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DEVELOPMENT REPORT

MASTER

**HIGH POWER DENSITY DEVELOPMENT
PROJECT SEVENTH QUARTERLY
PROGRESS REPORT
OCTOBER - DECEMBER 1961**

By
R.L. HOLLADAY

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GENERAL ELECTRIC

SAN JOSE, CALIFORNIA

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by

R. L. Holladay

Under Contract No. AT(04-3)-361

Approved By D/H Imhoff
D. H. Imhoff, Manager
Engineering Development

Approved By L. K. Holland
L. K. Holland
Project Engineer

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TABLE OF CONTENTS

	<u>Page No.</u>
SUMMARY	1
TASK IA- HIGH POWER DENSITY FUEL DEVELOPMENT	4
1. Task Objective	4
2. Fuel Irradiation	4
3. Instrumented Assemblies	6
TASK IB- FUEL FABRICATION DEVELOPMENT	18
1. Task Objective	18
2. Hot Swaged Fuel Rods	18
3. Powder Compacted Fuel Rods	22
4. Fused UO ₂ Product Evaluation	30
5. Erosion Testing Defected Powder Compacted Fuel Rods	37
6. Vibratory Compaction Studies	37
7. Design of Pool Facilities for Big Rock Point	41
8. Developmental Fuel Bundles for Big Rock Point	41
9. Task IB VBWR Fuel Assemblies	43
10. Mechanical Vibration Testing	44
TASK II - STABILITY, HEAT TRANSFER AND FLUID FLOW	46
1. Task Objective	46
2. Heat Transfer Tests	46
TASK III - PHYSICS DEVELOPMENT	47
1. Task Objective	47
2. Economical Methods of Power Distribution Analysis	47
3. Reduction in Axial Peaking Factor	48
4. Large High Power Density Conceptual Design	55
5. Scheduling Computer	57
TASK IV - COORDINATION AND TEST PLANNING	58
1. Task Objective	58
2. Coordination	58
3. Test Planning	58
ACKNOWLEDGEMENT	59

ILLUSTRATIONS

	<u>Page No.</u>
Figure 1 - Bundle 1F Spacer Condition	5
Figure 2 - Results of Temperature vs. Pressure Test	7
Figure 3 - Data from Recirculating Flow Change Test	8
Figure 4 - Results of Power Level Test	12
Figure 5 - Flux Distribution in Assembly 6E	14
Figure 6 - Hot Swaged Clad Sample	20
Figure 7 - Cross Section of Hot Swaged Fuel Rod	20
Figure 8 - Microphotographs of Hot Swaged Fused UO ₂	21
Figure 9 - Capsule Design for Isostatic Pressure	23
Figure 10 - Gas-Pressure Bonded Swaged and Rolled Fuel Rods	24
Figure 11 - Appearance of UO ₂ From Gas-Pressure Bonded Rolled and Swaged Rods	25
Figure 12 - Hot Isostatically Pressed Rolled and Swaged Rods	26
Figure 13 - Microphotographs of Isostatically Pressed Rolled Rod	28
Figure 14 - UO ₂ Clad Interface of Swaged Specimen	29
Figure 15 - Typical Stainless Clad Specimens from Run No. 2	31
Figure 16 - Second Phase Constituents in 3.5% Enriched Fused UO ₂	32
Figure 17 - Second Phase Constituents in 3.5% Enriched Fused UO ₂	33
Figure 18 - Effect of Particle Size on Amount of Gas Evolved From Fused UO ₂	34
Figure 19 - Effect of Exposure Time to the Atmosphere on Gas Evolved from Fused UO ₂	35
Figure 20 - Weight Change vs. Flow Test Time for Swaged and Rolled Erosion Samples	38
Figure 21 - Erosion Samples After 6000 Hours Flow	39
Figure 22 - Appearance of Erosion Samples	40
Figure 23 - Location of Developmental Assemblies in 50 MW Reactor	42
Figure 24 - Appearance of Slit Type Defects in Swaged Rods	45
Figure 25 - Thermal Flux Distribution	49
Figure 26 - Relative Power Distribution	50
Figure 27 - Relative Fuel Element Power	51
Figure 28 - Axial Peaking Factor Reduction	54

TABLES

Table I - Operational Tests on Instrumented Assemblies	10
Table II - Response of Instrumentation	17
Table III - UO ₂ Powder Analysis Before and After Hot Swage	19
Table IV - Capsule Data Run Number 1	27
Table V - Particle Size Distribution Before and After Offgassing	36
Table VI - Weight Measurements of Erosion Samples	37
Table VII - Relative Fuel Element Power	52
Table VIII - Reactivity Balance-Equilibrium Cycle	55
Table IX - Enrichment Requirements for 15,000 MWD/T	56
Table X - Power Peaking Factors	56

SUMMARY

During the seventh quarter, the twenty-four high power density fuel assemblies were irradiated in the VBWR to an average of about 2572 MWD/T. Also during this period, the instrumented assembly tests were completed, and the award of the contract for the Scheduling Computer was made to the Process Computer Section of the General Electric Company. A summary of progress by task is presented below.

Task IA - High Power Density Fuel Development

1. The 24 HPD fuel assemblies have continued to accumulate exposure during the quarter. Average burnup in MWD/T for the various groups was 2810 MWD/T for A and B group, 1770 for C and D, 2910 for E and F, and 3265 for the G group.
2. During July 1961, initial irradiation tests of the two instrumented assemblies were conducted. The tests demonstrated the applicability of the in-core flow and neutron flux instrumentation to steady state and transient measurements. They pointed out some areas of further development required on the in-core pressure drop system and temperature measurement. The in-core flowmeters were found to have an operational lifetime of approximately 6 to 8 weeks. All other instruments were found to have lifetimes in excess of 6 months.

Task IB - Fuel Fabrication Development

1. Hot swaging of 10-mil stainless clad fuel rods to 95 per cent theoretical density was demonstrated.
2. Swaged and rolled fuel rod samples were isostatically pressed using the BMI gas pressure bonding technique. The technique increased the as-fabricated density of swaged rods 1/2 to 1 per cent, while that of rolled rods 2 to 3 per cent. Clad grain growth and carbide precipitation was observed.
3. Small amounts of second phase inclusions in Spencer fused UO_2 have been observed and identified as uranium, uranium nitride, and high uranium oxides. Gas evolution determinations on Spencer fused UO_2 subjected to a $1600^\circ C$. treatment in hydrogen showed that the gas content was reduced from 0.55 to 0.02 cc per gram of UO_2 .
4. Four purposely defected erosion test samples were examined after 6000 hours flow testing in $536^\circ F$. water. The results of the tests indicate that swaged fuel rods with UO_2 densities greater than 90 per cent T.D. are highly resistant to UO_2 removal, through a clad defect, by the coolant stream.
5. Design of fuel inspection and handling equipment for the Big Rock Development fuel was initiated during the quarter.
6. Physics aspects of the first eight developmental fuel assemblies for Big Rock Point are being studied. Locations in the core have been tentatively chosen and enrichment has been specified.
7. Seven Task IB fuel assemblies were in the VBWR during the reporting period. Maximum exposure attained was approximately 2000 MWD/T. Two purposely defected

swaged fuel rods containing a 20-mil wide x 1/2-inch long slit were added to assembly 1S in mid-December. The two rods had been performing satisfactorily through the end of December.

8. Analyses of vibration mechanisms do not explain the August multirod test behavior. It is hoped that further testing will clarify this problem.
9. A single rod test is being planned which should give more informative results with regard to general understanding and prediction of vibration.

Task II - Stability, Heat Transfer and Fluid Flow

1. The nine-rod heat transfer test section was operated at mass velocities of 0.56×10^6 and 1.12×10^6 lb/hr-ft² for various qualities at 1000 psi. No burnout resulted from heat fluxes up to 75% of single rod burnout and maximum rod spacing.
2. The tests will continue in the coming quarter, and will include actual burnout points.

Task III - Physics Development

1. A study of economical methods of determining power distribution was performed to investigate the maximum number of mesh points necessary to yield a satisfactory description of the core. Results indicate that for a mesh spacing greater than about 6.3 cm. the power distribution calculation becomes inaccurate, although the accuracy of reactivity calculation is good. An attempt to replace the water reflection by a three-group current boundary condition was not successful.
2. The possibility of using axially distributed poison is being examined as a first step toward improving axial power distribution. A simplified calculation has shown some incentive for this, and more exact calculations are in progress.
3. Physics studies in connection with the recent 300 MW HPD conceptual design have been completed, and are included as a portion of a topical report covering this design, which was prepared during the quarter.
4. Proposals for the Big Rock data logging system were received from eight bidders. At the request of the AEC San Francisco Office, a technical evaluation of the proposals was prepared by APED and submitted for their consideration. The final evaluation and selection of the successful bidder was done by SAN, and the award was made to the Process Computer Section of the General Electric Company. The contract provides for a G.E. 312 computer and associated system, but excludes the computer programming.

Task IV - Coordination and Test Planning

1. A progress review was held with AEC representatives in San Jose on October 2, 1961. Physics development was reviewed in detail.
2. AEC Form 189, a financial and work forecast, was prepared and issued in December. Monthly reports for October and November were also prepared and issued.

3. Planning for tests in the Big Rock Point Plant was continued during this period. The R & D contribution to the Final Hazards Summary Report was finalized and incorporated into the report as issued in December. The R & D contribution to the Technical Specification, to be submitted as a part of the Consumers license application, was also prepared.

TASK 1A - HIGH POWER DENSITY FUEL DEVELOPMENT

1. TASK OBJECTIVE

A partial core consisting of 24 high power density fuel elements designed and fabricated by existing manufacturing processes is being irradiated in VBWR to study its performance and explore its feasibility. The effect of variations in structure, geometry, fabricating processes and mechanical features will be evaluated in terms of operation to 90 kw/l average power density and to 10,000 MWD/T average exposure. Instrumented assemblies capable of measuring flow, temperature, pressure drop, and flux level are being tested for performance and feasibility for later application in the Big Rock Reactor.

2. FUEL IRRADIATION

The twenty-four high power density fuel assemblies have been operating satisfactorily in the VBWR during the reporting period. Average burnup for each of the assemblies through December 20, 1961, is shown below:

<u>Assembly No.</u>	<u>MWD/T</u>	<u>Assembly No.</u>	<u>MWD/T</u>
1A	2707*	1E	3293
2A	2745*	2E	3089
1B	2758*	3E	3446
2B	3031*	4E	3178
1C	943	5E	3143
2C	1640	6E	2140
3C	944	1F	2613
4C	1987	2F	2365
1D	2623	1G	3055
2D	2515	2G	3311
3D	1764	3G	3207
4D	1739	4G	3488

*Exposure based on complete withdrawal of the control rod at all reactor powers.

Rework on Fuel Bundle 1F

During a November fuel move, the spacers on fuel bundle 1F were found skidded. The assembly was removed from the core. Details of the spacer condition are shown in Figure 1, indicating the possibility of some deformation of the spacer band. When the bundle was trial fitted into a channel, it would not freely enter the channel, indicating that some deformation of the bands had occurred. The fuel rods appeared to be in good condition.

The spacer bands will be straightened and the spacers relocated in January prior to further irradiation.

TOP SPACER SHOWING WIRE DEFORMATION
(APPARENT CRACK ON L.H. ROD IS FILM DEFECT)

BROKEN WIRE-TO-WIRE WELD ON TOP SPACER
(FROM LACK OF CORROSION FILM AT BREAKS,
IT IS ASSUMED BREAKS OCCURRED AS BUNDLE
WAS REMOVED FROM CHANNEL)

FIGURE 1
BUNDLE IF SPACER CONDITION

3. INSTRUMENTED ASSEMBLIES

In July 1961 a series of operational tests were performed on the instrumented assemblies. These tests, which occurred during the initial irradiation of the assemblies had as their objectives:

- (a) To demonstrate the operation of the instrumented assemblies under conditions of temperature, pressure and neutron flux found in boiling water reactors.
- (b) To obtain information on lifetime of the in-core instrument elements.

A summary of the tests is shown in Table I.

Results of the Operational Tests

(a) Single-phase Pressure Drop versus Flow.

Prior to criticality, the No. 1 recirculation pump was started and the pressure drop systems in the assemblies were adjusted. However, it was found that no pressure drop measurements could be made because of air dissolved in the backflow water of the pressure drop system. This water came from the reactor feedwater system. As the energy used to drive the recirculation pump was added to the reactor system, the water temperature increased. This caused the dissolved air to come out of solution and form bubbles in the pressure lines. Later, during reactor heat-up, the air was driven out of the reactor water. However, the system still had operating difficulties as discussed below.

(b) Temperature versus Pressure Test

The core inlet and outlet temperatures of the reactor coolant water were measured during the reactor heat-up operation and were compared to the steam dome pressure (measured with the reactor process system pressure gauge).

During the test, over the temperature range 350°F. to 550°F, the overall readings from the in-core thermocouples were approximately 1-1/2 to 2 per cent low. This included the inlet as well as the outlet temperatures. The data are shown in Figure 2.

(c) Recirculation Flow Tests

With the reactor operating at approximately 12 MWT, the recirculation flow was varied from 10,000 gpm to 20,000 gpm. Figure 3 shows portions of the actual traces of the upper and lower flow meters in Assembly 6E during the recirculation flow tests with traces of neutron flux, riser pressure drop and core inlet temperature. Note the similarity of the flow oscillations in the two flowmeters, particularly with respect to the phase relationships between the oscillations. Moreover, note the increase in amplitude of the oscillations in both flowmeters as the recirculation flow is increased. Finally, note that the amplitude of the oscillations in the upper flowmeter increase faster than in the lower flowmeter as the recirculation flow is increased. The riser pressure drop shown in Figure 3 seems to generally increase with

530-2

FIGURE 2 RESULTS OF TEMPERATURE VERSUS PRESSURE TESTS

REACTOR POWER = 12 MWT
 CONTROL ROD BANK = 900 DIV.
 CONTROL ROD 7 = 283 DIV.
 RECIRCULATION FLOWS:
 PUMP NO. 1 = 10,000 gpm
 PUMP NO. 2 = 3,400 gpm
 FEEDWATER = 46,500 lb/hr
 PRESSURE = 1,000 psig

REACTOR POWER = 12 MWT
 CONTROL ROD BANK = 900 DIV.
 CONTROL ROD 7 = 270 DIV.
 RECIRCULATION FLOWS:
 PUMP NO. 1 = 10,000 gpm
 PUMP NO. 2 = 5,500 gpm
 FEEDWATER = 46,500 lb/hr
 PRESSURE = 1,000 psig

REACTOR POWER = 12 MWT
 CONTROL ROD BANK = 900 DIV.
 CONTROL ROD 7 = 260 DIV.
 RECIRCULATION FLOWS:
 PUMP NO. 1 = 9,850 gpm
 PUMP NO. 2 = 6,950 gpm
 FEEDWATER = 46,500 lb/hr
 PRESSURE = 1,000 psig

FIGURE 3a
 DATA FROM RECIRCULATION FLOW
 CHANGE TEST

REACTOR POWER = 12 MWT
 CONTROL ROD BANK = 900 DIV.
 CONTROL ROD 7 = 253 DIV.
 RECIRCULATION FLOWS:
 PUMP NO. 1 = 9,850 gpm
 PUMP NO. 2 = 7,900 gpm
 FEEDWATER = 46,500 lb/hr
 PRESSURE = 1,000 psig

REACTOR POWER = 12 MWT
 CONTROL ROD BANK = 900 DIV.
 CONTROL ROD 7 = 265 DIV.
 RECIRCULATION FLOWS:
 PUMP NO. 1 = 9,700 gpm
 PUMP NO. 2 = 8,700 gpm
 FEEDWATER = 43,000 lb/hr
 PRESSURE = 1,000 psig

REACTOR POWER = 12 MWT
 CONTROL ROD BANK = 900 DIV.
 CONTROL ROD 7 = 273 DIV.
 RECIRCULATION FLOWS:
 PUMP NO. 1 = 9,650 gpm
 PUMP NO. 2 = 9,650 gpm
 PRESSURE = 1,000 psig

FIGURE 3b
 DATA FROM RECIRCULATION FLOW
 CHANGE TEST

TABLE I

OPERATIONAL TESTS ON INSTRUMENTED ASSEMBLIES

<u>Test</u>	<u>Objective</u>	<u>Reactor Condition</u>	<u>Measurements</u>
1. Single-phase pressure drop vs. flow test	To determine single-phase pressure drop as a function of flow. Provides a flow check on relative accuracy of flowmeters and pressure drop systems.	Power = 0 MWT, Recirculation flow approx. 4000 gpm	Channel flow, recirculation flow, fuel and riser pressure drop
2. Temperature vs. Pressure Tests	To determine accuracy of thermocouple readings during irradiation.	Power = 1.5 MWT, reactor heating, 10,000 gpm recirculation flow	Reactor pressure and all thermocouples.
3. Recirculation Flow Tests	To determine two-phase pressure drop at various flow rates. To determine flowmeter response rates.	Power = 12 MWT Recirculation flow increasing from 1200 to 2000 gpm.	Channel flow, in-core fluxes, inlet temperature, riser pressure drop.
4. Feedwater Transient Test	To determine response of instrumentation systems to changes in coolant inlet subcooling caused by increased feedwater flow.	Power = 12 MWT Recirculation flow 20,000 gpm	Channel flow, in-core fluxes, inlet temperature riser pressure drop.
5. Power Level Tests	To determine response of instrumentation systems at various power levels.	Power level increasing from 12 to 25 MWT, recirculation flow 20,000 gpm.	Channel flow, in-core fluxes, inlet temperature, riser pressure drop.
6. Flux wire irradiation tests	To determine calibration factor of in-core ion chambers, and to determine stability and sensitivity of ion chambers.	Power level = 30 MWT Recirculation Flow = 20,000 gpm	In-core neutron fluxes in all ion chambers.
7. Instrument Life Tests	To determine instrument lifetime and causes of failure for the specific design of the instrumentation used.	Not specified.	All instruments recording.

the flow; however, a repeatable relationship was not obtained. Since the flow readings are thought to be accurate, the pressure drop data, appears to be inconsistent. The performance of the ΔP system during loop calibration in single-phase and two-phase flow is presented in GEAP-3896-AEC, to be issued soon.

(d) Feedwater Transient Tests

With the power level at 12 MWT and the total recirculation flow at 20,000 gpm, the feedwater flow was increased on a slow ramp from about 45,000 to about 65,000 lb/hr. A slight reduction of the core inlet temperature was noted during the transient (approximately 2°F.) and the power level of the reactor increased by about 1 MW.

(e) Power Level Test

Recordings were made of flow, riser pressure drop, core inlet temperature and neutron flux as the reactor power level was raised from 12 to 25 MWT. Figure 4a is an example of the data taken from fuel assembly 6E. The high frequency character (about 8 cps) of the flow oscillations can be clearly seen. Figure 4b is an example of the neutron flux and flow oscillations taken from Assembly 2F. The high frequency components of the neutron flux and flow oscillations are 60 cycle electrical noise picked up in the readout system. Note that no obvious correlation exists between the neutron flux and flow oscillations. Again, the high frequency flow oscillation (approximately 8 cps) is apparent.

(f) Flux Wire Irradiation

Two flux wires (99.8 per cent Cobalt) were irradiated in instrumented assemblies 6E and 2F on August 18, 1961. The wires were inserted into guide tubes and irradiated at a reactor power of 30 MWT for 10 minutes. Upon removal from the core, the irradiated wires were stored in the spent-fuel pit in VBWR for several days. At the end of this time, one of the irradiated wires (from Assembly 2F) was lost. A weld bead which connected the cobalt wire to its leader wire fractured, causing the wire to be lost at the bottom of the spent-fuel pit.

A scan of the remaining wire and an estimate of the absolute flux level were made. Figure 5 shows the flux profile and the readings of the ion chambers based on the initially assumed calibration of 7×10^{-18} amps/neutron/cm² sec.

As a result of the wire irradiation, the ion chamber calibration factors were corrected. The new factors for assembly 6E are:

Upper Ion Chamber	1.50×10^{-18} amps/neutron/cm ² sec
Middle Ion Chamber	2.10×10^{-18} amps/neutron/cm ² sec
Lower Ion Chamber	2.72×10^{-18} amps/neutron/cm ² sec

The accuracy of these absolute flux determinations are limited to plus or minus 10 to 20 per cent. This is due in part to the technique of insertion and withdrawal of the flux wires during irradiation, and in part to the theoretical limitations in estimating the fraction of neutrons seen by the flux wire that are thermal neutrons.

FIGURE 4b
POWER LEVEL TEST

REACTOR POWER = 12 MWT
 CONTROL ROD BANK = 900 div.
 CONTROL ROD 7 = 101 div.
 PRESSURE = 1,000 PSIA
 RECIRCULATION FLOWS:
 PUMP NO. 1 = 9,850 GPM
 PUMP NO. 2 = 4,600 gpm
 FEEDWATER FLOW = 49,500 LB/HR

FIGURE 4a
POWER LEVEL TEST

REACTOR POWER = 23 MWt
 PRESSURE = 980 PSIG
 CONTROL ROD BANK = 900 div.
 CONTROL ROD 7 = 466 div.
 FEEDWATER = 77,000 LB/HR
 RECIRCULATION FLOWS :
 PUMP NO. 1 = 1,000 GPM
 PUMP NO. 2 = 9,500 GPM

530-28

530-5

FIGURE 5 THERMAL FLUX DISTRIBUTION IN ASSEMBLY 6E AUGUST 18, 1961
30 MWT REACTOR POWER, 20,000 GPM RECIRCULATION FLOW

Summary of Instrument Performance

The instrumented assemblies were installed in VBWR on June 20, 1961, and irradiation began on June 27. By December 16, 1961, they had the following exposures:

<u>Instrumentation No.</u>	<u>Fuel No.</u>	<u>Core Position</u>	<u>Exposure MWD/T</u>
A	6E	F-7	1946
B	2F	G-8	2172

In general, the flowmeters and ion chambers were very successful in giving reliable test data. However, each type of instrument was subject to operating difficulties.

(a) In-Core Flowmeters

The flowmeters suffered from deterioration of the permanent magnets. The design depended upon magnetic flux coupling of three permanent magnets located in the hub of the turbine rotor. For each revolution of the rotor, the changing magnetic flux caused three voltage pulses to be generated in the pickup coil. Thus, the flow rate was proportional to the frequency of the pulses. However, under high radiation fields, the magnetic field strength of the permanent magnets was reduced to a fraction of its original value. The voltage pulse signals became weaker until, eventually, it was difficult to distinguish between signal and 60 cps electromagnetic noise. Moreover, the loss in field strength of the magnets was unequal. This resulted in a signal waveform that was erroneously interpreted by the frequency-to-voltage readout converter. The converter sometimes counted two pulses and sometimes three. Thus, the frequency-to-flow calibration was eventually lost.

During August 1961 the upper flowmeter in Assembly 6E developed an open circuit in the pickup coil rendering it unusable. By October the signal strength in the remaining two flowmeters had deteriorated to the point where no readout could be made. Efforts to reinstate the flowmeters to reliable operation were largely unsuccessful due to the open-loop magnetic circuit design. A metal locator technique was tried to rescue the signal. This technique used an oscillator to drive the pickup coil and the change in inductance was monitored with a bridge circuit. Although this technique worked quite well during a bench test, insertion of a pickup coil into a flowmeter body caused enough magnetic shunting to make this approach unuseable. After a three-week reactor shutdown in October 1961, the two flowmeters were giving fairly strong flow signals, suggesting that there was a magnetic recovery during the shutdown. Low-noise differential amplifiers were used to increase the signal strength with limited success. A useable flow signal was obtained from assembly 6E for about one day and from assembly 2F for several weeks.

Design changes in the flowmeter are in progress in cooperation with the Potter Aeronautical Corporation. The proposed changes will result in a closed magnetic loop. The flow signal will be obtained by measuring the opened or closed state of the magnetic circuit as the turbine rotor revolves and will not depend upon the magnetic field strength of a permanent magnet.

(b) Ion Chambers

The only difficulty encountered with the ion chambers was that moisture (caused by condensing steam which had leaked from a flange on the reactor vessel head) penetrated the junction box for the electrical terminal strips. The moisture resulted in a leakage current which changed the effective calibration of the ion chambers. Otherwise, the ion chambers were operating satisfactorily when removed.

(c) Thermocouples

A problem of calibration was encountered with the thermocouple system at significant reactor powers. Steady-state measurements of the inlet and outlet core coolant temperatures were made with the reactor at 15 MWt and greater. The measured temperature rise across the core was 2 to 4°F. less than the subcooling temperature difference calculated from a heat balance. It was postulated that the difference was due to unequal gamma heating in the interior of the inlet and outlet thermocouples. The lower thermocouples were in a much higher radiation field than the upper thermocouples. The gamma heating was presumed to be causing a temperature gradient between the ungrounded, sheathed couple and the coolant.

No thermocouple calibration changes were noted as a function of accumulated exposure.

(d) Pressure Drop Systems

The calibrations of the pressure drop systems were erratic due to several factors:

- (1) Zero-shifts in the variable-reluctance transducers and bridge readout systems.
- (2) Errors caused by varying static pressure differences in the two, long vertical pressure lines connecting to each transducer. These errors were caused by steam void accumulations within the pressure lines and/or by transient temperature differences in the two lines. The back-purge system was able to reduce these effects, but did not eliminate them completely.
- (3) The pressure drop signal had a large amplitude fluctuation which did not have any obvious correlation with the in-core flow fluctuation. Figure 4a shows an example of the oscillation.

Instrument Response

The response of the instrumentation systems are tabulated in Table II. The slow system refers to the readout system designed for the instrumented assemblies and located in the VBWR control room. The fast readout system used for the operational tests described in Section 1, above, refers to Sanborn direct writing oscillograph recorders.

TABLE II
RESPONSE OF INSTRUMENTATION IN INSTRUMENTED ASSEMBLIES

<u>Type of Instrument</u>	<u>Slow Readout System</u>		<u>Fast Readout System</u>	
	<u>Response (1)</u>	<u>Accuracy</u>	<u>Response (1)</u>	<u>Accuracy</u>
Flowmeters	1/2 cps	$\pm 1\%$	10 cps	$\pm 1\%$
Differential Pressure	1/2 cps	(2)	(5)	(2)
Thermocouples				
Temperature, °F	1/2 cps	$\pm 2\%$	<1 cps	$\pm 2\%$
Core ΔT	1/2 cps	(6)	<1 cps	(6)
Ion Chambers	1/2 cps	$\pm 10\%$ (3)	~ 50 cps (4)	$\pm 10\%$ (3)

-
- (1) The response is expressed as the approximate frequency at which the gain is down 3 db.
- (2) See the remarks on the calibrations of the pressure drop systems in Section 2d above.
- (3) Based on best theoretical accuracy of wire irradiation calibration to determine absolute neutron flux level.
- (4) Limited to response of direct writing oscillograph recorder or other readout technique.
- (5) The response limit of the fast system was not determined and is a function of the arrangement of the pressure line installation.
- (6) Error of 2 to 4° F. compared to heat balance calculated ΔT of 5 to 25° F.

TASK IB - FUEL FABRICATION DEVELOPMENT

1. TASK OBJECTIVE

The purpose of Task IB is to study the performance in VBWR and in the Consumers Big Rock reactor of fuel assemblies which hold promise of having reduced fuel fabrication costs, long fuel life, and high power density. Approximately twelve fuel assemblies utilizing potential low cost fabrication processes will be designed, fabricated, and irradiated in the VBWR.

The work under Task IB will include investigations of potential reduced-cost fuel element fabrication processes and design, fabrication, irradiation, examination, and performance evaluation of the assemblies. Also included is the mechanical vibration testing of a dummy fuel test section simulating the initial Consumers reference fuel.

2. HOT SWAGED FUEL RODS

Hot swaging efforts reported last quarter included the fabrication of ten 16-mil clad fuel rods for VBWR irradiation. Hot swaged UO_2 densities ranged from 94.8 to 95.4 per cent of theoretical. Metallographic examination of the type 304 ELC stainless steel cladding and the -20 mesh fused UO_2 used in the fabrication of the fuel rods has been essentially completed with the following conclusions:

Cladding Material

The grain size as measured metallographically corresponded to No. 7 on the ASTM scale.* Some retained cold work was evident. Tensile testing indicates the clad to be approximately 1/8 hard. No precipitates were observed in the grain boundaries. Figure 6 is a typical photomicrograph of the cladding.

Uranium Dioxide

Figure 7 is a macrophotograph of a cross section of a hot swaged fuel rod with an apparent UO_2 density of 95.2 per cent of theoretical. Examination of the photograph reveals the extremely "close" packing of the particles. The void areas around the periphery of the fuel were caused from "pull-out" during polishing. Examination at higher magnifications revealed that particles have deformed to conform to shapes of nearest neighbor particles.

Metallographic examination to determine the degree of inter-particle sintering was inconclusive. Figure 8 shows photomicrographs of a typical area, in the cross section mentioned above. It appears from the unetched photograph that at least surface sintering has taken place. The photomicrograph of this same general area after etching indicates that no sintering has taken place. Probably the best evidence of interparticle sintering is the fact that after parting the clad of a hot swaged fuel rod one must actually break the UO_2 column to obtain a cross section as shown

*Mean value of 64 grains per square inch of image at 100X.

in Figure 7. Further evidence was given in the previous quarterly report where the clad was removed from fuel section leaving the UO_2 intact. Examination of Figure 8 reveals that higher oxides are present in some of the etched particles. This could very well contribute to the plasticity observed in some of the particles.

An analysis of the UO_2 before and after hot swaging the fuel rods for irradiation was performed and is shown in Table III. The high iron and chromium analyses is due to stainless steel shavings from a machining operation during fabrication and can be ignored. Note that the residual gas content of the powder was decreased from 0.43 to 0.26 cm^3 per gram of UO_2 by the hot swaging process.

Thin Wall Cladding

Two fuel rods clad with 10-mil stainless steel have been hot swaged to a UO_2 density of 94.5 and 95.5, respectively. The overall appearance of the fuel rods was excellent. Destructive testing of the rods is in progress.

TABLE III
 UO_2 POWDER ANALYSIS BEFORE AND AFTER HOT SWAGE

	<u>As Received</u>	<u>After Hot Swage</u>
N	105 ppm	105 ppm
C	65 ppm	30 ppm
Cl ⁻	<2 ppm	<2 ppm
F ⁻	<10 ppm	<10 ppm
Free U metal	~0.07%	~0.08%
O to U ratio	2.001	1.986
Particle Density (Ave.)	10.83 gm/cc	10.81 gm/cc
Gas Evolution (1450°C)	0.43 cc/gm	0.26 cc/gm
Al	15 ppm	20 ppm
Bi	2 ppm	2 ppm
B	<0.2 ppm	<0.2 ppm
Cd	<2 ppm	<2 ppm
Ca	10 ppm	10 ppm
Cr	<8 ppm	100 ppm
Co	<2 ppm	<2 ppm
Cu	3 ppm	5 ppm
Fe	20 ppm	500 ppm
Pb	<1 ppm	<1 ppm
Mg	<1 ppm	<1 ppm
Mn	<10 ppm	15 ppm
Mo	<3 ppm	<3 ppm
Si	10 ppm	20 ppm
Ag	0.5 ppm	0.5 ppm
Na	<35 ppm	<35 ppm
Sa	2 ppm	2 ppm
U	<35 ppm	<15 ppm
Zn	<50 ppm	<50 ppm
U-235	4.09 $\pm$ 0.04%	4.05 $\pm$ 0.04%

FIGURE 6
HOT SWAGED CLAD SAMPLE (150 X)

530-6

FIGURE 7
CROSS SECTION OF HOT SWAGED FUEL ROD (9 X)

500 X MAG
HOT SWAGED FUSED UO_2 UNETCHED

530-7

500 X MAG
HOT SWAGED FUSED UO_2 ETCHED (ETCHED 90%
 H_2O_2 , 10% HF)

FIGURE 8
MICROPHOTOGRAPHS OF HOT SWAGED FUSED
 UO_2 SHOWN IN FIGURE 7

3. ISOSTATIC PRESSING OF POWDER COMPACTED FUEL RODS

Some possible service problems which might arise in fuel rods which have been fabricated by the mechanical packing of UO_2 particles are: 1) cladding failures from mechanisms associated with cold worked material, 2) fuel washout if clad should fail, 3) waterlogging failures if pinhole defects should be present in the clad, and 4) low UO_2 thermal conductivity as compared to a pelletized fuel rod. Hot swaging is a process candidate for lessening the problems since the clad is at least partially annealed and the UO_2 column is partially sintered and hence approaching the characteristics of a pellet column.

A second process which could alleviate the proposed problems is the gas-pressure-bonding technique developed at Battelle Memorial Institute. Parembka and Associates at BMI cold pressed -20 mesh fused oxide into pellets and then loaded them into tubes for hot isostatic pressing for three hours at 10,000 psi and 2300°F. Resultant UO_2 densities ranged from 89 to 95 per cent of theoretical.

In view of these results, a program has been initiated with BMI to hot isostatic press fuel rods fabricated at APED by the cold swaging and tandem rolling processes*. The program consists of fabricating and pressing forty 12-inch capsules in four separate runs (10 rods each). The tentative pressing conditions are:

1) 10 rods	10,000 psi	2100°F	3 hours
2) 10 rods	10,000 psi	2200°F	2 hours
3) 10 rods	10,000 psi	2300°F	1 hour
4) 10 rods	10,000 psi	2400°F	$\frac{1}{2}$ hour

It is anticipated that the conditions other than No. 1 will be changed depending on the evaluation of rods from run No. 1.

Test Number 1

The first 10 rods consisted of pressing 5 capsules fabricated by swaging and 5 by tandem rolling. The swaged capsules were clad with type 304 ELC stainless steel and the rolled rods were clad with type 304 stainless steel. The capsule design is shown in Figure 9 and the hot isostatic pressing conditions were 3 hours at 10,000 psi and 2100°F.

Evaluation of Capsules

The evaluation of the capsules consisted of visual observance of capsules and UO_2 samples, UO_2 density determinations, metallography of clad and UO_2 and longitudinal tensile testing of clad specimens.

The appearance of the capsules after pressing was good except for collapse of the clad at the UO_2 end plug interface, visible in Figure 10.

Short segments of the capsules were removed for observation of the UO_2 column. The procedure consisted of removal of the clad by slitting each

* GEAP-3775, "Fabrication of Fuel Rods by Tandem Rolling" by J. W. Lingafelter.

FIGURE 9

CAPSULE DESIGN FOR ISOSTATIC PRESSING

ROLLED RODS

SWAGED RODS

FIGURE 10
AS-RECEIVED APPEARANCE OF GAS-PRESSURE-BONDED
SWAGED AND ROLLED FUEL RODS

ROLLED ROD

SWAGED ROD

FIGURE 11 ³
TYPICAL APPEARANCE OF UO_2 FROM GAS PRESSURE BONDED
ROLLED AND SWAGED RODS AFTER REMOVAL OF CLADDING

9 X MAG
TANDEM ROLLED FUEL ROD SECTION

9 X MAG
COLD SWAGED FUEL ROD SECTION

530-11

FIGURE 12
HOT ISOSTATICALLY PRESSED ROLLED AND
SWAGED RODS

side with a milling machine and lifting it away from the UO_2 . Figure 11 shows the UO_2 column to be intact. Subsequent handling of the column for photographs did not result in any crumbling or breakage, thus providing a good indication of interparticle sintering. Figure 12 is a polished cross section of a rolled and swaged rod isostatically pressed. It appears that there is a larger "average" particle size in the swaged specimen which might be the reason for the higher observed densities of the swaged rods. Note that there are circumferential cracks on the periphery of the fuel column. These cracks were observed in all specimens.

Figure 13 shows typical UO_2 areas in the rolled specimen. Surface sintering is apparent in both the etched and unetched samples.

Even though type 304 ELC tubing was used for cladding, some carbide precipitation was observed in the grain boundaries. Also, second phase constituents were observed in the periphery of the UO_2 fuel column. Figure 14 is typical of the swaged specimens. Note the large grain size in the clad (approximately No. 2 on the ASTM scale) and the granular structure of the fused UO_2 . The second phase constituents have not been identified, but possibly they are stainless steel particles which were scraped from the inside surface of the clad during the relative movement of UO_2 and clad while swaging. The second phase was observed in all swaged specimens and is continuous about the periphery.

UO_2 densities determined before and after pressing are reported in Table IV. Note that there was little increase in density of the swaged rods, and from 2 to 4 per cent increase in the rolled rods. This difference is presumed to be from the difference in starting densities.

Tensile testing of clad samples has not been completed.

TABLE IV
CAPSULE DATA RUN NUMBER 1

Rod No.	<u>UO_2 DENSITY</u>		<u>CLAD DIAMETER (AVE.)</u>		<u>UO_2 TYPE</u>	<u>CLAD TYPE</u>
	<u>Before</u>	<u>After</u>	<u>Before</u>	<u>After</u>	<u>-20 Mesh Fused</u>	
1-S	93.3	93.8	0.4075	0.4060	---	304 ELC
2-S	93.3	93.5	0.4073	0.4060	---	304 ELC
3-S	93.3	94.3	0.4043	0.4027	---	304 ELC
4-S	93.3	93.7	0.4063	0.4040	---	304 ELC
5-S	93.3	92.9	0.4092	0.4085	---	304 ELC
1-R	89	90.6	0.4013	0.4001	Fused	304
2-R	89	90.8	0.4015	0.3997	Fused	304
3-R	89	90.9	0.4015	0.4002	Fused	304
4-R	89	91.5	0.4012	0.4002	Fused	304
5-R	89	92.2	0.4046	0.4034	Fused	304

500 X MAG
HOT ISOSTATIC PRESSED FUSED UO_2 UNETCHED

500 X MAG
HOT ISOSTATIC PRESSED FUSED UO_2 ETCHED

530-12

FIGURE 13
MIRCO PHOTOGRAPHS OF ISOSTATICALLY PRESSED
ROLLED ROD

STAINLESS

UO₂

150 X MAG
NOTE CARBIDES IN STAINLESS CLAD GRAIN
BOUNDRIES AND SECOND PHASE IN UO₂

500 X MAG
NOTE GRAIN STRUCTURE IN UO₂ AND
SECOND PHASE (WHITE)

530-13

FIGURE 14
UO₂-CLAD INTERFACE OF SWAGED SPECIMEN

Test Number 2

The second test consisted of pressing 10 rods, which were clad with 16-mil type 304 ELC stainless steel, for two hours at 10,000 psi and 2200°F. All rods were fabricated by cold swaging. The capsule design was the same as test No. 1 except for a modified end plug design as shown in Figure 9. This modification was made in an attempt to prevent the clad collapsing at the end plug-UO₂ interface shown in Figure 10.

Evaluation of Capsules

Only a preliminary evaluation has been made at this time. The appearance of the rods was excellent with no apparent clad collapsing as mentioned earlier. Metallography samples of all rods have been prepared, but only the cladding has been studied. Figure 15 is typical of all samples and points out the large grains which are prevalent, and the carbides which have been precipitated in the grain boundaries. The measured grain size is 2 on the ASTM scale. Further evaluation of the capsules is in progress.

Test Number 3

The third test also consists of 10 rods fabricated by swaging, but the pressing conditions were changed in an attempt to reduce grain growth in the clad. Results on this test will be reported in the next quarterly report.

Test Number 4

Test No. 4 is in the planning stage, but tentatively will consist of cold-swaged rods, tandem-rolled rods and rods containing mixtures of ceramic-grade and fused UO₂. It is hoped the addition of the ceramic grade powder will increase the activity and hence yield higher pressed densities.

4. FUSED UO₂ PRODUCT EVALUATION

The metallographic and gas evolution characteristics of fused UO₂ is under investigation to determine the extent that second phase constituents and adsorbed gases are present. Two different lots of Spencer fused UO₂ are being evaluated: 3.5 per cent enriched UO₂ and 4.0 per cent enriched UO₂.

Metallography

Metallographic examination of fused UO₂ particles from each of the two lots has revealed similar second phase constituents. Free uranium, uranium nitride, and higher uranium oxides have been identified through X-ray diffraction and metallographic etching techniques. Photomicrographs showing the various second phases found in the two lots are presented in Figures 16 and 17. The second phases have appeared in only a very small percentage of the UO₂ examined. In the present low concentration, they are not believed to constitute an adverse condition.

Gas Evolution

The results of gas evolution determinations by vacuum extraction techniques on 4.0 per cent enriched fused UO₂ is shown in Figures 16 and 17. Figure 18

100 X MAG

100 X MAG

530-14

FIGURE 15

TYPICAL STAINLESS CLAD SPECIMENS FROM RUN NUMBER 2
(NOTE GRAIN SIZE AND CARBIDES IN GRAIN BOUNDARIES)

500 X MAG. ETCHED
HIGHER OXIDES (U_4O_9 AND U_3O_8) IN FUSED UO_2

100 X MAG. UNETCHED
URANIUM NITRIDE (WHITE ppt) IN FUSED UO_2

530-15

FIGURE 16
SECOND PHASE CONSTITUENTS IN 3.5% ENRICHED
FUSED UO_2

500 X MAG UNETCHED
URANIUM NITRIDE (WHITE ppt) IN FUSED UO_2

500 X MAG ETCHED
METALLIC URANIUM (LIGHT GRAY) AND URANIUM
NITRIDE (WHITE) IN FUSED UO_2

530-16

FIGURE 17
SECOND PHASE CONSTITUENTS IN 3.5% ENRICHED FUSED UO_2

530-17

FIGURE 18
EFFECT OF PARTICLE SIZE ON AMOUNT OF GAS
EVOLVED FROM FUSED UO_2

FIGURE 19
EFFECT OF EXPOSURE TIME TO THE ATMOSPHERE
ON GAS EVOLVED FROM FUSED UO_2

shows that an increase in gas content is associated with a decrease in particle size. Figure 19 shows the effect of exposure to the atmosphere on the gas content of fused UO_2 . The increase with time of the gas content must be at least in part due to chemisorbed gases.

Procedures for Eliminating Residual Gases

A room temperature evacuation and helium backfill does not effectively remove adsorbed gases from fused UO_2 powder. Samples evaluated immediately before and immediately after pumpdown all release approximately 0.5 cc per gram of UO_2 .

A second procedure under consideration consists of passing the UO_2 through a hydrogen sintering furnace at 1600°C . One batch of 4.0 per cent enriched powder was offgassed by this procedure and the offgassing (as measured by vacuum extraction) was reduced from 0.55 to 0.018 cc per gram of UO_2 . Samples of the powder have been left exposed to the atmosphere and are being analyzed for gas "pick-up." One sample, exposed 126 hours to the atmosphere, released 0.036 cc gram of UO_2 . The particle size distribution of the powder was changed during the offgassing and was probably caused from thermal shock. The particle size distribution before and after processing is shown in Table V.

Other processes under consideration are vacuum extraction at 1000°C and 500°C , respectively.

TABLE V
PARTICLE SIZE DISTRIBUTION BEFORE AND AFTER OFFGASSING

<u>Tyler Screen</u>	<u>Cumulative Weight Percent</u>	
	<u>Before</u>	<u>After</u>
-20 / 28	21.5	18.7
-28 / 35	35.1	32.7
-35 / 48	14.2	16.7
-48 / 65	3.7	4.1
-65 / 100	0.7	1.2
-100 / 150	7.4	7.6
-150 / 200	3.5	5.4
-200 / 270	5.3	2.2
-220 / 325	8.6	1.8
-325	---	9.6
TOTAL	100	100

5. EROSION TESTING DEFECTED POWDER COMPACTED FUEL RODS

Erosion flow testing of four purposely defected powder compacted fuel rod samples was completed this month after 6000 hours of testing. Prior testing up to 2000 hours duration had been performed on 24 samples, which served as a screen to select the four. The purpose of the erosion testing was to determine the nature and extent of erosion of UO_2 from a clad split by the coolant stream.

Results of the erosion testing of the 24 samples up to 2000 hours was reported in the Fourth Quarterly Report, GEAP 3717, and the Fifth Quarterly Report, GEAP 3830. These results indicated favorable erosion resistance of powder compacted fuel if the UO_2 density was greater than 90 per cent T.D. From the group of 24 samples, four samples were selected for an extended test. The samples chosen were fabricated by: cold swaging (92 per cent dense), cold swaging plus post swage anneal at $1000^\circ C$ (92 per cent dense), hot swaging (95 per cent dense), and tandem rolling (88 per cent dense).

The four samples accumulated a total of 6000 hours exposure to 7 fps flowing $536^\circ F$ recirculating water. The samples were examined and weighed after 1000, 2000, 4000 and 6000 hours of testing. The results of weight measurements after each examination are tabulated in Table VI.

TABLE VI
WEIGHT MEASUREMENTS OF EROSION SAMPLES

<u>SAMPLE</u>	<u>START</u>	<u>TOTAL WEIGHT, GRAMS</u>			
		<u>1000 hrs.</u>	<u>2000 hrs.</u>	<u>4000 hrs.</u>	<u>6000 hrs.</u>
Cold Swaged	70.4037	70.4053	70.4114	70.4032	70.3960
Cold Swg. Tann.	70.7694	70.7849	70.7870	70.7855	70.7800
Hot Swaged	71.7506	71.7616	71.7659	71.7661	71.7711
Rolled	57.7757	57.7440	57.7356	57.7205	57.7027

A plot of the weight change vs. testing time for each of the samples is presented in Figure 20. Photographs of each sample after 6000 hours of testing are shown in Figures 21 and 22.

The results of the erosion flow tests indicate that swaged fuel rods with UO_2 densities greater than 90 per cent T.D. are highly resistant to UO_2 removal through a clad defect by the coolant stream. The 88 per cent dense tandem rolled sample lost considerably more UO_2 than the swaged samples, but not to the extent as to be considered excessive.

6. VIBRATORY COMPACTION STUDIES

The compaction of UO_2 by electro-magnetic vibration techniques was only briefly studied during the reporting period. The work that was done concerned itself with an extension of the "next door neighbor" studies reported in the Sixth Quarterly Report, GEAP 3884.

FIGURE 20 WEIGHT CHANGE VS FLOW TEST TIME FOR SWAGED AND ROLLED EROSION SAMPLES TESTED IN 536°F, 7 FPS, FLOWING WATER

TANDEM ROLLED

COLD SWAGED

COLD SWAGED & ANNEALED HOT SWAGED

530-20

FIGURE 21
EROSION SAMPLES AFTER 6000 HOURS FLOW TESTING IN
536°F 7FPS FLOWING WATER (4X MAGNIFICATION)

TANDEM ROLLED

COLD SWAGED

COLD SWAGED AND ANNEALED

HOT SWAGED

530-21

FIGURE 22
APPEARANCE OF EROSION SAMPLES SHOWN IN FIGURE 21
AFTER SECTION THROUGH SLIT IN CLAD (4 X MAGNIFICATION)

The next door neighbor investigations have lead to the conclusion that the system of successive binary maxima will lead to compositions which are maximum for that particular system. For this method, the first fraction is combined with some second fraction and the composition of the binary maximum determined. Using this composition as the one fraction, the third fraction is combined and a new binary maximum obtained. This composition is now used with the next fraction, etc. The final composition will be the maximum for those particular sizes in a particular geometry. A topical report covering the work on electro-magnetic vibratory compaction is in preparation.

7. DESIGN OF POOL FACILITIES FOR BIG ROCK POINT

The design of fuel inspection and handling equipment for the Big Rock development fuel is approximately 10 per cent completed. Major items to be designed for fuel inspection are: (1) fuel elevator, (2) gamma scan device and recorder, (3) periscope and (4) several manually operated remote handling tools. The design is proceeding on the basis that the following operations may be performed on the Big Rock Development Fuel in the pool.

- A. Visual inspection of the fuel through the periscope at 1X and 6X magnification with provision for photos.
- B. Gamma scan of the fuel bundle.
- C. Gamma scan of individual fuel rods.
- D. Removal and replacement of individual fuel rods within the bundle.

8. DEVELOPMENTAL FUEL ASSEMBLIES

The first batch of eight developmental fuel assemblies for use in the Big Rock Point reactor have as a goal the achievement of reduced fabrication costs while maintaining the same performance requirements of 45 kw/l and 10,000 MWD/T as the initial fuel load. Locations for these assemblies have been chosen near the center of the core so that the initial power density in these elements will be equal to or greater than 45 kw/l. The planned locations are shown in Figure 23.

Since the performance of these assemblies is to be the same as the initial fuel load, the enrichment of these developmental assemblies has been chosen so that these elements will have nearly the same reactivity as the original fuel. The design of the developmental assemblies differs from that of the initial fuel in that the cladding thickness has been reduced from 19 mils to 10 mils and the number of fuel rods per assembly has been reduced from 144 to 121. As a result of these changes, the enrichment requirement of the development fuel is lowered to 2.7 per cent. A slightly lower water to fuel ratio is also anticipated in the developmental elements in order to preserve as nearly as possible the same operating reactivity coefficients as in the original core. Although the physics analysis of this fuel is not yet complete, a ratio in the neighborhood of 2.5 is expected.

The loading of the developmental assemblies into the Big Rock reactor introduces another factor which may affect the power distribution. Although the

530-22

FIGURE 23

BIG ROCK 50 MW REACTOR SHOWING LOCATIONS
OF THE FIRST 8 DEVELOPMENTAL ASSEM.

operating reactivity of the developmental fuel will be nearly equal to the remainder of the core, the fact that the thermal neutron absorption of the new elements is appreciably smaller is expected to lead to a higher thermal neutron flux in these elements. Since this may increase the local peaking in the original fuel, a study has been initiated to determine the power distribution in the Big Rock core with the developmental elements inserted. Cross section information for the two types of fuel is being generated for use in two dimensional diffusion theory calculations. The study of the Big Rock Point core under various loading and power conditions is expected to continue throughout the development period.

9. TASK IB VBWR FUEL ASSEMBLIES

The exposure accumulated through December 20, 1961, on the Task IB VBWR assemblies is presented below:

<u>Assembly No.</u>	<u>Description of Assembly</u>	<u>MWD/T</u>
1S	Swaged pellets, 12-mil clad, 3-defected swaged powder rods, 2- 83% dense vibratory comp. rods	1470
6S	Swaged pellets, 8 and 10-mil clad	1885
4S	Big Rock R & D prototype mechanical design, 14-mil clad	920
3S	Tandem rolled, 15-mil clad	1040
2S	3-pass cold swaged, 10 and 15-mil clad	1940
5S	2 and 3 pass cold swaged, hot swaged, 8, 10, and 16-mil clad	200
7S	2 pass cold swage, 16-mil clad vibratory compacted, 10-mil clad	815

During the reporting period, two new purposely defected swaged fuel rods were added to assembly 1S along with two non-defected 83 per cent dense vibratory compacted rods. These four rods are discussed below. The first defected swaged fuel rod (15-mil hole through the clad) has accumulated an exposure of 800 MWD/T over a period of four months, and continues to perform satisfactorily. Measured activity release from the defect has been 5 μ c/cc.

Four new fuel rods were added to assembly 1S on December 19, 1961: two vibratory compacted, one defected cold swaged, and one defected hot swaged. The two vibratory compacted rods had a UO₂ density of 83 per cent T.D., and were clad in 10-mil stainless. The two defected swaged rods each contained a 0.020-inch wide x 1/2-inch long slit in the stainless clad. Clad thickness in each case was 0.015 inch. One rod was a 90 per cent dense cold swaged rod. The second rod was a 95 per cent dense hot swaged rod. The defects

are shown in Figure 24. The defect in each case is located eleven inches above the lower end of the fuel rod, which is the approximate location of the maximum heat flux. Maximum heat flux to date has been $\sim 300,000$ Btu/hr-ft². The two defected rods have apparently performed satisfactorily during their first two weeks of irradiation. Maximum measured activity from assembly 1S after the two slit type defects were introduced has been 20 μ c/cc.

10. MECHANICAL VIBRATION TESTING

Analyses of possible self-sustained vibration mechanisms do not explain the multi-rod behavior that was measured in August. In particular, the quadratic relationship of amplitude to frequency is more characteristic of the shock loading of turbulent pressure fluctuations. This vibration might then be damped free vibration; however, the intensity of turbulent fluctuations generally increases in proportion to the mean velocity squared, but little dependence of amplitude on mean velocity was found in the measurements. It is hoped that a further test will help clarify this matter.

A single rod test is being planned which will have the following features:

- A. No rod coupling to obscure energy conservation.
- B. The range of mass velocity may be greatly increased.
- C. Pressure drop measurements may be related to rod motion, and
- D. The effect of turbulence level may be illuminated by altering the entrance geometry and inserting screens.

This instrumented rod will be one of those used in the multi-rod tests. The single rod results are expected to be more informative than the multi-rod work from the standpoints of understanding and general prediction of vibration.

HOT SWAGED ROD

COLD SWAGED ROD

530-23

FIGURE 24
APPEARANCE OF SLIT TYPE DEFECTS IN SWAGED RODS.
SLIT IS 0.020 INCHES WIDE X $\frac{1}{2}$ INCH LONG

TASK II - STABILITY, HEAT TRANSFER, AND FLUID FLOW

1. TASK OBJECTIVE

Hydrodynamic, thermodynamic, and nuclear effects on reactor performance and stability are to be evaluated analytically and also experimentally in the Consumers Big Rock Reactor. Assurance of reactor stability under conditions of R & D testing is an objective of the program, as is the use of these results and methods to evaluate performance of the 300 MWe HPD conceptual design.

Out-of-pile heat transfer and fluid flow development testing will obtain data in such critical areas as:

- (a) Burnout heat transfer
- (b) Pressure drop (single and two-phase flow)
- (c) Hydraulic stability (forced and natural circulation)

A wide range of operational and design parameters will be evaluated.

2. HEAT TRANSFER TESTS

The nine-rod burnout assembly has been operated at heat fluxes up to 75 per cent of single rod burnout. Operation has been at various qualities for mass velocities of 0.56×10^6 and 1.12×10^6 lb/hr ft² at 1000 psia with nominal rod spacing (maximum clearance). No burnout has occurred under these conditions.

Since these tests, the rod connectors, spacers and method of insulating from ground have been re-designed to increase assembly lifetime.

No tests were performed during December because of scheduling. However, further testing will begin on January 2, again with nominal spacing but at 1400 psia and both mass velocities. Following this, actual burnout points will be sought.

No data will be obtained with eccentric rod adjustments because thermal bow has been found to excessively distort the spacing.

TASK III - PHYSICS DEVELOPMENT

1. TASK OBJECTIVE

Long fuel lifetime and economically yet effective reactor control is the objective of physics control studies, one of four major areas of study in this task.

A second area is the study of hot spot reduction by power flattening. This will influence the degree of high power density that can be achieved within heat transfer limitations.

A Large High Power Density conceptual design study is to be carried out, and results of the Consumers Big Rock testing will be utilized and applied in the 300 MWe conceptual design evaluation.

The optimizing and scheduling of control is the objective of a computer and data-logging system which is to be studied for feasibility, designed, installed, and operated in conjunction with the Consumers Big Rock Plant.

2. ECONOMICAL METHODS OF POWER DISTRIBUTION ANALYSIS

During the developmental operation of the Big Rock reactor, a large number of full-core diffusion theory calculations will be made to determine power distributions and reactivity. For these analyses, numerical accuracy in excess of about 0.5 per cent in reactivity and 10 per cent in local power density is not essential. With the objective of effecting economy in these future calculations, a study was made to determine what savings could be achieved by simplifying the mesh description of the core without unacceptable sacrifices in numerical accuracy.

The two dimensional (x-y) power distribution in one quadrant of the Big Rock core was calculated employing the following variations in the geometrical description:

- A. Variation in mesh spacing from 3.1 cm to 18.8 cm.
- B. Replacement of the reflector with group independent neutron current boundary conditions.
- C. Replacement of the reflector with group dependent neutron current boundary conditions.

The results of this study indicate that essentially all of the above methods can yield adequate accuracy in the calculated reactivity, but none of the methods employing coarse descriptions of the mesh (spacing > 6.3 cm) yield high accuracy in the power distribution. The large thermal neutron flux gradients near the core-reflector interface are responsible for a large part of the disagreement between the several methods of analysis. As the core size is increased, the relative importance of the peripheral fuel elements diminishes, and the agreement would be expected to improve.

It is concluded that coarse mesh spacings and replacement of the water reflector with an albedo can introduce numerical errors in the calculated

fuel bundle power of the order 10-20 per cent. Such accuracy may be adequate for scoping calculations but would require additional precision for most applications. Some improvement can be achieved by decreasing the mesh spacing near the core-reflector interface. It should be noted that gains achieved by reductions in the total number of mesh points can sometimes be lost by the increased time required for convergence.

Methods and Results

In calculating the gross reactor power distribution, nuclear cross sections are averaged over each fuel bundle, by the appropriate flux weighting factors. Thus, an upper limit on the mesh spacing in the difference equations is set by the fuel bundle width. The lower limit on the mesh spacing is determined by both the dissimilarity which exists between regions in the reactor and the numerical accuracy which is required.

Figure 25 shows a comparison plot of the thermal neutron flux shapes for mesh spacing of 3.13 cm and 6.26 cm in the Big Rock core. The calculated reactivity differed in the two cases by 0.28 per cent. The fluxes show reasonably good agreement even in the high gradient regions near the reflector interface. The two-dimensional diffusion theory code PDQ was used for this and other calculations reported here.

Table VII summarizes a series of calculations for a typical uncontrolled core quadrant. The reference case is for the core at 288°C. with 20 per cent void volume in the moderator and 6 cm mesh spacing. The next column shows the results with a mesh spacing of one fuel bundle (18 cm). The power distribution can be seen to be significantly flatter in the latter case. The third and fourth columns are the results of 18 cm mesh spacing and the water reflector replaced by energy independent values of the boundary condition $\phi/\nabla\phi$ of 12 cm and 15 cm respectively. The final column shows the results for the group dependent current boundary conditions $\phi/\nabla\phi$ equal to 9 cm, 10 cm, and 33 cm, respectively. The results of Table VII can be seen more clearly in the curves shown in Figure 26. This is a plot of the power distribution along the x-axis for the five calculations tabulated in Table VII. The general lack in agreement in the flux shapes at the core boundary can be seen. In none of the approximations was an attempt made to preserve the thermal flux rise at this boundary. In all of the calculations described, the core contained one-half steel channeled elements and one-half zircaloy channeled elements with all control rods withdrawn.

A more severe test of the approximation of coarse mesh spacing was made with a core with all zircaloy channels and one-half of the control rods inserted. The fine and coarse mesh results are shown in Figure 27. Fortunately, the agreement for this core is better than that for the uncontrolled core. In general, it is concluded that none of the methods tested yields the accuracy required for most calculations and therefore mesh spacings of 6 cm or less and a complete description of the reflector are recommended for most future calculations.

3. THE USE OF DISTRIBUTED POISON TO REDUCE AXIAL POWER PEAKING

The overall power peaking factor may be considered to be made up of several components such as: axial factor, radial factor, local corner rod factor,

FIGURE 25 THERMAL FLUX DISTRIBUTION QUARTER CORE

FIGURE 26 RELATIVE POWER DISTRIBUTION QUARTER CORE

ALL ZIRCALOY CHANNELS

1.969 (2.083)	1.636 (1.623)	1.127 (1.126)	.753 (.660)
1.618 (1.614)	1.331 (1.258)	.901 (.878)	.600 (.521)
1.078 (1.100)	.854 (.894)	.562 (.688)	.433 (.382)
.691 (.627)	.531 (.545)		

530-26

REFERENCE X.XXX 6 CM. MESH
 TEST (X.XXX) 18 CM. MESH

FIGURE 27
 RELATIVE FUEL ELEMENT POWER

TABLE VII

Relative Fuel Element Power

1	2	3	4
5	6	7	8
9	10	11	12
13	14		

Geometry of Core Quadrant

Element No.	Refer- ence Case	Coarse Mesh	% Devia- tion	Coarse Mesh $\phi/\nabla\phi =$ 12 cm.	% Devia- tion	Coarse Mesh $\phi/\nabla\phi =$ 15 cm.	% Devia- tion	Fine Mesh $\phi/\nabla\phi =$ 9, 10, & 33 cm.	% Devia- tion
1	1.302	1.170	(10.1)	1.507	(15.7)	1.408	(8.1)	1.408	(8.1)
2	1.209	1.105	(8.6)	1.387	(14.7)	1.318	(9.0)	1.296	(7.2)
3	1.028	.973	(5.4)	1.176	(14.4)	1.159	(12.7)	1.074	(4.5)
4	.841	.778	(7.5)	.758	(9.9)	.805	(4.3)	.719	(14.5)
5	1.244	1.147	(7.8)	1.411	(13.4)	1.338	(7.6)	1.327	(6.7)
6	1.172	1.109	(5.4)	1.301	(11.0)	1.257	(7.3)	1.243	(6.1)
7	.974	.970	(0.4)	1.075	(10.4)	1.079	(10.8)	1.012	(3.9)
8	.764	.742	(2.9)	.671	(12.2)	.725	(5.1)	.650	(14.9)
9	1.104	1.062	(3.8)	1.194	(8.2)	1.173	(6.3)	1.144	(3.6)
10	1.092	1.134	(3.8)	1.020	(6.6)	1.030	(5.7)	1.132	(3.7)
11	.938	1.128	(2.0)	.748	(20.3)	.794	(15.4)	.961	(2.5)
12	.606	.834	(37.6)	.440	(27.4)	.503	(17.0)	.530	(12.5)
13	.887	.853	(3.8)	.743	(16.2)	.786	(11.4)	.757	(14.7)
14	.847	.997	(17.7)	.568	(32.9)	.626	(26.1)	.748	(11.7)

control rod perturbation factor, and overpower allowance. The axial peaking factor in typical boiling water reactors is in the neighborhood of 1.6 and usually represents the largest single contribution to the total peaking factor.

One fairly obvious method of reducing the magnitude of this factor is to add neutron absorbing material to the region of the core when the peak is expected to occur. This method has the advantage of being relatively easy to evaluate since the poison will remain fixed in location and magnitude throughout the operating cycle. Mechanical difficulties associated with the incorporation of such poison are expected to be minor.

Since the poison will remain in the core at the end of the operating period, there is a reactivity penalty involved. The magnitude of the penalty depends, of course, on the amount of correction desired. A study is in progress to determine the relationship between peaking factor reduction and reactivity penalty and thus to evaluate the desirability of incorporating this factor in future fuel element designs.

As a first step in this study axial flux calculations have been performed based on a single energy group and a uniform fuel composition. The center region of the fuel is assumed to contain sufficient poison to reduce K_{∞} to 1, thus providing a flat power distribution in this region. The unpoisoned K_{∞} was then computed to yield $k_{eff} = 1$. The power shape is then flat in the center region and has a cosine shape in the end sections.

On the basis of this simple model, the following expression can be derived for the axial peaking factor:

$$\left(\frac{P}{A}\right)_{axial} = \frac{\sqrt{\frac{k-1}{k_0-1}}}{\sqrt{\frac{k-1}{k_0-1} - \left(1 - \frac{2}{\pi}\right)}}$$

where k_0 is the required infinite multiplication factor without the use of poison and k is the value needed when poison is used. The difference $k - k_0$ represents the reactivity penalty due to the poison. In a typical boiling water reactor k_0 would be approximately 1.024 for a 6-foot long core, and approximately 1.011 for a 9-foot core. Figure 28 shows the axial peaking factor as a function of reactivity penalty for the 6 and 9-foot lengths.

This model admittedly represents an oversimplification of the problem. The effect of the reflector at the top and bottom will tend to reduce the peaking factor for both the poisoned and nonpoisoned cases. More important are the effects of steam voids and control rod locations in an operating reactor, since these produce geometrical variations in the composition of the reactor and thus invalidate the simple assumptions of the preceding analysis. Furthermore, the nonuniform irradiation of the fuel during power operation will result in spatial variations in fuel composition after the reactor has been in operation for some time.

Clearly a more sophisticated calculational method is required to take these factors into consideration. Such a calculation has been initiated and is currently being carried out with the use of digital computer codes.

FIGURE 28 AXIAL PEAKING FACTOR REDUCTION

4. PHYSICS ASPECTS OF THE HPD 300 MW CONCEPTUAL DESIGN

A number of changes have been made in the 300 MW HPD design since the previous physics study was performed. The major changes affecting the physics aspects of the design are:

- A. Reduction in the cladding thickness
- B. Increase in power density
- C. Reduction in the number of fuel rods per assembly
- D. Reduction in control blade thickness
- E. Use of B_4C filled rods for temporary control of the initial core load.

Physics analysis of the revised design has been performed during the current reporting period and is included in the topical report covering this concept.

The reactivity requirements of this core are summarized in Table VIII which gives the reactivity balance for the equilibrium cycle. This cycle is based on loading the reactor in quarter core batches with the fresh fuel being loaded at the periphery. Loading will occur at 7-1/2 month intervals assuming an 80 per cent capacity factor and a discharge-exposure of 15,000 MWD/T.

TABLE VIII

<u>Reactivity Balance - Equilibrium Cycle</u>	
	<u>% k_{eff}</u>
Cold to operating loss	3.5
Xenon and Samarium poisoning	3.5
Maneuvering and power shaping	2.0
Fuel depletion	<u>7.5</u>
TOTAL	16.5

In order to provide the required 16.5 per cent excess reactivity at the start of each operating cycle, the necessary multiplication of the fresh fuel batch is $k_0 = 1.275$. The initial enrichment for the equilibrium cycle is shown in Table IX, together with an estimate of the discharge composition. The enrichment of three rods at each corner of the fuel bundles has been reduced to 0.5 per cent to improve the power distribution. A slightly higher enrichment is needed for the initial fuel loading if the 15,000 MWD/T exposure is to be attained.

TABLE IX

Enrichment Requirements for 15,000 MWD/T

First Core Load

	<u>Central Rods (109)</u>	<u>Corner Rods (12)</u>
Initial U-235	2.75%	2.25%
Discharge U-235	1.34%	0.84%
Discharge Pu	0.60%	0.68%

Equilibrium Cycle

Initial U-235	2.65%	2.15%
Discharge U-235	1.25%	0.76%
Discharge Pu	0.61%	0.69%

The reactor is controlled by a system of cruciform control blades. These blades are arranged in a staggered array on 10-inch centers which permits a maximum blade width of 12 inches to be accommodated. Once the equilibrium loading cycle has been established this system provides adequate control for fuel enriched to reach an exposure of 15,000 MWD/T with the use of only 10-inch blades.

Controlling the initial core of the reactor presents a unique problem since at this time the core consists entirely of clean fuel. During the initial phase of operation it is planned to insert boron carbide filled tubes in the fuel bundles replacing some of the fuel rods. Approximately 12 of these poison rods per bundle will be required for initial control. These poison rods will be removed and fuel rods inserted in their place after the reactivity of the fuel has been sufficiently reduced due to burnup. A 12-inch control blade will be required during this period since the cold to operating reactivity swing is significantly greater with the poison rods inserted.

By reducing the water gap between fuel bundles to one-half inch, the corner rod peaking factor has been improved. Three fuel rods at each corner of the fuel assembly have a lower enrichment of 0.5 per cent than the remaining rods in order to further reduce this peak. Other peaking factors remain essentially the same as in the previous design. Table X lists the power peaking factors applicable to the present conceptual design.

TABLE X

Power Peaking Factors

Axial peaking	1.62
Radial peaking	1.30
Corner rod peaking	1.20
Control rod perturbation	1.15

The combined overall peaking factor is 2.91 to which must be added an overpower factor to allow for small transients, instrument error, etc.

More details concerning the physics aspects of the 300 MW conceptual design are included in a topical report which was prepared during the quarter.

5. SCHEDULING COMPUTER

The competitive proposals for the data logging computer system were opened at a public bid opening October 31. Out of over sixteen companies solicited, eight bids were received and were evaluated for compliance with the contractual and technical requirements of the system. At the request of the AEC San Francisco Office, a technical evaluation of the proposals was prepared by APED and submitted for their consideration. The final evaluation and selection of the successful bidder was done by SAN.

Award of the contract was announced in mid-December by the AEC, and was given to General Electric Company's Process Computer Section in Phoenix. The contract provides for a GE 312 computer and associated system, but excludes the programming required with this application. Programming of the system will be performed by APED.

TASK IV - COORDINATION AND TEST PLANNING

1. TASK OBJECTIVE

This task is concerned with the test planning and design, safety assurance, test equipment procurement, operational scheduling for the R & D program, and maintaining and issuing reports on progress of the program.

2. COORDINATION

A progress review of the R & D program was held in San Jose with representatives of the AEC from San Francisco, Chicago, and Washington on October 2. A detailed presentation of Task III was given along with a summary review of the balance of the program.

A financial summary and descriptive information for the AEC (Form 189) was prepared and submitted in December. Two monthly progress letters, Numbers 21 and 22 for October and November 1961, were prepared and issued.

Continuous liaison between the R & D program and plant design was maintained to assure plant capability to meet the program objectives. Provision to incorporate the scheduling computer into the plant facilities is one continuing facet of this coordination, as is the application of instrumented fuel assemblies in the Big Rock reactor.

3. TEST PLANNING

The schedule and range of tests to be performed as part of the R & D program have been divided into two phases: Phase I will comprise those tests that are generally within the initial core rating and operating limitations, while Phase II will involve higher power level and increased power density tests.

The limits to be observed in the Phase I tests are principally maximum reactor power (157 MWt), maximum power density (45 kw/l), and minimum burnout ratio (1.5). The tests will cover pressures from 800 to 1500 psi, recirculation flows from 60 per cent to 100 per cent of rated, and power levels of 60 per cent to 100 per cent of rated. These have been evaluated in detail with regard to establishment of thermal-hydraulic performance. However, physics limitations due to reactivity and power peaking are still under investigation.

Most of the work to date has been in support of the Final Hazards Summary Report and the plant Technical Specifications. The former has been formally issued and the latter is in the final stages of completion as of the end of this quarter.

Detailed definition of individual tests, along with additional evaluation in terms of operation and safety, will be undertaken during the coming quarter. The requirements imposed by the Final Hazards Summary Report and the Technical Specifications will be met by all Phase I tests which are planned.

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Project Engineer

L. K. Holland

Fuel Development

T. J. Pashos

Fuel

J. W. Longafelter
W. D. Fowler
W. R. DeHollander
E. H. Lees
C. M. Ryer
T. C. Rowland
R. K. Haling

Instrumented Assembly

A. G. Dunbar
A. C. Duckart
J. A. Hodde

Vibration Testing

E. P. Quinn

Stability and Heat Transfer Development

S. Levy

Heat Transfer

E. P. Quinn

Physics Development

D. L. Fischer

Physics Studies

R. K. Haling
N. H. Barth
W. L. Cranor
C. L. Miller
D. M. Ogborn

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L. K. Holland

Coordinating and Test Planning

L. K. Holland