JUL 25 1963

GEAP 3771-1

July 1, 1961

EURAC - 68
Informal Euratom-AEC Research
and Development Report

Cop. 2

MASTER

HIGH PERFORMANCE UO_2 PROGRAM FIRST QUARTERLY PROGRESS REPORT

APRIL - JUNE 1961

Contract AT(04-3) - 189

Facsimile Price \$ 3.60

Microfilm Price \$ 1.25

Available from the
Office of Technical Services
Department of Commerce
Washington 25, D. C.

GENERAL

ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT

SAN JOSE, CALIFORNIA

Encl 1

DISCLAIMER

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

DISCLAIMER

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

Informal Euratom-AEC Research
and Development Report

GEAP-3771-1

HIGH PERFORMANCE UO_2 PROGRAM
FIRST QUARTERLY PROGRESS REPORT

APRIL - JUNE 1961

PREPARED FOR EURATOM AND THE ATOMIC ENERGY COMMISSION

by

B. Weidenbaum

Under Contract AT(04-3)-189
Project Agreement No. 17

General Electric Company
Atomic Power Equipment Department
Vallecitos Atomic Laboratory
Pleasanton, California

July 1, 1961

APPROVED BY:

E. W. O'Rourke, Manager
Vallecitos Irradiation Services
Operation

A. N. Holden, Manager
Metallurgy and Ceramics

S. Naymark, Manager
Fuels & Materials Engineering

LEGAL NOTICE

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

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

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

EXTERNAL REPORT DISTRIBUTION LIST

	<u>No. of Copies</u>
U. S. Atomic Energy Commission San Francisco Operations Office 2111 Bancroft Way Berkeley, California	(3)
Division of Reactor Development U. S. Atomic Energy Commission Washington 25, D. C.	(4)
Deputy for Euratom U. S. Mission to Euratom Communities 13 Rue de la Loi Brussels, Belgium.	(4)
William D. Manly Oak Ridge National Laboratory P. O. Box X Oak Ridge, Tennessee	(1)
Assistant Director of Naval Reactors Division of Reactor Development Attention: I. H. Mandil USAEC Washington 25, D. C.	(1)
Manager Westinghouse Electric Corporation Atomic Power Division Bettis Plant, Clairton Site P. O. Box 1526 Pittsburgh 30, Pennsylvania Attention: A. Squire	(1)
General Manager Knolls Atomic Power Laboratory P. O. Box 1072 Schenectady, New York Attention: W. Cashin	(1)
Robert Macherey Argonne National Laboratory 9700 South Cass Avenue Argonne, Illinois	(1)
F. W. Albaugh Hanford Atomic Power Operation General Electric Company Richland, Washington	(1)

EXTERNAL REPORT DISTRIBUTION LIST
(continued)

	<u>No. of Copies</u>
USAEC Washington 25, D. C. Attention: Chief, Water Reactors Branch Civilian Reactors Division of Reactor Development	(2)
USAEC Washington, D. C. Attention: Chief, Water Systems Project Branch Army Reactors Division of Reactor Development	(1)
USAEC Chicago Operations Office 9700 South Cass Avenue Argonne, Illinois Attention: Director, Reactor Programs Division	(3)
General Nuclear Engineering Corporation P. O. Box 245 Dunedin, Florida Attention: J. M. West	(1)
Combustion Engineering, Inc. Nuclear Division Prospect Hill Road Windsor, Connecticut Attention: W. H. Zinn	(1)
Westinghouse Electric Corporation Atomic Power Department P. O. Box 355 Pittsburgh 30, Pennsylvania Attention: Dr. W. E. Shaupp	(1)
Allis-Chalmers Manufacturing Company Nuclear Energy Division Milwaukee 1, Wisconsin Attention: C. B. Graham	(1)
Allis-Chalmers Manufacturing Company Nuclear Power Department P. O. Box No. 8697 Washington 11, D. C. Attention: H. Etherington	(1)
Advanced Technology Laboratories Division of American-Standard 369 Whisman Road Mountain View, California Attention: S. A. Bernsen	(1)

EXTERNAL REPORT DISTRIBUTION LIST
(Continued)

	<u>No. of Copies</u>
Northern States Power Company Minneapolis 2, Minnesota Attention: D. F. McElroy	(1)
Consumer's Power Company 212 W. Michigan Avenue Jackson, Michigan Attention: A. Bethel	(1)

ACKNOWLEDGEMENT

This Research and Development Program, which is sponsored by Euratom and the AEC, is being conducted by the General Electric Company with the following personnel contributing to the project during this quarter.

Project Scientist	B. Weidenbaum
Direct Pressure Measurement Study	H. H. Klepfer M. B. Reynolds
Fuel Central Melting Study	T. J. Pashos
Fuel Assembly Design	M. F. Lyons
Burnout Safety Limit	E. Janssen
Water Chemistry	R. S. Gilbert
Loop Decontamination	S. Seigel
Loop Modifications	R. G. Kuhne
Design and Analysis	D. W. Howell D. E. Hafer E. A. Haglund D. J. Liffengren
Safeguards	W. N. Lorentz

TABLE OF CONTENTS

	<u>Page No.</u>
Introduction	1
Summary	3
Task I - Measurement of Fission Gas Pressure in Operating Fuel Elements	
1. Task Objective	5
2. Technical Considerations	5
3. Experimental	12
4. Current Status of Task I	14
Task II-A - Loop Modification	
1. Task Objective	15
2. Description of Existing Loop	15
3. Loop Component Changes and Additions	17
4. Preliminary Burnout Analysis	19
Task II-B - UO ₂ Fuel Performance as Related to Degree of Center Melting	
1. Task Objective	20
2. Fuel Fabrication	20
3. Burnout Heat Transfer Study	20
4. Experiment Design	22

ILLUSTRATIONS

	<u>Page No.</u>
Figure 1 - Fuel Pressure Transducer, Schematic	13a
Figure 2 - Fuel Pressure Readout Instrumentation, Schematic	13b
Figure 3 - Pressurized Water Loop Prior to Modification	15a
Figure 4 - GETR Elevations	16a
Figure 5 - Radial Locations of GETR Experimental Facilities	16b
Figure 6 - Pressurized Water Loop After Modification	17a

INTRODUCTION

A better understanding of the maximum operating characteristics that can be achieved with the use of UO_2 as a reactor fuel is the primary purpose of this program for Euratom and the Atomic Energy Commission. During this program, work will be undertaken in two areas that have been of concern to the reactor core designer for a long time, viz. fission gas release and central melting in fuel rods.

In this program, the rate of fission gas release will be measured by a direct method while the fuel rod is residing in the Vallecitos Boiling Water Reactor. A suitable transducer system will be developed for this purpose. Fuel rods with sintered pellets at two densities (95% and 90%) will be tested at a temperature near the melting point of UO_2 (2740°C) and at a temperature of 1200°C . At the lower temperature, no considerable amount of mass transfer of UO_2 is expected to occur, nor is an appreciable amount of grain growth anticipated. Fuel rods fabricated by swaging arc fused UO_2 powder will also be tested concurrently with the rods containing pellets of corresponding density (90%). These tests will furnish information regarding pressure buildup during the transient period, as well as during steady state operating conditions, of reactor operation.

The second question with which this program will be concerned has to do with the amount of central melting that can be tolerated in a UO_2 fuel rod while it is being used in a reactor. Fuel rod design has been, and even today is, arbitrarily limited by the attainment of a central temperature well below the melting point of UO_2 . Originally, the reason for this limitation was to avoid the release of fission from the UO_2 with the resultant buildup of pressure within the tube. There is then the eventual possibility of fuel rod failure by cladding rupture as a result of the high internal pressure developed during extended irradiation. Recently, consideration has been given to fuel rod designs with a plenum chamber or other means to reduce the pressure within the rod from fission gas release. Under these circumstances it may be reasonable to operate UO_2 fuel rods with a higher central temperature than used previously.

The central temperature limitation also imposes a limit on the specific power that can be attained for a given fuel geometry. Because fuel costs can be influenced by the specific power, a better understanding of the limiting central temperature for UO_2 fuel could possibly result in a lower fuel cost.

In this program, UO_2 fuel will be irradiated in the Pressurized Water Loop in the General Electric Test Reactor with progressively higher central temperatures. A series of four fuel assemblies will be irradiated at heat fluxes of 600,000 - 800,000 - 1,000,000 - 1,200,000 Btu/hr-ft² to attain different central temperatures. At the same time, of course, the extent of central melting will vary with little or no melting expected at the lowest heat flux. At the highest heat flux as much as one quarter of the fuel cross-section may be molten.

It is a primary purpose of this phase of the program to obtain a relationship between the extent of central melting and the length of time before the fuel rod fails as a result of interaction between the fuel and cladding. Care will be taken to minimize the likelihood of failure occurring from causes external to the rod. A suitable burn-out safety factor will be used to eliminate fuel failure from this cause. The burn-out safety factor to be used will be obtained experimentally under conditions simulating those that will be obtained during fuel irradiation in the loop. Similarly, care will be taken to maintain coolant chemistry that will lead to low corrosion rates and crud formation. These precautions should also be helpful in reducing the likelihood of fuel failure from ~~causes outside the fuel rod~~.

The three major tasks in the program and their objectives can be outlined as follows:

Task I - Measurement of Fission Gas Pressure in Operating Fuel Elements.

This task will measure directly the fission gas pressure in UO_2 filled fuel rods as functions of temperature and fuel configuration during transient and steady state reactor operation over an extended period of time.

Task II-A - Loop Modification.

The design, fabrication, and installation of the modifications to the Pressurized Water Loop in GETR required for the Task II-B experimental program is the objective of this task.

Task II-B - Performance of UO_2 Fuel.

This task will endeavor to obtain a relationship between the amount of central melting in a fuel rod and the burnup that can be attained before fuel-cladding interaction results in rod failure.

SUMMARY

The work performed on the three major tasks of the High Performance UO_2 Program during this quarter is summarized in the following.

Task I - Measurement of Fission Gas Pressure in Operating Fuel Elements.

1. Approval was received from the AEC-SAN for the fuel assembly design to be used.
2. The following components for the first fuel assembly containing two rods have been fabricated - VBWR Tube Bundle, Pressure Transducers, Pressure Vessel Penetration Flange, and Read-out Instrumentation.

The following items are currently being fabricated - Fuel Rod Parts, UO_2 Pellets, 95 percent density with 4.6% and 7.6% U-235.

The fuel assembly is scheduled for VBWR irradiation starting August 1961.

3. Fully enriched UO_2 powder has been ordered to fulfill the U-235 requirements for the remaining four rods to be fabricated. The necessary materials for these transducer equipped rods have been ordered.

Task II-A - Loop Modification

1. The reference design for the loop modification has been completed.
2. The detail design on the basis of the reference design is in progress. Procurement of material is being initiated for each item as soon as detail design drawings have been completed.
3. An ion exchange column of the flush and fill type with a removable head has been designed and tested to assure its suitability for service in the PWL.
4. A preliminary burnout heat transfer analysis indicated that the addition of a third pump in series with the existing pumps in the PWL system would increase the available coolant flow by 22 percent with a resultant increase of approximately 8 percent in the burnout safety limit. A third pump is therefore not being added to the loop system.
5. In connection with the hazards evaluation of the experimental program to be done in the PWL, a preliminary analysis has been made of three major possible accidents - (1) fuel element failure, (2) loss of flow, (3) pipe rupture.

Task II-B - Performance of UO₂ Fuel

1. A heat transfer, burnout study is in progress to establish the upper limit of heat flux for the experimental program in the PWL. The information obtained will also be useful for the hazards analysis of the program.
2. Requirements have been established for PWL operating parameters to assure that fuel rod failure will be the result of UO₂ melting and not of some limitation in the loop operation.
3. The design of the PWL test assembly and associated hardware is in progress.
4. Procurement of the ceramic grade and arc fused UO₂ powder required for the program is underway. Zircaloy-2 cladding for both the pellet filled rods or powder filled swaged rods is either at hand or on order.
5. Calculations are being made of the amount of central melting that can be expected at the heat fluxes to be run in the experimental program.

TASK I - MEASUREMENT OF FISSION GAS PRESSURE IN OPERATING FUEL ELEMENTS

1. Task Objective

This task has as its objective the direct measurement of the free fission gas pressure which develops in UO_2 filled fuel rods during extensive reactor operation as functions of operating temperature and fuel configuration. Suitably instrumented fuel rods are being developed and fabricated for this purpose. The irradiated fuel rods will subsequently be examined in the Radioactive Materials Laboratory to relate the fission gas release with the structural characteristics of the fuel.

2. Technical Considerations

The service life expectancy of a nuclear fuel element is a complicated function of both fuel material and cladding parameters. Apart from purely nuclear physics considerations, the characteristic of fissionable material which is most relevant to service life is the volume increase accompanying fission. When an atom fissions, two fission product atoms are produced each of which, on the average, is larger than the parent atom. In a solid fissionable material these fission product atoms must be accommodated either in interstitial or substitutional lattice positions. At very low burn-up the fuel lattice is able to accommodate the fission product atoms and is distorted locally about these atoms, but exhibits little or no macroscopic strain. As burn-up progresses the fuel material does exhibit macroscopic expansion increasing with burn-up and fuel operating temperature. The fact that approximately one out of every eight fission product atoms is an atom of krypton or xenon magnifies the problem since these atoms are normally insoluble in solids.

Practical fuel elements have, of course, been fabricated and used subject to the condition that operating temperatures be low enough to permit the fuel material lattice to retain the fission product atoms without gross expansion. However, as operating temperatures are increased, the limit of usefulness may be approached rather rapidly. As temperature increases, diffusion rates increase, permitting gaseous fission product atoms to diffuse to nucleation sites where they agglomerate to produce bubbles of gaseous phase. Also, as temperature increases the strength of the fuel material decreases, reducing the ability of the lattice to accommodate these

microscopic bubbles without gross swelling and escape of gas. There is then a limiting temperature above which a nuclear fuel is usable to retain fission gas. This temperature is not sharply defined and depends on a number of variables such as burn-up, fuel composition and fuel geometry. Since in metallic-clad fuel elements cladding failure may well result from excessive free fission gas pressure, this problem has been the subject of considerable study and investigation.

Pressure-Temperature Relationships in Fuel Elements

The pressure exerted by the free gas within a clad fuel element is a function of the quantity of gas, the void volume available for it to occupy, and the temperature distribution within the gas. The quantity of free gas in a nuclear fuel element is a function of the temperature history of the fuel element rather than the instantaneous temperature distribution; hence this subject will be deferred for later discussion. If the voids are all interconnected in the fuel body, the pressure is uniform throughout the void volume. If the free fission gas obeys the perfect gas law, the pressure, P , within the fuel element is given by the expression

$$P = \frac{nR}{\sum_i \frac{V_i}{T_i}}$$

in which n is the number of mols of free gas, R is the gas constant, and V_i and T_i are the volume and absolute temperature respectively of the i th void. This general expression is correct but of little use unless something is known of the void configuration in the fuel element. In some cases void configuration is known and the relation between pressure, temperature, and quantity of free fission gas may be established. If, for example, the fuel element consists of solid fuel pellets loosely fitted in a cylindrical metal cladding with minimum end clearance one may assume the void volume to be that of the annulus between fuel and cladding and the temperature to be approximately constant throughout the void volume. The expression for pressure then reduces to

$$P = \frac{nRT}{V}$$

In the case of a cylindrical tube filled with uniformly packed powder of void fraction, f , it may be shown that

$$P = \frac{nR}{2\tilde{n}f \int_0^A \int_0^L \frac{rdrdx}{T(r,x)}}$$

in which A and L are the radius and length respectively of the fuel segment and $T(r,x)$ is the absolute temperature. The temperature distribution may be determined from the thermal conductivity and power distribution in the fuel segment.

In general, neither the void configuration nor the temperature distribution in a fuel element is known accurately enough to permit calculation of the pressure existing in a fuel element during reactor operation. It is, then, desirable to measure directly fission gas pressure in selected, typical fuel segments if an understanding is to be obtained of the pressure existing in a fuel rod during transient and steady state reactor operation.

By connecting a pressure transducer to a fuel segment, its void volume is increased by an amount equal to the connected volume of the transducer.

The expression for pressure then becomes

$$P = \frac{nR}{\frac{V_1}{T_1} + \sum_i \frac{V_i}{T_i}}$$

where V_1 and T_1 are the volume and temperature respectively of the transducer. If the addition of the pressure transducer to the system is not to sensibly alter the observed pressure from the value which would exist in its absence, it is necessary that

$$\frac{V_1}{T_1} \ll \sum_i \frac{V_i}{T_i}$$

If $\frac{V_1}{T_1}$ is actually much more than $\sum_i \frac{V_i}{T_i}$, the measured pressure is

relatively insensitive to the void configuration and temperature distribution in the fuel and becomes a measure of the quantity of free fission gas present.

Another configuration of particular interest is a fuel segment of hollow fuel pellets shrink-fitted into a thick walled cladding tube. If the pellets were fitted carefully enough, the entire void volume might be confined to the central hole in the pellets. The gas temperature in this central hole, neglecting end effects, would then represent the central temperature of the fuel and might easily be determined by pressure measurement. The central temperature T_c would be given by

$$T_c = \frac{V_c}{\frac{nR}{P} - \frac{V_1}{T_1}}$$

where: T_c = Central Temperature
 V_c = Volume of Central Void
 V_1 = Volume of Pressure Transducer
 T_1 = Temperature of Pressure Transducer

Such an experimental temperature determination would be useful in extending fuel thermal conductivity data beyond its present limits.

Temperature Distribution in Fuel Segments

The spatial temperature distribution in a segment of nuclear fuel element influences not only the fission gas pressure but also the mechanical properties of the fuel material and consequently its ability to retain fission gas. The steady state temperature distribution within a fuel segment is a function of specific power and thermal conductivity of the fuel. Although both of these quantities, unfortunately, are temperature dependent, it is still possible to arrive at general relations for temperature distribution in common fuel element configurations.

For any region having volume V and bounded by surface S within a segment of nuclear fuel, one may write

$$\int_V \rho c \frac{\partial T}{\partial t} dV = \int_V m(r) dV - \int_S \underline{q} \cdot \underline{dS}$$

in which: ρ = Fuel Density in dV
 c = Specific Heat of Fuel in dV
 $m(r)$ = Heat Generation per Unit Volume in dV
 q = Heat Flux through Surface Element dS
 T = Temperature
 t = Time

Under steady state conditions $\frac{\partial T}{\partial t} = 0$

$$\text{and} \quad \int_V m(r) dV = \int_S \underline{q} \cdot \underline{dS}$$

Using this expression and the Fourier heat conduction relation, $\underline{q} = -K(T) \nabla T$, it is possible in principle to calculate the temperature distribution in any fuel segment if the temperature dependence of $K(T)$ and the spatial variation of $m(r)$ are known. In the case of a cylindrical fuel segment of thermal conductivity $K(T)$ and in which power generation is $m(r)$ it may be shown* that

$$m(r) = -K(T) \frac{1}{r} \frac{\partial}{\partial r} \left(r \frac{\partial T}{\partial r} \right)$$

$$\text{Integrating:} \quad \int m(r) r dr = -K(T) \left(r \frac{dT}{dr} \right) + C$$

$$\text{At } r = R_o, \quad \frac{dT}{dr} = 0.$$

$$\text{Hence} \quad \int_{R_o}^r m(r) r dr = -K(T) r \frac{dT}{dr} = P(r)$$

$$\text{Then, integrating again} \quad \int_{R_o}^r P(r) \frac{dr}{r} = - \int_{T_o}^{T(r)} K(T) dT$$

* By use of the Gauss Theorem and with the assumption $K(T)$ is independent of r .

If $m(r)$ and $K(T)$ are known from experimental data, T_r may be determined as a function of r by graphic integration. In the simplest case, in which m and K are constants and R_o is zero, (i.e. a solid cylinder) an analytical solution may be obtained.

$$P(r) = \frac{mr^2}{2}$$

and

$$\frac{m}{2} \int_0^r r^2 \frac{dr}{r} = -K \int_{T_o}^{T(r)} dT$$

where T_o is the temperature at the center, $r = 0$.

Integrating

$$T(r) = T_o = \frac{mr^2}{4K}$$

or

$$T(r) = T(R) + \frac{m(R^2 - r^2)}{4K}$$

in terms of the fuel surface temperature $T(R)$.

The quantity $\int_{T_R}^{T_o} K(T) dT$ may be easily shown to be equal to $\frac{1}{4\pi}$ times

the linear power density in the cylindrical fuel segment. It will be noted that for a given linear power density, the radial temperature drop across the fuel is independent of radius. The particular linear power density which can be obtained without exceeding the melting point of the fuel at the center depends upon the fuel surface temperature T_R , but will always decrease with increasing T_R regardless of the form of $K(T)$.

The Effect of Temperature on Fuel Behavior

As was mentioned above, the behavior of nuclear fuel material may be expected to be quite sensitive to its operating temperature because of the fact that increasing temperature increases the gaseous fission product mobility and decreases mechanical strength so that the fuel material is less able to contain "bubbles" of agglomerated gas. The actual temperatures existing in a fuel segment will vary throughout the segment. There

is superimposed on the radial temperature variation due to the thermal conductivity of the fuel material, an axial variation due to the variation in neutron flux with position in the reactor core. At a given position in the fuel rod, the maximum fuel temperature in a metal clad fuel element is equal to the coolant temperature plus the sum of the coolant-clad film temperature drop, cladding drop, fuel-cladding gap drop (where such exists), and finally the temperature drop across the fuel itself. Coolant temperature and the film temperature drop are effectively constants, dictated by the reactor design rather than that of the fuel element.

Of the temperature drops within the fuel element, that across the fuel-cladding gap alone may be minimized without altering other fuel parameters. Because heat transfer across the fuel-cladding gap is by gas conduction, this temperature drop is apt to be undesirably large unless the gap is minimized. When cylindrical solid fuel pellets or rods are fitted loosely into a metal tube cladding, there is a tendency for thermal expansion of the fuel to close the gap between the fuel surface and the inner surface of the cladding. If the expansion is great enough to establish solid contact between fuel and cladding, gas conduction is replaced by solid conduction. The gap temperature drop in this case is reduced to a negligible value. By proper adjustment of the cold gap clearance between fuel pellets and cladding this condition can be obtained in practice for the case of oxide fuel pellets in tubular metal cladding. There is a minimum value of cold gap clearance above which central melting occurs well before sufficient expansion takes place to establish contact between fuel and cladding. A plot of the power capability of a pellet type fuel rod against cold gap clearance show a sharp discontinuity at this critical gap width.⁽¹⁾

Post-irradiation examinations of a large number of UO_2 pellet-type fuel rods have shown that thermal stresses result in extensive cracking of the fuel pellets whenever significant power density is attained. As the power density is increased, central melting or "coring" occurs, beginning at the region of highest neutron flux intensity and extending outward axially and radially as the average power density increases.

(1) C. N. Spalaris, R. K. Baird, "High Heat Flux Experiment for Dresden Type Fuel Elements", GEAP-3005, April 7, 1958.

Fission gas release from the UO_2 fuel occurs in all cases, ranging from normally negligible release at all temperatures due to fission recoils to effectively complete release from regions in which melting occurs. Numerous post-irradiation measurements of fractional fission gas release have been made by puncturing irradiated fuel elements, collecting the free fission gas, and comparing with chemical burn-up data. As was indicated above, the pressure exerted by free fission gas in a fuel rod during operation is a rather complicated function of fuel configuration power density, and thermal history. Measurements, during operation, of fission gas pressure developed in typical fuel rod segments are expected to provide data relevant to future fuel element design.

3. Experimental

Six fuel rods with attached pressure transducers are to be irradiated in the VBWR as follows:

- (1) UO_2 pellets, 95% of theoretical density, $T_o > 2750^\circ\text{C}$
- (2) UO_2 pellets, 95% of theoretical density, $T_o \approx 1200^\circ\text{C}$
- (3) UO_2 pellets, 90% of theoretical density, $T_o > 2750^\circ\text{C}$
- (4) UO_2 pellets, 90% of theoretical density, $T_o \approx 1200^\circ\text{C}$
- (5) UO_2 powder, 90% of theoretical density, $T_o > 2750^\circ\text{C}$
- (6) UO_2 powder, 90% of theoretical density, $T_o \approx 1200^\circ\text{C}$

Central temperature T_o will be adjusted by adjustment of fuel enrichment and fuel-cladding gap.

The fuel rods in which fission gas pressure will be measured are to be irradiated in a special tube bundle in a VBWR channel. This tube bundle will accommodate from one to five fuel rods. To each fuel rod will be connected a pressure transducer located approximately 6 inches above the top of the reactor core assembly. Transducer lead penetration into the pressure vessel will be via the so-called thermocouple nozzle flange.

The fuel rods themselves will be based on a nominal fuel diameter of one-half inch, and an active length of approximately thirty inches. Because the investigation is concerned with fuel rather than cladding behavior, cladding will be thicker than is normal practice. Cladding material will be AISI Type 304 stainless steel, with a wall thickness of 0.050 inch.

A pressure transducer of the Booth-Cromer type has been developed for the pressure measurements. The principle of this device is shown in Figure 1. If the contact is so adjusted that electrical contact is just made with zero unbalance pressure on the diaphragm, a very slight pressure excess on the secondary side of the diaphragm will open the contact. The secondary pressure at the instant of contact opening is then equal to the primary pressure plus the slight differential required to open the contacts. This differential is negligible relative to the primary pressures to be observed. By suitable design the diaphragm motion may be so limited that the diaphragm is not plastically deformed, even by very large pressure unbalances. This feature permits a very large pressure range to be studied with relatively little changes in fractional accuracy.

The basic instrumentation needed with such a pressure transducer is shown in Figure 2. Its operation is as follows: assume that the selector switch (5) has connected the electronic relay (6) to the transducer associated with fuel rod number 1. With fill valve (14) open, pressure in the gas manifold (connected to the secondary side of all the Booth-Cromer transducers) will rise slowly and uniformly at a rate set by regulator (16). At the instant the manifold pressure slightly exceeds the pressure in rod number 1, the contacts in the transducer associated with this rod will open. Operating through the electronic relay (6) and relay center (7), the opening of these contacts will command printer (9) to print out the pressure indicated by transducer (10). The time indicated by clock (8), and the number of the transducer (in this case number 1) to which the selection switch is connected. Simultaneously the relay center will close valve (14) and open valve (13) permitting the manifold pressure to fall at a rate set by regulator (15). When the manifold pressure reaches a value slightly above atmosphere, the low limit switch (11) acting through the relay center will close valve (13), open valve (14), and cause the selector switch (5) to step to transducer (2) connected to rod number 2. The pressure will again increase until the contact in transducer (2) actuates the printer and initiates the cycle described above. This cycle will repeat indefinitely, indicating the pressure in the fuel rods in turn.

Figure 1. FUEL PRESSURE TRANSDUCER, SCHEMATIC

Figure 2. FUEL PRESSURE READ-OUT INSTRUMENTATION

4. Current Status of Task I

At the present time, the following components have been fabricated:

- a. VBWR Tube Bundle
- b. Pressure Transducers
- c. Pressure Vessel Penetration Flange
- d. Read-Out Instrumentation

The following are being fabricated:

Fuel Rod Parts

95% Dense Fuel Pellets

Fuel material is available for fabrication of 90% dense pellets and powder filled rods. It is planned to irradiate two rods (95% dense pellets) initially beginning with the August VBWR shutdown. This irradiation will be followed by simultaneous irradiation of the remaining four rods.

TASK II-A - LOOP MODIFICATION

1. Task Objective

Modifications to the existing Pressurized Water Loop in the General Electric Test Reactor will be designed, fabricated and installed to permit the experimental objectives of Task II-B to be accomplished. The modifications will be checked out and shakedown runs made in the loop before the start of the experimental program.

2. Description of Existing Loop

The Pressurized Water Loop in the GETR was designed to operate as a sub-cooled forced convection system. General facility characteristics of the existing system are as follows:

Maximum heat removal capacity - 500 KW

Maximum flow - 150 gpm without test element installed

Usable test space - 2.340" ID x 36" long

Design pressure and temperature - 1200 psig and 650°F
(Based on facility tube limitations)

The loop was constructed primarily of carbon steel. Exception to this are the in-pile facility tube, main circulating pumps, ion exchange resin bed liners, and instrument tubing and fittings, which were fabricated from 300 series austenitic stainless steel.

A schematic diagram of the existing system is given on Figure 3, General Electric Drawing 196E823. Flow is induced by two canned rotor pumps, which are connected in series. Flow passes from the pumps to a trim heater through an automatically actuated flow control valve. The trim heater allows for a final temperature adjustment before flow enters the test section. From the trim heater, flow proceeds in a horizontal piping run through a biological shield, the reactor pool, the reactor vessel wall, to a vertical facility tube within the reactor pressure vessel. The facility tube extends downward through the reactor core and vessel bottom into a sub-pile gallery. Both a top horizontal piping run from the inlet side of the facility tube and the facility tube itself are double wall and annular-construction to provide thermal insulation throughout the submerged areas. The annulus is pressurized with nitrogen to 130 psig to

INSTRUMENTATION TABLE									
NO.	ITEM	LOOP CONTROL PANEL			REACTOR CONTROL ROOM			AUTO	SCRAM
		READ OUT	RECORD	ALARM	RECORD	ALARM	RECORD		
1	PRI. LOOP FLOW	X		LOW	X	LOW	LOW		
2	LOOP PRESSURE	X		HIGH & LOW	X	HIGH & LOW	HIGH & LOW		
3	TEST SECTION INLET TEMP	X		HIGH	X	HIGH	HIGH		
4	TEST SECTION OUTLET TEMP	X		HIGH	X	HIGH	HIGH		
5	FUSION PRODUCTS MONITOR	X		HIGH	X	HIGH	HIGH		
6	PRESSURIZER WATER LEVEL	X		LOW & HIGH	X	LOW & HIGH	LOW & HIGH		
7	PRESSURIZER PRESSURE	X							
8	PRESSURIZER TEMP	X							
9	LOOP WATER CONDUCTIVITY	X							
10	PRI. HEAT EXCH. COOLANT INLET TEMP	X		LOW		LOW			
11	PRI. HEAT EXCH. COOLANT OUTLET TEMP	X							
12	PRI. PUMP COOLANT TEMP	X		HIGH		HIGH			
13	PRI. PUMP ELECT. CURRENT	X		HIGH & LOW		HIGH & LOW			
14	PRI. PUMP OUTLET PRESS.	X							
15	CLEAN-UP LOOP TEMP	X		HIGH		HIGH			
16	LOOP GROSS RADIATION	X		HIGH		HIGH			
17	CLEAN-UP LOOP FLOW	X		LOW		LOW			
18	DIFF. PRESS. ACROSS PRIMARY PUMP	X							
19	REGEN. HEAT EXCH. OUTLET TEMP	X							
20	FACILITY TUBE AIR SUPPLY			HIGH		HIGH			
21	DIFF. PRESS. ACROSS TEST SECTION	X		HIGH & LOW	X	HIGH & LOW	HIGH & LOW		
22	LOOP TEMPS.	X							
23	PUMP COOLANT FLOW	X							
24	ION EXCHANGER FLOW	X

REF. DWGS.

- 342AB63-INSTRUMENT SYMBOLS
- 196F.833-GETR BOILING WATER TEST LOOP
- 708ZD30-REACTOR CONTROL

LEGEND

PARAMETER (LETTER)	FUNCTION (LETTER)	VALVE	NUMBER	REACH
F FLOW	C CONTROL	V	I	R
P PRESSURE	T THROTTLE		Z	
T TEMPERATURE	S STOP		ETC.	
L LEVEL	E CHECK			
D SAMPLE	D DRAIN			
M MAKE-UP OR FILL	V VENT			
E EMERGENCY	R RELIEF			
I INSTRUMENT				

AIS-AMMETER INDICATOR SWITCH

—INSTRUMENT IN REACTOR CONTROL ROOM

(R)—ALARM LIGHT ADJACENT TO INSTRUMENT

—ANNUNCIATOR ON LOCAL PANEL & SINGLE WARNING LIGHT ON LOOP INSTRUMENT CUBICLE IN REACTOR CONTROL ROOM

—INSTRUMENT ON FACILITY OR LOCAL PANEL

SS-SAMPLE STATION
 * PRESSURE SWITCH, 3-15 PSI, SUPPLIED WITH LRS-36
 ** PRESSURE SWITCH, 3-15 PSI, SUPPLIED WITH FIS-50

NOTE: ALL VALVES AND PNEUMATIC OPERATED VALVES SHOWN IN THE DE-ENERGIZED POSITION. ALL MANUALLY OPERATED VALVES ARE SHOWN IN NORMAL OPERATING POSITION. ALL PNEUMATIC LINES TO BE 1/2 O.D. x .032 WALL SOFT COPPER TUBE

prevent water from seeping into the annulus in event of a leak. From the sub-pile gallery, the flow returns to a heat exchanger in the loop cubicle. Heat generated by the fuel is removed by the heat exchanger. A temperature controlled three-port proportioning valve and a bypass line are used in conjunction with the heat exchanger to provide temperature control by regulating the amount of flow that enters the heat exchanger. From the heat exchanger the coolant returns to the pump suction through a "Y" type strainer.

System pressure is provided by a pressurizer which connects to the circulating system at a point between the main heat exchanger and the pump inlet. Steam is generated in the pressurizer by six manually controlled 10 KW heaters. Pressure control is achieved by spraying sub-cooled water from the pump discharge into the steam phase in the pressurizer dome through an automatically controlled pressure actuated valve.

Water purity is maintained by a separate loop which is in parallel with the main loop across the main pumps. This loop contains a regenerative heat exchanger, a cooling heat exchanger, and two ion exchange resin beds. Piping and valves are arranged so that either or both of the ion beds can be used at any time. Auxiliary equipment also includes an emergency cooling system on the facility tube and a vent condenser on the pressurizer vent system. The emergency cooling system is used to supply water to the facility tube in the event of a loop depressurization. The condenser vent removes condensibles from the steam-gas mixture which is periodically vented from the condenser to the building exhaust stack. Drain, vent, isolation, and safety relief valves are installed throughout the system as required, and sample lines are run through coolers to a common sample station on the outside of the cubicle shield wall. Reach rods also extend to the outside of the cubicle to manipulate essential manually operated valves. De-mineralized and deaerated make-up water is provided by a system which is located near the Boiling Water Loop and which serves both facilities.

Figures 4 and 5 illustrate the Pressurized Water Loop position with respect to the reactor core and adjacent facilities.

Figure 5. G.E.T.R. ELEVATIONS

3. Loop Component Changes and Additions

Both the replacement of certain parts and systems in the loop, and modifications and additions to the loop will be necessary to effect the objective of the experimental program in Task II-B. A "Preliminary Piping and Instrument Diagram" for the modified Pressurized Water Loop is shown on Figure 6. The additions and modifications to the loop and containment cubicle, to a great extent, will be equipment to handle the after-effects of a failed fuel rod and to obtain more accurate information during the irradiation period. This would include removal of the defective fuel, removal of contamination in the loop, and initial containment and removal of contamination in the loop cubicle. The above considerations may be itemized as follows:

A. Replacement and Additions to the Main Loop

- a. Facility tube isolation manual operated valves will be replaced with air operated valves for remote operation.
- b. Miscellaneous vent valves will be over-hauled and/or replaced.
- c. A coupon station for decontamination studies will be added.
- d. A fission product trap will be provided to prevent UO_2 particles from infiltrating the loop after fuel failure.

B. Auxiliary Systems

- a. Decontamination System - A decontamination system to facilitate the decontamination procedure, and thereby minimize reactor down time due to high pressurized water loop activity, will be installed. This will be accomplished by designing the system to permit decontamination of the in-core facility tube, after which the reactor can re-commence power operation, and subsequent decontamination of the major portions of the loop while the reactor is at power.

The decontamination system will be designed for 100 psig pressure with the following major components:

- (1) One 125 gal. chemical mix tank.
- (2) Circulating pumps.
- (3) A strip heater to maintain chemicals at desired decontamination temperature.

Figure 6. RADIAL LOCATIONS OF G. E. T. R. EXPERIMENTAL FACILITIES

- (4) A standby heat exchanger capable of removing the heat generated by gamma heating of the facility tube.
- (5) Waste storage tanks.

b. Clean-Up System

(1) Ion Columns

A replacement in the clean-up system will be two cartridge type ion columns with provisions for glass wool crud filters. A remote flush and fill feature will be incorporated in the design in order to handle the high activity expected after fuel rod failure. Existing shielding casks will be modified.

(2) Fission Product Monitor

Operating experience has proven that the existing Fission Product Monitor is not completely reliable. This system will be removed. High activity detection will be accomplished with gamma monitors (discussed in instrumentation section) supplemented by laboratory water chemistry analysis.

c. Shielding

Additional shielding will be installed in the area between the Pressurized Water Loop and the General Atomic Loop operator station.

d. Water Chemistry Equipment

The existing sample system will be modified to permit the use of the most suitable current water chemistry techniques.

C. Instrumentation Changes and Additions

- a. Two temperature elements upstream and two temperature elements downstream of the facility tube will be added in order to improve fuel power output determination accuracy. These temperature elements will be connected to a common differential temperature recorder (DTR), in upstream and down stream pairs, to provide two independent differential temperature signals.

- b. Two thermocouples upstream and one thermocouple downstream of the facility tube will be added. These will be connected to existing temperature read-out instrumentation and will be used as a further check on facility tube differential temperature.
- c. An additional flow element will be installed in the main loop in series with the existing element. Existing instrumentation will be used to provide two independent flow readings.
- d. Radiation monitoring equipment will be installed. This equipment will consist of a 5 channel, 6 decade radiation monitoring system.

4. Preliminary Burnout Analysis

A preliminary burnout analysis was performed to determine allowable fuel bundle configuration, required mass flow rates, and temperature restrictions. Only steady state operating conditions were considered. A minimum design burnout ratio of 1.5 at the low flow scram point was assumed as a basis for this study. The range of parameters and geometries which were considered are as follows:

Pressure - 1100 to 1600 psia

Water temperature at the burnout point - 400°F to 525°F

Mass Flow Rate - $1 \times 10^6 \frac{\text{Lbs.}}{\text{Hr.} \cdot \text{Ft.}^2}$ to $8 \times 10^6 \frac{\#}{\text{Hr.} \cdot \text{Ft.}^2}$

Number of fuel rods - 2 to 5

The correlations which were used for this analysis were the Jens-Lotte correlation and the new Bettis correlation. The results of the analysis indicated that a compatible combination of parameters (within the range considered) and fuel bundle geometry could be selected without increasing the flow capacity, or increasing the pressure and temperature limitations of the loop. The addition of another pump in series with the existing pumps would increase the available flow approximately 22%. The resultant increase in burnout safety factor is approximately 8%. This increase in the burnout safety factor did not appear to be large enough to justify the installation of the third pump previously contemplated.

TASK II-B - UO_2 FUEL PERFORMANCE AS RELATED TO DEGREE OF CENTER MELTING

1. Task Objective

The experimental program will be directed to obtaining a relationship between the amount of center melting present in a fuel rod and the burnup that can be attained before failure occurs as a result of interaction between the fuel and cladding. Fuel rods which contain pellet fuel, and fuel rods made by swaging arc fused UO_2 powder will be evaluated and compared at four different heat fluxes to obtain the desired relationship.

2. Fuel Fabrication

Procurement of ceramic grade and arc-fused UO_2 powder was initiated. The quantities and enrichments ordered are sufficient to fabricate seven 4-rod PWL assemblies; two 8-rod VBWR assemblies, one powder filled and one pellet filled; spare rods of each type and enrichment to replace failed elements, and several prototype compacted powder rods to establish the powder rod fabrication process variables.

Zircaloy-2 cladding is available at APED suitable for use on the pellet filled rods. This cladding has passed all of the non-destructive examinations currently applied to Zircaloy-2 cladding for production use and is considered equal to or better than that which can be obtained by a new order. The tubing is 0.565" O.D. x .030" wall. Larger diameter tubing is required for the compacted powder rods in order to achieve the same finished fuel diameter. Procurement effort has just started on this larger diameter tubing.

3. Burnout Heat Transfer Study

An experimental study has been undertaken to determine the burnout heat flux over the range of operating conditions under consideration for the PWL experiments. Recent work at APED suggests that the actual burnout heat flux is lower than that calculated from either the Jens-Lotte⁽¹⁾ or Bettis burnout

(1) W. H. Jens and P. A. Lotte, Analysis of Heat Transfer, Burnout, and Pressure Drop for High Pressure Water. ANL-4627, May 1951.

correlations. The experimental study should establish a safe upper heat flux limit for the PWL tests and provide firmer grounds for obtaining safeguards approval to conduct the tests.

This program will be conducted in the large heat transfer loop facility at APED. All of the special equipment necessary to perform these burnout experiments has been fabricated, but the start of experimental work has been delayed approximately two weeks due to extended use of the heat transfer loop by other AEC programs.

A special insert has been fabricated to fit the loop test section consisting of 0.5" O.D., resistance-heated tubular test element to simulate the PWL loop fuel elements followed by an after-heater element that boils the cooling water and establishes the loop operating pressure. Three test elements have been fabricated varying from 29" to 36" in length. The demineralized water coolant flows through a concentric annulus around the tubular heater element with the lateral spacing maintained by electrically insulated pins that bear directly on the heated element surface. Both 1/8" and 1/4" cooling water annulus thicknesses will be investigated in the program.

To perform a burnout test, the loop operating conditions in terms of flow, pressure and test element inlet temperature are established using the after-heater element alone. Then the power to the special test element is increased until indications of incipient burnout in the test element are observed. Two measures of incipient burnout are employed, one being a rapid change in the resistance of the test element and the other a rapid rise in the test element wall temperature in the vicinity of the expected burnout point. The loop is instrumented to interrupt power to the test element at the onset of burnout to prevent the actual destruction of the test element. However, an indication of incipient burnout from both of the above measurements is required in order to consider the run a valid measure of the burnout heat flux.

The operating conditions to be investigated in this burnout study are listed as follows:

Pressure (psia)	Flow (lbs/hr-ft ²)	Maximum Subcooling*	
		1/4" Annulus BTU/lb	1/8" Annulus BTU/lb
1000	2 x 10 ⁶	75	170
	4 x 10 ⁶	40	90
	6 x 10 ⁶	25	60
1400	4 x 10 ⁶	40	90

* 5 values of inlet subcooling at each flow geometry starting at the listed maximum inlet subcooling.

An additional task has been added to this study in which provision will be made to measure the heating element temperature variation in the region where it contacts the external spacers. If this temperature variation can be demonstrated to be very small, the same spacing technique will be employed on the PWL rods. The use of external spacers will allow the use of unsegmented fuel rods and eliminate the possible disturbing effect of segment welds, etc., in the peak heating region. Fretting corrosion or wear at the spacer contact points are expected to be unimportant for the short period of time the elements are planned to be run.

4. Experiment Design

The PWL test element and insert design have been started. Water flow to cool the individual fuel rods will be conducted in a concentric annulus to minimize flow distribution variations. The exact annulus size depends on the results of the burnout heat transfer experiments but a 15/16" x .028" wall tube has been tentatively selected as the flow channel. This tube size will provide a radial water flow gap of ~0.150". A close packed bundle of four of these tubes will fit into the facility tube with sufficient room on the outside for cooling water flow over the facility tube wall.

Calculations have been started to predict the percentage of central melting that will occur in each of the test assemblies at the proposed operating heat flux. The most important parameter influencing the results of these calculations is the UO₂ thermal conductivity selected. An effort is being made to choose values that will accurately predict previous experimental results where central melting has occurred in both pellet and compacted powder fuels.

Considerable effort during the quarter has been directed at evaluating and recommending changes in the proposed PWL loop modification to insure that the loop will be truly capable of satisfying the experimental objectives. The primary consideration from the viewpoint of the experiment is that the fuel element failures encountered be the result of UO_2 central melting rather than from extraneous causes such as heat transfer limitations, inadequate coolant chemistry, etc. In this regard, tentative requirements have been set for operating instrumentation accuracy, fission product monitoring, and loop coolant chemistry. Not all of these requirements can be met due to economic or technical limitations but each is being studied with intent of providing alternative solutions that will accomplish the same objective.