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FUEL CYCLE PROGRAM

A BOILING WATER REACTOR RESEARCH AND DEVELOPMENT PROGRAM

First Summary Report

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1.0 INTRODUCTION AND SUMMARY:

1.1.0 Introduction

The Fuel Cycle Development Program is a basic development program of boiling and other water reactor technology. In scope, it covers the areas of oxide fuel fabrication, irradiation, and examination, the physics of water moderated reactors, and boiling water heat transfer and stability.

This is the first report to the commission of activities carried out under this program. It constitutes a review of the program from April 27, 1959 through July 31, 1960. The basis for the scheduling of the work is discussed in the substance of the report. Subsequent reports issued as monthly, quarterly and topical reports will summarize the progress of this program.

1.2.0 Program Summary to Date

During the period covered by this report, the Vallecitos Boiling Water Reactor, in which Task A and C programs are being carried on, was in the process of a company financed modification program which was designed to give greater experimental flexibility. Schedules for the fuel cycle program were examined and it was concluded that portions of the Task A program should be conducted immediately (May to December, 1959) in order to keep costs of the work as low as possible and to allow initiation of the fuel cycle program at the earliest possible date after the reactor was again returned to service.

The following chronological summary describes the content of work conducted under a contractor-financed advanced requisition, and

brings the technical progress outline up-to-date. Details are summarized in Sections 2 and 3.

<u>Time Interval</u>	<u>Task⁽¹⁾</u>	<u>Description of Work</u>
May, June, July 1959	A-I	<p>(1) Physics analysis for fuel cycle fuel in a three-region core in VBWR. Included were reactivity calculations, two-dimensional flux calculation enrichment studies and void and temperature coefficients.</p> <p>(2) Safeguards analysis of VBWR experiments. Analyses included:</p> <ul style="list-style-type: none"> a. Loss of coolant b. A-C power loss c. Reactivity accidents (initial) d. Analysis of activities occurring as a result of high power operation e. Transient heat transfer analysis of oxide fuel (preliminary) f. Preliminary study of stability considerations. <p>(3) Preliminary Design Analysis</p> <ul style="list-style-type: none"> a. Study of flow and flow distribution in the experimental cores. b. Review of heat transfer & burnout work to establish a basis for the design of fuel cycle fuel. c. Scoping out the range of variables for the test. <p>(4) Mechanical Design</p> <ul style="list-style-type: none"> a. Preliminary design of fixed poison channels (removable and replaceable with low absorber channels) b. Preliminary design of fuel c. Preparations and design supervision for manufacture of channels.

<u>Time Interval</u>	<u>Task⁽¹⁾</u>	<u>Description of Work</u>
August-September October-November 1959	A-I	a. Design completion & Purchase of stainless steel for fixed poison channels.
	A-II	
	A-I	b. Physics studies continued dealing primarily with the compatibility of fuel cycle and other fuels.
		c. Initial preparation of safeguards report for the advisory committee on reactor safeguards.
	A-IIIa	d. Initiate study of instrumentation for fuel assembly design
	A-IVa	e. Initiate study of a rod oscillator mechanism design for use with the stability program
		Restudy of content and goals of program for compatibility with reactor design.
November 1959 to February 1960	A	Continued work on the analysis of internal flows, orificing and heat transfer burnout in the test cores. Studies of coding for these problems were also carried out.
		Preparation and presentation of safeguards information pertaining to fuel cycle experiments to the advisory committee on reactor safeguards. (AEC)
		Review and detailed scoping of the experimental programs to verify content of the fuel cycle development program. Presentation of a scoping document to the commission.
		Fabrication of stainless steel channels and mockup in core to assure operability of the channel removal scheme.
March-April 1960	A-I	Preliminary preparation of procedures for some tests to be conducted under the fuel cycle development program.
		Preparation of procedures for startup tests appropriate to fuel cycle development.

<u>Time Interval</u>	<u>Task⁽¹⁾</u>	<u>Description of Work</u>
May, June, July 1960	A-II	Licensing hearings for VBWR experimental programs.
		Physics analysis of startup situations for the experimental cores.
		Detailed design of fuel assemblies
		Fuel materials list preparation
	A-IIIa	Ordering of UO ₂ , end plate castings Zirconium channels
		Preliminary design of instrumented fuel assemblies
	A-I	Planning and procedures for experimental program and startup
		Instrumentation for startup and experimental program
		Mechanical design and fabrication engineering of fuel assemblies.
		Design of the fuel examination program.
	A-II	Procurement of stainless steel & zircaloy tubing
		Non-destructive testing of tubing
	A-III	Detailed design of instrumented assemblies.
		Initiate procurement of instrument components for the instrumented assemblies
	B-I	Preparation and heat transfer burnout testing of a single rod heater at 1000 psia.
	B-IV	Preparation of two-phase pressure drop testing.
	B-I	Installation and startup of a 1000 psia observational burnout test section.
	B-I	Two-dimensional flow channel experiment installation.

2.0 TASK A - ADVANCED FUEL POWER LIMIT TESTS:

2.1.0 Analysis and Design of Experiments

2.1.1 Summary of Experimental Program

A scoping document, describing the essential nature of results that are expected from the experimental program and the approaches to obtaining the data was presented to the Commission in January 1960. Although relatively few changes have evolved in the intervening period, it is deemed wise to summarize here, for the record, the approach to Task A as it is envisioned at this time. The objectives are:

Design and build a total of 79 fuel assemblies for irradiation in the VBWR in order to:

1. Extend fuel life information on oxide fuel for boiling reactor service to high specific power operation (20 KW/Kg) with irradiation lifetime targets of 15,000 MWD/T. Coincidentally, fuel performance limit investigations designed to extend specific power still further are to be conducted to map out further paths of development.
2. Study the power stability and performance characteristics of an oxide fueled core with natural, forced and dual cycle circulation systems for the purpose of improving design limits of boiling reactor systems.

The irradiation testing program is anticipated to be as follows:

1. An initial two to three month period during which initial testing of power distributions, control rod worth and progressive steps in power until the design point power levels are reached. Concurrently, the preparation of test instrumentation control rod oscillator and other equipment will be tested, checked and made ready for operation.
2. A period of time (3-4 months) during which rod oscillator tests and the pumping investigations will be conducted. Fuel load will be the 4:1 AEC assemblies and General Electric driver assemblies.
3. The introduction to the reactor of the performance limit fuel assemblies with investigations conducted to establish the conditions under which the fuel will be tested. As few interruptions as possible to steady irradiation from this point on.
4. Steady irradiation to obtain the fuel burnups required.

The fuel to be tested in VBWR consists of:

Stainless steel clad fuel

10 - 16 rod fuel assemblies installed in control rod followers.

15 - 16 rod assemblies occupying regular fuel positions.

1 - 14 rod assemblies for use in an instrumented assembly

Zirconium clad fuel

36-16 rod fuel assemblies occupying regular fuel spaces.

2 - 14 rod fuel assemblies in instrumented assemblies.

3 - 16 rod assemblies for isotopic analysis in long term reactivity studies.

12 special fuel assemblies which are to be the subject of a yet unspecified agreement with the Commission.

The experimental objectives for the final one are outlined in Section 2.2.4.

Details of the fuel and its uses are given in Table 4 and discussed in Sections 2.2.0 and 2.3.0. A schedule of fuel irradiation is given in Section 5.

2.1.2 Fuel Life Test Program

The fuel life test assemblies are to be manufactured by present commercial methods, except in the case of the special fuel.

The program is designed to investigate the fuel life characteristics of core size quantities of these fuel assemblies. All of the assemblies, as presently designed, are capable of disassembly and removal of rods. A detailed study has been made of the irradiation examination program with the intent of producing as much information as possible per unit cost. Detailed reporting of the planning of irradiation examination will be reported in a subsequent report. Complete records of chemical analyses and metallurgical and fabrication histories will be provided in the course of the investigations and identifies the major types of measurements to be made.

TABLE 1 SUMMARY OF
MEASUREMENTS FOR FUEL LIFE INVESTIGATION

<u>Test</u>	<u>Type of Measurement</u>	<u>Comment</u>
Corrosion Test	<ol style="list-style-type: none"> 1. Visual observation 2. Recorded chemistry of water 3. Observation of performance of special assembly designed to introduce variables such as vibration & sliding for fretting corrosion investigation. 	Item 1 & 2 are for Zr. & S.S.
Fission gas pressure	Measurements by gas sampling of irradiated elements.	These measurements will be coupled with metallographic examination to relate such things as densification & internal void formation measurements. These made periodically during irradiation.
Hydrogen Migration	Chemical analysis and metallographic examination.	Migration of hydrogen & resulting concentrations in Zr. are known to change physical properties considerably.
UO ₂ Redistribution	Metallographic examination of reference fuel elements to determine the nature of redistribution and the effect on density of fuel.	Fuel element integrity and the expected performance of long life fuels are to be evaluated from this investigation.
Fuel life	<p>Complete detailed chemical, metallurgical and fabrication histories</p> <p>Detailed exposure records observations of failure and the nature of associated phenomena, mechanical distortion, corrosion, etc.</p>	The statistical nature of failures may be ascertained as the irradiation progresses.

2.1.3 The Performance Limit Program

The experimental objectives of this program are to explore some alternatives in UO_2 fuel element design which show the possibility of providing increased specific power for UO_2 fuel.

Two approaches are being made to reach the desired goals.

The first is to operate fuel rods, having a higher enrichment than the rest of the fuel assemblies, in a center melting condition. The second is to improve the thermal conductivities of the fuel by mechanical or other additives. In each case, incremental steps increasing specific power will be taken and the life of the fuel ascertained. Fabrication of the center melt test fuel is planned for November 15. The fuel having increased conductivity will be scheduled as agreement is reached between the Commission and General Electric on specific approaches.

2.1.4 Stability Tests

During an approximate three to four month period, the core fuel will be redistributed to form a core consisting of 60 to 65 fuel cycle assemblies and the remainder of the core of General Electric zirconium clad fuel of the same water to fuel ratio. The present scheduling of this work is shown in Section 5. The tests to be conducted in this period consist of rod oscillator tests and investigations of core performance under variable circulation system conditions.

A. Instrumentation

1. Three fuel assemblies will be instrumented to obtain flow temperature and, if possible, pressure drop information. In-core flux monitors will be provided.
2. A position controlled reactor control rod drive will be provided for cyclic and transient reactivity changes. Displacement will be sufficient to give about 10 per cent change in neutron flux and frequency range sufficient to cover the expected resonance ranges.
3. Instrumentation with proper frequency and signal responses for the rod oscillator tests and autocorrelation testing will be provided including:
 - a. reactor pressure
 - b. steam and water flow
 - c. temperatures of fuel, steam and water
 - d. neutron flux

B. Analytical Study

An analytical program is to accompany the stability test program. Content of this program is expected to be:

1. Prepare equations of motion, energy and reactor physics equations for handling the steady and transient conditions of operation for coding.

2. Interpret and generalize results of rod oscillator and stable power limit tests, also examine maneuvering tests.
3. Study models which have been proposed and make modifications which apply to the VBWR tests in its various modes of operation.
4. Investigate relationships obtained from out-of-pile loop tests for stability.
5. Study models and test data to attempt generalizations of results which apply to as large an area of boiling reactors as possible.
6. Examine the work of other stability studies which pertain to this program.

2.2.0 Design and Analysis of Fuel

The objectives of this subtask are to prepare the production drawings of the fuel in advance of the actual fabrication of the fuel assemblies and to provide the necessary analytical work in preparation of the VBWR core.

2.2.1 Core Configurations

Figures 1 through 3 show a layout of the VBWR cores as presently planned. The Start-up Core will go critical during August 1960 and will be modified from time to time until the High Power Density lattice of Figure 2 has been attained.

FIGURE 1
MODIFIED V-BWR CORE
STARTUP CORE

LEGEND

16 S 2.7 - 16 RODS PER ASSEMBLY
STAINLESS STEEL CLAD
2.7% ENRICHMENT

25 S 3.5 & 4.5 - 25 RODS PER ASSEMBLY
STAINLESS STEEL CLAD
3.5% & 4.5% ENRICHMENT

16 Z 3.5 - 16 RODS PER ASSEMBLY
ZIRCALLOY CLAD
3.5% ENRICHMENT

PROTOTYPE - PROTOTYPE FUEL ASSEMBLIES
FOR SAVANNAH II, CONSUMERS',
HAPO, ETC.

FF - FUEL FOLLOWER ASSEMBLY

SHADED REGIONS DESIGNATE FUEL CORE FUEL

FIGURE 2

HIGH POWER DENSITY LATTICE

- 16 S 2.7 - 16 RODS PER ASSEMBLY
STAINLESS STEEL CLAD
2.7% ENRICHMENT

FIGURE 3
MODIFIED VBNR CORE
HOMOGENIZED LATTICE

LEGEND

- 16 S 2.7 - 16 RODS PER ASSEMBLY
STAINLESS STEEL CLAD
2.7% ENRICHMENT
- 16 S 3.5 - 16 RODS PER ASSEMBLY
STAINLESS STEEL CLAD
3.5% ENRICHMENT
- 16 Z 2.7 - 16 RODS PER ASSEMBLY
ZIRCALOY CLAD
2.7% ENRICHMENT
- 16 Z 3.5 - 16 RODS PER ASSEMBLY
ZIRCALOY CLAD
3.5% ENRICHMENT
- FF - FUEL FOLLOWER (ROD)
- SHADED REGION DESIGNATE FUEL

Initially, the 10 fuel followers will be the only AEC Fuel Cycle assemblies in the start-up core. The High Power Density core, Figure 2, contains High Power Density fuel (HPD) which consists of 25 rods per assembly while the blanket of Fuel Cycle (FC) fuel consists of 16 rods per assembly. This configuration is necessary to achieve the high power densities. During January 1961, the "Homogenized" core of Figure 3 will be inserted. This core will be used primarily for the stability tests, described in Section 2.1.1, which will be conducted for approximately a 3 to 4 month period. Following the stability tests, the High Power Density core will be reinserted, and uninterrupted irradiation of fuel will occur from that point up to 15,000 to 20,000 MWD/(m)T burnup.

2.2.2 Details of Fuel

Figure 4 shows an exploded view of a typical fuel rod. It consists of the cladding tube within which the UO_2 fuel pellets are placed. Above the upper fuel pellet, a thin metal wafer is placed, on top of which is a short support tube. This tube provides a plenum into which the fission gases may expand. On each end of the fuel rod, two end plugs are welded. The upper end plug weld is the last assembly operation and is performed in a sealed weld box in a helium atmosphere. The pressure of the helium in the fuel rod at room temperature is one atmosphere. Section 2.3.1 describes the processing of the fuel and material which make up the fuel rods before assembly.

EXPLODED VIEW OF FUEL ROD

The configuration of the end plugs vary from one fuel element to another. Those shown in Figure 4 are threaded and are used as structural members in the fuel assembly. For the removable fuel rods, one end plug is provided with a small hole across the diameter of the plug to receive a cotter pin and the other end plug contains a hole to pull the rod out of the fuel assembly.

Two types of fuel assemblies are pictured in Figure 5. The assembly on the left is typical of the ten fuel follower assemblies. In these assemblies the four center rods are held in the upper and lower tie plates with hex nuts. The twelve outer fuel rods are pinned into place at the upper tie plate with cotter pins. The center four rods act as structural members for the assembly. Both the upper and lower tie plates are machined from stainless steel castings. The spacer is made of 20 gage (0.032 diameter) stainless steel wire. The spacers are held in place by stainless steel support wires welded to each tie plate. In each of the fuel follower rods, the wafer is replaced by a poison pellet. This prevents flux peaking in the moderator region around the handle of the fuel followers.

The assembly on the right in Figure 5 is typical of all fuel assemblies except the fuel followers. One important difference is the upper tie plate. Like the spacer, it is built up of 20 gage stainless steel wire. The handle is stamped from stainless steel sheet. The four corner rods in this assembly

Fuel Follower
Assemblies

All Other
Assemblies

EXPLODED VIEW OF FUEL ASSEMBLIES

are attached to the upper and lower tie plates with hex nuts and become the supporting members. The remaining twelve rods are removable and are pinned at the lower tie plate with cotter pins. The poison pellets have been eliminated in the fuel rods in these assemblies in favor of the metal wafers shown in Figure 4.

The core support, or grid plate, is a matrix made of stainless steel beams and rods. The rods are perpendicular to and penetrate the beams. Intersections of the beams and rods are dowelled for rigidity. Nose pieces at the end of the fuel assembly channels contain leaf springs that latch the channels to the grid beams.

Fuel assembly channels are placed in every other space in the grid plate, similar to a checkerboard. This eliminates flux peaking which would occur in the water gap between the channel walls and allows the adjacent fuel rods to be placed closer together to form a more homogeneous fuel lattice. Initially, the fuel assembly channels will be stainless steel, but as burnup of the fuel proceeds, the stainless steel channels will be replaced by zircaloy channels to increase the core reactivity.

2.2.3 Fission Gas Release

During the design of the fuel assemblies (approximately September 1959), an analysis was made to determine the magnitude of the fission gas pressure in the VBWR fuel rods. Knowledge of such pressure buildup due to fission gas release is necessary in order to prevent cladding rupture.

The following assumptions were made:

1. Ten per cent of the fission gases produced will be released. The balance of the gases are held inside of the UO_2 fuel pellets.
2. The volume available for the gas to collect consists of the space between the end of the fuel and the end of the rod in the cold condition.
3. The average fuel temperature is taken as the temperature of the gaseous fission products in the plenum.
4. The neutron flux is assumed to be uniform over the rod cross-section, but is considered to have a sine distribution axially.
5. The gaseous fission products are assumed to be perfect gases.

These assumptions were believed to have an element of conservatism since, in reality, the annular gap between the fuel pellets and cladding and a small percentage of the UO_2 volume are available to the fission gases. In addition, no credit was taken for the volume increase resulting from thermal expansion of the cladding. Another conservative assumption was the temperature of the gaseous fission products. They were assumed to be at the average fuel temperature. In reality, the fission gases would be about 75% of the average fuel temperature because of the temperature gradient between the fuel and the plenum.

The approach used to calculate the fission product pressure buildup was to obtain the theoretical number of atoms of the noble gases Kr and Xe produced at a burnup of 20,000 MWD/(m) ton. Since an axial sine flux distribution was assumed, each fuel rod was divided into ten lengths of 0.305 ft. and the number of atoms of Kr-83, 84, 85 and 86 and of Xe-131, 132, 133, 134 and 136 were calculated for each region. Then, using the equation of state for a perfect gas, the partial pressures of the Kr and Xe products were estimated (based on the 10% release of all fission products formed), and added to the partial pressure of the He gas originally present. The net pressure exerted on the cladding was estimated to be about 4,000 psi at 20,000 MWD/(m)T burnup.

An analysis of the stress history of the cladding was made on the basis of the buildup of fission gas pressure within the rods. The maximum tensile stress in the fuel cladding is 24,100 psi (this is the tangential tensile stress, σ_{θ}) and it occurs in the hot boiling condition in a fuel follower which has a heat flux of 465,000 Btu/hr - ft² and at 20,000 MWD/(m)T burnup.

The minimum allowable yield stress was computed using the octahedral stress criterion described by Timoshenko. * Using this criterion, failure of the cladding is said to occur when the yield stress is

$$\sigma_{yp} = \frac{\tau_{oct}}{0.47}$$

where τ_{oct} represents the octahedral shear stress and is

given by:

$$\tau_{oct} = \frac{1}{3} \left[(\sigma_{\theta} - \sigma_r)^2 + (\sigma_r - \sigma_z)^2 + (\sigma_z - \sigma_{\theta})^2 \right]^{1/2}$$

* Timoshenko, S. Strength of Materials Part II, Third Edition, 1956 p. 456.

FIGURE 3
STRESS HISTORY OF CLAD
ON FUEL FOLLOWER
AEC FUEL CYCLE PROGRAM

and where

σ_{θ} = maximum tangential stress

σ_r = maximum radial stress under the same conditions
as σ_{θ}

σ_z = maximum axial stress under the same
conditions as σ_{θ}

Using this approach, the octahedral shear stress is 9,133 psi which requires a cladding material with a minimum yield point of 19,000 psi at 700°F. This compares with the yield point of 41,600 psi (at room temperature) reported by the Wall Tube and Metal Products Company.

In June 1960, Spalaris and Mangrum published a report* in which the fission gas pressure buildup in the Savannah Core II fuel was analyzed. The following assumptions were made:

1. Internal pressure buildup due to helium in the plenum chamber was neglected.
2. The volume available to the gases in the fuel rod is external to the UO_2 fuel pellets.
3. If a local hot spot occurs in the fuel, fission gas would be released as if the fuel had been operated at this high heat flux from the beginning of core life.
4. The fission gas temperature is the weighted average of the temperatures of the surfaces with which it is in contact.

* Spalaris, C. N. and E. Mangrum, Jr., N. S. Savannah Fuel Design and Development Program, GEAP-3381, June 20, 1960.

The temperatures are weighted proportionally to the surface areas that the gas contacts.

5. The volume gain due to tubing deformation is so small as to be neglected.
6. Each fuel pellet may contain as high as 0.6 milligrams of moisture.
7. The fission gas mixture in the rod plenum acts as a perfect gas.

On the basis of these assumptions, the subsequent analysis, and the fission gas measurements made on the Savannah II fuel, fission gas yields up to 30% may be possible in the fuel cycle fuel rods at heat fluxes around 465,000 Btu/hr-ft². However, this method of analysis would not cause a maximum stress of greater than 24,100 psi nor a yield point stress greater than 19,000 psi since the assumed temperature of the fission gases was about 40% lower in this latter analysis.

2.2.4 The Physics Program

2.2.4.1 Objectives of Physics Effort

The AEC Fuel Cycle physics effort to date has been in support of Task A (Advanced Fuel Tests). The objective of this work has been to determine the nuclear characteristics of the uranium oxide core to be operated in the VBWR. The basis for determining the nuclear characteristics is to maximize the important experimental information that can be obtained during the test period in a way consistent with safe core design. This has required

quantitative investigation of a number of possible core arrangements with variations in fuel enrichment and fuel element geometry. The goal has been to achieve a core loading in which a region of high power density elements can be tested at the core center and in which all elements in the core are contributing important test information. In this way, an entire core is not being operated for the purpose of a few select "test" elements.

2.2.4.2 Review of Efforts to Date

a. General

Physics studies have been directed to the attainment of a satisfactory test core containing Fuel Cycle, Consumers HPD and other fuel. In order to minimize local flux peaks between fuel elements, and at control rods, the VBWR core has been modified to give as uniform a lattice of fuel rods as possible. This was achieved by eliminating the water gaps between fuel channels and employing a square, box-shaped control rod with a fuel element follower. With the control rod withdrawn, there is only a small perturbation in the core flux distribution.

In order to achieve high power densities at the core center, it was found that a core must be constructed with at least

two regions of different fuel types. This results from two basic considerations. First, in a uniform core with typical power distributions, the total heat removal capabilities of the VBWR were exceeded for reasonable core sizes and desired peak power densities (kw/l) at the core center. The second, and more significant consideration, from a physics standpoint, results from the high neutron leakage in a core such as the VBWR. The goal of high central power density favors a relatively tight lattice with as many fuel rods per square foot as possible. This introduces a large negative void coefficient, and for this reason, an entire core with this tight lattice would produce too large a reactivity in steam voids. The large reactivity in steam voids is undesirable because of the loss in reactivity available for burnup and the possibility for introducing instabilities.

This latter effect is to be investigated in the stability limit tests. Thus, the basic core consists of two regions, a central region with high power density (kw/l) and an outer region with lower power density,

more open lattice, but still relatively high specific power (kw/kg). Within each region, several variations in fuel elements exist, such as differences in clad material, in clad thickness, and differences in enrichment. In addition, a region of other test elements surrounds the two regions described above. From the standpoint of nuclear analysis, the core represents a relatively complex assembly of fuel types. This serves to increase the time and expense involved in calculating the nuclear performance of the core.

b. Power Distributions

The ideal power distribution would be one which is completely flat in each region containing a different fuel type. While this is not possible, it is possible to obtain relatively flat power distributions in each region and still maintain the significant requirements of each region. This is achieved through variations in enrichment and fuel loading coupled with reasonably flat thermal neutron flux shapes. Figures 7 and 8 show the flux and power distributions for the core with Consumers Development Fuel at the center,

FIGURE 6
Thermal Flux Distribution
118 Elements
8 Zr Channels
40 MW

surrounded by Fuel Cycle fuel elements, with General Electric test elements at the periphery. The power distribution in the Fuel Cycle region can be seen to be quite flat. Local flux peaks between elements have been eliminated through the uniformity of the lattice. As soon as detailed information is available on the fuel elements to be present at the time Fuel Cycle fuel is loaded into the core, more detailed two-dimensional power distributions will be calculated. An iterative calculation will be made in which the coupling between power distribution and void distribution is included. The power shape shown in Figure 8 assumes a uniform 15% average steam fraction in the core. This distribution is attainable by orificing the core properly.

c. Nuclear Characteristics of Fuel

Three types of fuel elements will be present in the Fuel Cycle zone of the VBWR. These include stainless clad fuel, zircaloy clad fuel, and fuel followers for the ten control boxes in the Fuel Cycle zone. Nuclear

characteristics have been calculated for each fuel type. Table 2 summarizes the most important nuclear parameters of these Fuel Cycle elements. The channel material between fuel bundles can be either 0.060" stainless steel or zirconium. The stainless channels will be initially placed in the reactor and as burnup progresses, zircalloy will replace the steel. In this way, approximately 8-10% reactivity can be gained as the stainless channels are gradually replaced by zircalloy.. A month-by-month calculation will be made of the core power distribution and reactivity status of the fuel elements. The reactivity coefficients of the fuel are discussed in a later paragraph.

TABLE 2

	<u>16-Z-2.7*</u>	<u>16-S-3.2</u>	<u>FF-2.7%e</u>
η	1.8084	1.8494	1.8085
e	1.0256	1.0252	1.0280
P	0.8622	0.8650	0.8512
f	0.7847	0.7507	0.6982
k_{∞}	1.2464	1.2311	1.1049
UO ₂ radius	0.375"	0.375"	0.375"
Clad thickness	0.022"	0.015"	0.020"
W/F ratio	3.9	4.0	3.5

* 16-Z-2.7 denotes 16 - 16 rods per element
 Z - Zr clad
 2.7 - 2.7% enrichment

d. Control Strength

The control system of the modified VBWR operates through the seven control drives in existence in the top head. Two possible systems were considered in early design studies. One consisted of a set of seven cruciform-shaped blades which would run between fuel elements. The other consisted of the 14 control boxes operated in tandem from the seven control drives. The cruciform blades produced a slightly stronger control system; however, the control boxes with fuel followers produced a more uniform lattice without water gap peaking. For this reason, some control strength was sacrificed for the more uniform flux distribution which is desirable in a core serving the purposes of this VBWR core.

Several calculations of the control strength for different core configurations have produced total worths varying between 18% and 20% Δk . With expected initial values of k_{eff} between 1.12 and 1.14, this provides ample shutdown margin with all rods inserted. However, with one rod withdrawn, the core was found to have little or no shutdown margin. Operationally, the reactor must be kept in a condition where all rods are removed an equal amount.

e. Reactivity Coefficients

The nuclear characteristics have been calculated for the several fuel element types as a function of temperature and void content. Some of these results are summarized in Table 2. From calculations, the temperature and void coefficients of the entire core with this fuel were found to be negative. Of more direct interest to the Fuel Cycle Program are the "characteristic" coefficients of the Fuel Cycle fuel. This "characteristic" coefficient is defined as that coefficient which would be obtained in a "just critical" array of that fuel type. This implies the leakage is sufficient to reduce k_{eff} to 1.0. Within this approximation,

$$\frac{\Delta k_{\text{eff}}}{k_{\text{eff}}} = \frac{\Delta k_{\text{oo}}}{k_{\text{oo}}} = \frac{\Delta(B^2 M^2)}{1 + B^2 M^2}$$

Table 3 summarizes the temperature and void coefficients characteristics of the Fuel Cycle elements.

TABLE 3

			<u>16-Z-2.7</u>	<u>16-S-2.7*</u>	<u>FF-2.7%e</u>
$\frac{\Delta k_{eff}}{k_{eff}}$	20°C	40°C	-.000104	-.000059	-.00035
$\frac{\Delta k_{eff}}{k_{eff}}$	40°C	60°C	-.000520	-.000100	-.000898
$\frac{\Delta k_{eff}}{k_{eff}}$	0%	3% void	-.001985	-.00205	-.002855
$\frac{\Delta k_{eff}}{k_{eff}}$	3%	6% void	-.002396	-.00215	-.003553

*Since this calculation was made the enrichment for this type of element has been changed to 3.2%. This is not expected to change the reactivity coefficients significantly.

2.2.5 Heat Transfer and Performance of Fuel Cycle Fuel

During this period of time, initial design computations have been made, a study of heat transfer burnout data conducted, and design orificing and heat balance calculations have been conducted for the fuel cycle fuel. From these studies, methods for handling the needs of the experimental program have been established and significant safeguards problems analyzed (See Section 2.2.6).

2.2.5.1 Study of Heat Transfer Data for Operation of Fuel Cycle Fuel

Burnout data for 1,000 psia operation in net boiling has come from the heat transfer facility at APED, and from extrapolation of data from the pressurized water program. (e.g. WAPD-TH-321). Recent data obtained at APED has been improved in accuracy and reliability. Figure 9 shows the curve used as a basis for burnout analyses of fuel cycle fuel. A continued program of investigation of net boiling burnout is being conducted under Task B. Results will be published as they become available.

The power distribution for the VBWR core to be used in the fuel cycle program for the fuel life investigation is given in Figure 8. Analyses were made of the hot channels of this core. Experience at VBWR indicates an axial power profile such that the maximum to average power output ratio is about 1.5. Calculations thus far have used a cosine distribution with maximum to average power output ratio of 1:57.

FIGURE 9
AEC TYPE I ASSM. AT 430 KW
VBWR AT 40 MW & 2000 GPM

V.A.S. 6-4-60

The core has been orificed for operation on the basis of the power delivered by the region. Each channel in a given region will receive, very nearly, the same flow. Hence, the hot channel for each region will produce peak power and be cooler by average flow. Heat flux, steam quality and the burnout heat flux cross plotted from Figure 9 are shown in Figure 10 vs. length for the hottest fuel cycle channel at 40 MW reactor power. The method of calculation will be summarized in a subsequent report.

The burnout margin is the nearest approach of the burnout curve and the actual heat flux for the channel. The analysis does not include any "hot spot" factors as such, since the hot channel is being investigated rather than the average channel. The burnout curve is not taken "below all available data". Consequently, the burnout curve represents the point at which the channel is expected to burn out. The burnout margin represents the margin between operating conditions and actual burnout, and includes any and all factors. The margin for the hottest channel is 615,000 BTU/Hr-Ft².

To judge the soundness of the approach, the following should be noted:

1. Test data is in each case for 8.5 foot rods (heated length) while VBWR has three foot fuel rods. Single rods have

an annulus of 0.1675 inch which is equivalent to the spacing in the 16 rod fuel cycle bundles. Multi-rod 0.433 inch O.D. tubes show at least equal burnout fluxes compared with the results of annuli.

2. Eccentric rod testing with spacing reduced to 0.096 inch shows no reduction in burnout for the 8.5 foot rods. Reducing spacing to 0.06 inch shows reduction of 8 to 15 per cent.
3. The data is well verified and a continuing program will provide additional data.

The effect of any divergence from the burnout curve has been investigated. A complete reporting of this work will be presented at a later reporting period.

2.2.6 Safeguards Study

An initial review of the then intended fuel cycle experimental core and its intended use with the VBWR was made with the licensing and regulation branch of the AEC in May 1959. The work conducted following this included scoping of the experiments, physics and thermal analysis of the core, preparation of safeguards issues and their solution. These were combined, along with the work of other programs in an extensive license application document SG VAL-2 Third Edition.

A formal presentation was made in January 1960 to the Advisory Committee on reactor safeguards, who returned a recommendation that a license be granted for the reactor and its operation. A license hearing was held on April and again in May 1960 for the purpose of licensing the VBWR, the experimental cores of this and other programs, and the experimental programs that are involved.

A license was officially received July 11, and amendments necessary for startup operation of the reactor were approved on August 4. Loading of the reactor is to proceed immediately, and fuel cycle fuel installed as rapidly as possible. The more important results of the safeguards analysis are summarized in the sections which follow.

2.2.6.1 Heat Sinks For Reactor Shutdown Cooling

Figure 11 summarizes the ways in which core decay heat may be dissipated for the protection of experimental fuel under shutdown conditions. Major heat dissipation sources are: The emergency heat exchanger and water in the fuel storage pool. Heat transfer from the exposed parts of the pressure vessel supplements this, and is able to carry the entire decay heating load after 70 hours of operation.

2.2.6.2 Loss of Flow Accidents

Because one of the fundamental goals of the fuel cycle development program is the study of operation of high

DECAY POWER AS FUNCTION OF COOLING
TIME - INFINITE IRRADIATION

FIGURE 11

power output forced circulation reactors, a considerable investigation has been made into the consequences and methods of operating the forced circulation system at high power levels. One such accident which was investigated, involves the loss of a pump at various power levels.

The VBWR may be operated with one or two pumps, and during the performance test program of the fuel cycle program it is expected that both the single variable speed pump system and the two pump system will be used. The limit of reactor power which may be used without damage to the fuel in the event of the failure of a pump has been set by the limiting safe condition of operation using only natural circulation in the external recirculation loop for cooling the core. This limit was found, based on analysis, to be 33 MW for one pump operation. Above this point the use of two pumps is planned. During the startup test program for the VBWR, the natural circulation in the external recirculation piping will be measured. The limits for one pump will then be re-evaluated.

The rate of flow decrease after a pump failure will depend on the kinetic energy of the recirculating water and the flow losses through the system and the pump. Figure 12 shows the flow decay for two pump discharge

FIGURE 12

valve settings (wide open and partly closed), and the two pump accidents (power loss with pump coastdown, and shaft seizure with immediate pump stoppage). In either event the flow will be reduced quite rapidly.

Counteracting the decrease in flow through the external system is the natural circulation driving force. In such an incident as described above, the final flow rate is determined by reactor power and the position of the pump discharge valves. Figure 13 shows the natural circulation performance of the system as a function of reactor thermal power.

The energy to be dissipated in the hottest spot of the VBWR core is shown as a function of time after the reactor is scrammed as shown in Figure 14. The data is expressed in terms of the biot number hr_m/k of 13.7 and 10.0. The biot number of the hottest channel is 11.9. Expressed in this manner the energy stored as well as the heat generated by the decay heating process is accounted for.

As described previously, the calculation considers that each region is orificed to give the same exit quality for each region of the multi-region core. Figure 15 shows the burnout heat flux for the hottest channel in the fuel cycle fuel and the hottest in the three region core (high power density region).

If the reactor has been operating with one pump at high power and high flow rates, the situation is somewhat more severe than indicated by the above quasi-steady state assumptions of the transient. This is because the change in flow rate at the point of loss of the pump is felt instantaneously at all points in the circulation system, while a compensating reduction in core inlet temperature does not occur until feedwater injected in the pump suction line reaches the core. The result is a temporary increase in exit quality and boiling length over the steady state condition. It is because of these considerations that one pump operation of the three region reference core is limited to 33 MW. Actually the limit is the high power density region which is limited to 130 KW/liter for one pump operation. A minimum burnout ratio of 1.5 will be maintained, as the VBWR operating license required, with a calculated 1,000 gpm natural circulation flow. This limit will be changed based upon initial test program results.

2.2.6.3 Brief Review of Reactivity Accidents

The experimental programs which are to be initiated in the VBWR have been examined to ascertain their effects upon reactivity additions to the reactor. During the course of the VBWR modification, changes to the control rod system and the process control instrumentation were made to assure satisfactory operation of the experimental

programs. A brief description of control rod system operation and the reactivity accidents which are of significance is made here to report the work which has been done to prepare for the fuel cycle program.

2.2.6.4 Control Drive Safety

The control drive circuits are arranged so that, although all rods may be inserted simultaneously at high speed, they may be withdrawn only one at a time and only at low speed. Control rod failures in this plant would be limited, due to the construction of the drive units, to the failure of a rod to go into the reactor or the withdrawal of a rod unintentionally. The failure of a rod to go into the core would not have serious consequences because the maximum loading will always be restricted so that a failure of one rod will allow the reactor to be critical only at low temperatures, and a soluble poison injection system is provided to shut the reactor down completely in such an eventuality. If a fault in the electrical equipment should cause the unintentional withdrawal of a rod, it would be restricted, by the design of the circuits, to one rod and to the normal withdrawal rate. The nuclear instrumentation, in this case, would scram the reactor down before a large excursion occurred. Most control circuit faults, however, would cause the rods to scram if they affected the position at all.

Analysis of reactivity accidents which have been investigated is believed to be complete for the VBWR, modified, and for its experimental programs. Reviews have been conducted with General Electric Company and AEC without uncovering faults in the analysis or the handling of the conceived problems. Table 4 summarizes these accidents.

2.2.6.5 Reactor Stability

The stable performance of VBWR with a plate core has been demonstrated by the natural circulation operation of the reactor at 30 MW. Extrapolation of the data obtained from this performance, comparison with BORAX and SPERT I reactors, and an examination of the hydraulic characteristics of the cooling circuit indicate that the VBWR can operate stably in natural circulation with a plate-type core at 50 MW and 1010 psia.

Operation of the modified reactor will now include the possibility of two basic types of fuel, those with 0.375" O.D. pellets and those with 0.50" O.D. pellets. The former have a 7.7 sec. time constant and the latter 17 sec. The smaller rods would have 15-20 grams U-235 per rod and the larger would have 22 grams U-235 per rod.

Unlike stable boiling water reactor operation in which the instantaneous power oscillates about some mean power level with amplitudes which are small compared to the average power, unstable reactor operation is characterized by one or more of the following:

TABLE 4

REACTIVITY ACCIDENT SUMMARY - VBWR

Type of Accident Conceived	Action	Counter Measures	Comments
1. Rod blowout	Rapid shoot-out of rod due to air pressure, while the drive carriage was disengaged from the air cylinder.	A double safety switch circuit prevents the drive carriage and the air cylinder from separating. In addition, sticking of shafts is checked by timing measurements of rod insertion.	The counter measures are believed effective in preventing shoot-out. Other types of shoot-out would require major failure of carriage parts which are highly improbable.
2. Stuck rod at reactor shutdown	One of the control rods fails to drive in when the reactor is scrammed.	Loading of the reactor is planned such that the reactor will always be subcritical at operating temperatures if the rod is stuck. A liquid poison system will then be used to obtain cold subcriticality.	The liquid poison system used in the modified VBWR is a versatile system capable of repeated use and repair during almost all modes of operation
3. Calculational errors	An excursion due to errors in temperature and void coefficients.	Each core is examined in detail prior to criticality. Critical experiments will check reactivity coefficients to determine that conditions are met.	A positive temperature coefficient of greater than 75 cents will not be used in operation. Void coefficients will always be negative.
4. Loading accidents	An excursion is allowed due to dropping a fuel assembly into place when it has a high reactivity worth.	1. Procedures are designed to avoid loading accidents. 2. Instrumentation is provided to scram the reactor in the event procedures are neglected. 3. Highly reactive fuel elements are kept in an inaccessible vault.	

TABLE 4 - REACTIVITY ACCIDENT SUMMARY-VBWR (con't)

Type of Accident Conceived	Action	Counter Measures	Comments
5. Steam flow stoppage	Multiple valve and control failures cause reactor pressure to rise and void coefficient to introduce reactivity.	Reactor goes on ramp rate excursion. The high flux scram actuates and either or both of the recirculation system and emergency exchanger dissipate the energy	No serious effects on fuel would be expected. The accident would be extremely difficult to obtain since simultaneous failure of main steam and pump turbine steam would have to take place.
6. Cold water accident	1. Feedwater flow is suddenly changed resulting in reactivity addition due to core void collapse.	1. The rate of opening feedwater valves is controlled to minimize the severity of this accident.	1. The accident was examined over the operational range of experiments and ascertained to cause no fuel melting or other damage in its worst occurrence.
	2. Recirculation flow valves open and collapse steam voids.	2. Recirculation valves open very slowly and temperature differences in recirculation lines are monitored to avoid this occurrence.	2. The problems have been examined over the operational range. Conservatively calculated problems introduced no serious results.
7. Startup accident	Inadvertent withdrawal of control rod without correction puts the reactor on an excursion.	Period scram, micro-microammeter scram and high flux scram provide protection.	Analyses showed that even when the first two counter measures were ineffective, the high flux scram would shut the reactor down satisfactorily.

- a. Amplitude of the power fluctuations (boiling noise) is large.
- b. The variation of the power from any initial oscillation takes on a divergent character.
- c. The frequency of oscillation shows rapid increase or some other unexplained phenomenon.

There are three modes of operation which have an effect on the stability of operation. First, operation as an internal natural circulation reactor nearly identical to the operation of VBWR before modification. Second, operation of the reactor as a direct cycle forced circulation reactor with relatively long boiling length. Finally, operation as a dual cycle reactor with forced circulation.

Stability in the steady boiling condition has been examined in BORAX, SPERT, EBWR and VBWR natural circulation reactors. The range of operating conditions, temperatures, pressures and core characteristics vary widely. Direct comparison of any of the tests is difficult and extrapolation of the information to other reactor designs is also qualitative at present. There are a number of principles demonstrated by these tests which are pertinent.

- a. Higher pressure levels are a definite aid in maintaining power stability.
- b. The time constant of the fuel is important in stability along with the system transport time. It is desirable to have a long fuel time constant. BORAX I has a 0.1 second and BORAX IV has a 0.5 second time constant. BORAX IV showed a definite increase in stability due to the long fuel response time. However, experimental variation of transport time has not been great enough to evaluate its effect in combination with a range of fuel time constants.
- c. The amount of reactivity held in steam voids has an effect upon the reactor responses. When sufficiently high reactivity is tied up in voids, the time responses of an initial oscillation have been shown (BORAX and SPERT) to be divergent.
- d. Boiling noise in steady boiling showed considerable amplitude for a period of time prior to the development of a self-oscillatory characteristics (BORAX).
- e. Behavior of the reactors was repeatable.

Table 5 summarizes the parameters which effect reactor stability and lists the VBWR modification cores for comparison.

The void and temperature coefficients for the various types of fuel are given in Table 3. Generally, tight lattices have larger negative temperature coefficients than the highly moderated cores and both may have small positive coefficients at low temperatures which are not harmful. Void coefficients are negative for all cores considered.

In a recent department study into system stability of boiling loops, several conclusions were made which are appropriate to the operation of the modified VBWR.

- a. Flow stability was tied to defined physical parameters, not rules of thumb such as exit voids.
- b. Forced circulation markedly reduces the amplitude of flow and pressure drop variation and extends the range of stable operation.
- c. The burnout power level is considerably lower with natural circulation than with forced circulation due to early occurrence of large amplitude flow system oscillations.

TABLE 5

BWR REACTOR CHARACTERISTICS AFFECTING STABILITY

REACTOR	BORAX I	BORAX IV	SPERT I	EBWR	PLATE	VBWR	DESIGN COMPARISON	
							ROD	VBWR
1. Pressure, psia	atm.	322	atm.	614	1000		1000	
2. Core length, feet	2	2	2	4	3		3	
3. Δt (subcooling) $^{\circ}\text{F}$	0-125 $^{\circ}\text{F}$	0	~ 180	0-150 $^{\circ}\text{F}$	0-200 $^{\circ}\text{F}$		0-200 $^{\circ}\text{F}$	
4. Water-to-fuel ratio	1.95/1	2.9/1	1.92/1	1.94/1 1.23/1	12.3/1		2.1/1 3.9/1	
5. Flow channel equivalent Diameter, D_e , inches	0.23	0.23	0.23	0.82 0.69	0.62		0.46 0.66	
6. Fuel time constant, sec.	0.1	0.5	~ 0.1	< 0.1	0.1		~ 7	
7. Typical press. coeff. of reactivity $\Delta k/k/\text{psi}$	~ 0.37	-----	-----	< 0.001	0.001			
8. Void coeff. $\Delta k/k/\%$ void	0.235	0.096	0.0224	-----	-----		-----	
9. Temperature coeff. $\Delta k/k/^{\circ}\text{F}$	1.2×10^{-2}	2.5×10^{-5}	6×10^{-3}	-----	-----		-----	
10. Channel shape	rect.	rect.	rect.	rect.	rect.		rod array	
11. Max. operated "steady boil" kw/l (core avg.)	35	45	-----	-----	-----		-----	
12. Max. rect. % in voids	2.5	2.5	-----	-----	-----		-----	
13. Boiling noise anal.	None	Yes	None	Not published.	Auto correl. shows no instability.		Planned	
14. Measurements to evaluate stability	Power & pressure traces.	Power & pressure traces. Transfer function measurements.	Power trace.	Power & pressure traces. Transfer function measure- ments.	Power trace (mixed core).		(Planned) Power & pressure traces. Transfer function measurements.	

The modified VBWR variable speed-variable head pumping system should exert a stronger control over the appearance of instability. The stability of the pumping system itself and the effects of cavitation situations which might arise have been investigated thoroughly.

2.3.0 Fabrication of Core

2.3.1 Stainless Steel Clad Assemblies

2.3.1.1 Description

Sixteen fuel assemblies have been planned with the following characteristics:

1. 0.015" stainless steel (type 304) cladding, 0.380" inside diameter.
2. UO_2 fuel pellets will have an enrichment of 3.2% U-235.
3. 16 fuel rods per assembly.
4. Active fuel length of 37 inches.
5. Each fuel assembly will have 12 removable rods.

One of the fuel assemblies just described will be fabricated per drawing number 932C365 G4. It will contain twelve fuel rods with unground fuel pellets varying in diameter from 0.378 inches to 0.368 inches. One of the sixteen fuel rods will be defective. The other fifteen fuel assemblies will be fabricated per

drawing 932 C 365 G1 and will contain fuel rods with fuel pellets ground to a diameter of 0.376/0.373 inches.

Ten control rod fuel follower assemblies have been planned per drawing number 932 C 378 G1 with the following characteristics:

1. 0.020 inch thick stainless steel (type 304) cladding with 0.380 inch inside diameter.
2. 0.375 inch diameter UO_2 fuel pellets with a 2.7% enrichment in U-235.
3. Sixteen fuel rods per assembly.
4. Active fuel length 30 inches.

2.3.1.2 Processing of Material

All stainless steel tubing for cladding was manufactured by the Wall Tube and Metal Products Company of Newport, Tennessee. They provided a certification with the tubing as shown in Table 6. The type H cladding will be used in the 10 fuel follower assemblies in the VBWR core. The lower yield point and ultimate strength of the Type H fuel occurs only on 0.020 inch thick cladding. This material was annealed after being cold worked. The analysis in section 2.2.3 requires a minimum yield strength of 19,000 psi at 700°F for this type of cladding.

Upon receipt of the tubing, each tube was identified by electro-etching a number on the outside. The APED Quality Control Unit conducted the following types of

TABLE 6

CERTIFICATE OF PROPERTIES OF STAINLESS STEELCLADDING AS PROVIDED BY WALL TUBE AND METAL PRODUCTS COMPANY

Designation	Chemical Composition									Physical Properties		
	C	Mn	P	S	Si	Ni	Cr	Cu	Mo	Yield (psi)	Ultimate (psi)	Elong.
AEC Type H Fuel Cladding	.055	1.70	.026	.015	.54	8.57	18.34	.26	.40	41,600	90,600	55%
AEC Type I Fuel Cladding	.055	1.70	.026	.015	.54	8.57	18.34	.26	.40	87,200 86,400 89,200	108,800 102,800 109,300	34% 33% 35%

tests on the material:

- a. An eddy current test which checked for circumferential defects such as whiskers, cracks, nicks, etc. Any tubing having a defect which exceeded 10% of the wall thickness was rejected.
- b. An ultrasonic test which checked for longitudinal defects in the tubing. Any tubing having a defect exceeding 10% of wall thickness was rejected.
- c. A statistical sampling of the tubing population was selected and checked for maximum and minimum wall thickness and inside diameter by measuring one end of each tube. A sample of each lot of tubing was kept for chemical and physical tests which will be run during September 1960.

When the UO_2 powder is received, ten 50 gram samples are extracted from each lot. Each sample is inspected for powder uniformity and compatability with the pellet process. A Composite 75 gram sample from each lot is later tested for O/U ratio, U-235 content and impurity content analysis. In addition, a certified analysis of O/U ratio, U-235 content and impurity content is required from the vendor. The following outline presents the important manufacturing processes of UO_2 from receipt from vendor state to the pellet stage.

- a. Receiving of UO_2
- b. QC inspection of UO_2
- c. Pellet sintering
- d. Pellet grinding
- e. Pellet density and diameter measurement
- f. 100% visual inspection
- g. Finished chemical analysis
- h. Witness of pellet storage
- i. Witness of pellet transfer

Similarly, during manufacture of the fuel rods, the following steps will be performed:

- a. Receiving of material
- b. QC inspection of tubes
- c. Machining operations of components
- d. Cleaning of components
- e. Temporary serial numbering tubes
- f. First end plug pressing
- g. Unloaded tube weight and inside length measurement.
- h. Pellet loading
- i. Loaded tube weight and inside length measurement.
- j. First end plug weld
- k. Second end plug pressing and welding in helium atmosphere.
- l. Permanent serial numbering tubes.

- m. Visual and X-ray weld inspection
- n. Visual rod bow inspection
- o. Dimensional rod measurement
- p. Visual surface defect inspection
- q. Rod leak test

Finally, for the manufacture of the fuel assemblies, the following processes will be performed:

- a. Component cleaning
- b. Component identification
- c. Component assembly
- d. Dimensional inspection

During these processes, detailed records will be kept. Table 7 shows a fuel rod data sheet and Table 8 shows a fuel assembly data sheet which will be colated into a topical report approximately December 1960. It will also include the following records on the UO_2 powder and pellets by lot:

- a. Isotopic analysis
- b. O/U ratio
- c. Chemistry and UO_2 purity
- d. Yield through segmentizing
- e. One 500 gram sample of as-received powder for storage
- f. Twenty as-sintered and twenty as-ground sintered pellets.

2.3.2 Zircaloy Clad Assemblies

Thirty-five assemblies will be manufactured per drawing number 932 C 365 G2 with the following characteristics:

1. 0.022-inch thick Zircaloy -2 cladding, 0.380-inch inside diameter.
2. 0.375-inch diameter fuel pellets made of UO_2 of 2.7% enrichment in U-235.
3. Sixteen fuel rods per fuel assembly.
4. Active fuel length of 37 inches.
5. Two rods per fuel assembly with planned defect holes through cladding.
6. Twelve removable rods per assembly.

Two fuel assemblies will be constructed identical to those described above except that one assembly will be made with a pellet to cladding clearance of 0.002 inch, and the other assembly will contain 12 fuel rods loaded with unground fuel pellets varying from 0.378 to 0.368 inches (drawing number 932 C 365 G5).

One zircaloy clad fuel assembly will be made per drawing number 932 C 365 G6 with the following characteristics:

1. 0.022-inch thick zircaloy -4 cladding with 0.380-inch inside diameter.
2. Twelve rods containing UO_2 fuel pellets (2.7% enriched in U-235) which have been sintered, but are unground and the diameters of which vary between 0.378" and 0.368". The remaining four fuel rods will contain ground pellets.

3. Sixteen fuel rods per fuel assembly.
4. Active fuel length of 37 inches.
5. One planned defect hole through cladding.
6. Twelve removable rods in the assembly

One zircaloy clad fuel assembly will be made per drawing number 932C365 G3. It will be identical to the preceding assembly except that the diametral gap between fuel pellet and cladding will be 0.002 inch. .

One zircaloy clad fuel assembly will be made per drawing number 932 C 365 G3 with the following characteristics:

1. 0.022-inch thick zircaloy -4 cladding with 0.380-inch inside diameter.
2. 0.375 inch diameter sintered and ground UO_2 fuel pellets (2.7% enriched in U-235).
3. Sixteen fuel rods per fuel assembly.
4. Active fuel length of 37 inches.
5. One planned defect hole through cladding in the assembly.
6. Twelve removable rods in the assembly.

The delivery of the zircaloy tubing is expected during September, 1960. The zircaloy-2 and zircaloy-4 were manufactured by the Carborundum Metals Company of Akron, New York, and the tubing was rolled by the Wolverine Tube Company of Detroit, Michigan. The manufacturers will certify that the tubing has been manufactured per specifications. The specifications of the chemical composition and mechanical properties are listed in Table 9.

TABLE 9

SPECIFICATIONS OF ZIRCALOY-2 AND ZIRCALOY-4 TUBINGG.E. - APED SPECIFICATION NO. B50YP2-S2

Alloying Elements	<u>Zircaloy-2</u>	<u>Zircaloy-4</u>
	(Weight %)	
Tin	1.20-1.70	1.20-1.70
Iron	0.07-0.20	0.14-0.24
Chromium	0.05-0.15	0.05-0.15
Nickel	0.03-0.08	
Total: Fe + Cr + Ni	0.18-0.38	

Maximum Impurities	(Parts per million)	
Boron	0.5	0.5
Cadmium	0.5	0.5
Calcium	30	30
Carbon	270	270
Chlorine	15	15
Cobalt	20	20
Hafnium	100	100
Nickel		75
Sodium	20	20
Titanium	50	50
Tungsten	100	100
Vanadium	50	50
Aluminum	75	75
Copper	50	50
Hydrogen	15	15
Lead	130	130
Magnesium	30	30
Manganese	50	50
Nitrogen	80	80
Oxygen	1000-1500	1000-1500
Silicon	100	100
Uranium	3.5	3.5

The processing of the zirconium-clad fuel assemblies will follow the identical pattern as outlined above for the stainless steel clad assemblies.

2.3.3 Special Fuel Assemblies

Twelve special fuel assemblies will be designed, fabricated and irradiation tested to obtain the following information:

1. The effect of gross center melting of UO_2 pellets.
2. The effect of thermal conductivity promoters such as TiO_2 and YT_3O_2 .

The design of these assemblies has not been started to date, however, it is expected that the first of these assemblies will be ready for insertion in the VBWR during the first quarter of 1961. The processing of these assemblies will follow a similar pattern as the stainless steel clad fuel assemblies.

2.3.4 Long-Term Reactivity Assemblies

Three of the zircaloy fuel assemblies will be designated to study long-term reactivity effects on fuel, cladding and manufacturing techniques. The design of these assemblies has not been started as of this date, however the present design of fuel will probably not require much revision.

Table 10 summarizes the information described in this section.

2.4.0 Instrumentation and Calibration of Three Assemblies

2.4.1 Parameters to be Measured

Three fuel assemblies will be instrumented for the stability tests. The parameters to be measured are as follows:

TABLE 10

FUEL CYCLE PROGRAM FUEL ASSEMBLIES

Fuel Assembly Designation	Identification	No. of Rods	Clad Material	UO ₂ Enrichment %	Clad Wall (in)	Pellet O.D. (in)	Clad O.D. (in)	Clad I.D. (in)	Pellet Clad Diametrical Gap - (in)	Assembly Drawing Number	Number of Defected Rods per Assembly	Active Fuel Length (in)	Number of Removable Rods	Manufacturer	Comments
AEC-1H	1-101	16	304 SS	2.7	.0215/.0185	.376/.373	.424/.416	.381/.379	.008/.003	932C378G1	None	30	None	Wall (4)	Fuel Followers
AEC-2H	1-102	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑
AEC-3H	1-103	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑
AEC-4H	1-104	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑
AEC-5H	1-105	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑
AEC-6H	1-106	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑
AEC-7H	1-107	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑
AEC-8H	1-108	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑
AEC-9H	1-109	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑	↑
AEC-10H	1-110	↓	↓	2.7	.0215/.0185	.376/.373	.424/.416			932C378G1	None	30	None		Fuel Followers
AEC-11I	1-111	↑	↑	3.2	.0165/.0135	.376/.373	.414/.406			932C365G1	None	37	12		One of these assemblies will be designated as an instrumented assembly
AEC-21I	1-112	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-31I	1-113	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-41I	1-114	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-51I	1-115	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-61I	1-116	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-71I	1-117	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-81I	1-118	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-91I	1-119	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-101I	1-120	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-111I	1-121	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-121I	1-122	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-131I	1-123	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-141I	1-124	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-151I	1-125	↓	↓	3.2	.0165/.0135	.376/.373	.414/.406			932C365G1	None				
AEC-161I(2)	1-126	↓	304 SS	3.2	.0165/.0135	.378/.368(1)				932C365G4	One				
AEC-161I(3)	1-126	↓	304 SS	3.2	.0165/.0135	.376/.373				932C365G4	None		12	Wall (4)	
AEC-1J	1-127	↑	Zr-2	2.7	.0235/.0205	.376/.373	.428/.420			932C365G2	Two		12		Two of these assemblies will be designated as instrumented assemblies
AEC-2J	1-128	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-3J	1-129	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-4J	1-130	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-5J	1-131	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-6J	1-132	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-7J	1-133	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-8J	1-134	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-9J	1-135	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-10J	1-136	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-11J	1-137	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-12J	1-138	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-13J	1-139	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-14J	1-140	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-15J	1-141	↑	↑	↑	↑	↑	↑			↑	↑	↑	↑	↑	↑
AEC-16J	1-142	↓	↓	2.7	.0235/.0205	.376/.373	.428/.420	.381/.379	.008/.003	932C365G2	Two	37	12	Carborundum-Wolverine (5)	
AEC-17J	1-143	16	Zr-2	2.7	.0235/.0205	.376/.373	.428/.420	.381/.379	.008/.003	932C365G2	Two	37	12		

TABLE 10 (Cont'd.)

Fuel Assembly Designation	Identification	No. of Rods	Clad Material	UO ₂ Enrichment %	Clad Wall (in)	Pellet O.D. (in)	Clad O.D. (in)	Clad I.D. (in)	Pellet Clad Diametrical Gap - (in)	Assembly Drawing Number	Number of Defected Rods per Assembly	Active Fuel Length (in)	Number of Removable Rods	Manufacturer	Comments
AEC-18J	1-144	16	Zr-2	2.7	.0235/.0205	.376/.373	.428/.420	.381/.379	.008/.003	932C365G2	Two	37	12	Carborundum-Wolverine(5)	
AEC-19J	1-145														
AEC-20J	1-146														
AEC-21J	1-147														
AEC-22J	1-148														
AEC-23J	1-149														
AEC-24J	1-150														
AEC-25J	1-151														
AEC-26J	1-152														
AEC-27J	1-153														
AEC-28J	1-154														
AEC-29J	1-155														
AEC-30J	1-156														
AEC-31J	1-157														
AEC-32J	1-158														
AEC-33J	1-159														
AEC-34J	1-160	16	Zr-2	2.7	.0235/.0205	.376/.373	.428/.420	.381/.379	.008/.003	932C365G2	Two	37	12	Carborundum-Wolverine(5)	
AEC-35J	1-161														
AEC-36J	1-162														
AEC-37J(2)	1-163														
AEC-37J(3)	1-163	16	Zr-4	2.7	.0235/.0205	.376/.373	.428/.420	.381/.379	.008/.003	932C365G5	None	37	12	Carborundum-Wolverine(5)	
AEC-1K(2)	1-164														
AEC-1K(3)	1-164														
AEC-2K	1-165														
AEC-3K	1-166														
AEC-4K	1-167														
AEC-5K	1-168														
AEC-6K	1-169														
AEC-7K	1-170														
AEC-8K	1-171														
AEC-9K	1-172														
AEC-10K	1-173														
AEC-11K	1-174														
AEC-12K	1-175														
AEC-13K	1-176														
AEC-14K	1-177														
AEC-15K	1-178														
AEC-16K	1-179														

- (1) Unground pellets
(2) 12 Rods
(3) 4 Rods
(4) Wall Tube and Metal Products Co., Newport, Tenn.
(5) Zircaloy manufactured by Carborundum Metals Co., Akron, N.Y.
Tubing rolled by Wolverine Tube Co., Detroit, Mich.

Three of these assemblies will be designated for the long term reactivity studies and twelve will be designated for center melting of UO₂ and thermal conductivity studies

1. Temperature of water at inlet of fuel assembly.
A thermocouple encased in a stainless steel sheath will be used.
2. Temperature of effluent water at outlet of fuel assembly. The water is at saturation temperature at this point.
3. Three pressure taps will be located in each instrumented fuel assembly. They will be located at the inlet of the fuel channel, at the top of the fuel bundle, and at the top of the fuel channel. The pressure taps will be of the pitot tube type. However, only the static pressure will be measured.
4. The neutron flux will be measured at three locations along the active length of the core, at approximately the $1/4$, $1/2$, and $3/4$ points with neutron flux ion chambers. In addition, a neutron sensitive wire will be irradiated along the total length of the core to calibrate the ion chambers and measure axial flux distribution. The ion chambers and wire will be located at two corners of the fuel channel.
5. The inlet flow will be monitored by placing a turbine type flow meter at the inlet of the fuel channel and if possible at the exit. These will be used to obtain the channel flow and steam void fraction.

2.4.2 Design of the Assemblies

The design of the instrumented fuel assemblies for the Consumers power plant tests has been completed. This design features the fuel channel and fuel rod bundle which are integral with the

instrumentation as pictured in Figure 16. The basic Consumers design will be used for the instrumented assemblies described in this section. However, the design will be improved by making the fuel assembly separately removable from the instrumented channel. Thus, the instruments mounted on the channel may be repaired, resulting in longer instrument life.

2.4.3 Instrument Calibration

The instruments in the fuel elements will be calibrated by placing the element in the APED heat transfer facility forced circulation loop, above the heated portion of the loop. This will permit controlled conditions of temperature pressure and coolant velocity for calibration of the instruments.

2.4.4 In-Core Ion Chambers

Operating experience of the APED designed ion chambers has been accumulated in the Dresden reactor. Out of twenty-eight chambers under irradiation since June 6, 1960, no failures have occurred to date, moreover, they show promise of six month's operation without failure. The ion chambers which will be put into VBWR will be tested in an autoclave under conditions of temperature and pressure they will see in the reactor. Every precaution will be taken to assure the quality of the chambers.

2.5.0 Stability Tests

2.5.1 Rod Oscillator Mechanism

A preliminary design of the rod oscillator mechanism has been completed. The specifications call for a frequency of from

INSTRUMENTED FUEL ASSEMBLY

one cycle per second to one cycle per minute and an amplitude variation of from 0 to two inches. This provides a change in reactivity of about 0.1% in the hot, boiling condition.

Schematically, the mechanism is pictured in Figure 17. A hydraulic cylinder provides the force to bear the oscillator arm against the shaft stop. The return force is provided by the operating air pressure in the lower portion of the control rod drive cylinder. As the pressure in the hydraulic cylinder increases, the driving force is transmitted through the connecting rod and oscillator arm to the shaft stop.

The rod oscillating mechanism assembly will be placed on control rod number seven and may be placed on other control rods. It is attached to the control rod drive support rods in three places by means of three split sleeves. The main frame pieces are positioned between the motor driven stop above and the shaft stop below.

During normal rod operation, the oscillator arm is in the up position and the rod oscillating mechanism assembly travels with the motor driven stop. For the stability tests, the control rod is withdrawn one inch further than the mean position about which the rod is to oscillate and the oscillator arm is lowered against the shaft stop. During oscillation the motor stop is held motionless and the oscillator arm is moved up and down by the hydraulic cylinder. Thus, the oscillator arm pushes down and the operating air pushes up. During scram, the shaft stop is driven away from

FIGURE 17

ROD OSCILLATOR MECHANISM

the motor driven stop and the rod oscillator mechanism assembly follows the shaft stop down. Subsequently, the limit switches on the motor driven stop open and the motor driven stop follows the shaft stop and rod oscillator mechanism assembly down.

Hydraulic pressure is provided by means of a motor and hydraulic pump as seen in Figure 18. A hydraulic servo valve similar to Moog series 900 may be used to control the flow of the fluid to the hydraulic cylinder. When the control rod has displaced from its mean position, a signal from the control rod position indicator is fed back to a controlling power amplifier. Here the position signal is compared to the input signal and a proportional controlling output signal is sent to the servo valve.

For a breakdown of the stability tests and how they will be tied into the overall program, the reader is referred to Sections 2.1.4 and 5.

2.6.0 Irradiated Fuel Examination Program

A preliminary study has been conducted to obtain the greatest amount of fuel examination per dollar, however, no schedule has been formulated to date.

3.0 TASK B - HEAT TRANSFER AND FLUID DYNAMICS:

3.1.0 Objectives of Task B

The overall objectives of Task B are twofold. First, to supply specific information on heat transfer and fluid flow in support of Task A. In particular, this might include such information as required for the design of the core and for the stability tests. Second, to obtain knowledge and develop an understanding of heat transfer and fluid flow phenomena both

FIGURE 18
ROD OSCILLATOR MECHANISM
CONTROL SYSTEM

to increase the performance capabilities of boiling water reactors and to provide a basis for improvement of boiling water reactor design.

Table 11 presents a detailed breakdown of the experiments and analyses falling under Task B.

3.2.0 Experimental Progress To Date

The only subtask for which quantitative data have been obtained is under the burnout tests. A 0.540-inch O.D. rod was placed concentrically in a 0.875 I.D. tube with a heated length of 8.5 feet. The following results were observed:

1. Modifying the surface of the rod by sandblasting has a slightly adverse effect of burnout at 1,000 psia.
2. Best-fit curves through data at 1,000 psia and 0.6, 0.8, and 1.1 lb/sec flow are shown in Figure 19. Subcooling was varied to establish these curves. The lower right end of each curve is the point of minimum subcooling and proceeding up and to the left increases the subcooling. All four curves appear to approach a common limit as the subcooling is increased. Further investigations at high subcooling are planned.
3. Best-fit curves through data at 0.8 lb/sec flow and at 600, 1000, and 1450 psia are shown in Figure 20. The pressure effect is clearly observable. A further examination of the pressure effect is planned. Equipment is in preparation for prototype two-phase pressure drop measurements. An observational test section has been placed into service which

TABLE 11

TASK B - HEAT TRANSFER AND FLUID DYNAMICS

BURNOUT	STEAM VOIDS	HYDRAULIC STABILITY	TWO-PHASE PRESSURE DROP
<ol style="list-style-type: none"> Single rod burnout as a function of L/D, annular clearance, mass rate, pressure, cosine power, distribution and loop characteristics. Multirod burnout (including prototype) sufficient to establish correlation with single rod data. Flow dynamics at burnout, both local and loop (flow stability). Investigation of certain methods for raising the burnout limit. 	<ol style="list-style-type: none"> Development of one or two methods to effectively measure steam voids. Make void measurements in subcooled and low quality zones. Make void measurements for correlation with data in one or more of other subtasks. 	<ol style="list-style-type: none"> Modify the method for controlling power so that step changes in power may be accomplished. Vary pump speed by controlling line voltage. Operate the loop near the threshold of stability to determine the effect of the above two controls. Compare with stability under natural circulation. Investigate the coupling between two vertical heated parallel channels. 	<ol style="list-style-type: none"> Pressure drop in a multirod configuration (including prototype). Pressure drop through basic geometrical configurations, with and without heat addition. (Obtain simultaneous void information.)

FIGURE 19
BURNOUT DATA
1000 PSIA

FIGURE 20
 BURNOUT DATA
 FLOWRATE 0.8 LB/SEC

permits direct observation of boiling and two-phase flow in a two dimensional channel. It shows promise for developing an understanding of the basic mechanism of burnout and associated two-phase flow phenomena.

4.0 TASK C - EXPERIMENTAL PHYSICS PROGRAM

4.1.0 Objective

The objective of this task is to obtain experimental data on nuclear parameters important to the successful attainment of long fuel exposures. Measurements will be made of the variation of isotopic concentration of fuel atoms with burnup over a span of about 10,000 MWD/(m)T. These measurements will be compared with analytical burnup models.

Resonance capture of neutrons in U-238 plays an important role in the long term reactivity effects of fuel. The available experimental information on resonance escape for typical boiling water fuel lattices will be extended through a series of critical experiments. It is the goal of both of these experimental programs to increase the present knowledge of long term reactivity effects in uranium oxide fuel. This information is essential to the optimum design of reactors with extended fuel lifetime.

4.2.0 Schedule

The physics effort in Task C during the next four months will be directed to the preliminary evaluation of isotopic concentration of Fuel Cycle fuel as a function of time. A series of studies will be made to predict the time dependent isotopic concentration of the Fuel Cycle elements. Studies will be initiated to show the effects of changes

in effective nuclear cross sections on the isotopic concentrations. This will lead to the evaluation of the sensitivity of the measurements to place limits on the effective cross sections.

The Task C effort will also include preliminary planning of experiments to measure resonance capture of conversion ratio of boiling water UO_2 fueled lattices.

5.0 PROGRAM SCHEDULE:

The program schedule on the following page gives the best estimate for completion of the Fuel Cycle Project tasks. The dates shown are tentative and may be subject to some variation.

Because of the long period between proposed submission and contract signature, the heat transfer facilities at San Jose, California were committed to other programs. As a result, Task B has been scheduled over an eighteen-month period. Two clarifying definitions are in order under Task B. "Basic" Burnout Tests designates experiments which will be run to raise the burnout limits and investigate the mechanism of burnout. "Basic" Hydraulic Instability Studies on the large loop are intended to establish fundamental relationships which control hydraulic instability.

FIGURE 21
AEC FUEL CYCLE PROGRAM - TENTATIVE PROGRAM SCHEDULE