencies predicted by the equations.
- (c) The pressure variation at a point 6" below the mean surface level consists of the summation of the first and second order pressure variations. The second order variation passes through its mean value.
- (d) The pressure variation at a point 36" below the mean surface level consists of second order pressure effects. Two transducers on opposite sides of the vessel were in-phase, as predicted by the equations.
- (e) The pressure recordings have the characteristics of randomness and superposition of many frequencies that are similar to neutron flux variations in VBWR.

Standing surface waves of this type are objectionable in a reactor because of their influence on (a) flux noise, (b) moisture carryover in the steam, and (c) steam carryunder to the pump suction. Continued analysis of waves in the air-water loop should indicate the types of baffles which are effective in damping the waves.

III. TASK A - ADVANCED FUEL POWER-LIMIT TESTS

B. Irradiation in VBWR

This task provides for the irradiation of the basic and special fuel assemblies in VBWR. Table I presents the VBWR Operating Schedule and Table II presents the VBWR Loading Schedule for the next quarter.

III. TASK A - ADVANCED FUEL POWER-LIMIT TESTS

C. Basic Fuel Program

The basic fuel program includes the irradiation and examination of a large number of stainless steel and Zircaloy clad fuel assemblies operated at high specific power to long life in a boiling water reactor. There are a total of 26 stainless steel clad basic fuel assemblies of which 10 are used as control rod followers. The control rod follower assemblies are designated as H assemblies and regular stainless steel assemblies are designated as I assemblies. There are 24 Zircaloy clad basic fuel assemblies, all of which are designated as J assemblies. Irradiation is in progress for all the basic fuel assemblies. Burnup and performance data on the Fuel Cycle basic fuel assemblies through June 10, 1962, is summarized in Table III.

During the VBWR May outage assemblies 9H, 1I, 8I, 9I, 11J, 12J and 25J were inspected in the VBWR storage pool using an underwater periscope. Peak heat fluxes and exposures for these assemblies are shown in Table III. No serious defects were found in the fuel rods or bundle hardware during the VBWR pool periscope examination. Only minor handling damage such as scratches on the corner rods, bent spacers, and loose spacer tie wires was evident. Figure 1, which illustrates the condition of assembly 12J, is typical of the good condition of the fuel bundles.

Concurrent with the pool inspection of these assemblies, four fuel rod changes were made. The irradiated rods removed were sent to the RML for detailed visual and destructive examination. A summary of these rod changes is given in the table below.

<u>Assembly</u>	<u>Rod Installed</u>	<u>Rod Removed</u>	<u>Burnup</u>
1I	41F	202F	3806 MWD/T
12J	R6D12	R2D88	3923 MWD/T
25J	R5H20	R8H5	726 MWD/T
25J	R6D10	R5H19	726 MWD/T

Detailed visual inspection of these rods at the RML revealed a 6 to 7 mil deep groove on rod No. R5H19, removed from fuel assembly 25J as shown in Figure 2. The groove appeared at a position on the rod consistent with the location of the middle wire spacer during operation.

Subsequently, assemblies 8I, 12J, 25J and 7L were given a detailed examination in the RML hot cell. These assemblies were chosen for their high burnup and high power operation. Examination of these assemblies showed that fretting corrosion had occurred on some rods in each of the Zircaloy clad bundles (12J, 25J and 7L) but no fretting corrosion had occurred in the stainless steel clad assembly 8I. The occurrence of fretting corrosion on the Zircaloy clad fuel assemblies is illustrated in Figures 2, 3, and 4. The greatest penetration, about 25 mils, occurred on the top end plug on rod No. R2D74 in assembly 12J shown in Figure 3. The greatest clad penetration, about 12 to 14 mils, occurred at the middle spacer location of this same rod. The remainder of the rods affected had not been penetrated by more than about 6 or 7 mils into the 22 mil cladding.

The following observations were made in this examination: (1) lack of rod support at the upper tie plate caused general fretting corrosion at the middle spacer in the 7L assembly; (2) rods with large clearances at the upper tie plate showed the most penetration; (3) rods with tight support at the upper tie plate suffered comparatively light fretting corrosion even with large clearances at the middle spacer; (4) the corner rods which are rigidly supported at both top and bottom underwent no fretting corrosion.

It appears that wire deformation as a result of fuel handling increases the clearance between the fuel rods and supporting spacer and upper tie plate wires. A typical wire spacer is shown in Figure 5. The clearance subsequently allows the rods to vibrate freely as reeds and fret against the stainless steel wires. It is expected that the fretting corrosion can be reduced or eliminated by spring clips at the upper tie plate which will provide improved rod support. The spring clips can be installed on the assemblies in the VBWR storage pool without disassembly of the fuel bundle. Fabrication of trial clips is in progress.

Fuel rods were removed from basic fuel assemblies 11I, 12J and 25J and are currently at the RML. They have been examined visually; length and diameter measurements made; axial distribution of gamma activity determined; fission gas release determined; burnup determined radiochemically; and sectioned transversely. Completed data from the past irradiation examination are summarized in Table V. Metallography, data reduction, and analysis is in progress.

The unpolished section made at the peak flux region for the sound stainless steel rod taken from assembly 8I is shown in Figure 6. An unpolished section made at the peak flux region for the sound Zircaloy rod removed from assembly 12J is shown in Figure 7.

Figure 8 shows the outside appearance of the intentionally defected Zircaloy-4 and Zircaloy-2 rods removed from assembly 25J. Each defect is a 20-mil hole in the clad near the peak flux region. Unpolished sections made at the peak flux and average flux positions on the defected Zircaloy-4 rod are shown in Figure 9. Similar sections for the Zircaloy-2 rod are shown in Figure 10. Several bright colors are evident on the pellets at the region of peak flux as noted by the light areas in the photographs. It is probable that discoloration of UO_2 was caused by reaction with water.

No deterioration other than fretting of the Zr-2 rods was noted in the preliminary examination of these fuel rods.

III. TASK A - ADVANCED FUEL POWER-LIMIT TESTS

D. Special Fuel Program

The Special Fuel Program includes the irradiation and examination of 12 fuel assemblies based on fuel concepts which show potential for improved fuel cycle economy through increased performance or lower cost. These assemblies will be operated at high specific power to long life in order to determine performance capabilities in relation to the basic fuel assemblies which represent current fuel design and fabrication processes. Some of the special fuel assemblies provide proof tests of fuel concepts which have been developed by other Atomic Energy Commission contractors. The special assemblies and their status are summarized in Table VI.

During the VBWR May outage, April 29 to May 26, special assemblies 1L, 5L, 6L, 7L and 8L were inspected visually at the VBWR storage pool through the underwater periscope. The peak heat fluxes and exposures are shown in Table VII. The assemblies appeared to be in good condition with only minor handling damage evident.

Concurrent with the pool inspection, four fuel rods were removed for detailed visual destructive examination. These are listed in Table VIII.

These rods have been visually examined, measured for length and diameter changes, scanned to determine the axial distribution of gamma activity, punctured for fission gas release measurements, and sectioned transversely. Samples of UO_2 have been analyzed radiochemically to determine the amount of U-235 burnup. Completed data from the examination are shown in Table IX. Metallography, data reduction and analysis is in progress.

No apparently significant deterioration other than fretting corrosion of the stainless steel lined Zr-2 rods were noted in the preliminary examination of these fuel rods.

A. Higher Thermal Performance

Current fuel design specifications require that the UO_2 fuel shall not be molten at the maximum thermal output expected. Three of the special fuel assemblies are to be operated to provide information with which to evaluate the significance of such a specification.

1. Assembly 1L

Assembly 1L is being operated in VBWR at a maximum heat flux of $500,000 \text{ BTU/hr-ft}^2$ in order to evaluate the lifetime performance of uranium dioxide fuel operated at a condition approaching but not reaching central melting. The fuel rods in this assembly operate at a central fuel temperature associated with incipient void formation. This had been interpreted as the center melting limit in early fuel designs.

This assembly has reached an accumulated exposure of $3,842 \text{ MWD/T}$ and continues to operate satisfactorily in the VBWR with one intentionally defected fuel rod. After an average exposure of $3,224 \text{ MWD/T}$ assembly 1L was visually inspected in the VBWR storage

pool through the underwater periscope. The assembly was found to be in good condition and a representative fuel rod was removed for detailed visual and destructive examination at the Radioactive Materials Laboratory.

The fuel rod visual inspection revealed no apparent adverse effects of operation. Comparison of pre- and post-irradiation dimensions indicated no change in diameter but an apparent shortening of the rod of 70 mils. An unpolished rod section made at the peak heat flux region is shown in Figure 11. The rod section illustrates the operation of the UO_2 at incipient void formation. Metallographic examination and determination of fission gas release are in progress; completed data are shown in Table IX.

2. Assemblies 2L and 3L

Fabrication of assemblies 2L and 3L was about 80% complete when the Atomic Energy Commission requested that work on these assemblies be stopped pending a review of the design and program objectives, especially with regard to possible duplication of other programs.

3. Center Melt Calibration Rod

Analysis of the data from the 1.38 inch diameter fuel rod operated in the VBWR is continuing to determine the performance conditions at which uranium dioxide will operate with central melting. It appears from examination of photomicrographs of rod sections and correlation with operating history that the thermal conductivity of UO_2 increases when the UO_2 temperature is reached at which columnar grain growth takes place. Uranium dioxide that melts during the initial operation of a fuel rod will totally or partially solidify with continued irradiation of the rod without a reduction in power production. This is attributed to the formation of high conductivity columnar grains. This solidification process during operation apparently creates a ring of very high density and essentially porosity-free UO_2 in the fuel as can be seen in Figure 12. Information on the calibration rod operation and orientation of sections is presented in the previous quarterly report, GEAP-3935. It is expected that the appearance of the calibration rod sections can be correlated with calculated performance based on the preceding theory. Results of the completed analysis will be reported in subsequent reports.

4. Improved Thermal Conductivity - Assembly 9L

Should UO_2 thermal performance be limited by center melting or fission gas release at high temperatures, thermal conductivity improvers might be used. The approach to thermal conductivity improvement being pursued is a dispersion of a small amount of molybdenum metal through the UO_2 . Two methods of dispersion are being investigated. One is to mix molybdenum fibers with UO_2 powder prior

to pelletizing. The other is to coat the individual UO_2 particles with molybdenum metal prior to pelletizing.

All three fuel rods for investigating the improvement in thermal conductivity by the use of metal fibers dispersed in the UO_2 are complete. The appearance of the pellets used in these rods is shown in Figure 13. A pellet density, 95% of theoretical, was achieved by vacuum hot pressing. Longitudinal and transverse sections of the pellets are shown in Figure 14. It can be noted that the molybdenum fibers are favorably oriented for radial heat transfer. The radial orientation was inherent in the vacuum hot pressing fabrication process.

Fabrication of pellets made from UO_2 particles coated with molybdenum resulted in pellet density which has been optimized at about 85% of theoretical. Higher temperatures required for further densification would result in pellet-graphite die reactions during the vacuum hot pressing. The achievable density was probably adversely affected by the particle size range which was between 100 and 140 mesh and the UO_2 particle density. The porosity in the UO_2 coated particles and the excellent appearance of molybdenum coating is evidenced in Figure 15. Pellets made from the coated particles are shown in Figure 16. Photomicrographs of sections of the pellets are shown in Figure 17. The internal structure of the pellet consists of a relatively unaltered structure of the UO_2 in a continuous network of the molybdenum coating.

The frame for retaining fuel rods containing these pellets was completed. Details of the hardware, including the quick disconnect feature which facilitates individual rod replacement, are shown in Figure 18. The large diameter central rod is a nonfuel tube which serves as mechanical support for the entire assembly.

B. Reduced Cladding Parasitic Absorption Concepts

Zirconium alloys provide a low neutron cross section fuel element cladding for pressurized water and boiling water power reactors. Zirconium alloys have good corrosion resistance to water at temperatures normally encountered in these reactors, but may be susceptible to damage on the inside cladding surface if water enters the fuel rod. Austenitic stainless steel alloys not only have good corrosion resistance to water but since they have good corrosion resistance to high temperature steam, they are not so susceptible to attack on the inside cladding surface. Austenitic stainless steel alloys, however, have the disadvantage of an appreciable neutron absorption cross section. One mil thick stainless steel has about the same macroscopic absorption cross section as 30 mil thick zircaloy. The approaches to reduced parasitic absorption in the fuel cladding being pursued are (a) improve zirconium alloys, (b) provide protection to the inside surface of a commercially available zirconium alloy, such as Zircaloy-2 and (c) reduce the stainless

steel thickness to that of a thin corrosion skin rather than a pressure tube. One special assembly is devoted to performance evaluation of each of the above approaches.

1. Zircaloy-4 Clad-Assembly 6L

Assembly 6L contains fuel rods clad with an improved zirconium alloy (Zircaloy-4). Irradiation of this assembly containing one intentionally defected fuel rod was started and it has accumulated an exposure of 213 MWD/T with a peak heat flux of 325,000 BTU/hr-ft².

2. Stainless Steel Lined Zircaloy-2

Assembly 7L contains fuel rods clad with Zircaloy-2 co-extruded with about a one mil stainless steel lining. This assembly has accumulated an exposure of 2,285 MWD/T with the peak heat flux up to 467,000 BTU/hr-ft². Visual inspection of this fuel assembly revealed fretting corrosion on the fuel rods at the location of the middle wire spacer. Damage to the upper tie plate has allowed the rods to vibrate freely and the fuel rod surfaces have been rubbing against the wires in the middle spacer resulting in fretting corrosion. Repair of the upper tie plate is not possible and the fuel rods are being transferred to a spare fuel rod bundle frame for continued irradiation. A selected fuel rod was removed for detailed visual, dimensional and destructive examination.

No further rod deterioration, other than the fretting marks, were noted in the visual inspection. Dimensional measurements indicate that the rod has increased about 6 mils in diameter and decreased about 70 mils in length. No significance can be attached to these fuel rod dimensional changes inasmuch as operation of the rod has been satisfactory. The appearance of a transverse section of the fuel rod at the peak heat flux zone is shown in Figure 19. Metallographic examination and fission gas release determinations are in progress, completed data are shown in Table IX.

3. Thin Clad Stainless Steel - Assemblies 8L and 10L

Assembly 8L contains pelleted fuel rods clad with 5 mil thick stainless steel tubes. The assembly has accumulated 1,896 MWD/T exposure at peak heat fluxes up to 488,000 BTU/hr-ft². The assembly is operating satisfactorily with the cladding of one rod intentionally defected with a 10 mil diameter hole at the fission gas plenum.

Visual examination of the fuel bundle through the underwater periscope at the VBWR pool revealed no apparent dimensional changes or deterioration of the fuel rods or bundle.

A selected fuel rod is being visually, dimensionally and destructively examined at the Radioactive Materials Laboratory. The visual inspection indicates no rod deterioration. The dimensional

inspection indicates no significant change in rod diameter but about a 60 mil decrease in rod length. As operation has been satisfactory, no significance can be placed upon the apparent dimensional change.

A transverse section of the fuel rod at the peak heat flux zone is shown in Figure 20. Metallographic examination and fission gas release determinations are in progress, completed data are shown in Table IX.

Assembly 10L will contain fuel rods clad with 5 mil thick stainless steel tubes filled with UO_2 powder compacted by vibratory techniques. Five mil thick stainless steel tubing is not self-supporting at reactor operating pressures; it must rely on the UO_2 in the rod for support against collapse. Under these conditions the form of the UO_2 in the rod may have a significant effect upon the performance of that cladding. A portion of the Assembly 10L rods will be filled with arc-fused powder and the remainder will be filled with high particle density UO_2 powder formed by a new process under development at HAPCO called "Dynapak". This new process involves the densification of UO_2 powder by drop forging. The resulting product yields particles greater than 99% of theoretical density.

Fabrication is delayed by the availability of thin-wall stainless steel tubing. Difficulties in manufacturing the tubing to meet specifications of grain size and yield strength have delayed delivery. It is expected the fabrication will proceed in July 1962.

C. High UO_2 Density and Reduced Fabrication Cost Concepts

Irradiation of assemblies containing fuel rods made by new fabrication processes, which offer potential increases in UO_2 density or cost reductions in the manufacture of UO_2 fuel elements, will provide engineering proof tests of the feasibility of these new methods. The following fabrication methods are scheduled for testing.

1. Isostatic gas pressing - Assembly 4L
2. Low temperature sintered pellets - Assembly 4L
3. Co-extruded UO_2 and stainless steel clad - Assembly 5L
4. UO_2 extrusions - Assembly 11L

1. Hot Gas Isostatic Pressed - Assembly 4L

Three fuel rods made by hot gas isostatic pressing of UO_2 and stainless steel clad (BMI process) have been irradiated in the VBWR during runs 137 and 138 to an average exposure of 316 MWD/T at a peak heat flux of about 365,000 BTU/hr-ft². The center melt calibration rod described previously was also included in Assembly 4L for convenience. At the end of run 138 the calibration rod was removed and in so doing the 4L assembly structure was inadvertently

damaged so that it could not be returned to the VBWR core. Since refabrication of the 4L assembly was scheduled for incorporation of the low temperature sintered pellet rods to be supplied by United Nuclear Corporation, no attempt was made to repair the damaged structure and it was stored in the VBWR pool for over five months. During the transfer of these rods from the damaged to the new 4L assembly, one rod broke into two pieces. During subsequent pool and RML examinations, all three rods broke into a total of about 15 pieces. The failures were transverse cracking at several locations on two fuel rods and at one location on the other rod. The failures were associated with a dark reddish brown corrosion product on the clad surface. The exact cause of failure cannot be determined until RML examinations are completed in late August. However, corrosion which appears to be of intergranular nature seems to be a significant factor. Future test work to determine the cause of failure will be planned when the results of metallographic examination are available.

2. Low Temperature Sintered Pellets - Assembly 4L

Fuel rods loaded with low temperature sintered pellets were received from United Nuclear Corporation. These were incorporated in the new assembly 4L but irradiation has been delayed by the failure of the hot gas isostatic pressed rods. Standard fuel rods have been obtained and will be used to replace the hot gas isostatic pressed rods in assembly 4L.

3. Co-extruded UO₂ Stainless Steel Clad - Assembly 5L

The co-extruded UO₂ stainless steel rods in Assembly 5L have continued to operate satisfactorily. The assembly has accumulated 2,675 MWD/T exposure at peak heat fluxes up to 446,000 BTU/hr-ft². After 2,350 MWD/T exposure the assembly was visually inspected through the underwater periscope in the VBWR pool. There was no evidence of rod or bundle distortion or deterioration.

A selected fuel rod was sent to the Radioactive Materials Laboratory for detailed visual, dimensional destructive examination. No rod deterioration was noted in the visual examination. Dimensional measurements indicate there was approximately 4 mils increase in rod diameter and 100 mils decrease in length. No significance can be attached to the dimensional changes since these fuel rods continue to operate satisfactorily.

A transverse section of the fuel rod at the peak power location is shown in Figure 21. Metallographic examination and fission gas release determinations are in progress; completed data are shown in Table IX.

4. UO₂ Extruded Fuel - Assembly 11L

Assembly 11L contains fuel rods which were filled with UO₂ extru-

sions. Six inch long UO_2 extrusions are being supplied by Allis Chalmers and CICAFA of France.

A total of 45 extruded UO_2 compacts were received from Allis Chalmers this quarter. Thirty-six of these were ground satisfactorily in the fuels laboratory to eliminate bow. Two extrusions were broken during the centerless grinding process and one was reduced to a diameter less than the lower tolerance limit. The finished compacts were dimensionally and visually inspected after grinding. Extrusion lengths fell in a range of 6-1/16 inches to 6-23/64 inches and the diameter varied from 0.4835 inch to 0.87 inch. The average extrusion density was 96% of theoretical. Typical appearance of the ground Allis Chalmers extrusions used in 11L are shown in Figure 21. Occurrence of surface defects in the 36 ground extrusions are summarized in the following table:

<u>Type of Defect</u>	<u>Number of Extrusions in which Defect Occurred</u>
Edge chips, 1/32 inch across	8
Chips and pits, cylindrical surface	5
Chips and pits, end surfaces	16
Low spots on cylindrical surface	4
Hair line cracks, cylindrical surface	2

Six fuel rods containing Allis Chalmers extrusions, six containing CICAFA extrusions and four containing UO_2 pellets of 5.46% U-235 enrichment have been fabricated. One of the rods containing Allis Chalmers extrusions and one of the rods containing CICAFA extrusions were intentionally defected by drilling a 13 mil hole through each of the plenums. The 11L fuel element is at VBWR for insertion in the core during the next scheduled outage.

D. Extended Life Concepts

The loss of reactivity as the result of fuel burnup may limit the lifetime of fuel. Initial reactivity for burnup is limited by the capacity of the control system to make the cold clean core subcritical. The reactivity change associated with fuel burnup can be reduced with burnable poisons. One special fuel assembly, 12L will contain fuel rods which incorporate boron as a burnable poison. The boron will be incorporated by (1) alloying small quantities of boron into stainless steel cladding and (2) intimately mixing a boron compound with the fuel.

Fabrication of this assembly is delayed by the availability of borated stainless steel tubing. The tubing vendor has again revised his delivery date and now promises delivery of the tube by August 6. The fabrication of the fuel rods containing burnable poison mixed in with fuel is in progress.

IV. TASK B - HEAT TRANSFER AND FLUID DYNAMICS

This task provides for out-of-core heat transfer and fluid dynamics work related to boiling water reactors, in support of the fuel tasks and reactor performance task.

1. Analytical Burnout Prediction

A topical report was published containing a superposition model to predict the critical (burnout) heat flux in forced convection flow. The model is applied to available experimental results and good agreement is obtained over a variety of geometries, flow rates, pressure and fluid enthalpies. The report is GEAP-3961, Prediction of The Critical Heat Flux in Forced Convection Flow, S. Levy, June 20, 1962.

The superposition approach consists of subdividing the complex problem of heat transfer at the critical heat flux into subproblems more amenable to analysis. Solutions are derived for the subproblems which are added algebraically to obtain the analytical representation of the overall process. Two cases are presented:

- (a) The case of subcooled flow, wherein the critical heat flux under non-flow or pool boiling conditions is first specified. To this heat transfer rate, the heat conducted from the surface by the subcooled non-flowing fluid and the heat convected from the surface by the subcooled flowing fluid are added.
- (b) The case of net vapor generation, wherein the pool-boiling critical heat flux is, again, first specified. To this the heat convected from the surface is added and the equivalent mass transfer rate is subtracted to account for the presence of vapor in the stream.

2. Multirod Burnout Tests

In order to supplement the single rod burnout work done under the Fuel Cycle Program a series of measurements of multirod burnout will be conducted. A test section containing seven electrically-heated rods was fabricated. It was described in GEAP-3935, Seventh Quarterly Progress Report.

During the current quarter the test section was inserted into the General Electric heat transfer facility at San Jose, California, and shakedown tests were run. The major performance problems encountered were:

- (a) Internal support of the outer six tubes, to prevent collapse under high pressure operation.
- (b) Development of wire spacers which will separate the seven rods and prevent excessive vibration due to large electrodynamic forces.

V. TASK C - EXPERIMENTAL PHYSICS

Task C of the Fuel Cycle Program provides for the study of long term reactivity characteristics of fuels irradiated in the VBWR. Isotopic analysis of uranium depletion and plutonium production is being experimentally determined and compared to theoretical predictions.

Two, 2.7% enriched, Fuel Cycle elements are being used for this task. Both elements contain five stainless steel guide tubes which allow flux measurements to be made in five locations within the element.

Irradiation of the two elements in the VBWR began in November 1961. At the end of December the elements were removed from the core and the first rods were removed for preparation of samples for isotopic analysis. Pellets were removed from axial locations, spanning the void range of 0-25%. Figure 23 shows the axial relative power and void distribution assumed for the rod sampled. The power distribution is based on the gamma scan of the irradiated fuel rod, and the void distribution is based on the power distribution and the recorded core flow rate.

Results of the comparison of measured and computed isotopic concentrations and burnup are presented here. Figure 24 presents the comparison of the measured and computed burnup as a function of axial position along the fuel rod. The computed burnup is based on the gamma scan axial power distribution, total core power from the operators log, and a two-dimensional analysis for the relative rod power to total core power. The measured burnup is based on the average of the fission product analysis of Cs-137, Ce-144, Ba-140, and Zr-95. The significant result from this comparison is the consistent underestimate of the calculated integrated power relative to that inferred from the fission product measurements. The uncertainties shown on the measured data represent uncertainties in the actual pellet sampled (i.e., ± 1 pellet) as well as a $\pm 5\%$ uncertainty in the measurement. The uncertainty of $\pm 10\%$ on the calculated data represents estimated uncertainties in total reactor power, in relative rod power to total power and in the axial power distribution.

Figures 25 to 27 present the comparison of measured and computed Pu-239, Pu-240, and Pu-241 atom fractions as a function of axial position along the rod. The computed data presented here are based on the measured burnup, so these uncertainties reflect accuracy of calculational methods and cross sections utilized. The uncertainties in the measured atom fraction of Pu-239, Pu-240, and Pu-241 are about $\pm 5\%$, $\pm 6\%$, $\pm 7\%$, respectively. In general, the calculations underestimate the Pu-239 atom fraction, overestimate the Pu-240 and Pu-241 in the high power region, and underestimate the Pu-240 and Pu-241 in the low power regions when compared to the measured data. This would indicate an underestimate of the capture in U-238 and an overestimate of the capture in Pu-239.

Figure 28 shows the measured vs. computed U-236 atom fraction as a function of axial position along the fuel rod. There is considerable scatter in the measured data, but the burnup as indicated from the U-236 concentration is relatively consistent with the measured burnup. The U-236 concentration

shown here is that amount present. Since the U-236 atom fraction before irradiation was about 2×10^{-4} , this is used as a zero value.

The same cross sections used for this analysis will be used for computation of the next set of data (i.e., at about 900 MWD/T). If similar differences are observed, an attempt will be made to justify altering cross sections.

A detailed topical report will be issued, describing methods used in the theoretical and experimental analysis.

Irradiated fuel rods have been removed from these Task C elements for the second set of pellet analyses. The average burnup for these pellets is about 900 MWD/T. These pellets are presently being analyzed by the Chemistry Laboratory. The gamma scan of these irradiated fuel rods is shown in Figure 29. The peak power here is about 3% lower than that for the rods with a burnup of about 500 MWD/T. Calculations are underway to determine theoretically the atom concentration for these pellets now being chemically analyzed.

The third set of rods is presently being removed at the Radioactive Materials Lab (RML). These pellets have an average burnup of about 1800 MWD/T.

Flux wire measurements were made in the five locations within element 18J. The measurements were made for a reactor power of 34 MW and a core flow of 16,000 gpm. Ni-Co wire was used in an attempt to get a measurement of the thermal and fast neutron flux. The wire was counted shortly after irradiation and the 2.5 hr Ni-65 activity observed. Several days later the wire was recounted and the 72-day Co-58 activity observed. Results of these measurements are shown in Figures 30 and 31. The five wire positions within the element are shown on the graph. These data indicate that this method of obtaining a thermal and fast flux distribution is relatively good. Figure 32 shows the ratio of thermal to fast neutron flux as a function of axial position along the fuel element. It is seen that this ratio varies by about 15% from no voids to exit voids. Additional flux measurements of this type are planned for the future for various reactor powers and flow rates.

Irradiation of these Task C assemblies is continuing and rods will be sampled in the future at about 2000 MWD/T intervals up to maximum burnup. The accumulated burnup to date for elements 17J and 18J is about 1800 MWD/T and 2000 MWD/T, respectively.

TABLE I

VBWR OPERATING SCHEDULE
JULY, AUGUST, SEPTEMBER 1962

July 1 to July 8 - Power Operation, ESADE Shakedown
July 8 to July 18 - Shutdown, Install ESADE Fuel, Plant Maintenance
July 18 to August 18 - Power Operation
August 18 to August 25 - Shutdown, Add Fuel, Plant Maintenance
August 25 to September 23 - Power Operation
September 23 to October 1 - Shutdown, Add Fuel, Plant Maintenance

TABLE II
VBWR LOADING SCHEDULE
JULY, AUGUST, SEPTEMBER 1962

	<u>July</u>	<u>August</u>	<u>September</u>
Consumers HPD	30	30	30
Fuel Cycle	52	50	48
Plutonium	1	1	1
(E)SADE	4	4	4
Pressure Build-Up	1	1	1
Dresden	7	7	7
Savannah I	3	3	3
Savannah II	6	6	6
DuPont	2	2	2
	<hr/>	<hr/>	<hr/>
Total Test Assemblies	106	104	102
	<hr/>	<hr/>	<hr/>
GE Driver Fuel	6	8	10
	<hr/>	<hr/>	<hr/>
Total Fuel Assemblies	112	112	112

TABLE III

BASIC FUEL IRRADIATION, STAINLESS STEEL CLAD ASSEMBLIES

Assembly No.	Specific Power KW/KG	Heat Flux BTU/hr-ft ²		Accumulated Exposure MWD/T
		Average	Peak	
1H	31.3	198,000	342,000	2952
2H	30.0	190,000	314,000	3009
3H	29.3	185,000	295,000	3454
4H	38.5	244,000	409,000	4225
5H	34.8	222,000	368,000	3729
6H	34.4	218,000	355,000	3543
7H	34.4	218,000	355,000	3640
8H	38.5	244,000	398,000	3861
9H	40.9	260,000	426,000	4436
10H	37.2	234,000	368,000	3634
1I	38.8	252,000	353,000	4050
2I	27.0	175,000	286,000	2509
3I	30.0	191,000	311,000	3594
4I	32.0	205,000	346,000	3489
5I	28.0	180,000	298,000	3100
6I	36.1	234,000	382,000	3267
7I	31.6	205,000	342,000	2972
8I	39.6	257,000	419,000	3964
9I	32.4	212,000	347,000	3944
10I	30.9	201,000	328,000	3370
11I	38.1	246,000	404,000	3863
12I	28	180,000	301,000	3364
13I	44.4	212,000	338,000	3023
14I	26.0	165,000	286,000	3569
15I	20.8	135,000	235,000	741
16I	39.5	191,000	307,000	3462

TABLE III (continued)

BASIC FUEL IRRADIATION, ZIRCALOY CLAD ASSEMBLIES

Assembly No.	Specific Power KW/KG	Heat Flux BTU/hr-ft ²		Accumulated Exposure MWD/T
		Average	Peak	
1J	30.5	191,000	310,000	2636
2J	25.0	158,000	268,000	2070
3J	23.8	149,000	250,000	2136
4J	29.9	189,000	318,000	2595
5J	33.3	210,000	352,000	2881
6J	43.8	274,000	450,000	3328
7J	32.7	203,000	338,000	2466
8J	35.0	220,000	335,000	2700
9J	31.8	200,000	320,000	2876
10J	24.6	155,000	248,000	2426
11J	46.5	288,000	482,000	4222
12J	44.5	277,000	479,000	4100
13J	41.6	259,000	420,000	3957
14J	43.6	272,000	439,000	3725
15J	35.8	223,000	361,000	2056
16J	41.2	257,000	410,000	2538
17J	36.0	225,000	368,000	1645
18J	40.8	255,000	413,000	1882
19J	34.0	212,000	334,000	2002
20J	22.6	142,000	242,000	2305
21J	26.1	164,000	256,000	1330
22J	29.8	187,000	311,000	1501
23J	29.0	182,000	304,000	1060
25J	49.0	308,000	492,000	1885

TABLE IV
BASIC FUEL POOL INSPECTION

<u>Assembly</u>	<u>Estimated Maximum Heat Flux BTU/HR-FT²</u>	<u>Estimated Average Exposure</u>
9H	479,000	4177
1I	407,000	3806
8I	422,000	3809
9I	382,000	3659
11J	408,000	3929
12J	479,000	3923
25J	492,000	1887

TABLE V
BASIC FUEL EXAMINATION IN RML

Clad	Peak Heat	Average Burnup		Dimensions					
	Flux	MWD/T		Diameter Inches			Length		
	BTU/hr-ft ²	Est.	Meas.	Pre	Post		Pre	Post	
				± .001	± .001	Δ D	± .03	± .02	Δ L
Stainless Steel	407,000	3806	4020	0.410	0.411	±.001	39.81	39.75	-.06
Zircaloy-2	479,000	3923	4400	0.425	0.426	±.001	40.69	40.67	-.02
Zircaloy-2 Defected	492,000	726	913	0.423	0.424	±.001	40.69	---	--
Zircaloy-4 Defected	492,000	726	1020	0.426	0.426	±.001	40.69	---	--

TABLE VI
SPECIAL FUEL ASSEMBLIES

STATUS

<u>Element</u>	<u>Concept</u>	<u>Status</u>	Peak Heat Flux	Accumulated Exposure
			BTU/HR-FT ²	MWD/T
1L	Center Melt	in VBWR	530,000	3842
5L	Co-Extruded UO ₂ and SS Clad	in VBWR	446,000	2675
6L	Zr-4 Clad	in VBWR	325,000	213
7L	SS lined Zr-2	being repaired	467,000	2285
8L	0.005" SS Clad Pellets	in VBWR	488,000	1896
2L & 3L	Center Melt	Fabrication 80% complete-stopped pending review at AEC request.		
4L	Isostatic Pressed Rods & Low Temp. Sintered Pellets	Isostatic pressed rods broke in handling after 316 MWD/T. Rods with low temperature sintered pellets re- ceived and inserted in assembly.		
9L	Thermal Conduc- tivity Improvers	Rods with M ₀ fibers in pellets complete. Fabrication of pellets from M ₀ coated UO ₂ in progress.		
10L	0.005" SS Clad over powder UO ₂	Fabrication delayed by tubing availability.		
11L	Extruded UO ₂	Fabrication complete.		
12L	Burnable Poison	Fabrication delayed by availability of borated stain- less steel tubing.		

TABLE VII
SPECIAL FUEL POOL

INSPECTION

<u>Assembly</u>	<u>Maximum Heat Flux</u> <u>BTU/HR-FT²</u>	<u>Average Exposure</u>
1L, "center melt"	530,000	3224
5L, SS-UO ₂ co-ext.	446,000	2350
6L, Zr-4	318,000	41
7L, Zr-2 - SS lined	467,000	2285
8L, .005" SS	488,000	1646

TABLE VIII
SPECIAL FUEL ROD CHANGES

<u>Assembly No.</u>	<u>Rod Installed</u>	<u>Rod Removed</u>	<u>Burnup</u>
1L	4.3-6	4.3-4	3224 MWD/T
5L	093	141	2082 MWD/T
7L	3	H	2285 MWD/T
8L	502	511	1646 MWD/T

TABLE IX
SPECIAL FUEL EXAMINATION

Rod	Peak Heat Flux BTU/HR-FT ²	Average Burnup		Dimensions					
		MWD/T		Diameter Inches			Length		
		Est.	Meas.	Pre	Post	ΔD	Pre	Post	ΔL
				$\pm .001$	$\pm .001$		$\pm .03$	$\pm .02$	
1L	530,000	3224	3470	0.515	0.515	0	40.21*	40.14	-.07
5L	446,000	2350	2820	0.403	0.396 0.412	$\pm .004$	40.94	40.94	-.1
7L	467,000	2285	2750	0.428	0.432 0.437	$\pm .006$	40.75	40.68	-.07
8L	488,000	1646	1860	0.385	0.384 0.388	$\pm .001$	45.87	45.81	-.06

*Estimated - Pre irradiation measurement not available.

SIDE C

SIDES D & A

**FIGURE 1 APPEARANCE OF ASSEMBLY 12J UNDERWATER PERISCOPE EXAMINATION
DURING VBWR MAY 1962 OUTAGE**

ROD # R5H19

ROD # R8H1, CENTER

FIGURE 2 FRETTING CORROSION OBSERVED IN ASSEMBLY 25J

ASSEMBLY 12J, ROD #R2D74, TOP END PLUG

ASSEMBLY 12J, ROD #R2D74, CENTER

FIGURE 3 MAXIMUM PENETRATION BY FRETTING CORROSION

RODS 7 & 8, CENTER

ROD F, CENTER

FIGURE 4 OCCURANCE OF FRETTING CORROSION IN ASSEMBLY 7L

FIGURE 5 WIRE SPACER

ASSEMBLY 11, PEAK HEAT FLUX

FIGURE 6 ASSEMBLY 11, ROD #202F SECTION

PEAK HEAT FLUX

FIGURE 7 ASSEMBLY 12J, ROD # R2D88 SECTION

ASSEMBLY 25J, ROD #R8H5, Zirc-4 DEFECT FUEL

ASSEMBLY 25J, ROD #R5H19, Zirc-2, FUEL DEFECT

FIGURE 8

PEAK HEAT FLUX

AVERAGE HEAT FLUX

FIGURE 9 ASSEMBLY 25J, ROD # R8H5, Zr-4

PEAK HEAT FLUX

AVERAGE HEAT FLUX

FIGURE 10 ASSEMBLY 25J, ROD # R5H19, Zr-2

PEAK HEAT FLUX

FIGURE 11 ASSEMBLY 1L, ROD #434

SECTION I

250X

VOID

O.D.

SECTION F

50X

CENTER

FIGURE 12 SECTIONS OF CENTERMELT CALIBRATION ROD

A. AS-HOT PRESSED MAG: 2X

B. CENTERLESS GROUND MAG: 4X

FIGURE 13 APPEARANCE OF HOT-PRESSED PELLETS CONTAINING UO_2 AND 20% MOLYBDENUM FIBERS. DENSITY IS 95% OF THEORETICAL. ASSEMBLY 9L

UNETCHED

MAG: 9X

A. LONGITUDINAL SECTION

ETCHED

MAG: 6X

B. CROSS SECTION

FIGURE 14 SECTIONS OF HOT-PRESSED PELLETS SHOWING ORIENTATION OF MOLYBDENUM FIBERS FAVORABLE TO RADIAL HEAT TRANSFER. ASSEMBLY 9L

A. AS-RECEIVED PARTICLES: SIZED BETWEEN
100/140 MESH.

MAG: 50X

B. SECTION OF COATED PARTICLES. COATING
THICKNESS: ~ 3 MICRONS. NOTE PORUS
APPEARANCE OF UO_2 .

MAG: 150X

FIGURE 15 INDIVIDUAL PARTICLES OF SIZED UO_2 COATED WITH MOLYBDENUM. ASSEMBLY 9L

A. AS-HOT PRESSED MAG: 2X

B. CENTERLESS GROUND MAG: 2X

FIGURE 16 HOT-PRESSED PELLET MADE FROM UO_2 PARTICLES COATED WITH ABOUT 3 MICRONS THICK MOLYBDENUM LAYER

A. LONGITUDINAL SECTION MAG: 16X

B. UNETCHED MAG: 250X

FIGURE 17 UO_2 PARTICLES COATED WITH ABOUT 3 MICRONS THICK LAYER OF MOLYBDENUM
HOT PRESSED INTO PELLETS WITH DENSITIES OF 85% OF THEORETICAL.

A. QUICK-DISCONNECT FEATURE ASSEMBLED

B. DISASSEMBLED

C. CENTRAL SPACER

D. BOTTOM PLATE SUPPORT

FIGURE 18 ASSEMBLY 9-L BUNDLE FRAME

PEAK HEAT FLUX

FIGURE 19 ASSEMBLY 7L, ROD # 11

PEAK HEAT FLUX

FIGURE 20 ASSEMBLY 8L, ROD # 511

PEAK HEAT FLUX

FIGURE 21 ASSEMBLY 5L, ROD #141

FIGURE 22 TYPICAL ALLIS CHALMERS EXTRUSIONS AFTER GRINDING

FIGURE 23 AVERAGE POWER & VOID DISTRIBUTION FOR ROD NO. 1

FIGURE 24 MEGAWATT DAYS PER TON OF URANIUM VS AXIAL POSITION ALONG ROD

FIGURE 25 Pu-239 ATOM FRACTION VS AXIAL POSITION ALONG ROD

FIGURE 26 $Pu-240$ ATOM FRACTION VS AXIAL POSITION ALONG ROD

FIGURE 27 $Pu-241$ ATOM FRACTION VS AXIAL POSITION ALONG ROD

FIGURE 28 U-236 ATOM FRACTION VS AXIAL POSITION ALONG ROD

FIGURE 29 GAMMA-SCAN OF IRRADIATED FUEL RODS. CURVE NORMALIZED TO 1.0

FIGURE 30 THERMAL NEUTRON FLUX DISTRIBUTION (Ni-65 ACTIVITY)

FIGURE 31 FAST NEUTRON FLUX DISTRIBUTION (Co-58 ACTIVITY)

FIGURE 32 THERMAL TO FAST FLUX RATIO

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