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DEVELOPMENT OF A THERMIONIC REACTOR SPACE POWER SYSTEM

FINAL SUMMARY REPORT

Prepared under
Contract AT(04-3)-840
for the
San Francisco Operations Office
U. S. Atomic Energy Commission

Date Published: June 30, 1973

GULF GENERAL ATOMIC COMPANY
P.O. BOX 608, SAN DIEGO, CALIFORNIA 92112

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INTRODUCTION

This is Gulf General Atomic's final report on the development of a thermionic reactor space power system.

Initial experimental work led to the award of our first AEC thermionic contract AT(04-3)-167, Project Agreement 14, on May 1, 1962, for the research and development of fission heated thermionic cells with an operating life of 10,000 hours or more. This work was primarily concerned with metallurgical studies and the construction of experimental thermionic converters. Two types of converters were fabricated: 1) electrically heated, and 2) fission heated where the fuel was either uranium carbide or uranium oxide.

A similar contract was awarded to the General Electric Company. A competition between GGA and GE ensued to develop a thermionic fuel element (TFE) and to design a thermionic reactor to be tested as an experimental reactor. This competition was climaxed on July 1, 1970 by the award to GGA of a five-year contract, AT(04-3)-840, renewable for three years, to develop an in-core thermionic reactor. At that time GGA had successfully operated out-of-pile converters in excess of 20,000 hours and a two-cell thermionic fuel element in excess of 2,500 hours.

The objectives for the Contract AT(04-3)-840 work were:

- To evolve and maintain space power reference designs. The principal projected uses were to have been a reliable man-rated system with a lifetime goal of five years for possible orbital space systems applications and a reliable lightweight one to two year life system for unmanned electric propulsion missions.

- Continued development of a thermionic fuel element (TFE) to a lifetime of one to two years for electric propulsion, with a goal of five years for secondary power applications. This included design, material and fabrication technology, and testing.

- Design, construction, installation, and operation of a thermionic reactor experiment. This effort was to have included measurement of the operational characteristics of a thermionic reactor core, or of more than one if necessary.

- Development of thermionic space power reactor technology in sufficient depth to allow user agencies in the late 1970s to commit to mission use in the 1980s.

The contract statement of work read as follows:

- Materials Technology

The objective of this task is to develop specific in-depth materials technology in support of the thermionic reactor development. The task will emphasize developing and characterizing the specific emitters, collectors, fuels, insulators, seals, and joints planned for use in the thermionic reactor. Fabrication, development, testing, analysis and post-test metallurgical examination will be involved.

- Converter and Fuel Element Development

The objective of this task is the development of thermionic fuel element (TFE) technology for a lifetime of at least one year and a goal of five years.

This task covers design, development, fabrication and testing of all components required in a thermionic fuel element and includes:

Single Cell Prototype Converters - These devices incorporate the essential components of the diode and will be used to study and evaluate basic processes associated with the converter, such as thermionic electrical performance, fuel venting, envelope integrity, design variations, and fabrication techniques. This configuration will allow detailed instrumentation and diagnostic procedures not possible in multicell tests.

Partial Length Fuel Elements - These devices include all the components of reference thermionic fuel elements, but are reduced to a length compatible with the irradiation facilities normally used. These partial length elements will be the principal TFE development test vehicle. Every TFE component will be contained in these elements.

Fuel Length, Reference Fuel Elements - This type of device is intended to be a complete reactor fuel element corresponding to the most up-to-date reference design. They will be used to verify results obtained with partial length elements and demonstrate performance and mechanical integrity prior to the reactor experiment test.

Converter and Fuel Element Testing - This involves primarily the in-pile (or out-of-pile) testing of converter assemblies fabricated under this task and the associated neutron irradiation requirement. The objectives will be the demonstration of long life, high thermionic output and performance stability. Diagnostic and performance analysis of the nuclear characteristics, thermal behavior, and electrical output will be conducted.

- Reactor Technology

The objective of this task is the understanding of the characteristics of a thermionic reactor. Included in this task will be at least the following: neutronic characteristics of the reactor core and reflector, reactor stability and dynamic behavior, control system design and logic, mechanical design and analysis, thermal/hydraulic analysis, electrical performance analysis, reliability analysis, and nuclear safety analysis.

- Reactor Experiment

The primary objective of the thermionic reactor experiment is to explore and evaluate the dynamic and steady-state operating characteristics of a thermionic reactor core. This task provides for a comprehensive reactor experiment program including the design, fabrication, installation, checkout and test operation of the experiment, including the facility. In addition, research and development associated with the construction line item will be performed under this task.

● System Technology

The objectives of this task are to study the applicability of thermionic reactors for various mission applications and to provide technology for selected components as necessary for such systems.

The system design analysis effort includes: tailoring the reactor design for specific potential missions; shielding design and analysis; power conditioning and electrical distribution network; heat rejection systems including pumps, heat exchangers, and radiators; specification of launch environmental conditions; power system reliability and safety; system integration considerations; and other factors.

System component development for thermionic reactors is to rely as much as possible on components which have been previously or concurrently developed for other space reactor systems. In cases where such technology does not exist or is not being developed adequately, technology development is to be undertaken in the thermionic program. It is contemplated that work on reactor components not required for the reactor experiment will not be initiated until after the reactor experiment. Hardware development of non-reactor components of the flight reference design power system is not required.

● Program Management

Program management includes: continuing review, planning, and technical direction of the program; recognition and response to problems which arise; schedule and cost control; reports and related matters.

Milestones set forth in the contract were as follows:

TFE Development

1. Place first full length TFE on in-pile test	March 1971
2. 10,000 test hours on in-pile partial length TFE	September 1971
3. 2,000 test hours on in-pile full length TFE	December 1971
4. 5,000 test hours on in-pile full length TFE	December 1972
5. 20,000 test hours on in-pile partial length TFE	June 1974

6. 10,000 test hours on in-pile full length TFE	December 1974
7. 30,000 test hours on in-pile partial length TFE	December 1976
8. 20,000 test hours on in-pile full length TFE	December 1976

Reactor Experiment

9. Select Reactor Experiment Site	August 1971
10. Submit PSAR	June 1972
11. Construction Authorization	January 1973
12. Submit FSAR	June 1975
13. Complete Construction & Installation
(Contractor Beneficial Occupancy Date)	January 1976
14. Full Power Operation (1.5 Mwt) (without TFEs)	January 1977
15. Insert first TFE	June 1977
16. Begin operation with full TFE core	June 1978

TFE development milestone numbers 1, 2, 3, and 4 were met according to the above milestone schedule. The Reactor Experiment milestones required appropriation of operating and facility money for the Reactor Experiment; the money was not appropriated, and the work was not done.

On January 5, 1973 GGA was notified of program termination by the AEC. At that time a six-cell prototypical thermionic fuel element had passed 8000 hours of operation in the Gulf General Atomic TRIGA test reactor.

This report is organized by tasks as follows:

Task 1	Thermionic Research
Task 2	Materials Technology
Task 3	Thermionic Fuel Element Development
Task 4	Reactor Technology
Task 5	Systems Technology

A bibliography of reports prepared for this program is presented in Appendix C.

TASK 1. THERMIONIC RESEARCH

The work contained within this task was in support of Task 3, Thermionic Fuel Element Development. For background information regarding this task, see the first part of Task 3.

1.1 OUT-OF-CORE TESTING (G. O. Fitzpatrick, M. K. Yates)

The objectives of this subtask included the operation of OC-A4, a diode using a graphite-cesium sorption reservoir, and the performance mapping of an out-of-core mockup of an in-core F series cell. During the past work periods, the mapping of OC-A4 was completed and a total of 26,296 test hours was accumulated.

1.1.1 Mark VIIA OC-A4 Test

The test of Mark VIIA OC-A4 included performance mapping with a liquid cesium reservoir followed by performance mapping with a graphite sorption reservoir, life testing for 23,772 hours, and testing in association with a remotely programmable driving supply for 1426 hours. A post-operational examination was conducted after a total of 26,296 hours had been accumulated.

The performance of the cell dropped slowly, totalling approximately 5% during the first 15,229 hours of life testing. The loss was attributed largely to emissivity changes of the collector due to contamination with carbon from the graphite reservoir. Low power diagnostic measurements at 15,229 hours were followed by a loss of 35% in output power, attributed to emitter contamination with titanium from the final closure braze. The diode was operable when shut down at 29,296 hours.

The post-operational examination showed that carbon transport from the graphite reservoir to the cell had occurred. Carbon or carbide layers 3-5 μ thick were visible on all the niobium surfaces of the cell. No carbide or carbon was found on the emitter. The reservoir graphite had partially delaminated in a manner typical of samples with high cesium loading, but did not appear otherwise damaged. The final closure braze had largely evaporated, and the cell assembly separated at that point during handling. Further details concerning the test are included in Appendix A.

1.1.2 Laboratory F Cell

The design of the electrically heated F series converter was completed and fabrication was initiated. The design is discussed in Section 3.1 and fabrication progress in 3.2.

1.2 DIAGNOSTIC METHOD DEVELOPMENT (G. O. Fitzpatrick)

The objectives of this subtask were to develop improved methods of determining the operating conditions of in-core thermionic devices. This objective was achieved through the continuous incorporation of new information from in-core thermionic tests, and following theoretical calculations using applicable diagnostic correlations.

1.2.1 Emitter Temperature Correlations

Prior to the operation of TFEs in-pile, most thermionic cells incorporated thermocouples to allow estimates of emitter temperatures. While these frequently degraded rapidly, they provided good estimates of initial operating conditions. Both E and F series TFEs, however, are fabricated without such thermocouples. In addition, they normally consist of a series string of cells, each operating with different emitter temperatures and output powers. The purpose of this subtask has been to devise ways of estimating individual emitter temperatures within these TFEs.

Emitter temperature estimation has been approached in two ways, the first based on a knowledge of the TFE's operating environment and its thermal characteristics and the second based on its electrical characteristics.

The second approach has been used to establish the initial operating condition of a TFE. Three aspects of the TFE's electrical characteristics are used: its output voltage in the obstructed mode (which gives an average emitter temperature), the device's open circuit voltage (which relates the average emitter and collector temperature), and the cell current at ignition

(which provides individual emitter temperatures). Details of these correlations and their use are given in Refs. 1 and 2. Ignition current measurements have proven most useful, providing as they do individual estimates of emitter temperatures.

The use of electrical characteristics in some operating conditions is not practical (ignitions disappear above 1900°K for example), and they are also subject to variations due to changes within a cell - gas in the inter-electrode space, work function changes, emitter swelling, etc. For those reasons after initial operating temperatures were established, further temperature estimates were based on knowledge of the cell's operating environment and thermal characteristics. **The mapping data acquired** with Mark VIIA OC-A4 (Ref. 2) was used to estimate the emitter temperature given the operating current density and electrode input power. The equation used was:

$$Q_E = 3.781 \times 10^{-5} T_e^2 + (1.0245 \times 10^{-3} J - 1.0104 \times 10^{-1}) T_e + 0.613 J + 71.88$$

where Q_E = Input power (Watts/cm²)

T_e = Emitter temperature (°K)

J = Current density (Amps/cm²)

Q_E in an operating device was estimated to be the sum of the collector heat flux and electrical electrode power in a cell. An empirical expression for cell collector heat flux was developed.

$$Q_C = \frac{A(T_{sh} + T_o)^b (T_{sh} - T_o)}{(1 - cT_{sh})}$$

where Q_C is the collector heat flux

A is empirically determined

T_{sh} is the sheath temperature (°K)

T_o is the pit water temperature (°K)

b is a constant determined by the containment gas

c is a correlation parameter

The correlation parameter was determined for each cell through use of ignition current temperatures determined at various input powers early in a test.

The core location of a test and the operating reactor power give further information on the input power expected and hence the emitter temperatures. For this reason an effort has been made to map the TITR core and to develop a correlation that could be used to estimate the input power in various types of thermionic devices. This is also necessary in order to specify the enrichments of devices in fabrication and specify the core locations for devices going on test.

A linear correlation was chosen as a first-order model. This correlation has been derived for the F series TFE design, which features 40 or 80 mil (1 or 2 mm) clad tungsten emitters and a fueled length of 2 inches (51 mm) and has the following form:

$$Q = C \times f_R \times f_Z \times f_r \times f_u \times f_f \times f_s$$

where Q - total input power in watts

C - a constant, derived from measured in-pile data ($C=3990W$)

f_R - reactor power factor, the reactor power level divided by the value used when C was evaluated ($f_R = 1.0$ at a reactor power of 1.59 Mw)

f_Z - axial power factor: A composite axial power profile was derived from axial mapping taken in the core using FC-1, a self-powered neutron detector, and operating TFEs.

This shape is shown in Figure 1-1. The mapping data showed this shape was approximately independent of reactor power level, enrichment, core location, and control rod position. However, the position of the flux centerplane relative to that of the TRIGA fuel centerplane does depend on control rod position. For most of the data with thermionic devices at power levels of ~ 1.5 MW the control rods are almost

Fig. 1-1. Composite Axial TFE Power Profile

out of the core. The flux centerplane was assumed to be ~ 0.5 inch (1.3 cm) below the fuel centerplane in those cases.

f_r - radial power factor: A small amount of radial power data was obtained during the FC-1 mapping. This data is shown in Fig. 1-2 where the peak power obtained during the East-West traverse and in the DE flux trap is shown. This data shows the extent to which the graphite pack skews the flux distribution, but not enough data was taken to allow the power in other parts of the core to be estimated. However, two-dimensional calculations of the core power distribution are performed in order to follow FLIP fuel element burnup. The results of the calculations are maps showing the power generated in each fuel element. One of these core maps is shown in Fig. 1-3. The values given in this particular map have been used as the radial power factor. If a value is not available for a given position, the average of the six surrounding positions is used.

f_u - Uranium 235 loading and clad thickness factor: This factor was calculated using one-dimensional codes assuming that the fuel smeared over the area defined by the 1.100 inch outside diameter by 0.040 inch or 0.080 inch tungsten clad emitter. The results in terms of relative power as a function of grams U-235 per 2 inch fuel slug are given in Fig. 1-4.

f_f - fuel factor, depends on fuel used
1.00 for UO_2
0.97 for 90 UC-10ZrC-4W or UC-4W

f_s - shim factor, depends on neutron shielding or shims added to device.

The use of this correlation in practice indicates it is accurate to within the experimental uncertainty in input power estimates based on the electrical characteristics of a TFE ($\pm 10\%$ based on $\pm 50^\circ K$ errors in emitter temperature estimates and $\pm 30\%$ errors in cell end loss estimates).

Fig. 1-2. Data From FC-1 On Peak Power Vs. Core Location

THERMIONIC TEST REACTOR

TRIGA FLIP CORE
TOP GRID PLATE

Figure 1-3. Relative Power Distribution (Local Power/Core Average Power)

Fig. 1-4.- Relative Power Vs. U-235 Loading

Further refinements in the accuracy of the values could be made by reducing emitter temperature errors below $\pm 50^{\circ}\text{K}$, making more exact end loss calculations, and using the measured data to refine Figure 1-3. Another source of uncertainty which could be reduced is that involving relative control rod heights. In much of the testing, the rods were not kept banked; they were usually set up to favor the operation of a 6F TFE. However, this mode of operation skews the power profile in the core and changes the relative worths of the various core positions, an effect that is not taken into account in the 2D calculations whose results are given in Figure 1-3. Beginning with the test of TFE 6F⁴, the control rods were kept banked.

1.2.2 TFE Performance Determinations

The performance of a TFE has been described in terms of a relative power number, P_r , which is the ratio of the corrected total electrode voltage produced by a TFE to that predicted by CPOP (Ref. 3) at a specified "standard" emitter temperature and current density. The following corrections are made to obtain the corrected TFE voltage.

1. Adjust voltage upward to make up losses due to off-optimum collector temperatures. (Correction based on initial operating measurements and collector optimum curves measured for OC-A4, Ref. 4.)
2. Correct the voltage to that which would be obtained if the spacing were 10 mils (Refs. 5 and 6).
3. Calculate the difference between the voltage obtained after corrections (1) and (2) and that predicted by CPOP for the TFE's actual emitter temperatures and current density.
4. Assume this voltage difference calculated in (3) would be the same at the "standard" operating conditions and subtract it from the CPOP predicted voltage under those conditions.

Changes in performance can result from a number of causes including electrode emissivity and work function changes, interelectrode spacing changes, interelectrode gas, poor thermal bonds in the collector structure, shorts or parasitic electrical discharges, and input power. Determination of sources of change are made by correlating data from IV curves, cesium optimums, collector optimums, input power based on sheath temperatures and reactor location, TFE response times, voltage probe changes, and neutron radiographs.

Recently a new technique of observing individual cell cesium optimums was developed. The electrode power and the efficiency of each cell varies as the cesium temperature varies, with maximum efficiency at the optimum cesium reservoir temperature. The efficiency of each cell is reflected in the sheath temperature opposite it, and hence measurements of these temperatures can be reduced to electrode voltage changes. An example of the results of reducing data taken during cesium optimum measurements of TFE 6F4 are shown in Figure 1-5. It is clear that the optimum temperature for cell 2 is significantly higher than cells 1, 3, 5, and 6, and cell 4 appears to have an unusually low optimum temperature. Observations of individual optimums such as these can provide information on electrode work function, emitter temperature, and current density changes in specific cells within a TFE.

1.2.3 Performance Map Correlations

The performance map obtained with Mark VIIA OC-A4 has been used to estimate thermal performance changes in in-pile tests. Thermionic performance has been compared to that predicted by CPOP at an interelectrode spacing of 10 mils (0.25 mm) corrected for other spacings according to slopes derived from TECO out-of-pile cell test data (Ref. 7).

A reference performance data set generated by NASA-LeRC was received in December (Refs. 8 and 9) but its comparison with existing performance maps has not been accomplished.

Fig. 1-5. Individual Cell Cesium Optimums

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TASK 2. MATERIALS TECHNOLOGY

This section summarizes the work carried out during this reporting period and in addition includes a brief summary of the current status and material problems to be resolved for the major converter components. Pertinent references are given in order to provide some degree of continuity for future workers in this field.

The development of the thermionic fuel-cladding system constitutes a major portion of the efforts of both the AEC and the NASA thermionic programs at Gulf General Atomic. Details have been described in various reports (see Bibliography in Appendix C). The recent status of thermionic fuel-cladding development can be found in a paper presented at the Third International Conference on Thermionic Electrical Power Generation (Ref. 1) and in another paper presented at the 1972 Intersociety Energy Conversion Conference (Ref. 2). More technical details are contained in the record of the First Thermionic Project Office Material Committee Meeting, which was held in September 1972*.

*Copies of this record were distributed by D. Beard and L. Price of the USAEC to various committee members.

On the basis of the in-pile test results obtained to date at GGA, it appears that the life of the carbide-tungsten thermionic fuel cladding system is limited to about several thousand hours at a cladding temperature of 1800° to 1850°K. The major problems for the carbide-tungsten system lie in the degradation of thermionic performance due to fuel diffusion through the cladding and the deformation and cracking of the cladding by stress generated through fuel swelling. The carbide fuel strengthens and also embrittles the tungsten cladding so that the cladding cannot take much deformation without developing cracks. The stress caused by fuel swelling probably also enhances fuel diffusion through the cladding and accelerates thermionic performance degradation. Solutions proposed for resolving these problems consist of: (1) to incorporate more surface area into the fuel body to facilitate fission gas release and thus to minimize fuel swelling, (2) to soften the fuel body by lowering the concentration of the tungsten additive or eliminating it so that less stress is imposed on the cladding when the fuel body swells, (3) to improve the fuel body design so that less stress produced by fuel swelling is transmitted to the cladding, and (4) to use chloride-arc-cast tungsten cladding to reduce fuel transport rates and thus to minimize thermionic performance degradation. All these have been implemented into carbide converter design and fabrication development but completion of fabrication for in-pile evaluation had not been accomplished at the termination of the program. These items should be considered as top priorities for any further carbide thermionic fuel development program.

The in-pile test results on the oxide-tungsten system obtained to date at GGA indicate that the life of the oxide fueled tungsten emitter of reference design (40 mil cladding) is about 8000 hours at a cladding temperature of 1800°K, the major problem being the swelling of the emitter which leads to interelectrode short. It appears that the life of the emitter could be increased to greater than 20,000 hours if the cladding thickness is increased to 80 mils or if the emitter diameter is reduced from the 1.1 inch prototypical size to 5/8 inch. Unlike the carbide fueled converters, the oxide fueled converters do not exhibit significant performance

degradation during long-term in-pile operation. Approaches for resolving the emitter swelling problem involve: (1) using lower fuel loading (72 vol % instead of 85 vol %) in the emitter cavity, (2) improving the fuel design so that very little of the stress generated by the fission gas bubble growth in the fuel body is transmitted to the cladding, (3) using high strength cladding material such as thoriated tungsten, and (4) lowering emitter operating temperature by incorporating oxygen additive (e.g., oxygenated molybdenum collector) into the converter. To evaluate these approaches, one oxide fueled capsule was under irradiation, while another oxide fuel capsule and an oxide fueled 6F TFE were being fabricated at the time the program was terminated. These approaches for reducing the swelling of the oxide fueled emitter under irradiation should receive immediate attention in the event the thermionic reactor program is reactivated. Consideration should also be given to evening out the radial fission power distribution in the fuel during the irradiation test. This can be achieved by using low fuel enrichment or by irradiation in a fast reactor.

The development of the ceramic components for the thermionic fuel element, such as the insulator seal, the trilayer and the electrical insulation, was carried out under the AEC thermionic program. Detail information on work concerning these components can be located through the Bibliography in Appendix C. The conference records for the Thermionic Conversion Specialist Conferences show the annual progress by GGA and other laboratories, TECO, ORNL, and LASL. The development of the insulator seal for thermionic fuel element application was described in two topical reports (Refs. 3, 4). The recent status for the insulator seal was given in a paper presented at the Third International Conference on Thermionic Electrical Power Generation (Ref. 5) and in another paper presented at the 1972 Intersociety Energy Conversion Conference (Ref. 2). The seal developed has operated successfully in converters and in thermionic fuel elements tested in thermal reactors under the AEC program at GGA. A cooperative effort with LASL for irradiating prototypical seals in EBR II was under planning. The development of Y_2O_3 seal for better resistance to fast neutron irradiation was initiated in 1972.

The development of the Nb-Al₂O₃ graded trilayer used in thermionic fuel elements was described in three topical reports (Refs. 6, 7 and 8). A number of problems remain to be resolved with regard to the fabrication and the in-pile operation of the Nb-Al₂O₃ graded trilayer. First, the niobium sheaths of the trilayers fabricated were often contaminated by oxygen and became embrittled, and the Al₂O₃ layers exhibited radial cracks. Secondly, there were indications that the electrical resistivity degraded when the trilayer was operating under an electrical potential in the presence of cesium vapor, both out-of-pile and in-pile even in a thermal reactor. Thirdly, cracks were observed in the ceramic layer when trilayer samples were irradiated in EBR II in a fast neutron environment. The first problem can be resolved by better control of the impurities in the gaseous environment and the schedule during gas pressure bonding and plasma-spraying. The second problem needs more detailed investigations on the mechanism for the electrical resistivity degradation. If this problem cannot be resolved, then all the TFE concepts involving the contact of cesium vapor with trilayer become infeasible. Solutions to the third problem require improved trilayer design so that the niobium will move with the ceramic when the latter swells under irradiation, and the use of ceramics of better resistance to fast neutron damage, such as Y₂O₃. A joint program with LASL for irradiating prototype trilayers in EBR II and a developmental program for Y₂O₃ trilayers were in progress during 1972. The solutions to these trilayer problems are crucial to the success of the operation of a thermionic fuel element.

The developmental status of the electrical insulation for the intercell regions in a thermionic fuel element can be found in the AEC contractual reports. Plasma-sprayed Al₂O₃ has been the reference insulation layer used in the thermionic fuel element. There is concern over the contamination and embrittlement of the thin convolution in the insulator seal skirt during the plasma-spraying of the Al₂O₃ insulation layer. It has also been observed that the insulation layer always develops cracks at the bottom of the convolution, probably caused by thermal cycling. Thus there is considerable room for improvement

in the Al_2O_3 insulation layer for the intercell region. Sputtered Al_2O_3 coating has been investigated for such application but more work is needed before it can be considered for thermionic fuel element application.

In addition to the material problem areas for the converter components, there exist a number of crucial material problem areas for other thermionic fuel element components. First, to prevent the reaction between the fission products released from the fuel body and the Al_2O_3 insulation coating around the fission product venting holes, and to prevent the plugging of fission product venting holes, traps containing Al_2O_3 were installed into the emitter transition regions of a carbide fueled 6F TFE to trap the active ingredients such as Sr and Ba. With the disposal of all the irradiated capsules and thermionic fuel elements (except 6F1 and 6F2) without any hot cell examination, the effectiveness of such traps during in-pile operation cannot be evaluated. For oxide fueled devices, the out-of-pile study of fission product trapping (such as for tellurium) was about to begin when the thermionic program was terminated. Such out-of-pile and in-pile studies of fission product trapping are of utmost importance to the long-term operation of a thermionic fuel element. Secondly, most of the Al_2O_3 support discs for the Bellville washers in the two 6F TFEs (6F1 and 6F2) examined were found to be disintegrated into individual grains. It appears that these discs were subjected to high temperatures and stress. A white phase was observed in the tantalum holder for these discs at the tantalum- Al_2O_3 interface. Thirdly, excessive grain growth occurred in the thin convolutions of the niobium skirts of the insulator seal in the intercell region. The intercell region contains a large number of refractory metal joints located close to one another. Combined with large temperature and stress gradients and possible contaminations by fission products and fuel components, this region offers a number of potential leaks between the fission gas space and the cesium space. In fact, there was evidence that such leakage did occur in the majority of the thermionic fuel elements tested.

Any future thermionic program planning should include a redesign of this crucial region.

Another major problem in the thermionic fuel element developmental program is how to reduce cladding temperature in an operating thermionic fuel element and how to minimize the temperature non-uniformity in a single emitter as well as among all six emitters. Although this is not a material problem, the life and performance of all the materials in a thermionic fuel element are strongly dependent upon the solution to this problem. Research and design efforts should be directed toward this area in any future thermionic program so that the material components will not be overburdened because of temperature uncertainty and penalty to performance caused by temperature non-uniformity. An additional area in need of attention is to study means for improving thermionic performance at low emitter temperature. This will help to lower the emitter operating temperature and thus to lessen the burden on emitter as well as other thermionic fuel element component materials.

A considerable amount of progress has been achieved during the last twelve years in developing thermionic fuel element component materials and in understanding the critical problem areas associated with long-term operation of the thermionic fuel element. It must be remembered that any material has its operating limits with respect to temperature and radiation damage. The development of a reliable system requires a thorough understanding of these limits and the designing of the system within these limits. The above described approaches raise these limits through the development of better materials (e.g., high strength cladding and better radiation resistant ceramics) and also provide wider safety margins below these limits through better fuel and TFE design. Such cooperative efforts in any future thermionic program could conceivably increase the thermionic fuel element life to beyond 20,000 hours for both space and underwater applications.

The work carried out during this reporting period under various subtasks is described below.

2.1 FUEL TECHNOLOGY

The objective of this subtask is to study the physical, chemical, and mechanical properties of carbide and oxide fuel materials for nuclear thermionic application and to develop new fuel materials and fabrication processes for reducing cost and improving the life and performance of thermionic fuel elements.

2.1.1 Carbide Fuel Phase Relationship (R. G. Hudson, L. Yang)

This work concerns two specific aspects of the phase relationship in carbide fuels. First, it is intended to find out the solubility of tungsten in UC and 90UC-10 ZrC as a function of temperature and C/U atom ratio. Second, it is intended to identify the nature of the metallurgical phases present when the solubility limit of tungsten is exceeded. The information is needed for the selection of fuel composition which is metallurgically stable toward the tungsten cladding. Due to the early termination of this work, studies were limited to an annealing temperature of 2073°K and UC specimens of C/U = 0.98, 1.01, 1.03 and 1.05 and tungsten concentrations equal to 0, 0.5, 1.0, 1.5, 2.0, and 4.0 wt. %. For specimens of C/U = 1.03, the tungsten concentrations were also extended to 6 and 8%.

The specimens were prepared by arc-melting of weighed amounts of uranium, tungsten and carbon. The arc-melted buttons were homogenized at 2523°K for 1 hour and then annealed at 2073°K for 16 hours before they were quenched in helium at a rate of 500° per minute to 800°K and then cooled to room temperature in helium. X-ray analysis of these specimens yielded the results shown in Table 2.1. The essential findings are summarized as follows:

TABLE 2-1
X-RAY ANALYSIS RESULTS OF UC-W SYSTEM

C/U Atom Ratio of Carbide before W addition	Wt. % of W added	Phases identified by X-rays				W
		Uranium Monocarbide	Uranium Dicarbide	η -UWC ₂	Orthorhombic UWC ₂	
0.98	0	x				
	0.5	x				
	1.0	x				
	1.5	x				
	2.0	x				x
	4.0	x				x
1.01	0	x	x			
	0.5	x	x			
	1.0	x	x			
	1.5	x	x			
	2.0	x		x	x	x
	4.0	x		x		x
1.03	0	x	x			
	0.5	x	x			
	1.0	x	x			
	1.5	x	x			
	2.0	x	x			
	4.0	x		x	x	x
	6.0	x		x		x
	8.0	x		x		x
1.05	0	x	x			
	0.5	x	x			
	1.0	x	x			
	1.5	x	x			
	2.0	x	x			
	4.0	x			x	

1. There were five different phases identified by X-rays. In addition to the monocarbide matrix containing dissolved tungsten, the dicarbide phase, and the tungsten phase, two UWC_2 phases were found. These were the orthorhombic phase (Ref. 9) and the η -phase (Ref. 10).
2. When the C/U atom ratio of the carbide was less than unity, only the monocarbide matrix and the tungsten phase were found. The tungsten phase appeared when the tungsten concentration was between 1.5 and 2 wt. %.
3. When the C/U atom ratio was higher than unity, the dicarbide phase was observed at low tungsten concentration. When the tungsten concentration was increased, the UWC_2 phases and the W phase appeared. For a C/U atomic ratio of 1.01, this happened at a tungsten concentration between 1.5 and 2 wt. % and the UWC_2 present consisted of a mixture of the η -phase and the orthorhombic phase. With further increase in tungsten concentration, the orthorhombic UWC_2 phase disappeared but the η - UWC_2 phase remained. For a C/U atom ratio of 1.03, the UWC_2 and W phases appeared at a tungsten concentration between 2 and 4 wt. %, and the UWC_2 present consisted of a mixture of the η -phase and the orthorhombic phase. With further increase of tungsten concentration to 6 and 8 wt. %, the orthorhombic UWC_2 phase disappeared while the η - UWC_2 phase remained. When the C/U atom ratio was increased to 1.05, no W phase was observed even at a tungsten concentration of 4 wt. %; the only dispersion present was the orthorhombic UWC_2 phase.

These results showed that for hyperstoichiometric carbide fuel the tungsten concentration at which the W phase appeared depended upon the C/U atom ratio of the carbide before the tungsten was added, and that the sequence of the appearance of the various dispersion phases with the increase of the tungsten concentration was: orthorhombic UWC_2 , $\eta\text{-UWC}_2$ + orthorhombic UWC_2 + W, $\eta\text{-UWC}_2$ + W. It appears that for the hyperstoichiometric specimens studied, higher carbon activity favors the formation of the orthorhombic UWC_2 phase, while lower carbon activity favors the formation of the $\eta\text{-UWC}_2$ phase, and that the $\eta\text{-UWC}_2$ phase and the W phase always coexisted. For the hypostoichiometric specimens, no UWC_2 phase was observed and the only dispersion present was the W phase.

The above described studies covered only a small part of the UC-W phase relationship. More extensive work is needed if the UC-W system is selected as the fuel material for the thermionic fuel element.

2.1.2 Compatibility and Trapping of Fission Products (G. Buzzelli)

The objective of this work is to evaluate the reaction between Al_2O_3 and the alkali earth fission products and the effectiveness of the trapping systems installed in TFE 6F5.

The experimental arrangement was described in the last thirdly report. Two tests were carried out previously, using tantalum capsules containing molybdenum discs coated with plasma-sprayed Al_2O_3 or Al_2O_3 chips as the trap and high purity strontium metal as the strontium source. The first test was carried out with a plasma-sprayed Al_2O_3 trap at 1350°K and a strontium pressure of 4×10^{-2} torr for 16 hours. The second test was performed with a trap at 1310°K and a strontium pressure of 4.5×10^{-3} torr for 32 hours. In each case, about 88% of the strontium atoms entering the trap was caught by the Al_2O_3 and

the tantalum housing of the trap. $\text{SrO} \cdot \text{Al}_2\text{O}_3$ was identified as the reaction product.

During this reporting period, another test was carried out with a plasma-sprayed Al_2O_3 trap at 1385°K and a strontium pressure of 1.5×10^{-2} torr for 101.5 hours (Sr source temperature = 810°K). For a 6F TFE operating at a fission power density of 60-70 W/cc, the rate of formation of Sr is estimated to be about 10^{13} atoms per second per emitter. Assuming complete release from the fuel body, the Sr flux at the 0.4 mm diameter entrance to the trap is 8×10^{15} atom/cm² sec. During steady state operation, the Sr pressure in the fuel chamber for maintaining such a Sr flux is about 10^{-4} torr. The Sr pressure in the third test capsule is thus about 150 times higher than that in the fuel chamber of an operating emitter, and the total number of strontium atoms entering the trap in 101.5 hours corresponds to that for about 15,000 hour operation of the emitter in a TFE.

After the completion of the test, the test capsule assembly was dismantled and the components were analyzed for strontium distribution by X-ray fluorescence spectroscopy and electron microprobe techniques.

Table 2-2 summarizes the strontium analysis results. It can be seen that the baffles absorbed about 95% of the strontium atoms entering the trap, while about 2% of these strontium atoms leaked through the trap. This compares to a 40% absorption by the baffles and 12% leakage through the trap for the first capsule test. The tantalum housing retained about 2% of these strontium atoms, compared to 47% for the first capsule test.

The individual baffle analysis indicates maximum absorption on the first baffle and decreasing quantities for subsequent baffles. This is in contrast to the first baffle experiment where there was no difference in the absorbed strontium levels for the various baffles. In the first test,

TABLE 2-2

DISTRIBUTION OF Sr IN FISSION PRODUCT TRAP
OF THE THIRD TEST CAPSULE CONTAINING MOLYBDENUM
BAFFLES COATED WITH PLASMA-SPRAYED Al_2O_3

<u>Location</u>	<u>Wt. of Sr Found, mg</u>	<u>% of Total Sr Atoms Entering the Trap</u>
Above Trap	0.18	1.9
Baffles		
No. 5	0.74	8.04
No. 4	1.02	11.1
No. 3	1.02	11.1
No. 2	2.24	24.5
No. 1	3.82	41.5
Tantalum housing of trap	<u>0.18</u>	<u>1.9</u>
TOTAL	9.20	100.0

Sr charge to capsule = 424 mg

Sr recovered = 410 mg

the strontium pressure was higher than this test by a factor of four, although the testing time was 16 hours, as compared to the 101.5 hours for the present test. It would appear that the absorbing capacity of the baffles was reached during the first test when the strontium flux was much higher, possibly because the arrival rate outweighed the rate of absorption into the Al_2O_3 by diffusion. This may explain the lower trapping efficiency of the first test. In the present test, the amount of strontium absorbed by the Al_2O_3 baffle decreased as the baffle moved away from the strontium source. This indicates that these baffles still possessed more capacity for absorbing strontium at the end of this experiment.

Another interesting observation concerns the absorption of strontium by the tantalum housing of the trap. It was observed in the first test that about 47% of the strontium atoms entering the trap was absorbed by the tantalum housing. In the present test, however, only 2% of these strontium atoms was absorbed by the tantalum housing. It was suspected that this difference might be related to the oxygen content of the tantalum, with tantalum of higher oxygen content trapping larger amounts of strontium. Analytical results showed that the tantalum for the housing of the trap used in the first experiment contained 78 ppm of oxygen after the test. On the other hand, the tantalum for the housing of the trap used in the present experiment contained 435 ppm of oxygen after the test. This is the reverse of what would be expected. Since there is a shortage of strontium inventory of about 4 mg in the present experiment, it is possible that these 4 mg of strontium may have diffused into the tantalum housing wall and were not removed by the acid leaching used for recovering the strontium from the tantalum housing. The tantalum housing for the present experiment will be sectioned and analyzed for strontium in the metal.

2.1.3 TFE Port Investigations (J. Kemme)

The objective of this task was to investigate the flow behavior of the present TFE port with respect to fission gas and cesium. This port, a 6 mil I.D. tube containing a 1 inch length of 4 mil diameter wire, forms an annular flow restriction in the fission gas venting path. In the event that leakage develops between this path and the interelectrode space of a TFE, the port is designed to limit the loss rate of cesium to about 1 g/10,000 hours. Because the port dimensions are small, there is some concern that it may become plugged with cesium or cesium compounds, a condition that would cause buildup of fission gas pressure and impair TFE performance. The approach used to study the behavior of the TFE port involved cold flow experiments with gas only and flow experiments with cesium only.

Cold Flow Experiments

The block diagram for the experimental apparatus and its description can be found in the last thirdly report. Five cold flow tests were completed. Measurements were made with a precision glass tube with an internal diameter of 0.076 mm and a length of 120 mm (0.003 inch I.D. by 4.7 inches long) and with a TFE port of the type described above (designated as port No. 1).

The glass tube was sized so that it would have flow characteristics equivalent to that desired in the TFE (equivalent to a 0.051 mm I.D. by 25 mm long tube). The TFE port that was tested was known to have a flow rate higher than desired in the TFE and was found to behave like a 0.060 - 0.076 mm I.D. by 25 mm long tube.

The results of the tests are shown in Figures 2-1 to 2-4. Figure 2-1 shows the results of measurements made with helium in the viscous flow range (near atmospheric pressure), where the flow rate in torr cc/min is plotted

Figure 2-1. High Pressure Helium Flow Data

Figure 2-2. Flow in Glass Tube at Low Pressures

Figure 2-3. Port Flow at Low Helium Pressures

Figure 2-4. Port Flow at Low Xenon Pressures

versus the pressure drop across the port. One set of measurements was made with the glass tube and another with the TFE port. The glass tube data was in good agreement with viscous flow calculations; the TFE port data was in good agreement with that predicted for a 0.063 mm I.D. by 25 mm long tube. The viscous flow equations were taken from Reference 11.

Figure 2-2 shows results obtained with the glass tube with helium at low pressures. As indicated, the flow was slightly less than that predicted by the molecular flow and basic port flow equations. The molecular flow equation was taken from Reference 11. The basic port flow equation was taken from Reference 12 and is a semi-empirical equation which is in agreement with experimental data over the entire range of Knudsen numbers for flow of a single gas species. This equation is:

$$\dot{N} = \Delta P \frac{\sqrt{2\pi}}{6} \frac{d^3}{\ell} \frac{1}{\sqrt{mkT}} \left[\frac{3\pi}{128} \frac{1}{K} + \frac{\pi}{4} \frac{1}{1+K} + \frac{K}{1+K} \right]$$

$\dot{N}$ = flow rate, molecules/sec

ΔP = pressure differential across the tube, torr

d = hydraulic diameter of tube, cm

ℓ = length of tube, cm

m = mass of the molecule, g/molecule

k = Boltzman constant

T = absolute temperature, $^{\circ}\text{K}$

$K = \frac{\lambda}{d}$ = Knudsen number

where $\lambda = 8.589 \frac{\eta}{P} \left(\frac{T}{M}\right)^{1/2}$ cm (from Ref.12)

and

η = viscosity, poise

M = molecular weight, grams/mole

When the port was tested at low gas pressures, the flow was greater than expected. The results obtained with helium are compared in Figure 2-3 with curves calculated from the molecular flow and basic port flow equations for a tube, 0.076 mm I.D. by 25 mm long. Similar information for tests made with xenon is shown in Figure 2-4.

The results obtained with the glass tube indicated that the cold flow apparatus was functioning properly, but that additional tests would be required to characterize the flow in TFE ports.

Cesium Flow Test

The apparatus used for the cesium flow test is described in the last thirdly report. The primary objective of this test was to measure cesium loss rates through a TFE port at cesium pressures present in the TFE. In Figure 2-5, some typical test measurements are compared with a curve calculated from the basic equation for port flow (for port #2). Although the loss rates at lower cesium pressures were higher than predicted, this was expected because the technique for checking production ports indicated that the test port was slightly less restrictive than those accepted for the TFE. When it was found that the measured loss rates were lower at higher cesium pressures, it was postulated that at these pressures enough cesium was collecting at the port exit to cause refluxing, a condition that would create a back pressure and thus restrict port flow. However, as will be described later, the tube attached to the port exit was examined after the test and very little cesium was found in the region above the port.

A second objective of the test was to determine the temperature conditions required to form a condensation plug in a TFE port. It was found that plugging did not occur when the port temperature was a degree or two above the cesium reservoir temperature. However, a plug formed within a few hours when the port temperature was a degree or two below the reservoir temperature. Also, once the plug formed, full flow was not restored for 120 hours, although the port

Figure 2-5. Cesium Flow Rate Data

temperature was gradually increased during this time until it was 50°K hotter than the reservoir.

The test was terminated after 720 hours. It was estimated that less than 5 mg of cesium transferred through the port while it was partially plugged. A more exact estimate, based on flow measurements and system pressure conditions, indicated that about 38 mg of cesium transferred during the remaining 600 hours. After the test, this estimate was checked by radiochemical analysis which showed that 46 mg of cesium collected in the tube that was attached to the port exit. This tube was designed to simulate the vent line leading to the charcoal trap in a TFE. The tube was analyzed in sections to determine the cesium distribution. The sections are identified in Figure 2-6. The amount of cesium in each section is given on the figure.

Because the tube at the top of the bend (see Figure 2-6) was below 390°K during the test, it was thought that condensation above the port would cause refluxing. Therefore, the location of essentially all of the cesium at the water-cooled end of the tube was not expected.

Based on the test results, the present TFE port should provide the desired cesium loss rate in the TFE, but the port should be kept above the cesium reservoir temperature to ensure that plugging does not occur. Questions concerning cesium refluxing in the vent line leading to the charcoal trap were not resolved.

The numbers given are the quantities of cesium found in each section at the end of the test.

Figure 2-6. Cesium Distribution in Port and Exit Tube Section After Hot Flow Test

2.2 CLADDING TECHNOLOGY

The objective of this subtask is to study the physical, chemical, and mechanical properties of chemically vapor deposited tungsten for thermionic emitter application, to establish techniques for quality control and to develop high performance and high strength cladding materials for improving the life and performance of thermionic fuel elements.

2.2.1 Study of Creep Behavior of Prototype Tungsten Emitter

(L. Yang, H. Johnson, D. Allen)

The objective of this work is to study the creep behaviors of both unfueled and fueled prototype tungsten emitters as a function of stress, fluorine impurity concentrations, and nature of the fuel materials. The experimental results generated will be compared with those predicted on the basis of uniaxial creep data by the viscoelastic computer program under Task 4.

During the previous reporting periods, the deformation of a prototype fluoride tungsten emitter containing 5 ppm fluorine was studied at 2000⁰K and under helium pressures of 60 and 120 psi. The test data indicated the absence of primary creep and that the exponential dependence between stress and creep rate at low stress level differs from that at high stress level. The stress-strain rate curve defined by the test data was used to calculate the deformation of the emitter. The calculated results for the emitter deformation under 60 psi helium pressure were in good agreement with the observed values. It was also shown that the growth of voids at the grain boundaries of the cladding was accelerated by the stress and that a higher concentration of voids was present at locations where the tensile stress is the greatest. In addition to unfueled emitter, the study was also extended to a carbide fueled prototype fluoride tungsten emitter of 40 mil cladding thickness. The emitter contains 6 ppm of fluorine and the bottom 2.5 cm

was fueled with two 90 UC-10 ZrC (C/U = 1.01, W= 4 wt. %) pellets. To insure a good contact between the fuel and the cladding, the cold gap was sized for a 2 mil diametric interference fit at 2000°K. The emitter was brought from room temperature to 2000°K in vacuum over a 48 hour period of time and was maintained at 2000°K at 24 hours. Upon cooling to room temperature, it was found that the emitter had expanded 2 mil diametrically in the fueled region. This is in good agreement with that predicted on the basis of the differential thermal expansion coefficient between the fuel and the cladding at 2000°K. The fueled emitter was then tested at 2000°K under 60 psi helium pressure for 1000 hours. The fueled region expanded diametrically about 1.4 mils, in agreement with the results obtained on an unfueled emitter, but the unfueled region and the emitter bottom showed no significant deformation.

During this reporting period, the emitter was tested for another 1000 hours at 2000°K and 60 psi helium pressure. This was followed by another 1000 hour test at 2000°K and 120 psi helium pressure. In both cases, no deformation was observed in the emitter structure, while previous tests of the unfueled emitter under similar conditions yielded significant deformation in both emitter diameter and emitter bottom. It appears that the tungsten cladding was considerably strengthened by the presence of the carbide fuel. Metallographic examinations showed the presence of a thin (~3 microns) UWC_2 layer at the fuel-cladding interface (Figure 2-7(a)) and porosities at the grain boundaries of the tungsten cladding (Figure 2-7(b)). Analysis of the cladding after the removal of the UWC_2 layer by polishing yielded a carbon concentration of 49 ppm, while the pre-test carbon concentration was only 8 ppm. The observed increase in the creep strength of the tungsten cladding is probably associated with the diffusion of carbon into the cladding. Such increase in creep strength, coupled with the formation of stress-induced porosities at the grain boundaries may lead to grain boundary cracking at low cladding deformation. This has been observed during the hot cell examinations

of irradiated carbide-fueled tungsten emitters (See Section 2.7). It is unknown whether the presence of oxide fuel will cause similar increase in the creep strength of the tungsten cladding. This is an area in need of more extensive work in any future thermionic fuel element development program.

2.2.2. Preparation of Uniaxial Creep Samples (J. Chin, R. G. Hudson)

Fluoride-chloride duplex tungsten samples are prepared for the study of their uniaxial creep rates at ORNL and TRW at temperatures and stress levels of thermionic interest. The data generated are used in the visco-elastic analysis of the dimensional stability of fueled thermionic emitters under irradiation. Deposits are made on square molybdenum mandrels of 6 inch length, from each of which four test samples can be obtained. Two low fluorine deposits (~5 ppm) were delivered to ORNL during this reporting period.

M 39122-2

1000x

(a) UWC_2 Layer at Fuel-Cladding Interface

M 39122-4

75x

(b) Porosities at Grain Boundaries

Figure 2-7. Microstructures of the Tungsten Cladding of a Carbide Fueled Prototype Emitter after Biaxial Creep Tests at 2000°K and 60 psi Helium Pressure for 2000 hours and 120 psi Helium Pressure for 1000 hours

2.2.3 CVD Tungsten Development (J. Chin)

This work concerns the development of an infrared analytical technique for examining WF_6 used in the chemical vapor deposition of tungsten. This analysis was complicated by the corrosive interaction between WF_6 and conventional materials used in analytical chemistry instrumentation.

Special sampling cells, Fig. 2-8, were constructed from Teflon and stainless steel and fitted with germanium CaF_2 , $NaCl$, or BaF_2 windows. Sample cells were evacuated through a liquid nitrogen trapped diffusion pumping system, Fig. 2-9.

Infrared absorption was monitored with a Beckman IR-4 instrument capable of examining wave lengths of 1.0 to 15 microns. Impurities examined in WF_6 were HF , HCl , and SiF_4 . Wave lengths required for these examinations were 2.0-10 microns. Known quantities of these materials were added to WF_6 for cell calibration. Infrared absorption versus concentration curves were then prepared for comparison with unknown samples. A typical infrared absorption curve is shown in Fig. 2-10.

Impurity contents found in eight vendor supplied bottles of WF_6 are shown in Table 2-3. The unexpected high concentration of HCl in several of the bottles were qualitatively related to difficulties experienced in depositing tungsten with controlled fluorine levels. The source of the HCl is thought to be WCl_xF_y or RCl_xF_y . Whichever is the source, these materials will interfere with the chemisorption, reduction and deposition reactions of WF_6 and H_2 necessary for the controlled fluorine deposition of tungsten. Fluoroform (HCF_3) was found in bottles from ORNL only.

When the bottles of WF_6 are nearly empty the impurity composition changes. These changes can lead to undesirable effects during chemical vapor deposition.

Fig. 2-8. IR Sampling Cells

Fig. 2-9. Liquid Nitrogen Trapped WF_6
Sampling System

Fig. 2-10. Typical Infrared Absorption Curve

TABLE 2- 3

IMPURITY CONTENT OF WF₆ SAMPLES MEASURED BY INFRARED ABSORPTION

Cylinder No.	Material Source	Impurity Content (PPM)			
		HF	HCL	SiF ₄	HCF ₃
R1114	Allied Chemical	205	3150	<300	-
R2687	Allied Chemical	<100	<100	<300	-
R2728	Allied Chemical	946	5380	<300	-
SG1028	Air Products	<100	<100	<300	-
SG9186G	Air Products	<100	1173	61	-
SG102020	Air Products	<100	1305	103	-
AEC 0148	ORNL	<100	4120	41	500-1000
AEC 1593*	ORNL	2900	2230	75	500-1000

*Bottle almost empty

2.3 SEAL, TRILAYER, AND ELECTRICAL INSULATION TECHNOLOGY

(J. Chin, C. Messick)

The objective of this subtask is to develop seals, trilayers, and electrical insulation which are stable under the thermal and irradiation environment of a thermionic fuel element.

2.3.1 Development of Y_2O_3 Seal

The reference seal for the converter in a thermionic fuel element consists of Lucalox ring, metallized with tungsten-2 wt% Y_2O_3 and brazed to niobium skirts with 60V-40Nb alloy. It provides electrical insulation between the emitter and the collector, and containment for the cesium vapor in the inter-electrode space. It is preferable that the seal remains leak tight during the designed life of the thermionic fuel element, although the thermionic fuel element design allows continuous operation even if the seal leaks. Such a reference seal operated successfully for 8000 hours in a prototypical converter. LASL EBR II irradiation results (Ref.13) indicated that Lucalox remained leak tight to helium after being subjected to a fast neutron exposure of 4.8×10^{21} nvt (> 0.1 MeV) which is equivalent to that for 20,000 hour operation of the 120 kWe thermionic reactor under consideration for nuclear electrical propulsion. To provide a safety margin for longer reactor life, seals containing ceramic materials of better resistance to fast neutron damage are required. Y_2O_3 which has a cubic lattice was shown by LASL EBR II irradiation results to be a promising candidate. The specific objective for this work is to develop a seal containing Y_2O_3 ceramic for fast thermionic reactor application.

Properties of yttria and rare earth doped yttria ceramics reported in previous GGA thermionic summaries brought out the following features:

1. Non-uniformity is hard to avoid in fabricating thermionic fuel element seal size ceramic bodies.

2. Yttria is mechanically weaker and more thermal shock sensitive than alumina.
3. The electrical resistivity of yttria is more erratic at 1073°K than that of alumina.

Attempts were made in this reporting period to overcome yttria non-uniformity and mechanical weakness problems by either additives to the ceramic or seal design modifications.

The results of the addition of two rare earth additives can be seen in the scanning electron photomicrographs (Figs. 2-11, 2-12). Schneider et al. (Refs. 14,15,16) theorize that the Y_2O_3 - La_2O_3 mixed oxide should be monoclinic and the Y_2O_3 - Ho_2O_3 mixed oxide should be cubic. The Y_2O_3 - La_2O_3 mixed oxide structure in Fig.2-11 is ill-defined. The Y_2O_3 - Ho_2O_3 mixed oxide structure, Fig.2-12, has many large tetrakaidecahedron grains. The addition of rare earths changed the microstructure of the yttria body. The effect of these changes on the properties of the yttria has not been determined.

Niobium and tungsten were added to yttria to form cermet. These metals were selected as yttria crack stoppers for improved mechanical properties. There are some deleterious reactions from these metal additions. Mixed WO_3 - Y_2O_3 oxide formation is known to cause some lowering in electrical resistivity (Ref.17) but this is not considered serious because of the limited amount of WO_3 present. Formation of Nb_2O_5 mixed oxides should also lower yttria electrical resistivity but this has not been verified. In practice, niobium cermet are strong and contain a minimum number of pores and cracks. These structures have worked well in graded yttria trilayer ceramic-to-metal transition zones. All ceramic cracking and tensile test failures occurred outside of this zone. The amount of niobium added to the yttria to form an insulating cermet is limited by the electrical conduction from touching niobium particles. This limit appears to be less than 5 weight per cent niobium for the materials and techniques now used for trilayer fabrication.

(5561-99C) 2000X

Fig. 2-11. Scanning electron photomicrograph of a fractured
 Y_2O_3 -6% La_2O_3 surface at 2000X

(5561-97C) 100X

Fig. 2-12. Scanning electron photomicrograph of a fractured
Y₂O₃-10% Ho₂O₃ surface at 100X

Recent yttria-based ceramic components were not completed in time for testing prior to program phase-out. One of these was a promising design for a ceramic-to-metal seal. The seal was designed with multiple, alternate layers of cermet and yttria as shown in Figs. 2-13 and 2-14. The intent of these multiple layers is to inhibit crack failure in the ceramic and enhance oxygen diffusion in yttria for higher density sintering.

2.3.2 Trilayer Development

The trilayer consists of concentric niobium cylinders bonded to an insulation layer between them. It provides electrical isolation between the collector and the liquid metal coolant and a thermal conductance path for heat rejection from the collector. A graded Nb-Al₂O₃-Nb trilayer was developed and was tested for thermal stability, property reproducibility, and irradiation stability. Metals and metal oxides were added to Al₂O₃ to stabilize its fine grain structure in an effort to improve its resistance to fast neutron damage. Cubic ceramics, such as Y₂O₃, were examined for the improvement of the resistance to fast neutron damage.

The objective of adding metals and metal oxides to stabilize the fine grain structure of alumina is to minimize the effect of fast neutron induced anisotropic expansion in hexagonal Al₂O₃. Select trilayer samples were prepared with tungsten, molybdenum, Pr₂O₃ and La₂O₃ added to the alumina. Fractured ceramic segments were examined before and after two hour, thermal treatments at 1973°K. The intent of this effort was to determine the effectiveness of the additives in inhibiting thermally induced grain growth in alumina, and thus the stability of the fine grain structure.

The cermets Al₂O₃-W and Al₂O₃-Mo were the most effective in inhibiting alumina grain growth (Table 2-4). Electrical resistivity decreased in all cermets but was not zero even in the 70 weight per cent tungsten sample. All samples were prepared by plasma spraying the metal-Al₂O₃ mixture and autoclaving the sprayed cermet at 1773°K and 10,000 psi for three hours.

(5565-13)

Fig. 2-13. Photomicrograph of a cross-section of multilayer $Y_2O_3 - (Y_2O_3 + Nb)$ ceramic-to-metal seal at 100X before autoclave densification

(5565-19) 100X

Fig. 2-14. Photomicrograph of a cross-section of multilayer Al_2O_3 -(Al_2O_3 + Nb) ceramic-to-metal seal at 100X before autoclave densification

TABLE 2-4

EFFECT OF ADDITIVES ON THE 2 HOUR, 1973°K STABILITY OF Al_2O_3 GRAINS

Sample No.	Ceramic Composition	Grain Size (μm) as Fabricated	Grain Size After 2 hr 1700°C (μm)	Appearance after Heat Treatment
5561-23A	Linde A Al_2O_3 + 6 wt per cent Pr_2O_3 (Dry Mix)	3-8	3-10	Some needle like grains
5561-23B	Linde A Al_2O_3 + 6 wt per cent La_2O_3 (Dry Mix)	3-8	12-25	
5561-29A	Linde A Al_2O_3 + 30 wt per cent Tungsten (Dry Mix)	1-5	1-5	
5561-29B	Linde A Al_2O_3 + 50 wt per cent Tungsten (Dry Mix)	1-4	1-4	
5561-29C	Linde A Al_2O_3 + 70 wt per cent Tungsten (Dry Mix)	0.5-4	1-4	
5561-43B	Linde A Al_2O_3 + 10 wt per cent La_2O_3 (Wet Mix)	3-8	3-11	
5561-45A	Linde A Al_2O_3 + 8 wt per cent Pr_2O_3 (Wet Mix)	2-8	3-16	Some needle formation
5561-45B	Linde A Al_2O_3 + 10 wt per cent Pr_2O_3 (Wet Mix)	2-7	2-17	Needle formation
5561-45C	Linde A Al_2O_3 + 12 wt per cent Pr_2O_3 (Wet Mix)	2-7	2-11	Needle formation random grains
5561-81A	Linde A Al_2O_3 + 5 wt per cent Molybdenum (Dry Mix)	2-12	2-12	Single ceramic grains filled trilayer gap
4966-53F	Linde C Al_2O_3 Control Sample No Additive	3-8	300+	

Molybdenum like niobium smeared during this plasma spraying. Thus Al_2O_3 with 10 weight per cent molybdenum and greater were electrically shorted. The tungsten did not smear and the cermet was not shorted at higher metal/ Al_2O_3 ratios. The 1973°K, 2 hour thermal treatment did not visibly influence the uniformity of the cermet grain structure, Fig. 2-15.

The rare earth additives were effective in inhibiting grain growth. Al_2O_3 with either 10% La_2O_3 or 12% Pr_2O_3 exhibited reduced grain growth in Al_2O_3 . Lanthania and praseodymia pinned grains were twice the size of the Al_2O_3 -W cermet grains but were equal to the molybdenum pinned alumina grains.

Needle grain growth was prominent in the Al_2O_3 - Pr_2O_3 samples but was not evident in any other samples (Fig. 2-16). Komeya and Inoue (Ref. 18) reported that needle structures in $\text{AlN-Y}_2\text{O}_3$ increased the ceramic flexural strength. The importance of these needle structures to trilayer ceramics in a thermionic reactor cannot be judged without further data.

2.3.3 Thermal Stability of Al_2O_3 and Y_2O_3 Trilayers

The specific objective for this effort is to evaluate the long-term stability of the electrical resistivities of Al_2O_3 and Y_2O_3 trilayers at high temperatures under a continuously imposed electrical potential. The results should indicate whether these trilayers can maintain stable electrical performance in an operating thermionic fuel element and whether significant amounts of ionic transport occur in the Al_2O_3 and at the Nb- Al_2O_3 interface which may cause degradation of the structural integrity of the Al_2O_3 and the Nb to Al_2O_3 bond.

A recent thermal stability test of Nb- Al_2O_3 -Nb and Nb- Y_2O_3 -Nb trilayer samples at 1073°K under 10 VDC potential was terminated after a test time of 1524 hours. These electrically loaded trilayer segments were:

1. Nb- Al_2O_3 -Nb graded trilayer
2. Nb- Y_2O_3 -Nb graded trilayer
3. Nb- Al_2O_3 -Nb TECO cermet

(5561-29AT) 10,000X

Fig. 2-15. Scanning electron micrograph at 10,000X of a fractured surface of an Al_2O_3 -30%W cermet after a 2-hour, 1473°K thermal treatment

(5561-45CT) 5000X

Fig. 2-16. Scanning electron photomicrograph at 5000X of a fractured surface of an Al_2O_3 -12% Pr_2O_3 sample

Fig. 2-17 shows the electrical resistivities of trilayer segments as a function of test time. Measured resistivities for the Nb-Al₂O₃-Nb, Nb-Y₂O₃-Nb graded trilayers were nearly constant for the 1524 hour test. Resistivities measured across the TECO cermet were erratic when first measured but the overall trend was for a gradually increasing electrical resistivity. There were no detectable differences in the microstructural appearance of samples electrically loaded from those not loaded (Figs. 2-18 to 2-20). What appeared to be chains of touching spheres could be seen throughout the TECO sample. There was no evidence that this caused electrical shorting paths across the cermet. The Nb-Y₂O₃-Nb trilayer was more stable at 1073°K to a 10 VDC potential in this test than in any previous test.

2.3.4. Electrical Insulation Coating Development

The specific objective for this work is to develop electrical insulation coatings for intercell regions to prevent cesium vapor arcing and for other TFE components, such as bus bar, thermocouple sheath, and cesium reservoir, to prevent electrical shorting. Plasma-sprayed Al₂O₃ coating is currently being used in thermionic fuel elements under test. Alternate methods for preparing insulation coatings, such as sputtering and ion-plating, were also evaluated.

A newly developed technique for plasma spraying alumina was investigated. Plasmadyne, the developer, sent samples of Al₂O₃ sprayed on 410 stainless steel of 10 mil thickness. These newer coatings were applied in vacuo at estimated gas velocities of mach 3. Older TFE coatings were applied in air with maximum gas velocities estimated to be mach 2. Bright field and polarized light views of the new coating can be seen in Figs. 2-21a and 2-21b. The outer layer with a different polarized light appearance was softer (DPH 365) than the dull appearing inner layer (DPH 500). Both coatings are much softer than fully dense Al₂O₃ (DPH ~3000). The newer plasma-sprayed alumina looks interesting but because of its softness and non-uniformity, should be investigated further before being used in place of the older mach 2 coatings.

Fig. 2-17. Log of Resistivity versus Time for Selected Trilayer Samples

(a) With 10 VDC potential

(5561-19L)

(b) Without 10 VDC potential

(5561-19U) 2000X

Fig. 2- 18. Scanning electron micrograph at 2000X of a fracture ceramic surface of a graded Nb-Al₂O₃-Nb trilayer after a 1073°K, 1514 hr test with and without a 10 VDC potential across the trilayer

(a) With 10 VDC potential
(TECO 45L)

(b) Without 10 VDC potential
(TECO 45U) 2000X

Fig. 2-19. Scanning electron micrograph at 2000X of a fractured ceramic surface of a TECO Al_2O_3 -30% Nb cermet with and without a 10 VDC potential across the trilayer

(a) With 10 VDC potential

(5561-93L)

(b) Without 10 VDC potential

(5561-93U) 1000X

Fig. 2-20. Scanning electron micrograph at 1000X of a fractured ceramic surface of an Nb-Y₂O₃-Nb graded trilayer with and without a 10 VDC potential across the trilayer

(a)

(b)

Fig. 2-21.(a) Bright field and (b) polarized light photomicrographs of a cross-section of a mach 3 plasma-sprayed alumina coating at 250X

Evaluation of sputter coating as a means of electrical isolation continued this last reporting period. Plasma-sprayed Al_2O_3 over niobium was sputter coated in two trials to seal the coatings. Scanning electron photos of these surfaces showed either very little sealing or a washing out of the plasma-sprayed structure by sputtered alumina coating. (Fig. 2-22). The appearance of the surface was that of the sputtered coating only. This duplex coating has an advantage over straight plasma spraying in areas where the coating must be leak tight, where the thickness of the plasma-sprayed coating is needed and where substrate bending is not a problem.

Sputter alumina was applied to niobium and NaCl substrates to determine the effect of substrate temperatures on the properties of the sputtered alumina. Fig. 2-23 (a), (b) shows scanning electron micrographs of coatings applied at 425 and 750°K. The higher temperature substrates appear to promote the formation of larger grains and continuous nucleation. Grain growth on the cooler substrate appears to be at sites where nucleation has already occurred.

The as-deposited alumina on NaCl was amorphous according to electron diffraction examination, Table 2-4. These coatings changed to $\theta\text{-Al}_2\text{O}_3$ plus some $(\text{Al}_2\text{O}_3)_5\text{H}$ after a 2 hour, 1473°K thermal treatment, Fig. 2-24(a), (b). This work agrees with the findings of Dragoo and Diamond (Ref. 19) on structural changes in evaporative alumina coatings.

One ion-plated alumina sample on niobium was prepared. The appearance of the coating was similar to sputtered coatings. The properties of ion plated alumina coatings and the conditions of deposition are not known in enough detail to compare it with sputter coatings.

It appears from the results of all physical vapor deposited alumina coatings tried in this program that sputter coatings can be applied to thermionic fuel element components as amorphous or γ alumina coatings. Conversion of these coatings takes place in a predictable manner with thermal treatment. Coating adhesion is good and can be applied to full scale ceramic-to-metal components. Compression stress strain tests of two of these seals

(a) (5562-13) 2000X

(b) (5562-13S) 2000X

Fig. 2-22. Scanning electron photomicrographs at 2000X of a plasma-sprayed alumina surface (a) before and (b) after sealing the surface with sputtered Al_2O_3

(a) (5565-37B) 5000X

(b) (5565-37A) 5000X

Fig. 2- 23. Scanning electron micrographs at 5000X of sputtered alumina applied to a substrate heated to (a) 425°K and (b) 750°K.

(a)

Fig. 2-24. Transmission electron micrograph at 19,720X of an amorphous, sputter coated Al_2O_3 coating (a) before a 2 hour, 1473°K thermal soak

(b)

Fig. 2-24. Transmission electron micrograph at 19,720X of an amorphous, sputter coated Al_2O_3 coating (b) after a 2 hour, 1473°K thermal soak

before and after sputter alumina coatings were applied (Fig.2-25) did not adversely affect the properties of the substrate and did not spall off the surface. This means these coatings could be applied to critical parts of the thermionic fuel element at intermediate fabrication stages without a detrimental effect on the coating or subassembly.

Fig. 2-25. Compression Test, Ceramic-to-Metal Seals before and after sputter Coating with Al_2O_3

2.4 JOINT TECHNOLOGY (J. Grebetz)

The objective of this subtask is to develop joining techniques that would improve the structural integrity of thermionic fuel element components and reduce cost. Earlier work in this program defined electron-beam welding parameters for effecting joints in refractory metal TFE components.

2.4.1 Diffusion Bonding Study

The object of this program is to refine the diffusion bonding of the tantalum transition to F-series tungsten emitters so that a joint of highest integrity can be produced with minimum cost.

Earlier work in this area provided lapping/chemical cleaning procedures that improved surface finish. (See Historical Review below.) Scanning electron microscopy was used to determine the most effective post-lapping cleaning treatment that would ensure the removal of lapping compounds without increasing surface roughness to coarser finishes than those created by the lapping process. As a result of this work, new lapping equipment was obtained to remove manual work from the lapping process and reduce its cost.

Work conducted during the past reporting period was to evaluate the effects of time, temperature and pressure with the overall objective of reducing fabrication costs by shortening the bonding cycle. Impetus was given to this work by observation of joint porosity in bonds examined during post-irradiation hot cell examination of fueled emitters. These bonds showed what appeared to be diffusion induced void formation on sections of bond perpendicular to the joint interface and a curious absence of such porosity on the portions of the emitter-transition piece joints that are on a 45° angle. The history of the development of the bond will serve to introduce the subject of current work in proper perspective.

Historical Review: The present method of diffusion bonding tungsten

to tantalum was initiated in 1963 with the fabrication of OC-4, and a series of improvements have been applied to some parts of the process since that time. The first study of diffusion bonding of these materials for thermionic applications was conducted by Elsner, Horner and Weinberg. The parameters devised were:

Joint temperature - 2073°K

Joint pressure - 2000 psi

Cycle - 1. Heat to 2073°K @ 500 psi

2. Raise pressure to 1800 psi for 1 hour

3. Raise pressure to 2000 psi for 1 hour

4. Cool to room temperature at 500 psi in ~ 40 minutes

System pressure: $< 5 \times 10^{-5}$ Torr

A joint temperature of 2073°K represents close to a practical limit since the midpoint of the C Series and F Series emitter tubes was raised to $\sim 2273^{\circ} - 2173^{\circ}\text{K}$ with the joint at 2073°K . The method of joint preparation was lapping with 600 grit silicon carbide abrasive.

During this period, configurations for lapping thin walled cylinders of tungsten to tantalum were studied. Various emitter tapers from 45° I.D. cones to 45° O.D. cones were studied. The best configuration was a flat (90° to the cylinder axis) surface with a trace (10 mils wide) conical chamfer at 45° on the inside of the cup. The chamfer allowed the machinist to lap the emitter in a true circular pattern.

In 1964 the lapping compound was changed to 15 micron diamond powder to ensure that any residual contamination from lapping would be carbon and a post-machining degassing treatment at 2273°K for 6 hours was added.

In 1969-70 the bonding chamber system pressure specification was revised to a value of $< 5 \times 10^{-6}$ Torr and bonds have typically been made at pressures of 2×10^{-6} to 8×10^{-8} Torr. During this period a 1200°C stress relief anneal was added at the end of the bonding run to accommodate the differential thermal stresses that occur upon cooling. In addition a more thorough cleaning

specification was applied to the tantalum component, and the lapping procedure was revised. The lapping compound was changed to six micron diamond paste (natural diamond) to give a smoother surface finish, and lapped surfaces were inspected at magnifications from 10 to 60 diameters. The post bonding degassing treatment at 2273°K was eliminated and a 14 hour stress relief anneal at 1223°K was added.

Failures of diffusion bonds have been traced to several causes:

A. Microcracks in the Tungsten at the Tungsten-Tantalum Interface.

The microcracks are generally caused by the O.D. and end grinding sequences; for example, when the open end of the emitter is ground prior to the finish grinding of the tungsten cup O.D. (as it should be) and the last grinding pass is taken in a direction to allow the grinding wheel to pass from the flouride tungsten substrate at the open end (as it should not be), damage to the grain-to-grain bonding results. If such a structure is lapped to a transition piece and bonded, the structure generally cracks along a grain boundary. Evidence of this weakness is always found in the form of grain boundary pull-out at the lapped tungsten surface near the O.D. of the part. Parts that exhibit this behavior are now lapped upon a flat plate until all pull-out is removed, then lapped to a transition piece for bonding and re-inspected.

One example of an apparent inherent weakness in a flouride tungsten billet was observed during 1970. This billet was lapped and bonded twice-- leaks occurred each time at the bond interface. Subsequent lapping followed by examination of the lapped tungsten surface at 20X revealed that grain boundary pull-out continued to occur even though several mils of material were removed during each lapping sequence.

B. Excessive Surface Roughness

It is believed that a rough surface texture results in "laps" in the surface of the tungsten that are not completely filled by tantalum as it creeps at the interface during the bonding step. It is highly probable that a rough surface is also one that is contaminated by embedded lapping materials (oil and diamond particles) that had not been removed by the chemical cleaning

process employed when coarser lapping compounds were employed.

C. Misalignment of Components

Component misalignment has caused bond failures to occur. Present machined accuracy of components and set-up techniques have eliminated this problem from our recent experience.

D. Temperature-Joint Loading Calibration

Some failures of bonds have been traced to pyrometer calibration or calibration of the force application system that yielded lower temperatures or lower pressures than those specified. This type of failure is accidental or related to operator error.

Current Work: A series of bonding study specimens were prepared and bonded at three different temperatures, 1673° , 2073° and 2194°K , with the overall objective of reducing fabrication costs by shortening the bonding cycle. Samples were bonded for three different times at temperature, 1 minute, 10 minutes and 100 minutes. A simple cylindrical sample configuration, described in Reference 19, was used because of its machining ease and easy characterization. Bonded samples were evaluated by metallographic examination, electron beam microanalysis and scanning electron microscopy. A sample of an F Series emitter tungsten-to-tantalum diffusion bond, prepared by the latest fabrication methods for surface preparation, was evaluated as a comparison sample by the same techniques. The results of these studies are summarized below:

A. Current practice in bonding F Series emitters results in an interface that is essentially void free. This result is a direct consequence of the improvements made in the lapping and post-lapping cleaning processes.

B. For cylindrical bond specimens, a gradient of void exists:

- (1) Void near the O.D. of specimens was practically eliminated for samples bonded for 2 hours at 2073°K .

- (2) Void present near the centers of specimens was determined to be related to the original surface roughness of the tantalum member by scanning electron microscopy. Void in these regions was regularly distributed with a spacing characteristic of the spacing of machining and lapping grooves, and many voids showed internal surfaces characteristic of being abrasively machined rather than being formed by a diffusional process.

Because of the void gradient, and because void density was unrelated to concentration gradients observed by electron beam microanalysis, it was concluded that void density was not related to volume diffusion effects.

2.4.2 Evaluation of Work

It was concluded that the presence of void could be related only to the original surface roughness of the samples that were bonded. Variations in void density between bonding temperatures and times for different samples were related to creep behavior and not to diffusional processes. Variations in void density along a sample diameter were a function of the varying creep rate (volume flow) of material along the diameter (increasing creep rates near the O.D. of the samples where volume restraint or sample edge strength was low, decreasing to low values at the centers of test samples where bulk flow was restrained by the surrounding sample volume). Direct parallels to these effects exist for hot forging operations in steel working where grain refinement may be poor in the centers of large forgings and adequate near the edges of the same forgings.

It was further concluded that the current bonding techniques for tungsten-tantalum joints for F and LF Series emitters produce an optimum joint.

2.5 SEAL AND TRILAYER IRRADIATION (J. Chin, C. W. Messick)

The objective of this subtask is to study the effect of fast neutron irradiation on the dimensions, structures and properties of seals and trilayers for thermionic fuel element application. The samples fabricated by Gulf General Atomic are irradiated in EBR II in heat pipe operated capsules designed and built by LASL. Post-irradiation examinations are carried out in the LASL hot cell.

2.5.1 Capsule K-3, 4

Irradiation of capsule K-3, 4 in EBR II has been completed. Post-irradiation examinations have been initiated in the LASL hot cell. The GGA samples irradiated in capsule K-3, 4 consist of a number of trilayers of the LASL size, a list of which was given in a previous thirdly report (Ref. 20). Most GGA samples were irradiated to a fast neutron fluence (> 0.1 MeV) of $2.0-2.6 \times 10^{21}$ nvt at 973°K . Out-of-pile ceramic impurity contents and microhardness measurements are shown in Tables 2-5 and 2-6. Examination results of irradiated K-3, 4 samples will be reported in a LASL topical report. A few remarks here summarize some of the post-irradiation examination results and what these results suggest in terms of trilayer stability to fast neutrons.

Irradiated alumina in all trilayer samples expanded at a rate predicted from measurements made on irradiated free standing alumina bodies. Niobium and Nb-1Zr trilayer skin swelling is estimated to be less than one-third of the alumina swelling. Trilayer samples made with high modulus Nb-1Zr either failed at the ceramic metal interface or caused cracks to form in the ceramic. Trilayers made with thin walled niobium skins followed the alumina swelling and did not crack. Where thick walled Nb skins were used, as in the case of the GGA graded trilayer (12 mil graded cermet, 26 mil collector and 28 mil thick inner sheath), the graded ceramic-to-metal joint was intact but the 5 mil thick alumina layer cracked.

IMPURITY CONTENT OF K-3 TRILAYER STARTING MATERIALS

2-63

TABLE 2-6

MICROHARDNESS MEASUREMENTS, K-3 UNIRRADIATED CONTROLS

Sample No.	Insulator Type	Microhardness in DPH				
		Collector		Ceramic	Sheath	
		<u>.028*</u>	<u>.002*</u>		<u>.002*</u>	<u>.028*</u>
4967-51	Graded Nb-Al ₂ O ₃	120	119	2520	180	183
4967-53	Graded Nb-Y ₂ O ₃	119	119	835	183	178
4967-55A	Al ₂ O ₃ -3% W		100	3220	177	
4967-55B	Al ₂ O ₃ -1% Y ₂ O ₃		119	3615	185	
4967-57A	Al ₂ O ₃ (Linde A)		117	3265	175	
4967-57B	Al ₂ O ₃ (Mellor 3μ)		107	3275	154	
4967-57C	Al ₂ O ₃ (301-E)		102	3230		

* Distance from the ceramic metal interface in inches

Thus, in all cases when the trilayer skins could not follow ceramic movement, ceramic failure occurred. The trilayer wall should thus be made of a thin low-modulus material with a good thermal expansion match to the ceramic, which will still allow passage of the assigned DC currents with minimum Joule loss and yet can follow dimensional changes of the ceramic due to fast neutron irradiation.

Trilayers made with cubic ceramics such as Nb-Y₂O₃-Nb graded trilayers showed a slight amount of swelling which was approximately what one would expect from niobium swelling only. The cubic ceramic trilayers showed less evidence of neutron damage than the Al₂O₃ ceramic trilayers but swelling was slightly greater than free standing bodies.

2.6 FUEL-CLAD IRRADIATION (M. Yates, E. Thacher, D. Allen, A. Marshall, L. Yang)

The objective of this subtask was to study the dimensional stability and compatibility of thermionic fuel-clad systems as a function of design and operating parameters of thermionic fuel elements. To allow more flexibility in the incorporation of test variables and more reliability in the instrumentation for measuring the temperature of the test sample, a material irradiation capsule was selected as the test vehicle. Capsule irradiation of carbide fuel-clad systems will be carried out under a NASA-sponsored program. Capsule irradiation of oxide fuel-clad systems is included under this subtask.

Previously, two capsules, FC-1 and FC-2, were fabricated for the evaluation of the effect of cladding temperature and thickness on the dimensional stability of oxide fueled tungsten emitters(Ref.21). The operational histories and test summaries are given in previous thirdly reports. Significant amounts of emitter distortion were observed in most of the FC-1 and FC-2 emitters. However, these results were not consistent with the results of the recent TFE tests, 6F3 and 1F1, as described in Section 3 of this report. One possible reason for the discrepancy is the difference in the pressure of the cover gas over the oxide fuel in these devices. The helium cover gas pressure was about one atmosphere over the fuel in the FC-1 and FC-2 capsules, as compared to the pressure of fission gases of 5-20 torr over the fuels in 6F3 and 1F1.

To continue the evaluation of the irradiation behaviors of oxide fueled emitters, two capsules, FC-3 and FC-4, were planned for the FY-73 program. The status of these capsules is summarized below.

2.6.1 FC-3 Oxide Fuel Capsule

Even though the observed dimensional changes in 6F3 and 1F1 are below those seen in FC-1 and FC-2, further improvements were desired. For this reason, an oxide fueled capsule designated as FC-3 was designed. This capsule contained four emitters; the major characteristics and schematic layout of the capsule and the emitters were given in prior thirdly reports (Refs. 22,23). The capsule assembly

was completed in November 1972, and the capsule was put on test on the 17th of November. A total of 1379 hours was accumulated by the time the reactor was shut down on January 22, 1973. The test history and the results of the neutron radiographs taken after shutdown are given in Section 3.5.

Thermal analysis of the FC-3 capsule was performed with varying input power and helium and neon in the space between primary and secondary containments.

Figure 2-26 illustrates the calculational model. The principal heat transfer (radial, cold) gap (emitter to heat rejector) is 2.11 mm (0.083 inch). The heat rejector to stainless containment gap is 0.13 mm (0.005 inch), and the gap from emitter transition to heat rejector is 0.13 mm (0.005 inch). The emitters incorporate a 0.51 mm (0.020 inch) thick stem. Clearance at the bottom of the emitter is 3.81 mm (0.15 inch).

The average emitter temperature, maximum emitter temperature and maximum heat rejector temperature as a function of emitter heat flux are presented in Figure 2-27. Data points represent four computer runs at varying input power using the two gases, helium and neon, in the secondary containment.

Figure 2-28 illustrates the detailed temperature map for the case of the emitter average temperature at 1822° K. Note that average temperatures, average gap thicknesses at temperature and emitter collector heat fluxes are given. The radial (I) coordinates (1 through 13) and the axial (J) coordinates (18 through 44) are the same as those indicated on the scale illustration in Figure 2-26.

Fig. 2-26. Scale Layout of Emitter in FC-3

Fig. 2-27. Capsule Temperatures

THE RADIAL (I) DIRECTION IS HORIZONTAL
THE AXIAL (J) DIRECTION IS VERTICAL

	1	2	3	4	5	6	7	8	9	10	11	12	13
44	1361	1659	1690	1706	1717	1732	1750	1750	973	794	737	373	
43	896	860	846	838	833	826	816	814	808	800	733	373	
42	858	853	842	835	830	823	814	811	805	789	723	373	
41	871	870	860	853	846	835	808	800	778	747	698	373	
40	954	952	937	933	937	932	926	925	920	800	727	373	
39	988	986	963	954	953		929	928	923	805	731	373	
38	1006	1006	987	972	968		943	941	927	799	728	373	
37	1006	1006	998	988	982		979	979	978	780	720	373	
36	1006	1006	1002	998	990		1000	998	998	773	716	373	
35	1006	1005	1002	999	995		1008	1005	1005	768	713	373	
34	1005	1005	1003	1001	1001		1071	1071	1071	756	704	373	
33							1141	1141	1140	759	707	373	
32							1232	1231	1230	767	712	373	
31							1416	1415	1414	806	741	373	
30							1554	1564	1562	838	764	373	
29				2488	2351	2135	1694	1692	1690	892	802	373	
28				2586	2454	2245	1818	1816	1813	956	845	373	
27				2645	2513	2304	1878	1875	1872	990	866	373	
26				2663	2531	2323	1895	1893	1890	1000	872	373	
25				2641	2511	2305	1882	1880	1877	990	866	373	
24				2571	2449	2256	1852	1850	1846	968	852	373	
23				2332	2247	2112	1803	1801	1798	923	822	373	
22	1873	1872	1871	1870	1853	1827	1781	1777	1774	878	790	373	
21	1607	1671	1702	1719	1730	1745	1763	1764	1761	862	780	373	
20	1368	1669	1700	1716	1727	1742	1760	1761		799	740	373	
19	899	863	849	841	836	828	819	816	810	803	735	373	
18	861	856	845	838	833	825	816	814	808	792	725	373	

EMITTER AVG. TEMP. 1822.35 K
 EMITTER MAX. TEMP. 1893.23 K
 HEAT REJECTOR AVG. TEMP. 888.08 K
 HEAT REJECTOR MAX. TEMP. 999.92 K
 FUEL CENTERLINE MAX. TEMP. 2663.39 K
 AVERAGE EMITTER HEAT FLUX 36.06 WATTS/SQ. CM.
 AVERAGE EMITTER TO HEAT REJECTOR GAP .0853 IN.
 AVERAGE TRANSITION PIECE TO HEAT REJECTOR GAP .0071 IN.
 AVERAGE HEAT REJECTOR TO PRIMARY CAN GAP .0054 IN.

Fig. 2-28. Fuel Clad Thermal Analysis for FC-3 Capsule

2.6.2 FC-4 Oxide-Fuel Capsule

In addition to FC-3, another oxide-fueled capsule, designated as FC-4, was planned for the evaluation of other fuel forms and thoriated tungsten cladding. The basic ground rules adopted for this capsule were based on the FC-3 ground rules, which were discussed in Reference 20.

Several changes in mechanical detail from the FC-3 configuration were incorporated in FC-4. These are documented on drawing 278-SK-2079E and its supporting subassemblies and details. The cell fuel assembly and emitter assembly designs are documented on the drawings tabulated below in descending sequence from the capsule top. All cells were fueled with UO_2 .

<u>Cell No.</u>	<u>Assembly</u>	<u>Drawing No.</u>
1	Emitter	278-SK-2053-28
	Fuel	278-SK-2151-1
2	Emitter	278-SK-2053-32
	Fuel	278-SK-2166-1
3	Emitter	278-SK-2053-30
	Fuel	278-SK-2169-1
4	Emitter	278-SK-2053-26
	Fuel	278-SK-2050-1

Cell 1 fuel is insulated from the clad and on each end by 0.030 in. of ThO_2 . The intent of this is to create a minimum temperature difference of $100^\circ K$ between the fuel and the clad. Thus, the more plastic, weaker fuel would swell into the fuel central cavity rather than distort the clad. It also seemed possible that reduction or elimination of fuel ratcheting might occur because of the physical isolation of the fuel from the clad. Thoria was chosen as the insulator because:

- a. Its thermal conductivity is low enough to allow a thin sleeve.
 - b. Its vapor pressure at the temperatures involved is low.
- Therefore, it would not redeposit significantly.

- c. UO_2 diffusion through the ThO_2 was estimated to be small.
- d. Thin ThO_2 sleeves are fabricable and relatively cheap.

The ThO_2 insulators on the fuel ends are required to reduce an anticipated increase in axial heat conduction in the fuel caused by the ThO_2 sleeve. The cell 2 fuel design is illustrated in Figure 2-29.

The fuel design for cell 2 (Figure 2-30) has six wedge-shaped slots equally spaced around its periphery. The purpose of the slots is to provide a volume at the clad-fuel boundary into which the fuel may swell. The total slot volume is enough to accept a maximum anticipated fuel volume increase of about 15%. At a constant fuel volume fraction of 0.72, the number of slots is a trade-off between the optimum slot shape (small included angle, small radius at vertex), a fabricable slot shape (larger included angle, appreciable vertex radius), and the need to have a fuel central cavity diameter greater than 0.040 inch.

To prevent fuel redeposition inside the slots, each was sized so that it could be lined with 0.005 inch thick W-25Re foil. The foils are installed so that they are always in contact with the clad, but can be easily pushed back into the fuel slots, flush with the fuel surface, as the fuel expands during start-up. This should prevent the slot edges from placing concentrated loads on the clad inner surface.

The fuel for cell 3 (Figure 2-31) is a stack of ten fuel pellets and nine equally spaced W-25Re disc fins. The fins are 0.020 inch thick and have a 0.050 inch diameter central hole to allow fission gas passage. Each pellet is 0.182 inch thick and has a central hole. The hole diameter depends on the percent of theoretical density of the fuel. The fuel volume fraction of this configuration is also 0.72.

Fig. 2-29. Fuel Design for Cell No. 1 of FC-4 Capsule

Fig. 2-30. Fuel Design for Cell No. 2 of FC-4 Capsule

Fig. 2-31. Fuel Design for Cell No. 3 of FC-4 Capsule

The disc fins take advantage of the redeposition characteristics of the fuel. They are sized to minimize the redeposited fuel thickness on the clad. Since there is then less fission-gas-producing material next to the clad, less clad deformation is expected. The fins run colder than the fuel, so the fuel tends to redeposit on them as well as on the clad.

The clearance between the disc fin outer diameter and the clad inside diameter was made equal to the differential expansion between the fins and the clad. This prevents loading of the clad by the fins.

The last cell in the fuel-clad capsule contains a conventional UO_2 fuel configuration sized for 0.85 FVF. However, the fuel cladding is thoriated tungsten coated with chloride tungsten instead of fluoride-chloride duplex tungsten.

The thoriated tungsten provides higher creep strength than fluoride tungsten, while the chloride tungsten layer provides the high vacuum work function needed for improved thermionic performance.

2.7 HOT CELL EXAMINATIONS (H. Johnson, G. Buzzelli, L. Yang)

The objective of this subtask is to study the dimensions, macroscopic appearances, microstructures, fission product distribution, and joint integrity of the components of capsules, converters and thermionic fuel elements after in-pile operation. The information is fed back into the material and thermionic fuel element development programs for the improvement of the life and performance of thermionic space power systems.

During this reporting period, the main effort was devoted to the examinations of the 90 UC-10 ZrC - 4 wt % W fueled TFE 6F1 and TFE 6F2. The results are described as follows.

2.7.1 Examination of 90 UC-10 ZrC-4 wt % W Fueled Thermionic Fuel Element 6F1

The essential components and operating history for 6F1 were described in Table 2-6 of the last thirdly report. The fission product release and distribution in 6F1 were studied during the last reporting period and the results obtained were also included in the last thirdly report. Macroscopic and metallographic examinations of various 6F1 components were carried out during this reporting period.

2.7.1.1 Emitter deformation. The maximum diametric expansions for the top five emitters are as follows: No. 1 (top emitter) 23 mils, 1/2 inch from emitter bottom; No. 2, 17 mils, 1/2 inch from emitter bottom; No. 3, 5 mils, 1/4 inch from emitter bottom; No. 4, 17 mils, 1/2 inch from emitter bottom; No. 5, 18 mils, 1 inch from emitter bottom. These variations of emitter deformation probably reflect the temperature profiles of these emitters. Cracks were visible in the claddings of all these five emitters, mostly along grain boundaries (Figure 2-32).

Figure 2-32 Typical Example Illustrating
Cracks in the Cladding of 6F1
Emitters. (Emitter No. 4, 2X)

2.7.1.2 Macroscopic Appearance of Emitter Cross Section. Typical examples are shown in Figures 2-33 and 2-34 for emitter No. 1 and emitter No. 3, respectively. The top fuel pellet of emitter No. 1 was loose and fell out of the emitter cavity during handling; therefore, only seven fuel pellets were shown in Figure 2-33. It can be seen that the fuel pellets showed little or no swelling near the top of emitter No. 1. On the other hand, the fuel pellets near the bottom of emitter No. 1 exhibited such a large amount of swelling that the fuel central cavity disappeared. This indicates the presence of a very large axial temperature gradient in emitter No. 1. The appearance of the cross section of emitter No. 3 (Figure 2-34) implies a much lower axial temperature gradient in this emitter and the fuel central cavity showed the typical hourglass shape.

2.7.1.3 Fuel-Cladding Interaction. Figures 2-35 and 2-36 illustrate the fuel-cladding interaction layer. In each emitter the interaction layer becomes thinner near the emitter stem where the temperature is lower. The maximum thickness for the four emitters examined are: No. 1, 8 mils; No. 2, 12 mils; No. 3, 8 mils; No. 4, 7 mils. There does not appear to be a relationship between fuel enrichment and the thickness of the fuel-cladding interaction layer.

2.7.1.4 Microstructures of Fuel. The fuel body in all the emitters examined remained highly porous, although the fuel density appeared to be slightly higher in the fuel body near the cladding (Figures 2-37(a), (b) and (c)). The lack of densification in spite of the high operating temperature of the fuel is probably caused by the presence of closed pores containing fission gases. The slightly higher fuel density near the cladding may be caused by the vapor transport of fuel material toward the cooler region. There is no evidence of segregation of tungsten-containing phases at the surfaces between fuel pellets and at the fuel-cladding interfaces (Figure 2-38) as was observed in the UC fueled 1F2 emitter (see last thirdly report).

Figure 2-33 Longitudinal Cross Section of No. 1 Emitter
of TFE 6F1 (5X)

Figure 2-34 Longitudinal Cross Section of No. 3 Emitter
of TFE 6F1 (5X)

Chloride
Tungsten
Cladding

Fluoride
Tungsten
Cladding

Interaction
Layer

Fuel

Figure 2-35 Typical Macroscopic View of
Fuel-Cladding Interaction Layer
(Emitter No. 1, 30X)

Figure 2-36 Typical Microscopic View of
Fuel-Cladding Interaction Layer
(Emitter No. 3, 75X)

(a) Near Fuel Central Cavity

(b) Midway between Fuel Central Cavity and Cladding.

(c) Near Cladding

Figure 2-37 Typical Fuel Microstructures in TFE 6F1(Emitter No.2, 100X), Showing High Degree of Porosity.

Figure 2-38 Fuel Material Near Cladding
of the Bottom of Emitter No. 1
of TFE 6F1, Showing Absence of
Segregation of Tungsten-Containing
Phase (150X)

2.7.1.5 Microstructures of Cladding. The cladding in general exhibited columnar grain structures for the fluoride tungsten and equiaxial grain structure for the chloride tungsten (see Figure 2-36). Grain growth in fluoride tungsten was observed in regions where large cladding deformation was observed. In these cases, the boundary between the chloride tungsten and the fluoride tungsten vanished and porosities appeared at the grain boundaries, which eventually led to grain boundary cracking. (See Figures 2-39 and 2-40)

2.7.1.6 Bellville Washer Support. For all the emitters examined, at least part of the Al_2O_3 disc supporting the Bellville washer disintegrated into individual grains (Figure 2-41). In addition, the tantalum support reacted with the Al_2O_3 disc to form a white interaction layer (Figure 2-42). It appears that the Al_2O_3 disc was subjected to temperatures much higher than the design value.

2.7.1.7 Collector-Trilayer. Gaps were found between the ceramic layer and the niobium outer sheath of each of the collector-trilayer structures examined. Typical examples are shown in Figure 2-43. In some cases, gaps were also observed between the ceramic layer and the niobium collector. These gaps were probably formed by the sintering of the plasma-sprayed Al_2O_3 during the in-pile operation of the TFE. The presence of these gaps impeded the heat transfer between the collector and the niobium outer sheath and caused the collector to operate at temperatures much higher than the design value. The microhardness of the niobium at the I.D. of the ceramic (collector side) was higher than that at the O.D. of the ceramic (outer sheath side) (Knoop hardness = 200-230 versus 140-150), while the microhardness of the niobium collector and the niobium outer sheath (Knoop hardness = 100-120) was lower than these values. This indicates that contamination of the niobium surrounding the Al_2O_3 layer occurred during the operation of the TFE and that the collector side suffered more contamination because of its higher operating temperature.

Figure 2-39

Grain Growth in
Fluoride Tungsten
and Porosities at
Grain Boundary of
Tungsten Cladding
of Emitter No. 2
of TFE 6F1, 75X

Figure 2-40

Grain Boundary
Cracking in
Tungsten Cladding
of Emitter No. 2
of TFE 6F1, 100X

Figure 2-41 Al_2O_3 support for Bellville Washer of Emitter No. 2 of TFE 6F1. Note Part of the Disc Disintegrated into Individual Grains. 200X

Al_2O_3 Disc

Interaction Layer

Tantalum Holder

Figure 2-42 Interaction Layer Between Al_2O_3 Disc in Figure 2-41 and Tantalum Holder

Nb Outer Sheath

Gap
 Al_2O_3

Nb Collector

(a) Collector-Trilayer of Emitter No. 1

Nb Outer
Sheath

Gap Al_2O_3

Nb
Collector

(b) Collector-Trilayer of Emitter No. 2

Figure 2-43 Collector-Trilayer Structures in TFE 6F1. Note Gap Between Al_2O_3 Layer and the Nb Outer Sheath. 30X.

2.7.1.8 Intercell Region and Fission Product Venting Orifices. The plasma-sprayed Al_2O_3 insulation layer remained white except at locations surrounding the fission product venting orifices where a dark region existed (Figure 2-44). This dark region was believed to be caused by the interaction of fission products, such as Sr and Ba, with the Al_2O_3 . Similar interactions between Al_2O_3 and Sr and Ba fission products were observed in the two cell E-series TFE 2E3. The reaction shown by 6F1 was less severe than that by 2E3, probably because the fission product release was less from the 6F1 fuel. The low fission product release from 6F1 fuel was attributed to the fact that most of its open porosity was lost when being operated at very high temperatures because of poor heat transfer across the fuel-cladding gap formed by fuel sintering. Figure 2-44 also shows the presence of cracks in the Al_2O_3 layer at the bottom of the convolution of the insulator skirt. These cracks were probably formed by stress induced through thermal cycling. Their presence may impair the ability of the Al_2O_3 insulation layer to prevent cesium arcing between the outer sheath and the insulator skirt. Figure 2-45 shows the surfaces of the Nb-1 Zr sheath around the fission product venting orifices in emitter No. 2. There was no heavy deposit on these surfaces, although the areas directly facing the orifices showed a darker color. The lack of heavy deposit is consistent with the observed low fission product release from 6F1.

To insure that the low fission product release was not due to the plugging of the Al_2O_3 venting tubes, each intercell region was mounted, sectioned and polished to reveal the cross section of these venting tubes. Figure 2-46 is a typical example of such cross sections. No significant plugging was found in all the samples examined.

Figure 2-44 Intercell Region Above Emitter No. 2 of TFE 6F1. Note Dark Region Around Fission Product Venting Orifice and Crack in Al_2O_3 Layer at Bottom of Convolution of Insulator Skirt (2X)

Figure 2-45 Surfaces of Nb-1 Zr Sheath Surrounding the Fission Product Venting Orifices of Emitter No. 2 (2X)

Figure 2-46 Typical Example of the Cross Section
of the Al_2O_3 Fission Product Venting
Tube (Emitter No. 1, 5X)

2.7.1.9 Joints. No major defects were observed in all the joints examined. A white phase, however, was found in the weld joints between niobium and tantalum (see Figures 2-47 and 2-48) in the intercell regions above emitters No. 2, 3, 4 and 5 (Emitter No. 6 was not examined). A similar white phase was also present in the niobium convolutions of the insulator skirts of these emitters (see Figure 2-49), which grew from the side in contact with the cesium vapor toward the side coated with the plasma-sprayed Al_2O_3 insulation layer. This white phase was probably formed by some contaminants, such as oxygen, in the cesium vapor and its formation was sensitive to temperature. The niobium-to-tantalum welds above emitter No. 1 did not contain this white phase, probably because the temperature there was cooler than that above the other emitters because of heat loss through the electrode lead. Samples were sent to LASL hot cell for the determination of the composition of this white phase and the nature of the contaminant, but the work was not completed due to the termination of the thermionic program.

Figure 2-50 shows the tantalum-to-tungsten diffusion bond in emitter No. 2. Some Kirkendall voids were found on the tantalum side in the flat portion of the bond but the tapered portion of the bond contained fewer Kirkendall voids. The bond in general appeared to be in good condition. Examinations of the tantalum-tungsten diffusion bond in emitters No. 1, 3, 4 and 5 yielded the same results. These observations are in general agreement with the results obtained in previous hot cell examinations made on other thermionic devices. The tantalum-to-tungsten diffusion bond appears to be a promising joint for long-term application in a thermionic fuel element.

2.7.2 Examination of 90 UC-10ZrC-4 wt% W Fueled Thermionic Fuel Element 6F2

Due to the termination of the thermionic program early in January 1973, hot cell examination of 6F2 was limited to macroscopic examinations of the emitters, the Al_2O_3 insulation layers in the intercell regions, and the Al_2O_3 support for the Bellville washer, and the determination of the fission gas Kr-85 released into the charcoal trap.

The essential components and operating history of 6F2 are summarized in Table 2-7.

Figure 2-47 Weld Joint Between Tantalum Transition of Emitter No. 2 and Niobium Collector of Emitter No. 1. Note White Phase on the Niobium Side of the Weld (100X)

Figure 2-48 Weld Joint Between Tantalum and Niobium Skirt of Insulator in Emitter No. 2. Note White Phase Around the Weld (75X)

Figure 2-49 Niobium Convolution of Insulator Skirt of Emitter No. 2. Note White Phase Grew in from the Cesium Side Toward the Plasma-Sprayed Al_2O_3 Layer Side (75X)

Figure 2-50 Tantalum to Tungsten Diffusion Bond in Emitter No. 2. Note Kirkendall Voids on Tantalum Side of the Interface (100X)

TABLE 2-7

ESSENTIAL COMPONENTS AND OPERATING HISTORY FOR 6F2

Fuel**	Emitter	Insulator Seal	Trilayer	Collector	Average Emitter Temp ($^{\circ}$ K)	Testing Time and Average Burnup	Failure Mode
90 UC-10 ZrC -4 wt% W $C/U^* = 1.012$, 1.034, 1.040, 1.030, 1.028 1.030.	Chloride- Fluoride Duplex Tungsten 40 mil thick, 2 inch active length, 1.1 inch nominal O.D. $F^* = 5$, 13, 5, 5, 4, 4.	Lucalox-Nb 40 Nb-60V Braze, W/Y ₂ O ₃ Metallizing	Nb-Al ₂ O ₃ Graded Cermet	Nb	1900	7685 hours 6.6×10^{19} Fission/cc	Envelope Leak and Shorting

*Listed from the top cell toward the bottom cell.

**Bottom of the last fuel pellet dished.

2.7.2.1 Determination of Fission Gas Release into Charcoal Trap. A total of 79.0 millicuries of Kr-85 was collected from the charcoal trap after bakeout at 250°C. This corresponded to 3.5% of the Kr-85 formed during the irradiation. This was much higher than that present in the charcoal trap of 6F1 ($2 \times 10^{-5}\%$). The release, however, was much less than that observed for carbide fuels irradiated in high pressure (~ 1 atm) inert gas environment (40-60%). This is probably caused by the loss of open porosity in the fuel body upon irradiation in low pressure (<20 torr) inert gas when the fuel temperatures reached very high values because of poor heat transfer across the fuel-cladding gap.

2.7.2.2 Macroscopic Examination of the Emitters. All six emitters had cracks in the cladding. A typical example is shown in Figure 2-51. The cracks appear to follow the grain boundaries.

2.7.2.3 Intercell Region Al_2O_3 Insulation Layer. The Al_2O_3 insulation layer of the intercell region was darkened around each fission product venting hole. The area of the darkened region was much bigger than that in 6F1. This is consistent with the observed larger release fraction (3.5% versus $2 \times 10^{-5}\%$) of fission gas (presumably also other fission products) through these venting orifices into the charcoal trap. The dark layer is believed to be formed by the interaction between Al_2O_3 and the fission products Sr and Ba. A typical example is shown in Figure 2-52. The Nb-1 Zr sheath surface facing the intercell region in Figure 2-52 is shown in Figure 2-53. It has a general dark appearance, probably due to the condensation of some of the vented fission products such as Cs, Sr and Ba.

2.7.2.4 Al_2O_3 support for Bellville Washer. All the Al_2O_3 supports for the Bellville washers in 6F2 were found to be cracked, presumably by the stress imposed when the emitter structure expanded during heating. A typical example is shown in Figure 2-54. This is an area in need of better design in any future TFE developmental program.

Figure 2-51 Macroscopic Appearance of Emitter
No. 5 in TFE 6F2 (2X)

Figure 2-52 Al_2O_3 Layer in the Intercell Region
Above Emitter No. 2 in 6F2. Note Dark
Region Around Fission Product Venting
Orifice (2X)

Figure 2-53 Nb-1 Zr Sheath Surface Facing the
 Al_2O_3 Insulation Layer of the
Inter-cell Region Shown in Figure 2-52.
Note Dark Deposit (2X)

Figure 2-54 Al_2O_3 Support for the Bellville
Washer of Emitter No. 5 of 6F2
Note Cracks. (2X)

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TASK 3. THERMIONIC FUEL ELEMENT DEVELOPMENT

The thermionic fuel element (TFE) development task goal is to provide a thermionic fuel element for use in thermionic reactor power systems to produce electrical power for space and terrestrial applications. The application presently most influencing its development objectives is space nuclear electric propulsion. Detailed information on TFE planning and test schedules to meet these development objectives is given in Ref. 1.

The expected power output from a TFE for long-term applications is 800 watts to 1 kWe. Specific lifetime goals in the development program were operation of a six-cell TFE (6F) for 10,000 hours by July 1974, and 20,000 hours by January 1977. Present lifetime on a 6F has surpassed 8,000 hours, and 8,000 hours has been achieved on a single-cell F-series. 10,000 hours has been achieved on experimental in-core tests and on a 2-cell TFE of smaller diameter.

The principal development areas are emitter dimensional stability and performance stability. The primary objectives of nearly all tests being operated or designed is to provide data relating to one or both of these areas. Each test also provides data on other development areas including fission product venting characteristics, insulator stability, envelope integrity, etc. Development has been underway on TFE design and fabrication procedure modifications aimed at reducing TFE cost. A summary of development work alone in each major development area and present status is given in Section 5 of the TFE Plan, Ref. 1.

The thermionic fuel element development task encompasses the design, engineering analysis, fabrication, encapsulation, and testing of TFEs and components. Fission heated TFEs are tested in the core of the Thermionic Test Reactor (TITR) at Gulf General Atomic. Data summarizing the current status of fuel elements is given in Sections 3.2 and 3.5 and in Appendix A.

3.1 ENGINEERING AND HARDWARE DESIGN (E.F. Thacher)

The objective of this subtask is to produce drawings and specifications (except materials and process specifications) and engineering analyses in support of the fabrication of development hardware. This includes design of single-cell, two-cell, and full-length elements and of fuel-clad capsules for testing in the TITR.

The approach taken is to prepare designs, to develop mathematical models of the designs and their environment, and then calculate heating rates, temperatures, stresses, strains, and electrical performance. Design versions are based on the analytical results and test data.

The tables in Appendix A summarize past effort on Subtask 3.1. Major work completed in this reporting period was: (1) conceptual design of TFE 2F⁴, design of fuel-clad capsule FC-4 (Section 2.6.2), analysis of TFE 6F⁵, design of the UO₂ fuel press, preliminary design of TFE 2LF², modification of BRITL code to analyze thermionic emitter distortion, and laboratory F-cell design.

3.1.1 F-Series TFE Design Analysis (E.F. Thacher, D.T. Allen, D.J. Heisser).

3.1.1.1 TFE 2F⁴

Conceptual design of the fuel for F-series TFE 2F⁴ was completed. This carbide-fueled device was to be used to test means for reducing emitter deformation. The methods employed were increasing fuel surface-to-volume ratio, improving the vented surface distribution in the fuel, and reducing fuel tungsten content. TFE details, other than for the fuel, were standard for the 2F-series.

Figure 3-1 depicts the fuel concepts selected for TFE 2F⁴. No detailed design work on these concepts was done. The advantages of

STANDARD INTERIOR
UC PELLETS. BOTTOM
AND TOP PELLETS ARE
DISHED ON TOP AND
BOTTOM FACES. FOUR
PELLETS REQ'D.

PLATELET CONCEPT
STACK OF 17 REQ'D

SCALE 2:1

Fig. 3-1. Fuel Pellet Designs for TFE 2F4

the platelet concept shown in Fig. 3-1 are high surface-to-volume ratio, better distribution of vented surface in the gas-producing volume and a potential for using pellets of differing enrichment to produce more axially uniform volumetric heat generation. The standard geometry fuel was intended to act as a control to help judge the effectiveness of the platelet concept.

Both fuels in Fig. 3-1 contain the same low percentage of tungsten, perhaps $\leq 1\%$. The exact percentage had not been selected when work was stopped. The expected effect of lowering the tungsten content below 4% (which was standard) was reduction of the fuel creep strength. Measured creep data indicated that reduction by a factor of 100 might be achieved. A weaker fuel body would accommodate fuel swelling by allowing the fuel to move inward into the central cavity rather than moving outward to increase the cladding diameter.

The dimensions shown in Fig. 3-1 for the platelet concept yield a vented surface-to-volume ratio of 9.9 in.^{-1} . This can be compared to a surface-to-volume ratio of 1.5 in.^{-1} for the standard fuel shape. The pellet thickness shown for the platelet design is about the smallest considered fabricable.

3.1.1.2 TFE 6F5

The relative effects of fuel configuration on stress and deformation at startup were determined from elastic calculations for the "shaped" and the reference fuel forms of TFE 6F5. All of the 6F5 fuel forms were designed with 0 to 0.003 in. interference between fuel and cladding at the design temperature. The analysis assumed an interference of 0.003 in. at 1900°K . The fuel shapes in the 6F5 converter are described in Ref. 2.

Cladding stresses appear to be significantly reduced by the slots in the fuel, but the peak stresses in the fuel are increased. Fuel effective stresses at points

of concentration are no higher than 494 MN/m^2 (71,700 psi). The F37 configuration has lower calculated peak cladding stress than the F39 configuration, but the latter has 19% less elastic cladding expansion, indicating that the average hoop stress is lower.

An elastic thermal stress analysis in the plane normal to the axis was performed for both the F37 and F39 fuel configurations. The F37 fuel has a cross-shaped void in the center. The F39 fuel has a small central hole with radial slots from the outside inward. The calculations were made with the computer program FEATS (Ref. 3) and modeled as a case of generalized plane strain. The initial cold gap between fuel and cladding was accounted for by adjusting thermal expansion expressions to zero at the calculated isothermal temperature at which the gap closed.

For comparison of the FEATS results with the stress condition in the axisymmetric fuel configuration used generally before 6F5, a calculation was made for the axisymmetric fuel form with the same power and fuel volume fraction and the identical initial radial gap and fuel properties. Resulting temperatures and deformation of the emitter are summarized in Table 3-1.

In comparison to the axisymmetric fuel form, the ΔT across the fuel and cladding is 49% greater for F37 and 60% greater for F39 configurations. The fuel-weakening configurations for F37 and F39 result in 23% and 38%, respectively, less emitter expansion than the axisymmetric standard fuel shape.

The worst stress condition for F39 appears in the emitter-cladding interface at the outside corner of the radial slot. This appears to be an artificial result of the computer program's requirement for continuity from cladding to fuel. The high tensile and shear stresses indicate that, in fact, the fuel and cladding will slide on each other as the TFE is started up.

TABLE 3-1

EMITTER TEMPERATURE AND DEFORMATION

	<u>F37</u>	<u>F39</u>	<u>Axisymmetric</u>
Max. Fuel Temperature	1986°K	1976°K	1955°K
Min. Emitter Temperature	1864°K	1845°K	1873°K
Maximum ΔT	122°K	131°K	82°K
Max. Emitter Diametral Expansion	0.28 mm (0.0109 in.)	0.22 mm (0.0085 in.)	0.34 mm (0.0133 in.)
Min. Emitter Diametral Expansion	0.26 mm (0.0103 in.)	0.21 mm (0.0083 in.)	0.34 mm (0.0133 in.)

The value for maximum cladding stress in the F39 configuration is probably closely coupled to this unreal "locking" of fuel and cladding. Relief of the shear stress along the fuel-cladding interface would result in lower stresses in the cladding. Unlike the axisymmetric configuration or F37, the F39 fuel will be forced to move along the fuel-cladding interface during thermal cycling. This motion will serve to reduce stresses in the cladding below the values calculated.

For both shaped configurations, the maximum fuel stress is located in the notch of the radial cut. Table 3-2 summarizes the maximum calculated stresses in fuel and cladding for the three configurations studied.

The lower cladding stresses in F37 and F39 indicate the effectiveness of weakening the fuel. The higher fuel stresses in F37 and F39 result from stress concentrations caused by the grooves.

For the F37 configuration maximum effective stresses increase 131% in the fuel and decrease 69% in the cladding compared to the axisymmetric configuration. The F39 fuel as modeled has 56% higher maximum fuel stress and 61% lower clad stress than the axisymmetric form.

3.1.2 TFE Cost Reduction

3.1.2.1 UO₂ Fuel Press

Design of a UO₂ fuel press which will reduce the UO₂ fuel fabrication cost was completed. Figure 3-2 is a sketch of the fuel press.

The knock-down design of the fuel press die allows a quicker removal of the completed pellet from the press than does the conventional solid die configuration, thereby reducing fuel fabrication time.

TABLE 3-2
EMITTER MAXIMUM STRESSES

(+ tension, - compression)

	<u>F37</u>	<u>F39</u>	<u>Axisymmetric</u>
Fuel			
Inplane Principal Stress	-845 MN/m ² (-122,600 psi)	-707 MN/m ² * (-102,600 psi)	368 MN/m ² (53,300 psi)
Axial Principal Stress	-138 MN/m ² (-20,000 psi)	-555 MN/m ² * (-80,500 psi)	368 MN/m ² (-53,300 psi)
Maximum Shear Stress	352 MN/m ² (51,000 psi)	303 MN/m ² * (44,000 psi)	0
Effective Stress	494 MN/m ² (71,700 psi)	333 MN/m ² * (48,300 psi)	214 MN/m ² (31,100 psi)
Cladding			
Inplane Principal Stress	+670 MN/m ² (+97,100 psi)	+965 MN/m ² ** (+140,000 psi)	+2050 MN/m ² (+297,300 psi)
Axial Principal Stress	+728 MN/m ² (+105,500 psi)	+831 MN/m ² ** (+120,500 psi)	+2050 MN/m ² (+297,300 psi)
Maximum Shear Stress	207 MN/m ² (30,000 psi)	531 MN/m ² ** (77,000 psi)	0
Effective Stress	404 MN/m ² (58,600 psi)	515 MN/m ² ** (74,700 psi)	1305 MN/m ² (189,320 psi)

* This is the maximum stress in the fuel, if it can be assumed that the unreality of the F39 model only affects the results for the stress map in the peripheral position of the fuel.

** Actual stress will be less due to fuel-cladding slip.

Fig. 3-2. UO₂ Fuel Press

The center pin is used with the die to form a central hole in the fuel and reduce subsequent machining operations. The center pin is made more easily removable by hardening it so that it deforms elastically less than the fuel. Fuel "spring back" releases pressure on the pin when the die segments are knocked away.

3.1.2.2 TFE 2LF2

The proposed design for TFE 2LF2 differs from that of 2LF1. The stem assembly of 2LF1 is essentially a modified F-series design; the stem assembly of 2LF2 has been redesigned completely to reduce its cost and to reduce the amount of heater power necessary to control the cesium reservoir temperature. The insulator seal has also been redesigned for lower cost and the lead adaptor trilayer section has been eliminated. Figure 3-3 illustrates all of these changes.

In order to simplify the cesium reservoir design and eliminate as much machining as possible, the reservoir is made of a piece of niobium tubing which is pinched off and welded at its upper end after cesium loading. The cesium vapor transfer passage is a tantalum tube pinched off at its upper end. This tube is welded to the reservoir bottom plug and the resulting assembly is welded into the reservoir tube. When the reservoir assembly is EB welded to the manifold block, the lower reservoir wall forms a vacuum wall insulator to keep the transfer tube hot and prevent cesium condensation inside it. The reservoir assembly is designed to maintain the cesium liquid-vapor interface as the coldest surface in the TFE.

Preliminary calculations indicated that reservoir heater power consumption will be about 25 watts. The total heat leakage into the stem assembly from the TFE will be about 103 watts, (about double the leakage of the short stem F-series TFEs) with 17 watts of this total going up the reservoir stem and 86 watts up the positive current lead. Significant reduction in total leakage can still be made. Our means for doing this would be reducing the positive current lead thickness.

Fig. 3-3. Stem Assembly

The fission product port is heated by conduction from the manifold block through the helium gas. The port and fission product exhaust tube are configured so that the port entrance temperature is 50 - 100°C hotter than the surface of the liquid cesium, while the port exit is at least as hot as this surface. The design objective is to prevent blockage of the port by condensing volatile fission products on the inside surface of the section of fission product tube upstream of the port and to keep cesium from condensing inside the port itself.

The positive current lead, no longer integral with the reservoir as in the F-series TFE, has about 40% of the electrical resistance of the F-series lead over its length with the dimensions shown on Figure 3-3. The temperature of the Cu/Nb joint is about 600°K, with 86 watts flowing into the positive current lead.

The 2LF2 insulator seal was redesigned to reduce cost and to keep the stress in the flexing member below yield. Lowering seal cost was accomplished by replacing the machined convolution with a commercially fabricated bellows and by using a simply configured brazed seal. This concept is illustrated in Figure 3-4. A savings of about \$1,000,000 over the standard F-series seal for 1000 units was estimated, based on a bellows unit cost of \$75,00. The low bid for the bellows manufacture was \$18/unit for lots of 100, indicating that the savings may be even larger. Owing to project termination, further investigation of the low bidder's understanding of bellows fabrication with niobium was not initiated.

3.1.3 Modification and Use of BRITL Computer Code to Analyze Thermionic Emitter Distortion

To expand and update previous work on fuel-clad interaction and clad distortion (Reference 4), a one-dimensional (radial) analysis of TFE fuel and emitter was performed including the following phenomena:

Fig. 3-4. 2LF2 Insulator Seal

Fuel: Restructuring due to vaporization and condensation and surface diffusion, volume swelling, and both thermal and radiation induced creep.

Clad: Volume swelling and elastic-plastic-creep deformation due to fuel contact pressure.

The calculations were performed with the help of the BRITL computer program (Reference 5), which had been developed to analyze UO_2 fuel pins in the gas-cooled fast reactor. The program was modified to incorporate the latest creep equation for tungsten and a fuel swelling model believed to be more accurate in the temperature range of interest for thermionic emitters.

For fuel swelling, the model proposed by Warner and Nichols (Reference 6) was used. Since the burnup levels were less than 4×10^{20} fissions/cm³, only the initial swelling rate, which is independent of burnup, was used. The dependence of swelling rate on temperature was calculated by interpolation between the initial slopes of the five curves in Reference 6.

The clad creep model used was

$$\dot{\epsilon} = 1.933 \times 10^{22} \sigma^{(-13.071 + 1.183 \ln \sigma)} e^{(-120500/1.986 T)},$$

where $\dot{\epsilon}$ = creep rate (in/in/hr),

σ = equivalent stress (psi), and

T = temperature ($^{\circ}\text{K}$).

This model is the best fit of ORNL and TRW data and the results of the biaxial creep test (Reference 2). Since the model deviated considerably at stresses under 400 psi, the creep rate below 400 psi was assumed linearly related to stress, equal to zero at zero stress.

Initial calculations were performed for a reference case identical in dimensions, fission power distribution, and clad temperature to those estimated for TFE 1Fl. The deformation rate calculated under these conditions was about 1% of that measured during irradiation of 1Fl. It does not appear that such a large discrepancy can be explained by errors in the model or in estimating the operating conditions. The most likely explanation is that small thermal cycles ($\pm 10^{\circ}\text{K}$) in the course of testing increase hoop stresses in the clad periodically. Because of the highly non-linear dependence of creep strain on stress, even short periods of time at high stress levels can produce a distortion of the clad very much larger than that which would result from the continuous slow swelling of the fuel due to fission gas.

To evaluate the effect of thermal cycles, the swelling rate of the fuel was increased by factors of 100 and 1000. These rates of fuel swelling correspond to rates of temperature rise of 7 and 70°K per hour.* The results of these calculations are given in Table 3-3.

While these results are not rigorous (they are based on equilibrium swelling and deformation rates and do not include the build-up in stress levels when the temperature begins to rise), they do indicate the potential for a large deformation of the emitter due to a repetition of small thermal cycles. The effects of these thermal cycles are expected to be cumulative, since the fissioning fuel cannot support significant tensile stress when the temperature is reduced without forming voids. While the fuel temperature is near the lower extreme, the voids will be closed by vaporization and condensation and surface diffusion so that dense fuel will be present to stretch the clad when the temperature is again raised.

*Since the fuel has a higher thermal expansion coefficient than the clad, there is a net growth in the dimensions of the fuel relative to those of the cladding when the temperature is increased. In terms of stress and deformation of the cladding, the effects of fuel growth from differential expansion are expected to be indistinguishable from the effects of the same growth due to swelling.

TABLE 3-3

EFFECTS OF THERMAL CYCLES

Rate of temperature rise ($^{\circ}\text{K/hr}$)	7	70
Emitter deformation rate ($\Delta D/D/\text{hr}$)	1.0×10^{-4}	9.1×10^{-3}
Time to rise 10°K	1.4 hr	8.5 min
Relative deformation per 10° cycle ($\Delta D/D$)	1.4×10^{-4}	1.3×10^{-3}
Frequency with which cycles would have to occur to deform clad as observed in 1F1 (days)	4	32

Several cases were run with the BRITL program varying clad thickness, emitter temperature, distribution of power density in the fuel, and the diameter of the central hole. They are compared with the reference case, TFE 1F1, in Table 3-4. The parameters of 1F1 were 1 mm clad thickness, 1780°K emitter temperature, a power distribution peaked at the outer edge, and a central hole diameter of 3.05 mm. While the distortion predicted for 1F1 is only 1% of that observed, the relative effects may be indicative of the importance of the parameters varied.

The effect of thickening the cladding is generally as expected although the magnitude of the reduction in deformation is somewhat larger than observed in 6F3. The results emphasize the important effect of emitter temperature on distortion. Modifying the power density through the fuel to a uniform profile (simulating fast flux operation) resulted in no change in clad deformation. It had been expected that the reduction in fission rate next to the clad would reduce deformation somewhat.

The unexpected increase in clad deformation from increasing the central hole diameter may possibly be due to the difference in temperature sensitivity between fuel swelling and creep models. As the hole diameter increases the fuel mean temperature decreases (for constant heat generation). This lowers both the creep and swelling rates. If the creep rate decreases significantly while the swelling rate does not, the clad will experience a larger contact pressure causing increased deformation.

3.1.4 Laboratory F-Cell Design (E. J. Steeger)

The "Laboratory F-Cell" is an F-series thermionic diode with its emitter heated by an electron bombardment heater. The purpose of the cell is to investigate the thermionic performance of the F-series design

TABLE 3-4

EFFECTS OF EMITTER PARAMETERS ON DISTORTION

<u>Parameter</u>	<u>Change In Emitter Distortion</u>
Reference case (1F1) 2 mm clad	-57%
1680°K emitter temperature	-88%
1880°K emitter temperature	+162%
Uniform power density	0%
Increased central hole dia to 14.2 mm	+20%

in a radiation-free environment and without the complicating effects of fuel swelling, fission production evolution, etc. The cell was designed to operate over a wide range of temperature by controlling collector temperature with a heater and cooled niobium collector heat sink.

The GGA thermal analysis computer program, TAC2D, was used to determine the thermal performance and the computer program TACTIC was used to determine the performance with electron cooling. The axial temperature gradient along the cell was varied to simulate the gradient along a fueled cell assembly by introducing a stepped gas gap between the collector and the collector heat sink.

A thermal stress analysis was made with particular emphasis on the end closures of the collector heat sink to allow for thermal expansion between the collector and heat sink without over-stressing the closures.

3.1.5 Evaluation of Work

Efforts to reduce the cost of TFEs through better tooling (UO_2 fuel press) and improved design (TFE 2LF2) are important to increase the applicability of thermionic fuel elements. Cost is a major factor for both aerospace and undersea applications. Significant progress in this area was made during the last four months.

The work in fuel-clad modeling indicated that thermal ratcheting coupled with fuel swelling may be more significant than fuel swelling alone in causing distortion of UO_2 fueled emitters. Further work in modeling the thermal cycles typical of those observed during testing is necessary to evaluate this mechanism.

3.2 TFE FABRICATION (H. Horner, J. Kay)

Thermionic fuel elements which incorporate the technology required to support long-life testing in TRIGA are fabricated under this task. The fuel elements are prototypic of those planned for use in thermionic reactor applications.

The technical approach used applies developed technology and technology being developed under Task 3.3 to the fabrication of thermionic fuel elements incorporating one, two, or six thermionic converters assembled in series. The F-series converter is the unit cell for these fuel elements and is produced at a constant rate. Fuel elements are completed as the cells become available. Materials and process specifications are prepared within this task.

Prior work in fuel element fabrication resulted in completion of 50 F-series cells made into 12 F-series TFEs. Three two-celled E-series fuel elements were also fabricated. The technology used for fabrication of F-series and E-series is an extension of technology used earlier to fabricate C-series converters for the unit cell reactor concept and single cell test converters, called Mark VI cells, that were built and tested to demonstrate the feasibility of long-term operation in a reactor environment. This work is summarized in References 7, 8, and 9. The total cells and TFEs fabricated are summarized in Tables 3-5, 3-6, and 3-7.

3.2.1 Recent Fabrication Accomplishments

Since the last thirdly reporting period, TFEs 6F6, 1F5 and 2LF1 were completed bringing the total number of F-series TFEs fabricated under this contract to 15. The sixteenth TFE, 6F7, was assembled into a sheath tube and readied for final sheath tube bonding, degassing and cesium loading.

TABLE 3-5

Cell and TFE Fabrication Summary

Cells	F1	F2	F3	F4	F5	F6	F7	F8	F9	F10	F11	F12	F13	F14	F15	F16	F17	F18	F19	F20	F21	F22	F23	F24	F25	F26	F27									
Major Components																																				
Emitter	0	0	0	0	0		0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0									
Collector	0	0	0	0	0		0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0									
Fuel	0	0	0	0	0		0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0									
Insulator Seal	0	0	0	0	0		0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0									
Assemblies																																				
Emitter Fuel Loading	0	0	0	0	0		0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0									
Cell Assembly	0	0	0	0	0		0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0									
TFEs																																				
	1F1				6F1				1F2				6F2				1F3				2F1				6F3				2F2				2F3			
Cell(s)	0				0				0				0				0				0				0				0							
Sheath Tube	0				0				0				0				0				0				0				0							
Cesium Reservoir Assy.	0				0				0				0				0				0				0				0							
Stem Sub-Assembly	0				0				0				0				0				0				0				0							
TFE Assembly																																				
TFE Assembly	0				0				0				0				0				0				0				0							
Sheath Tube Bonding	0				0				0				0				0				0				0				0							
Outgassing																																				
Final Processing	0				0				0				0				0				0				0				0							

0 Completed prior to this reporting period

TABLE 3-6

Cell and TFE Fabrication Summary

Cells	F28	F29	F30	F31	F32	F33	F34	F35	F36	F37	F38	F39	F40	F41	F42	F43	F44	F45	F46	F47	F48	F49	F50	F51	F52	F53		
Major Components																												
Emitter	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	*	*	*		
Collector	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	*	*	0		
Fuel	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0		
Insulator Seal	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0		
Assemblies																												
Emitter Fuel Loading	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	*	*	*	*		
Cell Assembly	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	*	*	*	*	*	*		
TFEs																												
	6F4					1F4					6F5					1F5					6F6					6F7		
Cell(s)	0					0					0					0					0					*		
Sheath Tube	0					0					0					0					*					*		
Cesium Reservoir Assy.	0					0					0					0					*					*		
Stem Sub-Assembly	0					0					*					0					0					*		
TFE Assembly	0					0					*					0					*					*		
Sheath Tube Bonding	0					0					*					*					*					†		
Outgassing																												
Final Processing	0					0					*					*					*					†		

0 Completed prior to this reporting period

* Completed during this reporting period

† On hold for disposition at time of
reporting - work terminated.

TABLE 3-7

Cell and TFE Fabrication Summary

Cells	F54	F55	F56	F57	F58	F59	F60	F61	F62	F63	F64	F65	F66	F67	F68	F69	F70	F71	F72
Major Components																			
Emitter	*	*	-	*	-	-	*	*	-	-	-	-	-	-	-	-	-	-	-
Collector	0	0	*	*	*	*	*	-	-	-	-	-	-	-	-	-	-	-	-
Fuel	*	*	*	None															
Insulator Seal	*	*	*	*	*	*	*	*	*	*	*	-	-	-	-	-	-	-	-
Assemblies																			
Emitter Fuel Loading	*	*																	
Cell Assembly	*	*																	
TFEs																			
	2LF1	1F6	1F-Lab	Work on 2F4, 1F7 & 6F8 Terminated															
Cell(s)	*	*	-	-															
Sheath Tube	None	-	-																
Cesium Reservoir Assy.	*		†	-															
Stem Sub-Assembly	*		†	None															
TFE Assembly	*		†	†															
Sheath Tube Bonding	None		†	†															
Outgassing	*		†	†															
Final Processing	*		†	†															

- In work at time report was prepared
- 0 Completed prior to this reporting period
- * Completed during this reporting period
- † On hold for disposition at time of reporting - work terminated.

Four emitters were loaded with UO_2 fuel forms of differing geometries to satisfy requirements for fuel-clad capsule FC-3. Work on emitters for fuel-clad capsule FC-4 was initiated.

An emitter and cell components were prepared for the F-series laboratory cell. All components were machined except for finish machining of heater wire grooves in the collector, heat sink body. All components for this device were boxed separately so that easy continuation of work would be possible.

Ceramic-to-metal seals were prepared for TFE 2LF2 and the emitters were approximately 80% completed.

Emitter fabrication for NASA clad-capsule CC-1 and for NASA Exhibit B, D and D/2 sized capsules was initiated by preparation of molybdenum mandrels required for deposition of tungsten. Twenty mandrels were fabricated and processed, and deposits were prepared for six D sized fuel cups.

During this work period, eight emitters for F-series cells were completed bringing the total F-series emitter fabrication during this program to 58 excluding the four fabricated for fuel-clad capsule FC-3. An additional eighteen F-series emitters were in work at various stages at the time of program cancellation.

Seven trilayers were completed during this period and an additional eight were in various stages of work at program cancellation.

Eleven ceramic-to-metal seals were completed for device application. Of these, four were of the configuration required for the LF-series TFE. An additional batch of eight seals was processed to the point of readiness for assembly brazing.

Program close-out work in fabrication (other than the technical work described above) involved the disposition of TFE components and material,

disposition of records (history packages), disposition of control paper work, and disposition of AEC Equipment.

3.2.2 Evaluation of Work

Efforts in thermionic fuel element fabrication over the past several years have resulted in the development of a controlled system capable of repetitive fabrication of TFEs of the F-series design with complete end item documentation. This controlled system is expandable to handle the numbers of fuel elements required for a thermionic reactor experiment and other thermionic reactor fuel element fabrication tasks. The features of this controlled system are described in the Thermionic Reactor Quality Assurance Plan, Appendix B, and the Q. A. Plan provides a time period for implementation of Q.A. functions as a Thermionic Reactor Experiment program would evolve into test and ground prototype reactor construction.

During the past year, several component fabrication areas that caused schedule and cost problems were subjected to close scrutiny and process or equipment changes were implemented or initiated. Some of these efforts were related to development of components of modified design for the LF-series TFE and are discussed in Section 3.3, TFE Fabrication Development. Others are discussed below.

A severe trilayer contamination problem caused levels of oxygen in niobium components of the trilayers to be in the range of 300 to 600 ppm after gas pressure bonding. Yields of only 50% of acceptable parts were suffered because of this problem. Emphasis was placed upon protection of the components during gas pressure bonding and upon an improved plasma spraying technique involving automatic part rotation and gun traverse motions and use of water cooling inside the sprayed substrate. Substrates were increased in length by a factor of 2.5 to improve spraying and gas pressure bonding economies. Swaging of a niobium sheath over the sprayed

trilayer substrate prior to gas pressure bonding was eliminated (to avoid gas entrapment) and replaced by an assembly technique using ring inserts for welding at the ends of the trilayer blank. These changes in processing were not fully evaluated; however, the yield of trilayers had increased to about 75% acceptable for TFE use at the time of contract termination.

Maintaining schedule on chemical vapor deposition of tungsten emitter blanks (H_2 reduction of WF_6) has been demanding with a single emitter deposition rig. A CVD tungsten deposition rig capable of producing three emitters in one deposition run was made fully operational during the year. The production runs on the rig resulted in components containing 3-10 ppm flourine and deposition efficiency increased from approximately 35% to approximately 60% utilization of WF_6 . Approximately one-half as many manhours were used for each billet with the three-emitter production rig as with a single emitter deposition rig.

Schedule delays and high unit cost problems were eased for ceramic-to-metal seal fabrication by installation of seal production equipment in the TFE fabrication facility. Major pieces of equipment consisted of a tungsten mesh element hydrogen metallizing furnace capable of metallizing batches of 20 ceramic rings in wet hydrogen at $1850^{\circ}C$, a $1200^{\circ}C$ air firing furnace facility for metallizing preparation, application and seal mechanical assembly, and a seal brazing furnace. The brazing furnace was capable of automatically programming batches of six seals through the complex brazing cycle (Ref.10) required for vanadium-niobium seal brazing at $850^{\circ}C$ without human intervention or control. Brazing cycles were completely reproducible and two brazing cycles could be made per shift. Seal production capability per manhour for brazing and metallizing operations was increased by a factor of about 10 by installation of the equipment and the possible production rate was increased by factors of from 12 to 20.

A controlled plating system designed for the specific geometry of F-series sheath tubes was placed into service during the year. The facility has provisions for repetitive nickel plating of sheath tubes (using a process covered by a written specification) so that a precise quantity of nickel can be applied to the tube. The facility and procedure have reduced plating rejections and increased plating quality with respect to adherence, thickness, appearance and thickness uniformity.

Efforts in the process modifications described had reduced the work content of affected operations and increased product quality and reproducibility. Process specifications for seal fabrication, sheath tube nickel plating and tungsten fabrication using the three-emitter rig are a part of the records prepared for storage as part of the AEC contract termination.

Fabrication of TFE 2LF1 has demonstrated the effectiveness of the low cost features of the design, and the fabricability of this TFE concept.

3.3 TFE Fabrication Development (J. Grebetz)

The objective of this task is to develop the fabrication technology required to construct thermionic fuel elements. The approach is to apply the most recent advances in manufacturing technology in a manner that will lead to the efficient and economical fabrication of thermionic devices, while maintaining their quality and reliability.

Past efforts in this task have included the development of all fabrication techniques and processes and the development of quality control procedures necessary for fabrication of thermionic fuel elements. The work has included: 1) development of thermionic fuel element assembly techniques by well characterized electron beam welding procedures that specified welding fixtures and parameters; 2) experimental characterization of component abilities to withstand the TFE environment in thermal cycling and thermal stability testing; 3) development of machining procedures for TFE components including tungsten emitters, ceramic-to-metal seals, trilayers and thin-walled, close-tolerance sheath tubing; 4) ultrasonic testing procedures to evaluate sheath tube-to-trilayer and trilayer ceramic-to-metal bonding; and 5) TFE final assembly and processing techniques including evaluation methods to ensure that final processing does not degrade trilayer ceramic-to-metal bonding.

Work during the current reporting period included final assembly of 2LF1, refinement of a low cost brazing process for ceramic-to-metal seals, and completion of a parametric study of the tungsten/tantalum diffusion bonding process. Work on 2LF1 is covered in a general discussion of low cost component development (3.3.1).

3.3.1 Low Cost Component Development

Techniques for the assembly of the low cost F-series TFEs were defined during the assembly of TFE 2LF1. Work performed this period included definition of the process for preparing low cost copper-nickel brazed ceramic-to-

metal seals, definition of welding parameters and fixtures for "slip-over ring" welding, and TFE final assembly and degassing procedures.

Low Cost Ceramic-to-Metal Seals. Seals were prepared by brazing simple grooved rings to metallized ceramic rings of rectangular cross-section using copper-10wt-% nickel braze foil. The configuration for brazing is described in Ref. 11. Warpage of the ceramic rings during metallizing caused braze gaps greater than 0.007 inch in order to assemble components for initial brazing experiments. Thermal cycling of these components with large gaps after brazing and finish machining caused leakage through ceramic-metal interfaces. Use of lower metallizing temperatures (a reduction of approximately 300°C) and better metallizing fixturing reduced the braze gaps to approximately .001 inch and yielded seals capable of withstanding 5 cycles to 800°C without loss of vacuum or structural integrity. A thermal cycling experiment was performed to determine if the stress lowering convolution machined integrally with the low cost seal would be subject to low cycle fatigue failure as indicated by analytical studies. Fifteen thermal cycles to 850°C were performed upon a convolution operating in a "real life" cyclic stress environment without leakage failure. Eight ceramic-to-metal seals were prepared for 2LF1 and 2LF2 applications.

Slip Over Ring Electron Beam Welding Parameters. Slip ring welding parameters were defined using samples that mocked up the trilayer of the LF series TFE and the thin wall (.010 inch thick) CB 1% Zr slip over ring. Welding was accomplished using a niobium holding fixture to compress the slip over ring uniformly onto the trilayer. Weld parameters were devised that did not affect the metallurgical integrity of the niobium-alumina bond.

TFE 2LF1. Final assembly of 2LF1 was accomplished by a straight-forward application of welding parameters and techniques devised with experimental samples. Assembly consisted of welding units consisting of a lower trilayer and an upper trilayer with an emitter, slip over ring and an insulator sandwiched in between. The weld fixture needed no fixturing other than standard collets to hold the 1.314 inch O.D. trilayers in proper concentric relative position and alignment. Five welds were made for each setup compared to the

one or two welds made for each setup for F series TFEs.

Low Cost Emitter Preparation using the "double diffusion bonding" technique described in Ref. 11 . By using a flat surface rather than one with a tapered joint, a reduction of a factor of six in lapping time was realized in joint preparation and bonding time was cut by one-half because two bonds were prepared during the same bonding run. Work performed on tungsten-tantalum diffusion bonding parameters and reported in Section 2.4 showed that no differences in joint integrity should be expected by employing this new process.

3.3.2 Evaluation of Results

The new development processes refined during the current reporting period proved successful in providing low cost fabrication techniques for the manufacture of thermionic fuel elements. The LF series seal fabrication was simplified, compared to F series, by reducing the complexity of the ceramic-metal joint and by limiting accurate machining to a past brazing operation. The number of setup operations required for TFE assembly was reduced by a factor of three for final welding of the LF TFE compared to the F series TFE.

3.4 INSTRUMENTATION AND ENCAPSULATION (D. Schwarzer)

The objective of this task is to instrument the TFE with thermocouples, voltage probes, and the high current bus bars. The TFE is then outgassed under high vacuum and encapsulated for testing in the TTR core.

Pre-assembly of the primary containment feedthrough seal and the secondary containment was initiated. The pre-assembly of the required instrumentation through the primary and secondary containment seals permits these units to be assembled and stored, reducing the elapsed time from receipt of the TFE until it is ready for testing in core. In conjunction with the pre-assembly, modified outgassing and encapsulation methods were developed. These new methods were initiated to reduce set up times while maintaining high quality.

During this report period, the instrumentation and encapsulation of TFE 6F5 and capsule FC-3 were completed. Instrumentation for TFE 6F6 was initiated.

3.5 IN-CORE TESTING (G. O. Fitzpatrick, M. K. Yates, D. E. Schwarzer)

Thermionic devices and fuel-clad capsules were tested in the ThermIonic Test Reactor (TITR) at Gulf General Atomic. The objectives of the in-core tests were to demonstrate improved lifetimes through upgrading components and to determine the combined effects on performance of fuel and fission product diffusion and the nuclear environment. Testing included device startup, performance mapping, periodic logging of data, diagnostic measurements, neutron radiographs, and analysis and interpretation of the data obtained.

Thirty-eight thermionic devices were life tested in-core between 1962 and 1973. Table 3-8 summarizes the test hours, longest tests, and failure modes of these devices. The devices were fueled with either carbide or oxide fuel. Considerable progress was made in improving in-core converter lifetimes with the longest tests increasing from 1000-2000 hours in 1965 to 10,000 hours by 1970. The number of devices tested stayed at about the three or four per year level until mid 1971, when it was increased to 10-15 per year. This increase was allowed by the installation of a new core in TITR (coincident with the startup of the first full-length TFE, 6F1) which allowed 10-15 simultaneous tests. This increase in the number of tests resulted in an increase in the information available to the development program and shortened the feedback time. The test operational hours for the period from July 1971 to January 1973 are plotted versus year in Figure 3.5.

The lifetimes and performance of all the prototype F-series TFEs are given in Table 3-9. Table 3-10 summarizes the design and operating parameters and lifetimes of recent test devices and gives the latest information from the neutron radiographs. Complete summaries of these tests as well as of all in-core and out-of-core tests since 1969 are given in Appendix A. The tables given in this Appendix summarize all pertinent operating parameters as well as the current neutron radiograph results.

TABLE 3-8

SUMMARY OF GGA THERMIONIC IN-PILE LIFE TESTS*

	Experimental Cells	Prototype Cells		Thermionic Fuel Elements			
	Mark VI	Mark VIIA	Mark VIIB	E Series (Two Cell)	F Series (One Cell)	F Series (Two Cell)	F Series (Six Cell)
Number of Tests	15	3	4	3	4	3	5
Total Test Hours	45449	4604	15915	27280	22152	16551	23972
Average Test Hours	3030	1535	3979	9093	5538	5517	4794
W/cm ²	6.3	5.2	3.7	4.5	3.8	4.0	2.6
Longest Test Hours	9754	2006	7881	12535	8560	8021	8062
W/cm ²	6.0	6.2	3.4	4.0	4.0	3.4	3.0
Failure Modes							
<u>Carbide Fuel</u>							
None	1	-	1	-	1	2	2
Short Circuit	1	-	-	-	2	-	1
Envelope Leak	10	2	2	1	-	-	1
<u>Oxide Fuel</u>							
None	1	-	-	1	-	-	1
Short Circuit	1	1	1	1	1	1	-
Envelope Leak	1	-	-	-	-	-	-

*As of January 22, 1973

Fig. 3-5. In-Core Testing from July 1971 to January 1973

TABLE 3-9
PROTOTYPE TFE TEST SUMMARY

<u>Device</u>	<u>Fuel</u>	<u>Average Relative Power</u>	<u>6F Lead Power (Watts)</u>	<u>Test Hours</u>
6F1	UC-ZrC	0.20	248	3890 ^a
6F2	UC-ZrC	0.60	516	7685
6F3	UO ₂	0.89	647	8062 ^b
6F4	UC-ZrC	0.94	664	2956 ^b
6F5	UC-ZrC	0.91	825	1379 ^b
2F1	UC-ZrC	0.67	679 ^c	8021
2F2	UC-ZrC	0.92	895 ^c	5574 ^b
2F3	UO ₂	0.98	894 ^c	2956 ^b
1F1	UO ₂	1.11	937 ^c	8560
1F2	UC	1.05	852 ^c	5618 ^d
1F3	UC	0.49	426 ^c	3378
1F4	UC-ZrC	0.78	821 ^c	4928 ^b

^aTest operated with zero lead power until 6376 hours.

^bAs of 1/22/73.

^cCorrected to provide 6F lead power under equivalent operating conditions.

^dTest operated negative power until 6376 hours.

TABLE 3-10

IN-CORE TESTS (10/16/72-1/22/73)

Test	Electrodes	Emitter Area cm ²	Fuel	Hours	Emitter Temp. No. °K**	Electrode Power Density W/cm ² **	TFE Lead Power W**	Performance	Maximum Emitter Dia. Increase		Failure Mode
									Hours	mils/mm	
TFE 6F3	W(W110)/Nb	6x49.0	UO ₂	8062	1 1770 2 1820 3 1750 4 1740 5 1740 6 1770	3.0	647	Efficiency low due to gas in cesium vapor. Possible sheath shorting.	8062	7/0.17 9/0.24 10/0.25 7/0.18 11/0.27 8/0.20	*
TFE 6F4	W(110)/Nb	6x49.0	90UC-10ZrC -4W	2956	1 1660 2 1720 3 1720 4 1690 5 1690 6 1690	3.6	664	High initial relative power, falling primarily because of contamination in Cell 2. Stable as predicted at 2956 hours.	2956	<2/0.05 <2/0.05 <2/0.05 <2/0.05 <2/0.05 <2/0.05	*
TFE 6F5	W(110)/Nb	6x49.0	90UC-10ZrC -4W	1379	1 1780 2 1810 3 1790 4 1770 5 1780 6 1790	3.7	825	As predicted and stable	1379	4/0.11 <2/0.05 8/0.20 <2/0.05 3/0.07 5/0.14	*
TFE 2F1	W(110)/Nb	2x49.0	90UC-10ZrC -4W	8021	1 1850 2 1870	3.4	181	Poor: Electrode Contamination & fission gas in cesium space	8021	20/0.51 <2/0.05	Fission gas leak & elec. contam.
TFE 2F2	W(110)/Nb	2x49.0	90UC-10ZrC -4W	5574	1 ~1820 2 ~1820	4.4	277	Poor: Electrode Contamination & fission gas in cesium space after 4195 Hours	5574	<2/0.05 3/0.08	Fission gas leak
TFE 2F3	W(110)/Nb	2x49.0	UO ₂	2956	1750-1850	4.8	267	Low efficiency due to fission gas in cesium space	2956	<2/0.05 13/0.33	*
TFE 1F1	W(110)/Nb	49.0	UO ₂	8560	1780	4.0	139	As predicted & stable prior to partial interelectrode short	8560	14/0.37	Partial inter-elec. short
TFE 1F4	W(110)/Mo+O ₂	49.0	90UC-10ZrC -4W	4928	1880	4.3	114	Poor: Due to electrode contamination	4928	<2/0.05	*
TFE 2E2	W(110)/Nb	2x26.3	UO ₂	11084	1820	5.6	203	As predicted. Possible slow drop due to emitter swelling.	11084	9/0.22 18/0.46	*
Mark VIIA IC-D3	W(100)/Nb	26.0	UO ₂	0-281 281-1577	1820 1250	4.4 0	85 0	Operated open circuit.	1577	12/0.30	Interelectrode Short
Mark VI IC-I6	W(110)/Nb	11.6	UO ₂	6334	1820	4.9	56	As predicted & stable. Emitter temp. being cycled.	6334	7/0.17	*
FC-3	W(110)/Nb	49.0	UO ₂	1379	1 1800 2 1820 3 1750 4 1750	-	-	Fuel-clad capsule test.	1379	<3/0.08 <2/0.05 2/0.05 4/0.10	*

*Operable 1/22/73; **Average Over Test Life

Twelve devices were tested in the reactor during this reporting period. They include:

- 3 Full-length TFEs (six cell), 6F3, 6F4, and 6F5
- 4 Two-cell TFEs, 2F1, 2F2, 2F3, and 2E2
- 1 One-cell TFE, 1F4
- 1 Prototype cell, Mark VIIA IC-D3
- 1 Experimental cell, Mark VI IC-I6
- 1 Fuel-Clad Capsule, FC-3

Tests of two new devices, TFE 6F5 and the fuel-clad capsule FC-3, were initiated during this reporting period, and operating and diagnostic correlations were derived for each. Two tests were terminated, TFE 2F1 and Mark VIIA IC-D3. Nine devices were on test prior to the shutdown on January 22, 1973; all eight thermionic devices were producing power at the electrodes.

The test hours achieved with 6F TFEs include 8062 hours on TFE 6F3 at an average power output of 647 watts, 2956 hours on TFE 6F4 at an average power output of 664 watts, and 1379 hours on TFE 6F5 at an average power output of 825 watts. Tests of partial-length TFEs reached 11084 hours on TFE 2E2, 4928 hours on TFE 1F4 with an average 6F equivalent lead power of 821 watts, 8021 hours on TFE 2F1 with an average equivalent 6F lead power of 679 watts, 5574 hours on TFE 2F2 with an average equivalent 6F lead power of 895 watts, and 2956 hours on TFE 2F3 with an average equivalent 6F lead power of 894 watts.

Off-design testing of two devices was conducted during this period. The emitter temperature of IC-I6 was raised from a normal temperature of 1800°K to a value of 1900°K for a period of ~5 hours once each week in order to ascertain the effect of this type of cycle (thermal ratching) on clad swelling. The result was swelling rate increase equivalent to 1/2 mil (0.12 mm) per cycle. IC-D3 was taken open circuit at 280 hours. An emitter temperature of 2300°K resulted. At 330 hours the emitter shorted to the collector, resulting in an emitter temperature drop to 1200°K. Testing was terminated at 1577 hours.

Figures 3-6 and 3-7 show the relative power histories of all current tests. Details of each test history are included in Appendix A.

Fig. 3-6. Performance History of Oxide Fueled Devices

Fig. 3-7. Performance History of Carbide Fueled Devices

Included in this section is a brief summary of results which have been achieved in the areas of performance, emitter dimensional stability, and parasitic discharges. Where applicable, oxide and carbide fueled tests are discussed separately. A summary of in-pile electrical discharge results is also included.

3.5.1 Thermionic Performance

There are two important aspects of thermionic performance which can be examined. These are the initial levels of performance and performance stability of a device.

In general, oxide fueled tests have exhibited consistently good initial thermionic performance and have maintained this performance throughout their tests. Carbide fueled tests have tended to have lower and less consistent initial performance and have exhibited less stable performance during testing. The longest UO_2 fueled tests, IC-I4 (9754 hours), 2E2 (11084 hours), and 2E1 (12535 hours) had stable performance within $\pm 10\%$ (in the case of 2E1 until a short at 7035 hours). The performance of the longest UC-ZrC fueled test, IC-C11 (7881 hours), decreased $\sim 15\%$ over its lifetime. This difference in performance trends is ascribed to the different effects of the fuel constituents which diffuse through the tungsten cladding.

The two most recent carbide fueled tests, TFE 6F4 (2956 hours) and TFE 6F5 (1379 hours), incorporated design changes intended to reduce fuel operating temperature. Both devices performed stably near design levels until test termination.

3.5.1.1. Initial Performance

The initial performance level of a device is based on its electrode power output at the time of the initial performance mapping. The mapping was usually done one to two weeks after test startup when its performance had stabilized. The initial thermionic performance of most of the tests operated

to date is plotted against emitter temperature at a current density of 7 Amps/cm² in Figure 3-8 . The electrode materials, spacing, performance relative to CPOP, and method of determining emitter temperature are given in Table 3-11 . TFE 2F1 and 6F2 have not been included since they apparently had special initial condition problems which were not considered representative. In order to provide a better comparison, data acquired for each test is corrected to show the device's performance with its collector at its optimum temperature. Out-of-pile cell (OC-A4) mapping data was used to make this correction. No correction was made to account for differences in spacing and this factor should be considered when comparing individual tests.

The curve labeled CPOP is a calculated curve which is based on out-of-pile W(110)/Nb electrode data (Ref. 12). The CPOP results were those used in systems and reactor calculations. The most significant source of differences in performance in in-pile tests revealed by this data is the emitter material. Devices with W(WC16) emitters performed with a relative power 8% better than those with W(WF6) emitters. There was also an indication that oxide fueled devices performed ~ 4% better than carbide fueled devices, and cells with molybdenum collectors produced ~7% more power than those with niobium collectors. The variation in performance which cannot be attributed to these differences amounts to $\pm 4\%$, the equivalent of emitter temperature estimate errors of $\pm 25^\circ\text{K}$. This is consistent with the errors expected and indicates the initial performance of the devices is reproducible within measurement errors if certain devices are not included (TFE 2F1, 6F2).

3.5.1.2 Performance Stability

Changes in the performance of the devices tested may be attributed to one or a combination of the following factors:

1. Leaks between the cesium or fission gas spaces and the containment gas (envelope failures).

Fig. 3-8. Thermionic Performance Comparison

TABLE 3-11

DEVICES LISTED IN INITIAL
THERMIONIC PERFORMANCE COMPARISON

<u>Device</u>	<u>Emitter</u>	<u>Collector</u>	<u>Fuel</u>	<u>Spacing (Hot)</u>	<u>T_e Basis</u>	<u>P_R</u>
<u>IN-PILE DEVICES</u>						
J1	W(WF ₆)	Mo	Carbide	8 Mils	T.C.	.98
I4	W(WF ₆)	Mo	Oxide	8 Mils	OC-5	1.04
I5	W(WC1 ₆)	Nb	Carbide	9 Mils	J _{ig}	1.02
I6	W(WC1 ₆)	Nb	Oxide	8 Mils	J _{ig}	1.01
D1	W(WF ₆)	Nb	Carbide	8 Mils	φ _e	.90
C11	W(WF ₆)	Nb	Carbide	9-12 Mils	T.C. & J _{ig}	.82
C13	W(WF ₆)	Nb	Oxide	9 Mils	J _{ig}	.97
C14	W(WC1 ₆)	Nb	Carbide	10.5 Mils	J _{ig}	.98
2E1	W(WC1 ₆)	Nb	Oxide	10 Mils	J _{ig}	.96
2E2	W(WC1 ₆)	Nb	Oxide	10 Mils	J _{ig}	1.05
2E3	W(WC1 ₆)	Nb	Carbide	10 Mils	J _{ig}	.94
1F1	W(WC1 ₆)	Nb	Oxide	10 Mils	J _{ig}	1.06
1F2	W(WC1 ₆)	Nb	Carbide	10 Mils	J _{ig}	.94
1F4	W(WC1 ₆)	Mo + O ₂	Carbide	14 Mils	J _{ig}	.85
2F1	W(WC1 ₆)	Nb	Carbide	10 Mils	J _{ig}	~1.0
2F2	W(WC1 ₆)	Nb	Carbide	10 Mils	J _{ig}	.92
6F3	W(WC1 ₆)	Nb	Oxide	14 Mils	J _{ig}	.97
6F4	W(WC1 ₆)	Nb	Carbide	10 Mils	J _{ig}	.94
6F5	W(WC1 ₆)	Nb	Carbide	10 Mils	J _{ig}	.99
<u>OUT-OF-PILE DEVICES</u>						
OC-5	W(WF ₆)	Mo	-	8 Mils	T.C.	.97
EC-7	W(WF ₆)	Nb	-	9 Mils	T.C.	.80
LC-9	W(WC1 ₆)	Nb	-	9 Mils	T.C.	.98
LC-10	W(WC1 ₆)	Nb	Carbide	8 Mils	T.C.	.93
LC-11	W(WC1 ₆)	Nb	Carbide	8 Mils	T.C.	.93
OC-A4	W(WF ₆)	Nb	-	8 Mils	T.C.	.87
<u>COMPUTER PROGRAM</u>						
CPOP	<110>	Nb	-	10 Mils	-	

2. A leak between fission gas and cesium spaces resulting in fission gas and fission products in the interelectrode space.
3. Failure of the bonds in the collector-sheath tube structure resulting in high collector temperatures.
4. Contamination of the electrodes during fabrication.
5. Electrode contamination due to fuel component diffusion through the emitter clad.
6. Interelectrode spacing changes.

Envelope failures were a frequent problem in early Mark VI and Mark VII tests, accounting for fifteen out of twenty failures. They were almost always associated with a braze joint, and consequently, an all-welded design was used in the fabrication of TFEs. The new design solved the problem as indicated by the fact that only one out of eight failures, TFE 2E3, has been the result of an envelope leak.

Leaks between fission gas and cesium spaces were common, having occurred in TFEs 6F1, 6F2, 6F3, 2F2, and 2F3. The sources of the leaks were not ascertained in all cases, although emitter cracking is one observed source. The primary result of a leak of this type is a loss in thermal efficiency. Particularly in the case of oxide fueled devices, such as 6F3 and 2F3, thermionic performance is maintained. In the case of the carbide fueled devices, 6F2 and 2F2, there is evidence that electrode contamination may have been associated with such a gas leak, possibly due to fuel migration through clad cracks (discussed in more detail in a following section). In the oxide devices, where contamination associated with leaks has not been a problem, thermionic performance losses due to gas pressure buildup in the interelectrode space may be minimized by venting such gases to larger traps through larger ports. To prevent cesium loss, larger reservoirs may also be needed.

Poor bonds in the collector-sheath tube structure have been definitely identified as a performance change mechanism in TFE 6F1. An early style trilayer employing a pure alumina layer between niobium sleeves was used in that device. Later TFEs incorporated a graded niobium/alumina cermet trilayer structure, which appears to have solved the problem.

Contamination of electrode surfaces during fabrication is a problem that may have been present in two forms. Most TFE tests have initially performed at a higher level than predicted and have suffered an initial performance loss resulting in stabilization with performance near that predicted. The performance loss occurs over a period of one to several hundred hours. This improved performance has been attributed to the presence of oxygen sources within the TFE, possibly due to tungsten oxide residue on the emitter as the result of electropolishing the emitter. Tests of TFEs 1F2, 2F1, and 6F2 had low initial performance which could be improved by short periods of operation with high emitter temperature. This suggested the possibility that the electrodes may have been contaminated during the fabrication process. Work was conducted to ascertain the existence and extent of any such contamination. The possibility of contamination due to handling or the environment (oil, etc.) was eliminated. The possibility of metal spray contamination during the welding process was not eliminated.

Electrode contamination due to fuel component diffusion has been visible in the carbide fueled devices. Much effort has been concentrated on the solution of this problem and a generally satisfactory one has been found in the use of duplex tungsten clads and fuels with closely controlled stoichiometry. The most recent carbide tests, TFE 2F2 and TFE 1F4, indicate a new mechanism for fuel transport to the emitter surface may be important. Thermionic performance remained high and stable past 4,000 hours in the case of TFE 2F2 and 3500 hours in the case of TFE 1F4 but began to fall as the result of contamination after these points. Fuel sintering was observed prior to these times in each test, but swelling to full diameter occurred during the same period as the performance loss. This association between performance loss and fuel swelling confirms observations made with earlier F series carbide fueled tests (Ref. 2). Out-of-pile tests (Ref. 2) indicate the clad stresses resulting from fuel swelling promotes grain boundary void information, which would result

in an increased fuel transport rate. In order to prevent sintering, TFEs 6F4 and 6F5 were designed to minimize fuel temperatures, in the case of TFE 6F4 by back filling the fuel spaces with a heat conducting gas (40 torr argon) and in the case of TFE 6F5 by providing an interference fit between fuel and clad at operating temperatures. Some sintering has been observed in TFE 6F4 at 2956 hours indicating that gas over pressure may indeed be an effective means of preventing sintering.

Interelectrode spacing changes appear to have a minimal effect on thermionic performance prior to interelectrode shorting. A small loss in performance in TFE 2E2 (~10% at 11084 hours) was attributed to spacing changes and there are indications that a similar loss may have occurred in IC-I6. The primary problem associated with spacing changes is the attendant errors in ignition current based emitter temperature estimates. Further discussion of the emitter swelling problem is included below.

3.5.2 Emitter Dimensional Stability

3.5.2.1 Carbide Fueled Emitters

Fuel Behavior: Fuel sintering has been observed in almost all of the F series TFEs fueled with carbide fuel except in TFE 6F4 and in some emitters of TFE 6F5. During this reporting period:

1. The fuel in the 6F4 emitters showed no evidence of sintering as of 2956 hours.
2. The fuel in emitters 2 and 3 of 6F5 sintered ~1/2% by 1379 hours. No sintering was observed in the other emitters.
3. The fuel in 2F2 was sintered ~ 1.0% (on the diameter) at 4195 hours. It had returned to full diameter by 5574 hours and was swelling into the central cavity and stem area.
4. The fuel in 1F4 had sintered ~ 1% on the diameter at 3549 hours. At 4928 hours the top two pellets remained sintered ~ 1% but the bottom two had returned to full diameter.

The observed sintering may be due to an initial fuel-clad gap and the lack of any conducting gas in the fuel cavity (see Ref. 2). Under these conditions heat is transferred from fuel to clad primarily by radiation with a fuel-to-clad temperature differential as high as 500°K. The sintering history data on 6F2 (Appendix A) indicates that there is an enrichment effect on the sintering behavior. The fuel in the lowest enrichment emitters sintered the most and remained sintered the longest and that in the highest enrichment emitters sintered the least.

From the data on emitter dimensional stability which is discussed in the next section it appears that the sintering of the carbide fuel is detrimental to dimensional stability. Therefore, several approaches are being followed to reduce this problem. These include:

1. Changing the fuel-clad gap to insure an interference fit at temperature (adopted for 6F5 and later TFEs). The presence of sintering in two emitters in 6F5 at 1379 hours indicates that some sintering may still result.
2. Backfilling the fission gas space with a small amount of gas. This technique requires that the gas pressure be low enough to prevent the loss of a significant amount of power if the gas should leak into the TFE, but high enough for good thermal bonding of fuel and clad (adopted for 6F4 and 6F6). Forty torr of argon was used as a backfill gas in TFE 6F4. No sintering was observed as of 2956 hours, indicating this may be an effective technique.
3. Pre-sintering the fuel (planned for a fuel-clad capsule experiment).

Fuel-clad reaction layers were observed in most of the emitters of carbide fueled TFEs where the fuel enrichment is higher than 40%. In general these layers were present at the time of the first radiograph (1200-1300 hours) and did not change significantly after that time. Usually 0.13 mm (5 mils) of clad reacted. Reaction layers were generally not observed in emitters whose enrichment is less than or equal to 20% at times around 5000 hours (≤ 0.08 mm of clad reacted). This dependence of the reaction layers upon the fuel enrichment apparently results from the fact that the fission rate near the clad increases significantly as the enrichment goes up. In an actual

thermionic reactor, the fission power density is uniform over the volume of the fuel pellet. Comparable fission distributions can be obtained in TITR by using low enrichments. As a result of this fact, and the fact that the reaction layer does not include more than about 0.13 mm of clad, this reaction layer is not felt to be a major problem in the TFE development program.

Emitter Dimensional Changes: A comparison of all of the early GGA data on carbide fueled emitters is given in Ref. 2. Data from most of the F series tests is given in Fig. 3-9 as the maximum diametral increase plotted against burnup. The average emitter temperature values refer to the average values at the time of the last set of neutron radiographs. The dimensional changes which have been observed in most of the long-term carbide fueled tests (6F1, 6F2, 2F1, and 1F2) appear to be associated with the return of the fuel pellets to full diameter following sintering. It is postulated that the as-built open porosity of the fuel disappears or becomes closed porosity during the sintering process. Later the fission gas pressure in the fuel builds up since the gas cannot be released, and the fuel begins to swell. The fuel is running at high temperature during this period since it is not generally in contact with the clad. This results in a rapid fuel swelling rate (rates of 1-2% per 1000 hours swelling on the diameter have been observed), and the stresses on the cladding are high enough to cause clad swelling and rupture. Internal gas pressures in the emitters due to plugging of the fission gas vent tubes may be contributing to the swelling problem.

Work is proceeding to solve both the fuel sintering and gas pressure problems. The fuel sintering work has been discussed under Fuel Behavior and is directed at preventing the fuel from sintering into a dense body with little open porosity. Laboratory tests have shown (Ref. 11) that baffles in the fission gas space can trap fission products which might plug the vent tubes or channels. This design change was incorporated in TFE 6F5.

The data from the last two 6F TFEs (TFE 6F4, 2956 hours; TFE 6F5, 1379 hours) appear encouraging. No dimensional changes

Fig. 3-9. Diametral Increase Versus Burnup

were observed in TFE 6F4 ($\leq 0.18\%$ on the diameter), and the dimensional changes observed in TFE 6F5 (from $\leq 0.18\%$ to 0.44%) are attributed to the initial fuel-clad interference fit. These data points were deleted from Fig. 3-9 because of the low values of burnup that were achieved. However, no fuel sintering was observed in TFE 6F4, and, therefore, high fission gas release and improved dimensional stability should result.

No fuel sintering was observed in four of the 6F5 emitters and the same conclusions can be drawn. However, further irradiation time is necessary to confirm these predictions. Two emitters in 6F5 contained fuel slotted in manners intended to relieve stress (Ref. 2). No clad swelling was observed in one of these and ≤ 2 mils (0.05 mm) in the other, although the other four emitters exhibited significant swelling resulting from the interference fit. Such stress relief slots may serve to prevent clad swelling after sintering has occurred.

3.5.2.2 Oxide Fueled Emitters

Fuel Behavior: The fuel in the oxide fueled TFEs has usually redeposited and a central cavity formed by the time of the first neutron radiograph (1000-2000 hours). After this time very little change in fuel geometry is seen (TFEs 2E1, 2E2, 1F1, 6F3). Cracks are sometimes observed in the fuel, but these are expected to be due to differential thermal expansion on cooling.

Emitter Dimensional Changes: All of the early GGA data on UO_2 fueled emitter dimensional stability was summarized in Ref. 2 . Data from most of the current tests is given in Fig. 3-10 as the maximum diametral increase in per cent plotted against burnup. The average emitter temperature values refer to the average values at the time of the last set of neutron radiographs.

The last neutron radiographs of TFE 2E2 were taken at 11084 hours. The difference in the swelling rates between the two emitters in 2E2 is ascribed to a $100^\circ K$ temperature difference between them. The observed changes are very similar as those observed in TFE 2E1 (Ref. 2), a TFE of the same design as 2E2. No shorting has been observed in this TFE even though the swelling of the bottom emitter has closed 72% of the initial gap (assuming circumferential uniformity).

Fig. 3-10. Diametral Increase Versus Burnup

The longest test to date of an F-series 1 mm clad emitter is that of 1F1, which was operated 8560 hours. Partial shorting occurred (a 25% loss in electrode power density) at the neutron radiography shutdown at 8228 hours, and the test was removed from core shortly thereafter. The partial short occurred when the gap was 56% closed, apparently because the swelling was eccentric. A plot of the diametral distortion as a function of time and axial location is shown in Fig. 3-11. The point of maximum swelling is at or somewhat below the midplane of the emitter, where the temperature is the highest.

The last set of neutron radiographs of TFE 6F3 were taken at 8062 hours. The swelling of the six emitters varied from 0.5 - 0.9% at that time. If the swelling remained uniform, partial shorting would not be expected before 20,000 hours for the most rapidly swelling emitter and before 40,000 hours for the least rapidly swelling emitter. The only quantity which varies significantly from emitter to emitter is the enrichment; however, plotting swelling rates versus enrichment shows no obvious correlation. Differences in emitter fluorine content or fuel stoichiometry are not significant in this TFE and, therefore, cannot be used to correlate the data. In earlier data it appeared that the relative position in the core might correlate the data. This no longer appears to be the case. Even though the calculated emitter temperatures cannot be used to correlate the data, it is possible that the data could be correlated by emitter temperature because of relative emitter-to-emitter temperature uncertainties ($\pm 50^{\circ}\text{K}$).

Plots of diametral distortion as a function of time and axial position are given in Fig. 3-12 for TFE 6F3. Figure 3-13 shows the axial profile of each emitter at the 8062 hour radiograph. No significant differences are noted from emitter to emitter or between these emitters and the TFE 1F1 emitter. This lack of difference from emitter to emitter in 6F3 is surprising because of the differences in the fuel shape inside the emitters (see Appendix A). The TITR flux shape effects the shape and the distribution of the isothermal central cavities in UO_2 fueled devices. This results in an increase in the amount of fuel in the lower half of the emitters in the bottom cells of the TFE and a decrease in the amount of fuel in the lower half of the emitters in the top cells of the TFE. Since the maximum

Fig. 3-11. Emitter Distortion in TFE 1F1 During the Life Test

Fig. 3-12. Emitter Distortion in TFE 6F3 Emitters During the Life Test

Fig. 3-13. Emitter Distortion in TFE 6F3 at 8062 Hours

swelling occurs in the lower half of the emitter, this increase in the amount of fuel present there could increase the swelling rate. This might be the reason for the increased swelling in emitter number five, but the pattern is not uniformly followed. It is possible that the problem could be resolved and a correlation determined if two dimensional flux, temperature, and stress profiles were calculated for this TFE.

The clad thickness appears to have a definite effect on swelling rate. The 1 mm clad emitter in 1F1 distorted at a rate between 1.5 to 2.5 times as fast as the 2 mm clad 6F3 emitters. The 1 mm clad emitter in 2F3 was swollen 1% at a burnup of 3.2×10^{19} f/cc whereas no swelling was detected on the 2 mm clad emitter ($\leq 0.18\%$), even though these emitters should have been operating at the same temperature.

3.5.3 Electrical Discharge Testing

In-pile electrical discharge tests have been conducted with the 1F TFEs 1, 2, 3, and 4. (See Ref. 2 and Appendix A for test details).

Testing with TFE 1F1 was concluded at 8560 hours. At that time the TFE was stably supporting a one ampere discharge current with an emitter to sheath potential difference of 9 volts. The sources of the discharge current were not known.

Testing of TFE 1F2 was conducted between 2120 and 5618 hours with an emitter to sheath tube potential difference of 5.0 volts. The resulting current grew slowly from 12 milliamps to ~ 330 milliamps. Following a shutdown at 5618 hours the emitter and sheath tube were found to be shorted, as were the emitter and collector. Although no correlation between the two shorts other than time has been found it is possible that the high collector temperatures which would be produced where the emitter touched the collector may have contributed to a trilayer short.

Discharge testing of TFE 1F3 was terminated after 551 hours of testing with an applied voltage of 5.0 volts, following a collector-sheath tube short. It should be noted that the cesium and fission gas spaces in this cell were deliberately connected.

TFE 1F4 stably maintained a current of 1 ma at 5 volts for 1507 hours, followed by 280 hours at 10 volts with stable current of 4 ma. At that time the polarity was reversed and the current rose slowly to 300 ma. At 2320 hours a bistable discharge mode developed with currents in the high current mode passing through the collector sheath insulator. Testing was terminated at 2400 hours.

The sheath insulator in TFE 1F1 performed well. It was of an early design and was not graded. The insulators in TFEs 1F2 and 1F3 were of the GGA graded cermet type. It was not clear in these tests whether the observed shorts

were through these insulators or through plasma-sprayed insulating coatings. As noted in Section 3.7 the graded sheath insulator tested in cesium out-of-pile performed well for 1921 hours, and the evidence indicates the current flow in that test may have been through the insulating coating. The peculiar switching mode behavior observed in TFE 1F4's sheath insulator, a TECO cermet, is similar to that observed in out-of-pile tests of GGA insulators which had been contaminated (Ref. 13). The nature of this behavior is not understood. It should be noted that the resistances of the two TECO trilayers tested out-of-pile quickly dropped to and stabilized at resistances near 0.2 Ω .

In summary, in-pile and out-of-pile tests have demonstrated the feasibility of the trilayer insulators and alumina insulating coatings at up to 10 volt levels for up to one year. There is evidence that breakdown can occur, particularly in cesium, however, and further work is needed to understand the mechanisms and prove the techniques.

3.5.4 Neutron Radiography

Work with the neutron radiography unit has shown that dysprosium foils provide superior resolution to indium foils, whereas indium foils show fuel cavities and cracks more clearly. The use of both during radiography is thus desirable. The work has also shown that:

1. For dysprosium pictures, foil order, backing shields and position in the radiation fixture has no significant effect on sharpness. Foil exposure time has a significant effect. An optimum exposure for the existing system was obtained.
2. For indium pictures foil order, cadmium backing, and exposure time have no significant effects. Indium backing is beneficial in thicknesses to 1/16 inch. Dysprosium also has a beneficial effect, but to a smaller degree.

A topical report (Ref. 14) has been written which summarizes the design and use of the neutron radiograph equipment and the techniques used in data acquisition and reduction.

3.5.5 Evaluation of Work

Two of the primary problem areas associated with TFE development are performance stability and emitter dimensional stability. The initial performance of most devices tested has agreed with predictions within the errors in estimates of the device's operating condition. Prior to interelectrode shorting the thermionic performance of oxide fueled devices has proven stable to within the measurement errors of $\pm 10\%$, in the case of TFE 2E2 reaching 11084 hours. Carbide fueled tests generally have proven less stable, a fact attributed to fuel component diffusion through the tungsten emitter clad. Much of the fuel diffusion problem appears to have been solved by control of fuel stoichiometry and cladding structure as demonstrated by TFE 2F2, which performed stably past 4000 hours. In that case a performance loss was experienced only after the carbide fuel had sintered, returned to full diameter, and had swelled and possibly cracked one emitter clad.

Present system designs require that the emitter dimensional strains do not lead to electrical shorting at burnups of $1.3 - 1.6 \times 10^{20}$ fissions/cc (20,000 hours, 1 mm clad thickness). The data from the oxide fueled test 1F1 (0.25 mm spacing) indicates that further improvements in dimensional stability are necessary to meet the 20,000 hour goal with 1 mm clad thickness. The data from 6F3 (0.36 mm spacing) indicates that the 20,000 hour goal can be met or exceeded, but the effect on the system design of using the thicker 2 mm clad with its lower fuel inventory must be considered. TFE 2F3 has confirmed the effectiveness of a 2 mm clad in reducing the swelling rate when compared to a 1 mm clad. Data acquired with IC-I6 demonstrated the fact that temperature cycles can induce clad swelling. Operation in a mode which minimizes such cycles may prove effective in reducing the swelling rate. The use of different fuel designs, such as those incorporated in the oxide fuel clad capsules FC-3 and FC-4, may also prove useful in reducing swelling.

The data from the carbide fueled tests (6F4, 6F5, 2F1, 2F2, 1F4) indicates that the 20,000 hour goal may be feasible if the fuel sintering problem can be solved. The results obtained with TFE 6F4 (which has a 40 torr argon fuel cover gas) are encouraging since no sintering, swelling or performance loss was observed in 2956 hours of testing.

3.6 TRIGA OPERATION (D. Schwarzer, G. Fitzpatrick, M. Yates)

The objective of this task was to supply a neutron environment for the testing of thermionic fuel elements and fuel-clad capsules and to provide a source for neutron radiography. A total of eleven devices were tested in TITR, nine of them simultaneously.

During the November 13-17 period, the reactor was shut down for neutron radiography of in-core tests, normal reactor maintenance, and thermionic instrumentation maintenance and upgrading. Also tests Mark VIIA IC-D3 and TFE 2F1 were removed from core and TFE 6F5 and capsule FC-3 installed during this period.

A reactor scram caused by a thermionic low current instrumentation trip on November 22 kept the reactor down until November 29. The long down period was necessary due to the holidays and the diagnostic investigation required to discover where the malfunction occurred. The malfunction was found to be an intermittent shorting of switches in the cross-bar section of the data acquisition system causing spurious signals. The cross-bar was replaced and the malfunctioning unit sent out for repair.

The reactor was shut down on December 7 for relocation in-core of TFEs 6F5, 2F2, and 2F3.

Due to cancellation of the thermionic program, the reactor was shut down on January 22, 1973. All nine in-core tests were neutron radiographed and disassembly of the instrumentation initiated.

The reactor was operated for a total of 1700 hours out of a possible 1982 hours giving a duty cycle of 86%. During the calendar year of 1972, the reactor was operated for 7983 hours out of a total of 8784 hours with a reactor duty cycle of 91%.

3.7 TFE DEVELOPMENT TEST SUPPORT

The objective of this task was to conduct parasitic electrical discharge tests and to provide test support as needed for TFE development.

The electrical discharge tests were designed to study possible parasitic losses within the TFE, particularly in the event of a cesium leak from the interelectrode to the fission gas spaces. The approach used in the investigation was to:

1. Determine what conditions are necessary to prevent electrical breakdown.
2. Determine any adverse effects of discharges on the outer sheath or the converters as a function of voltage and cesium vapor pressure.
3. Provide data, both through the literature and by experiment, on breakdown voltages and discharge current-voltage characteristics.

The work has shown that an unprotected TFE may be effectively shorted by an electrical discharge in the event of a cesium leak to the fission gas spaces; however, the use of alumina insulating coatings on exposed surfaces and alumina tubes in the fission gas vent holes will reduce electrical losses to an acceptable level. Tests of a TFE intercell region with such protection conducted at normal TFE operating sheath temperatures and cesium pressure showed no damage following sustained application of TFE open circuit voltages (14 volts) for over one hundred hours. A more detailed summary of the tests results are given in Ref. 11.

Three life test specimens were tested during this work period, LT-1A, and LT-2A, and LT-2B. LT-1A uses a GA graded cermet trilayer (Ref. 15). LT-2A and LT-2B use TECO cermet trilayers (Ref. 16).

Testing was conducted with the samples at 1075°K with a cesium pressure of 5 torr and an applied voltage of five volts. LT-2A and 2B were run with opposite polarities and LT-1A was operated with the insulating coating at a positive potential. The histories of these tests are shown in Figs. 3-14 and 3-15. .

The resistance of LT-2A and LT-2B fell during the first one hundred hours of testing, stabilizing near 0.2 ohms in each case. The IV characteristics of these devices were linear, indicating a resistive short. Both devices were operated with currents near 6-1/2 amperes until test termination at 1921 hours.

The resistance of LT-1A dropped slowly during the test, stabilizing near 40 Ω at 1400 hours. At 1627 hours the voltage was raised to 10 volts, resulting in an ignition and an accompanying resistance drop to approximately 15 ohms. It remained stable at this level until test termination at 1921 hours. A representative IV curve taken at the 1920 hours is shown in Fig. 3-16.

The post-operational appearance of the insulating coatings and trilayer insulators are shown in Figs. 3-17, 3-18, and 3-19. The chip in the insulating coating visible near the edges occurred during sample disassembly. No shorts were visible in the trilayer sections examined. One or two trilayer cracks were visible in each case, but these may have resulted from sample preparation. The sheath insulation in LT-2A and 2B was leak tight and LT-1A had a helium leak rate of 1.2×10^{-7} std cc/sec. A microprobe analysis of the sheath insulators, alumina insulating coatings, and a weld exposed to cesium revealed no cesium at or above the 1/2% level.

The three sample trilayers were divided into quarters and halves and their resistances measured.

The results were:

<u>LT-1A</u>	<u>LT-2A</u>	<u>LT-2B</u>
Half $10^7 \Omega$	Half 0.4 Ω	Half 0.6 Ω
Half $10^7 \Omega$	Quarter 14.4 Ω	Quarter 1.0 Ω
	Quarter $10^6 \Omega$	Quarter 0.4 Ω

Fig. 3-14. Cesium Discharge Lifetest Histories

Fig. 3-15. Cesium Discharge Lifetest Histories

Fig. 3-16. LT-1A Discharge IV Characteristics

Plasma Sprayed Alumina Insulating Coating

GGA Graded Insulation Topped by Insulating Coating (300X)

Fig. 3-17. Post-Operational Appearance of LT-1A

Plasma Sprayed Alumina Insulating Coating

TECO Cermet Insulator Topped by Insulating Coating
(300X)

Fig. 3-18. Post-Operational Appearance of LT-2A

Plasma Sprayed Alumina Insulating Coating

TECO Cermet Insulator Topped by Insulating Coating (300X)

Fig. 3-19. Post-Operational Appearance of LT-2B

These indicate the shorts observed in LT-2A and LT-2B were distributed to some degree. The resistance of LT-1A after testing was quite high. This, in conjunction with the form of the IV curve, suggests the discharge current observed in that case occurred through the alumina insulating coating.

The LT-1A test confirmed earlier tests which demonstrated the feasibility of graded sheath insulators and alumina insulating coatings to reduce discharge currents to acceptable levels. In conjunction with in-pile tests, they also suggest the need for further development, both of the graded sheath insulators and the alumina insulating coating. The TECO cermet insulators in particular need further effort to determine the source of and solution to their low resistances in the presence of cesium.

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TASK 4. REACTOR TECHNOLOGY

The nuclear analysis effort over the contract period was concerned with five principal activities:

1. Nuclear cross section data evaluation
2. Analytical methods development
3. Critical experiments and other physics measurements
4. Nuclear design of thermionic reactor cores
5. Nuclear analysis in support of the TFE test program.

Cross section studies were concentrated on the refractory metals and other nuclides important to thermionic reactor design which had not been extensively studied in other reactor programs. New evaluations were carried out for niobium, molybdenum, rhodium, samarium, rhenium, praseodymium, and dysprosium. As a result of the critical experiment program discussed below, discrepancies in the cross section data for U-235, tungsten, niobium, molybdenum, beryllium, oxygen, samarium, and dysprosium were identified and, in some cases, corrected in later evaluations.

The analytical methods development effort led to the development of a number of computer programs which were extensively used in design studies and critical experiment analysis. These include the GAF/GAR/GAND (References 1, 2 and 3) system for calculating neutron energy spectra and preparing multigroup cross section data, the GANDY (Reference 4) and FSDP3 (Reference 5) codes for calculating cross sections in the unresolved resonance energy region, the GAPER (References 6 and 7) transport perturbation theory program, and the 3DT three-dimensional transport theory code. These programs continue to be useful in fast reactor analysis.

The earliest reactor physics experiments carried out consisted of spectrum measurements on non-multiplying and subcritical assemblies (Reference 8). The design of these one-dimensional assemblies of tungsten and U-235 formed the basis for several later spectrum experiments sponsored by other agencies, and initiated the extensive fast spectrum program at GGA. The tungsten measurements revealed a spectral temperature dependence at high energies which had not been noted previously.

The major undertaking in the experimental reactor physics program was the design, construction, and licensing of the Thermionic Critical Facility (References 9, 10 and 11). This work included the design and fabrication of a full complement of neutronicallly homogeneous fuel elements for Assembly No. 1. A comprehensive series of experiments encompassing nearly the entire spectrum of fast reactor physics measurements was performed with Assembly No. 1 (References 12, 13, 14 and 15). In addition to the conventional critical mass measurement, the isothermal temperature coefficient, reactivity worths of individual fuel elements, and neutron lifetime (β/ℓ) were measured. The neutron spectrum was measured by the pulsed neutron, time-of-flight method in a novel and calculable slab geometry configuration (Reference 13). Additionally, the reactivity worths of ~30 materials of interest to thermionics were measured in a unique experiment in which reactor flux perturbations were minimized (Reference 15). Reflector worths were also determined for thin slabs. The Doppler reactivity coefficients for samples of U-238, W, Ta, Nb, Dy_2O_3 , and Sm_2O_3 were measured in a calculable spherical configuration by a new method in which the sample temperature was varied using a flowing stream of helium gas (Reference 16).

Assembly No. 1 of the Thermionic Critical Experiment was studied analytically in considerable depth, and the comparisons between calculation and experiment yielded considerable information relevant to thermionic reactor core design.

Nuclear analysis of thermionic reactor systems was carried out for a wide variety of reactor designs for space and terrestrial applications (References 17 and 18). It was shown that a single TFE design, originally optimized for a fast reactor, could be used in systems with fast, intermediate, or thermal neutron energy spectra with satisfactory characteristics of critical size, power distribution, control margin and temperature coefficient. The latest study in this series is described below.

Finally, enrichment requirements and power distributions were calculated for all TFE and capsule tests irradiated in the TRIGA reactor.

The reactor design and analysis effort over the contract period has resulted in conceptual reactor designs for a variety of space and terrestrial electric power applications as well as for reactor experiments to be conducted in test facilities. This work was performed to establish reactor configurations for incorporation in power system designs and to analyze selected design parameters to establish performance parameters and feasibility of the particular concept.

Early reactor design work was focused on the "pancake" reactor concept (References 19 and 20) where the thermionic fuel elements are unit cells. The more recent work adopted the concept of the converters placed in series in a tube to form "flashlight" fuel elements (References 21, 22 and 23). The pancake reactor configuration employed cross flowing liquid metal coolant while the flashlight reactor employs liquid metal coolant flow parallel to the axes of the flashlight fuel element.

Reactor applications included electric propulsion as well as auxiliary power (Reference 24). Reactors designed for electric propulsion are, generally, composed entirely of thermionic fuel elements and are characterized as fast neutron spectrum reactors. Auxiliary power reactors typically include the addition of driver fuel in the core to minimize core size. Both fast and moderated spectra have been studied, dependent upon the choice of driver fuel material.

Water-cooled reactors were also studied for undersea application (Reference 18). These designs have a core composed of a mixture of thermionic fuel elements and UzRH fuel elements with surfaces exposed to the coolant of stainless steel construction.

Reactor designs for reactor ground tests were prepared at different times during the program. These reactors were intended to provide engineering data for use in design of reactors for space power (Reference 25).

The reactor test-conceptual design and cost task was concerned with the preparation of designs for a reactor test as well as cost estimates for the reactor test program.

The most detailed study of a reactor test was performed assuming use of an existing facility at Los Alamos, New Mexico (References 25 and 26). The reactor and facility were designed to facilitate removal and replacement of the fuel elements and other core components.

4.1 NUCLEAR ANALYSIS (M. Merrill, A. Marshall, D. Mathews)

The objective of this task was the development and understanding of the neutronic characteristics of a thermionic reactor, including evaluation and refinement of cross section data, development of analytical methods for reactor physics design, and nuclear analysis in support of systems studies.

Past work has included nuclear analysis of fast and moderated thermionic reactors in support of systems studies. In addition, the task has been concerned with the analysis of data from critical experiment work performed at GGA and with the planning of future critical experiment programs.

4.1.1 Cross Section Maintenance

The processing of the Version III ENDF/B cross section data was completed for all nuclides of interest to thermionics except beryllium, for which modification of the existing processing codes would be required. The intended testing of the Version III data by re-calculation of the reactivity worth measurements in Assembly 1 of the Thermionic Critical Experiment (Reference 15) was not initiated.

4.1.2 Analytical Methods Development

Work was terminated on the development of a multi-region version of the MICROX cross section preparation code (Reference 27). Completion of this task would have required about three man-months of additional effort.

4.1.3 Critical Experiment Planning

The work intended for this task in the second half of the fiscal year was not initiated. The Thermionic Critical Facility is being dismantled and de-licensed.

4.1.4 Reactor Design Studies

Nuclear analysis of a fast spectrum thermionic reactor for electric propulsion applications has been carried out over the past 17 months. A complete summary of this work is presented here, which includes some material from previous summary reports as well as the most recent work. Also included are the results of nuclear calculations for the Thermionic Reactor Experiment (TREX).

4.1.4.1 Introduction

The standard electric propulsion thermionic reactor was designed to produce 120 kWe with 162 UCZrC fueled thermionic fuel elements (TFEs).

Elevation and plan views of the reactor core are shown in Figures 4-1 and 4-2. The core consists of 162 TFEs of about 3.3 cm outer diameter arranged in a hexagonal lattice. The design of the UCZrC fueled TFE closely resembles that of the 6F series of prototype thermionic elements tested at GGA. A drawing of the 6F cell is given in Figure 4-3. The TFEs contain six thermionic diodes 5.1 cm in length arranged axially and separated by intercell connector regions 2.3 cm long (see Figure 4-4). Axial reflector pieces of BeO 10 cm long are included on each end of the TFEs. Additional TFE data are given in Tables 4-1 and 4-2.

Reactor control is provided by 11 cm thick BeO shutters which open out to increase core leakage (Figure 4-2). The core and reflector shutters are separated by a niobium pressure vessel. NaK coolant flows in the spaces between the TFEs.

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Figure 4-1. Reactor Elevation View

Figure 4-2. Reactor Plan View

Figure 4-3. Arrangement of Components for F Series Cells

Fig. 4-4. Thermionic Fuel Element

TABLE 4-1

REACTOR AND TFE MATERIALS

Emitter

Tungsten: Inner 0.7 mm vapor deposited from WF₆Outer 0.3 mm vapor deposited from WCl_x

Surface: (110) crystal orientation, 4.9 eV work function

Collector Nb

Electric lead Mo

Outer sheath Nb-1% Zr

Fuel loading 90 UC-10 ZrC plus 4 wt.% W

Range of carbon/uranium ratio $1.01 \leq C/U \leq 1.03$

Average % theoretical density 77%

U²³⁵ inventory (total for reactor) 195 kg

Maximum emitter cavity loading 69% of theoretical density

Insulator-seal Brazed Nb-Al₂O₃ (Lucalox)-NbSheath insulator (trilayer) Gas pressure bonded Nb-Al₂O₃-Nb

TABLE 4-2

TFE DIMENSIONS

	<u>I.D.</u> <u>(cm)</u>	<u>O.D.</u> <u>(cm)</u>
Emitter	2.59	2.79
Collector	2.86	2.59
Alumina sheath insulator	2.59	3.13
Inner sheath	3.13	3.23
Outer sheath	3.23	3.34
Interelectrode gap	0.029 cm	
Emitter area	49 cm ²	
Fuel length per cell	5.08 cm	
Unfueled length per cell	2.29 cm	
Length - six cells	44.20 cm	
Upper reflector length	10.16 cm	
Lower reflector length	10.16 cm	
End cell fittings & miscellaneous	<u>3.70 cm</u>	
TOTAL LENGTH	68.20 cm	

4.1.4.2 Calculational Methods

Multigroup neutron cross section sets were obtained with the GGC-5 code (Reference 28). For most calculations, an 11 broad group structure was used which is given below in Table 4-3.

The energy group structure was chosen to provide approximately equal flux in each energy group at the core-reflector interface. Some adjustment of the energy groups was then made to adequately treat the various types of reactions in the resolved and unresolved resonance range and in the fission source range.

The general procedure followed was to run separate GGC calculations for the core region and the radial reflector. The initial approximation is thus made that the neutron energy spectrum in these regions is that appropriate to an isolated homogeneous mixture of the materials in the region with a fission spectrum source and out-leakage represented by an energy-independent buckling. The adequacy of this approximation has been investigated with many-group one-dimensional calculations from which the actual region-dependent spectra can be approximately determined and used for cross section averaging in place of the GGC spectrum calculation.

The material volume fractions for the core region are given in Table 4-4. The GGC calculation for this region is complicated by the large number of resonance absorbing materials and the very heterogeneous arrangement. The principal resonance absorbers are U-235, U-238, tungsten, tantalum, and niobium. The latter three materials occur in a variety of shapes in the emitter, the collector-sheath assembly, and the assorted connectors, spacers, springs, etc., of the intercell region. Only a few of these shapes resemble the spheres, slabs or cylinders for which escape probabilities are available in GGC.

TABLE 4-3
NEUTRON ENERGY GROUP STRUCTURE

<u>Group</u>	<u>Energy Interval (eV)</u>	<u>Lethargy Interval</u>
1	$14.9 \times 10^6 - 4.06 \times 10^6$	-0.4 - 0.9
2	$4.06 \times 10^6 - 3.01 \times 10^6$	0.9 - 1.2
3	$3.01 \times 10^6 - .907 \times 10^6$	1.2 - 2.4
4	$.907 \times 10^6 - .408 \times 10^6$	2.4 - 3.9
5	$.408 \times 10^6 - .202 \times 10^6$	3.2 - 5.0
6	$.202 \times 10^6 - 67 \times 10^3$	3.9 - 5.0
7	$67 \times 10^3 - 11.71 \times 10^3$	5.0 - 6.75
8	$11.71 \times 10^3 - 1.58 \times 10^3$	6.75 - 8.75
9	$1.58 \times 10^3 - 275$	8.75 - 10.5
10	275 - 2.38	10.5 - 15.25
11	2.38 - 0.0	15.25 - ∞

TABLE 4-4
REFERENCE DESIGN COMPOSITIONS

<u>Region*</u>	<u>Material</u>	<u>Volume Fraction in Region</u>
Fueled	UCZrC**	.5465
	W	.0891
	Nb	.2223
	Al ₂ O ₃	.0257
	NaK	.0931
Unfueled	W	.1968
	Nb	.1333
	Al ₂ O ₃	.0798
	NaK	.0931
Filler	BeO	.7998
	Nb	.0060
	NaK	.0931
Vessel	Nb	1.0
Shutters	BeO	.950
	Nb	.020
Axial Reflector	BeO	.7998
	Nb	.1071
	NaK	.0931

*Refer to Fig. 4-5 for region location.

**90 UC-10 ZrC (4 wt. % W), 72% theoretical density, 10% void.

None of the resonance absorption treatments presently available in GGC-5 are adequate to treat the core region lattice. The most detailed spatial model available (the GAROL option for the resolved resonance range) allows only two regions, while to treat only the TFE fuel and its surrounding materials in one-dimensional cylindrical geometry requires a minimum of three regions (fuel, emitter, collector) and additional regions in the thick fuel body would be very desirable to improve on the flat source approximation in the escape probabilities. The TFE fuel bodies are spaced sufficiently close that some form of Dancoff correction should be employed in the escape probabilities, but the standard formulation of this effect does not allow for the presence of fissile material and resonance absorbing materials between the regions being treated.

In the calculations actually done, most of this complexity was necessarily ignored. All resonance calculations were done with the GAMNIT-GANDY options of GGC-5, and because of expense, only one calculation was done for each resonant nuclide regardless of how many different geometries it occurs in. The resonance group cross sections of U-235 and U-238 were evaluated for the fuel body geometry, those of tungsten for the emitter, and those of niobium for the collector-sheath assembly. For tungsten and niobium, which occur as thin walled cylinders in the emitter and collector, the resonance calculations were done in cylindrical geometry by defining an equivalent solid cylinder with mean chord length.

$$\frac{1}{l} = \frac{4V}{S_{\text{outer}}} = \frac{2(r_o^2 - r_i^2)}{r_o}$$

Scattering materials inside the cylinder, such as the carbon in the UCZrC fuel are homogenized into the equivalent cylinder as an admixed moderator, e.g.,

$$N_{\text{carbon in W}} = N_{\text{carbon in UCZrC}} \times \frac{V_{\text{fuel}}}{V_{\text{W}}}$$

The choice of these approximations was carried over from earlier work. The accuracy of the method has not been confirmed by more detailed calculations for the present case.

The GGC for the radial reflector was run assuming a homogeneous mixture of the reflector materials. The material volume fractions for the reflectors are also given in Table 4-4. A separate GGC was run for the axial reflector.

The various calculations described below were done with the one and two-dimensional transport theory codes LDF (Reference 29) and TWOTRAN (Reference 30). The P_1S_4 approximation was used in all cases, except for the detailed R θ study which used the P_0S_4 approximation.

The one-dimensional cases may be run with almost any reasonable number of energy groups, while the two-dimensional cases were limited for reasons of economy to the standard 11 group structure.

The transverse leakage is represented in LDF either by an adjusted fission spectrum ($X' = X - L/S$)* or by reading in a single transverse dimension from which geometric bucklings are computed and used with the cross sections of each region to compute DB^2 for that region. For many radial cases for which the axial leakage did not significantly affect the desired results, the option of supplying a single transverse dimension was used. The value specified, 64 cm, which implies a 10 cm reflector savings, has been found to give reasonable results.

The two-dimensional calculations were done in r-z and r- θ geometry. Because of limitations of core storage and budget, only portions of the core were represented in these models. For

* X' = the adjusted fission spectrum, X = the actual fission spectrum, and L/S is the neutron leakage per source neutron.

example, the largest r-z model, used for power zoning, represents half the reactor in the axial direction. The differences between upper and lower grid plates and axial reflectors were considered to be less important in this problem relative to the need to have reasonably small mesh spacing in the core regions.

Individual temperature coefficient components were computed using the GAPER, and GAPER-2D codes (References 6 and 7). GAPER and GAPER-2D are, respectively, one-dimensional and two-dimensional P_1 transport perturbation theory codes. These codes use the direct and adjoint fluxes and currents, as computed by 1DF or TWOTRAN, to calculate the change in the inverse multiplication factor $\Delta(1/k)$ due to various perturbations in the macroscopic cross section sets.

4.1.4.3 Reactivity Requirements

The reactivity requirements assumed for the purpose of initial sizing studies were as follows:

Temperature defect, burnup	0.02 $\Delta\rho$
Shutdown margin	0.04 $\Delta\rho$
Uncertainty	0.03 $\Delta\rho$
Control worth	0.09 $\Delta\rho$

These requirements imply that the cold clean reactor should have a calculated multiplication factor of $k_{\text{eff}} = 1.05$ with all shutters closed. The temperature and uncertainty allowances are estimates based on previous studies. The shutdown margin includes an allowance for one control drive failure.

The calculations carried out to establish the critical size involved a sequence of four reference designs which will not be described in detail here. In the course of these studies, however,

many results were obtained on the effect of various design changes on the core reactivity. These are summarized in Table 4-5. The excess reactivity is seen to be very sensitive to variations in reflector design, which affect leakage, and to uranium density. The sensitivity functions are for small changes; large changes may be quite non-linear. The excess reactivity of the final reference design was determined from a two dimensional (RZ) calculation for the unzoned core. This calculation included the following design changes, from the previous reference design, which affect reactivity:

1. Replace tantalum with niobium in the intercell region (+0.0106 $\Delta\rho$).
2. Reduce pressure vessel side wall thickness to 0.090 inches (+0.0087 $\Delta\rho$).
3. Replace control drums with full length control shutters.

The calculational model geometry is shown in Figure 4-5. Table 4-6 gives the volume fractions of the various materials used in the calculation. This calculation gave a multiplication factor for the unzoned core of 1.066. Two effects not included in the calculation are the effect of increasing the number of energy groups from 11 to 25 (+0.0027 $\Delta\rho$) and the effect of including in the two-dimensional mesh the coolant plenum and pressure vessel head above and below the axial reflectors (+0.0043 $\Delta\rho$). Including these effects would increase the multiplication factor to $k_{\text{eff}} = 1.074$.

Calculations were also made for the zoned core. The fuel zoning specified is shown in Figure 4-6 which gives the fractional theoretical density of the UCZrC fuel material in each axial and radial position. The multiplication factor of the zoned reactor was calculated to be 1.061. With the two corrections indicated

TABLE 4-5

EFFECT OF PARAMETRIC VARIATIONS ON REACTIVITY OR CRITICAL SIZE*

Variation	Reactivity Change, $\Delta\rho$	No. of TFEs for Equal Excess Reactivity
Change fuel from UCZrC to UC	0.03-0.04	
Remove BeO filler pieces between drums and in NaK annulus	-0.036	
Increase control drum diameter from 4" to 5"	0.007	
Increase core diameter by adding TFEs (constant reflector thickness)	0.018 per cm	
1 mil change in TFE pitch spacing	-0.00028	
Replace 6 cell TFE by 7 cell TFE	0.024	
Increase axial BeO reflector thickness from 4" to 5"	0.0032	
Use enriched W	0.018	
Replace niobium vessel and grid plates with stainless steel	0.02	
1% change in fuel volume fraction	0.0039	
Use 0.9 of reference fuel density		228
Use 0.8 of reference fuel density		350
Double reference emitter thickness		286

* These results are from a variety of electric propulsion reactor studies and should be considered to apply only approximately to the reference design. Some cases had control drums and others had shutters.

Fig. 4-5. Calculational model

TABLE 4-6

NUCLEAR CHARACTERISTICS OF ELECTRIC PROPULSION FAST REACTOR

Reactor Diameter/Height: cm	69.9/64.5
U-235 Loadings: kg	178
Maximum nvt > 0.1 Mev per 10^4 hr	2.4×10^{21}
Median Energy of Flux/Fission/Absorption: kev	390/235/170
Infinite Multiplication Factor, k_{∞}	1.72

Neutron Balance

Absorption per Source Neutron in:	U	.5100
	W	.0547
	Nb	.0401
	NaK	.0013
	Other	<u>.0006</u>
		.6067
Leakage per Source Neutron:	Axial	.1246
	Radial	<u>.2687</u>
	Sum	1.0000

1	4	7	10	13	16	19	Zone Numbers
.6930	.6930	.6930	.6930	.6930	.6930	.6930	Fraction TD
1.0813	1.0665	1.0291	.9786	.9230	.8880	.9774	Power
2	5	8	11	14	17	20	
.6393	.6424	.6515	.6641	.6763	.6744	.6213	
1.0780	1.0717	1.0541	1.0300	1.0025	.9865	1.0402	
3	6	9	12	15	18	21	
.5875	.5889	.5959	.6059	.6149	.6110	.5614	
1.0543	1.0470	1.0307	1.0088	.9855	.9773	1.0389	

Fig. 4-6. Relative power and fuel density distribution

above, this becomes $k_{eff} = 1.069$ which is the present best estimate for the zoned reference design and gives a margin of nearly $0.05\Delta k$ above the estimated requirement for temperature and burnup.

In the two-dimensional calculations mentioned above, the TFEs are all oriented in the same direction to facilitate the zoning calculations. In the actual design, one-half of the TFEs are inverted and in the initial design the TFEs were staggered in the axial direction. It was later decided that it was preferable to align the TFEs and stagger the end reflector pieces. This has the advantage of increasing the reactivity ($0.01\Delta\rho$) relative to the previous reference design and being more consistent with the calculational model used above.

4.1.4.4 Power Distribution Studies

The RZ calculation, described above for the zoned core case, provided the two-dimensional power distribution for the reference core. As shown in Figure 4-6, the diodes are represented radially as rings of unit cells in a hexagonal array and axially as the 5.1 cm fueled and 2.3 cm unfueled zones.

The axial power distribution in the core zone was flattened, relative to the unzoned core, by varying the density of UC-ZrC in each diode zone as shown in Figure 4-6. These densities represent the fraction of theoretical density inside the fuel cavity, with the average density maintained at 64.8% theoretical density. Finding the correct densities is an iterative procedure involving a computer code called ZONE which evaluates thermionic performance for an input power distribution and specifies fuel density adjustments to obtain uniform emitter temperatures. If the density changes required were significant, a second nuclear calculation would be required.

The relative power distribution in the diodes in the zoned core is given in Figure 4-6 normalized to an average power in the diodes of 1.0. The highest axial max/min power ratio was 1.08. Further variations in the diode fuel density would reduce this max/min power, but would result in a somewhat lower average fuel density, with a corresponding loss in core reactivity.

The power distributions discussed here were computed with the shutters closed. The power distribution with the shutters in an operating position, i.e., partially opened, has not been evaluated.

The RZ calculations cannot accurately treat the power distribution in the TFEs adjacent to the reflector. In order to determine the adequacy of the RZ calculation in this region, an R θ calculation was made which explicitly described a TFE adjacent to the reflector. The geometric model used is shown in Figure 4-7.

A series of one-dimensional calculations were made which confirmed the adequacy of the mesh size and of the homogenization of the emitter and collector used in the R θ model. The one-dimensional calculations also compared the power distributions from a .25 group calculation (5 thermal groups) with an 11 group calculation (1 thermal group). The one-thermal group calculation badly underestimated the power peak at the core reflector interface.

Because the R θ problem was larger than any previously run case, and because of no prior experience with the CDC-7600 computer on which it was planned to run, the first R θ calculation was run with one thermal group using the P_0S_4 approximation. The P_0 approximation probably slightly overpredicts the power peak in the outer TFE, while the use of one thermal energy group is known to produce a definite underprediction of the peak. A P_1S_4 , multi-thermal group R θ calculation was planned for a future calculation

Fig. 4-7. R θ Geometry for an Explicit TFE in the Electric Propulsion Reactor

to complete the study. The transverse leakage was represented by using a core height of 64 cm to compute the buckling. This leakage approximation was verified by comparing a one-dimensional radial problem, using an input 64 cm core height for the buckling, with an RZ calculation.

Figure 4-8 compares the $R\theta$ relative power, for a traverse across the radial center line of the outer TFE, with that obtained from an 11 group, P_1S_4 RZ calculation. Relative power is defined here as the power in a mesh interval divided by the lowest mesh interval power in the TFE. It is seen in this comparison that the RZ calculation overpredicts the power distribution relative to the $R\theta$ calculation. It is expected that a similar comparison with multi-thermal groups would also show that the power peaking predicted by an RZ calculation is conservative; further investigation is required, however, to verify this prediction. A contour plot of the power distribution from the $R\theta$ calculation is given in Figure 4-9.

4.1.4.5 Control Worth Studies

An investigation was made early in the design study to find a control system that would provide adequate control with a minimum reflector thickness. Calculations were done for two control concepts: rotating BeO drums with B_4C absorber segments (the original), and BeO shutters which open out to increase core leakage.

Calculations for a preliminary design with 5-inch diameter drums indicated a control worth of $0.06 \Delta\rho$ with natural boron and $0.09 \Delta\rho$ with B^{10} for an absorber segment one-sixth of the drum diameter in thickness. It would be desirable to use natural boron instead of B^{10} for reasons of cost. Fully enriched boron (93.1 wt.% B^{10}) costs around \$3/g. However, using a larger poison segment of natural B_4C would reduce the excess reactivity in the drums-out configuration.

Fig. 4-8. Comparison of Relative Power in a TFE Adjacent to the Reflector from an RZ Calculation and a Rθ Calculation

Fig. 4-9. Contour Plot of Relative Power in a TFE Adjacent to the Reflector

The possibility of adding hydrogen to the poison segments of control drums was considered with the idea of slowing down a greater fraction of the reflector neutrons into the energy range where the boron cross section is largest. This idea was tested with some calculations in which hydrogen atom densities of 0.03 and 0.06 atoms/bn-cm were assumed in the poison segments in place of the carbon (0.06 is roughly the hydrogen density in H_2O or $ZrH_{1.7}$; an actual material suitable for the temperature requirements has not been selected). The density of natural boron was unchanged. The results are as follows:

	<u>k_{eff} (drums out)</u>	<u>k_{eff} (drums in)</u>	<u>$\Delta\rho$</u>
Original design	1.0384	0.9782	0.0593
$N_H = 0.03$	1.038	0.9531	0.8558
$N_H = 0.06$	1.0377	0.9375	0.1030

The method is seen to be quite effective. With the drums full out, the excess reactivity is hardly affected, since neutrons which reach thermal energies far out in the reflector have little chance of returning to the core.

Another control alternative is to use opening shutters in place of rotating drums. Since the motion of the shutters opens up large streaming paths, multi-dimensional calculations are required to determine the control swing.

Calculations of the control shutter worth have been done in RZ geometry since the streaming with the shutters open is primarily in the axial direction. A correction for the radial streaming is obtained from an R θ geometry calculation. The reference shutters-closed configuration was calculated for the unzoned core before the inclusion of the three design changes discussed above. The calculational results are the following:

	<u>k_{eff}</u>	<u>Δk</u>
1. RZ, Unzoned Core, Shutters Closed	1.033	
2. RZ, Unzoned Core, Shutters Open 1.5 in.	0.981	0.052
3. RZ, Unzoned Core, Shutters Open 3.0 in.	0.936	0.097
4. Rθ, Unzoned Core, Shutters Open 3.0 in.	0.907	0.126

The Rθ calculation, Case 4, includes transverse leakages (DB² cross sections) from Case 3. Further iteration of leakages between Cases 3 and 4 should be done, but the final control worth should be closer to 0.12 Δk. Most of the design changes, e.g., removal of Ta, thinner pressure vessel and longer axial shutters, will change the multiplication factor of the shutters-open case less than that of the shutters-closed case. It is, therefore, concluded that ample control margin is available in the shutter design.

Since shutter control provides adequate control margin with a minimum reflector thickness, the shutter control concept was chosen for the reference design.

4.1.4.6 Water Flooding Studies

During shipment or launch of an electric propulsion reactor, the control shutters would be either removed or in the full-open position (7.6 cm displacement of center-of-reflector mass). If an accident occurs in which the reactor is immersed in water and either the core or the leakage gaps between shutters are filled with water, a potential criticality problem exists. One-dimensional radial calculations (10 fast groups, 5 thermal groups) were done for several conditions of water flooding to investigate this problem. The results are as follows:

Core	Reflector	Multiplication Factor k_{eff}
NaK Filled	Shutters closed - dry	1.074
H ₂ O Filled	Shutters closed - dry	1.127
H ₂ O Filled	Shutters open 1 cm, H ₂ O in gap	1.125
H ₂ O Filled	Shutters open 4 cm, H ₂ O in gap	1.081
H ₂ O Filled	11 cm H ₂ O reflector	1.055

It seems apparent that some means would have to be provided to avoid criticality in the case of accidents which lead to flooding. In the pre-launch phases of a mission, this could take the form of a poison material between the core and shutters. For situations involving re-entry and flooding after removal of this poison, some means of breaking up the core beyond simply blowing off the shutters may be required. The alternative of incorporating sufficient resonance and thermal poisons to assure sub-criticality after intact re-entry and flooding may penalize the excess reactivity of the unflooded system to an unacceptable degree, although this has not been established by calculations.

4.1.4.7 Other Nuclear Characteristics

Table 4-6 gives some miscellaneous nuclear characteristics calculated for the electric propulsion reactor.

4.1.4.8 240 kWe Reactor Studies

Some one-dimensional calculations were done for a 240 kWe electric propulsion design as part of a separate study. Three concepts were investigated: 1) a 240 TFE reactor, 2) a moderated 240 TFE reactor, and 3) two coupled 120 kWe reactors.

The 240 TFE reactor is simply an enlarged version of the 162 TFE reference design. It has adequate excess reactivity if the fuel density is reduced by 10-12% from the 162 TFE design. Control is satisfactory with 4.5 in. diameter drums with $B^{10}C$ segments occupying one-third of the drum diameter. (The spacecraft design in this case favors drums over shutters.) Because of the higher power density, lower fuel density, and higher radial flux peak, the maximum fast fluence at 10^4 hr is about 30% higher than in the 162 TFE design.

If it were necessary to reduce the fluence for a long mission, it could be brought to the level of the reference design (2.4×10^{21} nvt per 10^4 hr) by the inclusion of about 10 volume percent ZrH in the core. In this case the fuel density is reduced about 14% from the 162 TFE reference design. The cell geometry assumed for this case is shown in Figure 4-10.

The use of multiple 120 kWe thermionic reactors was considered as a method of obtaining higher powers without building larger cores. In addition to development cost savings, this concept avoids the problems of higher fast fluence and reduced control margin encountered in larger diameter cores. An upper limit estimate of the reactivity coupling between adjacent 120 kWe reactors was made by replacing the vacuum boundary condition at the outer reflector surface by a reflecting boundary condition. This calculation, which essentially represents a close-packed infinite array of reactors, gives a reactivity increase of $0.12 \Delta k$ for reactors controlled with poison drums. It can be concluded that an arrangement of two or more drum-controlled reactors would be very weakly coupled and is, therefore, a feasible concept.

4.1.4.9 TREX Temperature Coefficient Studies

Nuclear calculations were done for the Thermionic Reactor Experiment with the objective of comparing the dynamic behavior of an all-TFE TREX with that of a design using a smaller number of TFEs and a driver fuel.

Fig. 4-10. Geometry of ZrH-TFE lattice of Moderated 240 kWe Reactor

The latter design is attractive from the standpoint of economy, but it must be determined to what extent its dynamic behavior is affected by the relative magnitudes of the reactivity temperature coefficients of the TFEs and the driver fuel. The objective was to find a design for which the dynamic response in one or more types of transient experiments is dominated by the reactivity feedback from the TFEs.

As a start, the Doppler coefficients of TFEs and driver fuel were calculated for a variety of design options. For all cases, it was assumed that there are 60 TFEs and that the amount of driver fuel is adjusted to give a multiplication factor of 1.05. Design options included (1) locating the TFEs at the core center or at the core edge, (2) using 3-inch or 5-inch thick BeO-B₄C control drums and (3) using a close-packed driver fuel lattice or a cluster type driver element. The cluster type driver element is a group of seven driver pins inside a tube of the same outer diameter as a TFE. The driver fuel pin for both close-packed and cluster-type elements is fully enriched UO₂ clad with stainless steel with an outer diameter of 0.368 inches. The geometry of both fuel types is illustrated in Figure 4-11.

Doppler coefficients were determined by (1) obtaining direct and adjoint fluxes and currents for the desired configurations from one-dimensional transport calculations, (2) obtaining macroscopic cross sections at a variety of temperatures, and (3) using the fluxes, currents, and cross sections in a first order perturbation theory calculation to obtain the reactivity perturbation.

The principal results for all cases are summarized in Table 4-7. Detailed results for Case 7 are given in Table 4-8. Note that the definition of a "prompt" coefficient depends on the nature of the transient considered, e.g., whether the initial perturbation is in the electrical load, the position of a control drum, or the coolant temperature. For the present purposes of comparison, we define the TFE prompt coefficient as the Doppler coefficients of U-235 and U-238 and half of the Doppler

CLOSE PACKED DRIVER GEOMETRY

CLUSTER DRIVER GEOMETRY

Fig. 4-11. Driver Fuel Element Arrangement

TABLE 4-7

PRINCIPAL RESULTS - TREX DRIVER FUEL STUDIES

Case	Fuel Position		Driver Type	Control Drum Diameter (in.)	Core Diameter (cm)	Reactor Diameter (cm)	Doppler Coefficients ($^{\circ}\text{C}^{-1}$) @ 650 $^{\circ}\text{K}$		
	TFEs	Driver					TFE, "Prompt"	TFE, Isothermal	Driver, Isothermal
1	Center	Outer	Close-Pack	3	34.36	50.06	$+2.2 \times 10^{-8}$	-5.1×10^{-7}	$+1.6 \times 10^{-7}$
2	Center	outer	Cluster	3	44.35	60.05	$+2.0 \times 10^{-8}$	-3.5×10^{-7}	-1.4×10^{-7}
3	Outer	Center	Close-pack	3	32.20	47.90	$+8.7 \times 10^{-8}$	-8.2×10^{-7}	$+2.7 \times 10^{-5}$
4	Outer	Center	Cluster	3	43.20	58.90	$+4.3 \times 10^{-8}$	-7.3×10^{-7}	$+3.5 \times 10^{-8}$
5	Center	Outer	Close-pack	5	31.98	57.84	$+0.8 \times 10^{-8}$	-8.5×10^{-7}	-3.2×10^{-7}
6	Center	Outer	Cluster	5	38.81	63.61	$+1.9 \times 10^{-8}$	-5.6×10^{-7}	-3.4×10^{-7}
7	Outer	Center	Close-pack	5	31.00	58.86	$+11.0 \times 10^{-8}$	-13.5×10^{-7}	$+2.5 \times 10^{-8}$

TABLE 4-8
DOPPLER COEFFICIENTS FOR CASE 7

<u>Nuclide</u>	<u>Temperature ($^{\circ}$K)</u>	<u>Doppler Coefficient ($^{\circ}$C^{-1})</u>
U ²³⁵ (TFE)	650	+ 6.57 x 10 ⁻⁷
	1500	+ 3.04 x 10 ⁻⁷
	2500	+ 1.86 x 10 ⁻⁷
U ²³⁸ (TFE)	650	- 10.68 x 10 ⁻⁸
	1500	- 5.71 x 10 ⁻⁸
	2500	- 3.79 x 10 ⁻⁸
W (TFE)	650	- 8.81 x 10 ⁻⁷
	1500	- 4.41 x 10 ⁻⁷
	2500	- 2.88 x 10 ⁻⁷
Nb (TFE)	650	- 10.09 x 10 ⁻⁷
	1500	- 4.23 x 10 ⁻⁷
	2500	- 2.46 x 10 ⁻⁷
TOTAL (TFE)	650	- 13.46 x 10 ⁻⁷
	1500	- 6.17 x 10 ⁻⁷
	2500	- 3.86 x 10 ⁻⁷
U ²³⁵ (Driver)	650	+ 2.71 x 10 ⁻⁸
U ²³⁸ (Driver)	650	- 1.81 x 10 ⁻⁹
TOTAL (Driver)	650	+ 2.52 x 10 ⁻⁸

coefficient of tungsten. Approximately half of the tungsten is in the intercell region and has a more delayed time response than the emitter itself. The definition of the prompt coefficient given here is particularly appropriate for transients initiated by electrical load changes or step reactivity perturbations. The isothermal coefficient of the driver fuel is composed of the U-235 and U-238 Doppler coefficients and would have a time response comparable to the TFE prompt coefficient for step reactivity perturbations.

The principal conclusion to be noted from preliminary analysis of these results is the desirability of having the TFEs in the outer region and having as thick a reflector as can be allowed by power peaking considerations. With outer TFEs, the close-pack driver arrangement gives substantially larger TFE coefficients than the cluster arrangement.

In most small fast spectrum reactors, the axial expansion temperature coefficient is of comparable magnitude to the Doppler coefficient. If the driver fuel in the TREX design is not axially segmented in some way, the axial expansion coefficient of the driver fuel is expected to dominate the dynamic response of the reactor. A brief study was made to determine what effect axial segmentation had on the axial expansion coefficient of the driver fuel. Two-dimensional calculations were made using the core configuration given in Case 7 of the Doppler coefficient study. It is assumed in these studies that the driver fuel is held rigid at the top and bottom of the core and permitted to expand at the point of segmentation.

It was observed that a gap at the axial center of the driver fuel gave an expansion coefficient ($\Delta\rho/\Delta T(^{\circ}\text{K})$) of about $+4 \times 10^{-7}$. For a gap at a point 11 cm above the axial center, the coefficient is $+9 \times 10^{-8}$. Previous calculations for unsegmented driver fuel (Reference) gave an expansion coefficient of about -1×10^{-6} . It is obvious from these calculations that a significant reduction in the driver axial expansion coefficient can be obtained by segmentation. Further studies are necessary, however, to optimize the segmentation in order to reduce the axial expansion coefficient to a minimum.

4.2 REACTOR DESIGN AND ANALYSIS

The objective of this subtask is the mechanical, thermal, and hydraulic analysis and design of the reactor core, reflector, core vessel, and control drive system. The requirements of the design are set by system specifications.

The approach to reactor design at this stage of reactor development is to do sufficient analysis to assure the basic feasibility of the conceptual designs and to develop analytical tools for use in these calculations, when necessary.

Past work in this subtask included conceptual designs of fast reactors at 100 and 300 kWe power levels and moderated reactors in the 5-100 kWe power range. Thermionic reactors containing fast driver fuels were also studied.

The most recent past work has been concerned with the 120-kWe NEP (nuclear electric propulsion) reactor in the side thrust system configuration. This is a fast liquid metal cooled reactor with 162 TFEs producing 22 volts. The reactor is controlled by movement of the radially located beryllium oxide reflector. Eighteen reflector elements rotate towards and away from the core to regulate openings for neutron leakage which controls the reactor. The latest status of the 120 kWe NEP system design, including reactor parameters, is described in Reference 9.

4.2.1 Layout and Specifications for the 120 kWe NEP Reactor (D. Allen)

The axial configuration of the core and reflector was defined in greater detail. This more detailed specification is required for more precise nuclear analysis of the core and for system shielding analysis. Since the core and axial reflector are completely defined by the TFE detail, the NEP Reactor TFE has been designed.

The typical TFE is illustrated in Figure 4-12. In order to produce the reactor output voltage of 22 V, 6 such TFEs must be connected in series. To accomplish the series connection, the string of six converters in each TFE is oriented alternatively as shown in Figure 4-12, or upside down from that configuration. Two types of TFE result - the N and P types - identified according to whether the top lead of the TFE is connected to the negative or positive side (collector or emitter) of the top converter. Two-thirds of the TFEs are connected in pairs (one N and one P-type) through a connection housing at the end opposite the end welded to the reactor vessel. The TFEs are connected in the groups of six as shown in Figure 4-13. To cross-connect all of the TFE strings at their mid-potential, they are grounded to the NaK coolant. The 27 groups of 6 TFEs are located in radial rings in the core so that nuclear power input to the TFEs in each group will be approximately the same. This connection scheme is illustrated in the inset to Figure 4-13.

To accomplish the maximum fuel density in the core, the TFEs must be arranged so that the fueled regions of each emitter are aligned axially. This configuration has a calculated advantage in multiplication factor of 0.01ΔK over the alignment of TFEs so that the planes bounding all of the converters line up (See Section 4.1.4.2). Since the fuel is not axially centered in the converter, the resulting interface between converters and the axial reflector blocks is irregular. The TFE end details in Figures 4-14 and 4-15 indicate the configuration resulting. The reflector blocks are alternatively 88.3 mm (3.475 in) or 115.0 mm (4.525 in) long. The overall axial layout of the reactor by zones, significant to the nuclear analyst, is given in Figure 4-16.

It can be noted from the TFE end details of Figures 4-14 and 4-15, that the reflector blocks are shaped to accommodate the TFE leads, the pump-out lines for the interelectrode space and the fission gas port tube. The TFE insulator-seal is configured to the same design as the converter insulator seal. Three tubes emerge from the TFE. They are

the cesium tube to be connected to the cesium reservoir, the fission product tube to be connected to the fission product trap and the fission product space pump-out tube, which is pinched closed after fabrication. The details of the connection housing at the ends of two-thirds of the TFEs can be seen. The connection is to be made with electron beam welds from the end on. All but the last weld are circular. The last weld, sealing the TFEs at that end is an ogival weld.

Design work on the details of the reactor and NEP TFE configurations should not proceed further until their fabrication practice is demonstrated.

4.2.2 Mechanical Design for 120 kWe Reactor (E.J. Steeger)

4.2.2.1 Effect of Isothermal and Self Heating on the Reflector Element Assembly

If the reflector elements are heated isothermally for bake-out or heated by coolant circulation in the reactor, differential dimensional changes between the niobium can and the BeO will take place. These are shown as Figures 4-17 and 4-18.

Insertion of a Bellville spring at either or both ends of the BeO will serve to accommodate the differential axial dimension changes. If the spring preload is greater than the expected load during the launch sequence, then contact between surfaces will be maintained. Figure 4-19 shows the design of a spring and the spring stress and deflection with load.

A thermal analysis of the reflector element assembly with self heating was done in order to determine the temperatures during reactor operation. The GGA computer code TAC2D was used for this analysis. The self heating of BeO was taken as $0.04 \text{ W/cm}^3/\text{megawatt(th)}$. Since part of the element is looking at a section of insulation and part is looking at the heat exchanger (radiator), the problem was broken up for solution.

Fig. 4-12. NEP Reactor Thermionic Fuel Element

ARRANGEMENT OF THE 27 GROUPS OF 6 TFEs
(LOOKING DOWN ON THE REACTOR)

Fig. 4-13. TFE Connection for 120 kWe NEP Reactor

Fig. 4-14. TFE Stem Assembly Layout

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Fig. 4-15. Bottom TFE Connection

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Fig. 4-17. Reflector Axial Expansion Versus Temperature

Fig. 4-18. Reflector Lateral Expansion Versus Temperature

Fig. 4-19. Bellville Spring Design

Also, the design of the element allows a 0.38 mm (0.15 inch) clearance between the BeO and the can when at ambient temperature.

Since some contact between the BeO and the can will take place, the following assumptions were used for the analysis:

1. Elements in the "open" and closed positions.
2. Elements looking at the thermal insulation and the heat exchanger.
3. One-fourth of the BeO surface in contact with the Nb can.
4. A gap between the BeO surface and the Nb can.

Figures 4-20 through 4-23 show the results of the analysis. The ambient temperature for all cases was 1055°K.

The highest BeO and niobium can temperatures occur with the elements in the closed position with the assumption that a radiation gap exists between the BeO and the can.

If the worst case is chosen, the BeO temperature will be about 1300°K with a radial temperature gradient of 14°K. The can will have a radial temperature difference of 60°K.

The temperature difference across the can of 60°K would result in a displacement inwardly of .010" (.025 cm) at the center of the element.

For the condition of 60°K ΔT across the niobium can, the BeO has a ΔT of 14°K. The tensile stress induced by the ΔT is

$$S_T = \sigma_{E\Delta T} = 40.7 \text{ MN/m}^2 \text{ (5900 psi)}$$

CONDITION - (a) SHUTTERS CLOSED LOOKING AT INSULATION

(b) 1/4 OF BeO SURFACE IN CONTACT WITH Nb CAN

Fig. 4-20. Temperature Distribution in Reflector

CONDITION: SHUTTERS CLOSED LOOKING AT INSULATION

Fig. 4-21. Temperature Distribution in Reflector

CONDITION - (a) REFLECTORS IN "OUT" POSITION LOOKING AT THE
HEAT EXCHANGER

(b) GAP BETWEEN BeO AND Nb CAN

Fig. 4-22. Temperature Distribution in Reflector

CONDITION - REFLECTORS IN "OUT" POSITION LOOKING AT THE HEAT EXCHANGER.

1/4 OF BeO SURFACE IN CONTACT
WITH CAN

Fig. 4-23. Temperature Distribution in Reflector

If the BeO is assumed to be in contact with the can during this condition then the radial ΔT is 8°C resulting in a stress of 23.2 MN/m^2 (3370 psi).

4.2.2.2 Control Element Under Launch Shock and Vibration

For the control element and its support structure, the most severe projected launch loading is the requirement to withstand 15 Gs along the three axes. No sinusoidal excitation at or above resonant frequencies are called out in the qualification specification.

The configuration and mass distribution are shown in Figure 4-24.

If it is assumed that the BeO "pucks" carry no load, sideways static loading of 15 Gs results in bending stresses in the 0.76 mm thick can of 22.8 MN/m^2 (3310 psi). The factor of safety for this condition, based on the proportional limit of Nb-1Zr is eleven. The angular deflection at the end supports under the acceleration is 0.048° , and the centerline deflection 0.17 mm (0.0066 in.).

An axial acceleration of 15 Gs produces a tensile stress in the can of 9.0 MN/m^2 (1308 psi) which corresponds to a factor of safety of twenty-eight. Maximum shear stress where the element supports meet the bearing shaft is 1.8 MN/m^2 (2577 psi) which is a factor of fourteen above the limit.

VIEW AA

TOTAL WT.	<u>Kg</u>	<u>LB</u>
BRACKETS	1.47	3.24
CAN	1.58	3.50
BeO	19.50	43.00
END PLATES	<u>1.81</u>	<u>4.00</u>
	24.3	53.7

Fig. 4-24. Reflector Element Weight

The control element bearings are subjected to a stress of 3.6 MN/m^2 (518 psi) in compression due to lateral acceleration, and shear and compressive stresses of 4.2 MN/m^2 (602 psi) and 17.3 MN/m^2 (2467 psi), respectively, under axial acceleration.

The calculated natural frequency of the control element is to be 174 Hz. The sinusoidal input specified for qualification spans the range of 4 to 35 Hz. However, random acceleration is specified over a range up to 2000 Hz.

Launch load conditions on the control element and its supporting parts appear to be moderate. No potential problem areas should be expected for this component from a mechanical standpoint.

4.2.2.3 Design of Control Element Snubber

The control element deceleration snubber is designed to act within the last 10° of control element travel to absorb the inertial energy stored in the system and stop the drive in the control rod full out position. The item as designed is illustrated in Figure 4-25.

The cylinder is divided into two chambers with one portion a bellows. Initially, with a gas pressure of 34.5 kN/m^2 (5.0 psia) the force exerted by the piston rod against the snubber arm almost balances the spring torque. As the control element moves toward an "in" position, the piston will travel 8.9 mm (0.35 in.) (10° of control rod movement) to a stop. An orifice through the piston allows the gas to come to equilibrium within the system.

As the control elements are spring driven during scram, the snubber arm will contact the piston rod 10° before the full out position. The gas to the right of the piston is essentially trapped and will be compressed. If the orifice is designed correctly, the inertial energy can be dissipated in flow work as the gas is transferred through the piston and there will be no tendency to bounce.

Fig. 4-25. Control Rod Deceleration Snubber

Fig. 4-26. Snubber Energy Absorption Versus Travel

Fig. 4-27. Control Rod Snubber Pressure Versus Displacement

The snubber was designed to absorb the energy from a 52° "in" position with a 8.9 mm (0.35 in) travel. From any position less than 52°, the snubber will not bottom from energy absorption, but will stop the control element and then go to the full-out position as the gas in the two chambers reaches equilibrium.

Figure 4-26 shows the energy absorption versus piston travel with the energy stored in one control element indicated and Figure 4-27 is the calculated pressure in the snubbing chamber.

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TASK 5. SYSTEMS TECHNOLOGY

Many studies of thermionic power systems have been made over the last 10 years. Early studies were largely parametric in nature, and were aimed at establishing design constraints on thermionic converters that could serve as fuel elements in nuclear reactors. These studies (References 1, 2, 3, and 4) were concerned initially with reactor parameters, but were later broadened to include the other portions (shielding, heat rejection, and power conditioning) of the system. The emitter and overall fuel element dimensions of the E- and F-series TFEs were chosen on the basis of these studies.

Applications studies were performed to determine requirements for thermionic reactors for nuclear electric propulsion (References 5 and 6). These studies showed that a power level of 100 kWe was optimum for unmanned missions, but that the optimum was very broad, so that power levels up to 300 kWe were suitable.

Point design studies were made of thermionic power plants for several applications. One of the first studies (Reference 7) dealt with a 300 kWe unit-cell reactor, in which the fuel elements were single thermionic cells. This study included transient and control analyses and reliability analyses, which were later extended into further studies (References 8 and 9). Other point design studies treated 40 kWe systems for application to manned space stations (Reference 10) and systems ranging from 5 to 134 kWe for unmanned applications (References 11 and 12) including nuclear electric propulsion. Studies were also made comparing alternate point designs in the same power range (References 13 and 14). The comparisons included reactors with fast driver fuel elements (UN or UO_2) or moderating driver fuel elements (U-ZrH) as well as those in which all of the fuel is in the TFEs. The most recent design study dealt with thermionic reactors for undersea applications (References 15 and 16).

5.1 SYSTEM DESIGN

The objective of this subtask is to design thermionic reactor power systems for manned and unmanned auxiliary power applications in space, for unmanned space electric propulsion applications, and for undersea application.

The approach used is to study reactor and system parameters within application guidelines and constraints and within engineering and technology constraints of system components. A baseline design is selected to serve as an initial model for more detailed analysis of system components and their interaction. Improvements made to the baseline design result in a reference design upon which detailed engineering, operational and integration analyses are based.

Previous work includes a design study of 40 kWe power systems for a manned space laboratory. Two systems were designed: one for rigid boom-mounting on the space laboratory and the other for deployment at the end of a 3.2 km (2-mile) tether. Baseline designs were also prepared for 5 and 10 kWe systems for unmanned auxiliary power and for a 120 kWe power system for electric propulsion of unmanned spacecraft.

The major effort of the current fiscal year is the reference design of a reactor power system at 120 kWe for nuclear electric propulsion. The side thrust NEP spacecraft and the thermionic reactor power system are illustrated in Figure 5-1. This system design is defined by the specifications recently published in Reference 17. Nuclear and structural analyses of the reactor are discussed in Sections 4.1 and 4.2.

Fig. 5-1. Sidethrust NEP Spacecraft and 120 kWe NEP Thermionic Reactor Power System

In this reporting period, a design study was concluded for the application of thermionic reactors as undersea power plants. Point designs were completed for a 30 kWe compact power plant and a 5 kWe, natural-convection-cooled reactor plant for sea bottom application. The designs are described in a topical report (Reference 16) and summarized briefly in the following section.

5.1.1 Undersea Power Systems (D. Allen, C. Carney, W. Leech)

Baseline designs are complete for two undersea power plants. The first power plant embodies a compact, 30 kWe thermionic reactor. This plant is called NEMO (Nuclear Electricity for Marine Operations). The reactor employs the TRIGA* type U-ZrH rods as driver fuel. TRIGA reactors are the most widely used research reactors, with more than forty in operation around the world. NEMO has 33 thermionic fuel elements (TFEs) and 73 U-ZrH rods. Core cooling is by forced convection, and heat exchange to sea water is via a tube-in-shell heat exchanger. The result is a compact power plant. The NEMO power plant is shown in Figure 5-2. The core and pressure vessel are located centrally. They are surrounded by the lead gamma shielding and suspended from the roof of the secondary containment in a volume of borated water which serves as neutron shielding. Reactor auxiliaries are also submerged in the shield water.

The overall schematic flow diagram for NEMO is shown in Figure 5-3. The redundant use of pumps in both primary and secondary circuits can be seen. Double containment of primary coolant is accomplished by using double-walled tubes in the heat exchanger. The isolation valves on the seawater circuit can be closed simultaneously with a reactor trip as an additional containment feature. Reactor afterheat can be conducted and convected to the seawater through the shield water without overheating the core.

* TRIGA is a registered trademark of the Gulf Oil Corporation.

Fig. 5-2. NEMO Power Plant

The reactor control is accomplished by poison-backed drums on the periphery of the core. Shutdown rods driven into the central region of the core are also provided. The 18 drums are operated by 6 motors. The three rods are driven by spring force and held out by hydraulic pressure. Figure 5-4 is a control schematic for NEMO.

The second undersea system is a 5 kWe power plant similar in design to the widely used TRIGA family of research reactors. The reactor core is predominantly U-ZrH fuel rods cooled by water in natural convection. This power plant incorporates 9 TFEs and conditions the electrical power output to user requirements. Flow in both reactor primary and secondary coolant loops is by natural convection. Heat transfer to the sea is also by natural convection. This power plant is expected to find application to sea bottom installations and is termed the Thermionic Reactor for Installed Oceanic Service (TRIOS). The design was conceived with particular emphasis placed on reliable operation with features to facilitate infrequent maintenance. The power plant elevation is shown in Fig. 5-5.

Significant features of each system are compared in Table 5-1.

Shielding analyses were conducted for the NEMO and TRIOS systems. The geometrical models used are shown in Figures 5-6 and 5-7. Neutron fluxes and doses are calculated using a one-dimensional transport code. Gamma doses were calculated using a point-kernel model with exponential attenuation and appropriate build-up factors. Both primary and secondary gamma rays were considered.

The results of the studies are given in Table 5-2, which gives the dose at two different points outside the shield for each system.

Fig. 5-3. NEMO Power Plant Coolant Flow

Fig. 5-4. NEMO Reactor Control

Fig. 5-5. TRIOS Power Plant Elevation

(NOT TO SCALE)

ALL DIMENSIONS IN MILLIMETERS

Fig. 5-6. NEMO Shield Configuration

All dimensions in millimeters (mm)

Fig. 5-7. TRIOS Shield Configuration

TABLE 5-1

UNDERSEA REACTOR DESIGN SPECIFICATIONS

	<u>NEMO</u>	<u>TRIOS</u>
Thermal Power	750 kWth	600 kWth
Electrical Power (unconditioned)	30 kWe	5 kWe
Voltage	2.9 volts	as
Current	10,248 amperes	required
U ²³⁵ Core Loading	35.7 kg	21.2 kg
Thermionic Fuel Element Replacement Period	10,000 hours	10,000 hours
System Mass		
Dry (drained shield tank)	10,985 kg (24,166 lb)	29,000 kg (64,000 lb)
Total mass	31,957 kg (70,306 lb)	49,910 kg (110,000 lb)
Submerged weight	10,747 kg (23,644 lb)	15,500 kg (34,000 lb)
Overall Dimensions		
Diameter	2,438 mm (8 ft)	5,700 mm (18.7 ft)
Length	2,438 mm (8 ft)	9,160 mm (30 ft)
Maximum Surface Dose Rate	130,000 mrem/hr	
Seawater Throughput		
Flow rate	17.6 kg/sec (139,400 lb/hrs)	
Temperature increase	20°K (36°F)	

TABLE 5-2

UNDERWATER REACTOR DOSE SUMMARY

<u>NEMO</u>		
DISTANCE FROM CENTERLINE mm	GAMMA DOSE RATE mr/hr	NEUTRON DOSE RATE mrem/hr
1267.	25.1×10^3	56.0×10^3
2567.	26.6	~ 0.0
<u>TRIOS</u>		
2000.	5.92×10^3	<1.0
3000.	4.45×10^1	~ 0.0

5.2 SYSTEM ANALYSIS

The objective of this subtask is to develop and improve analysis methods, and to perform analysis on the thermionic reactor system. Areas of analysis include performance, safety, dynamics and control, reliability, vibration, radiation shielding, and operations.

The approach used for this subtask is to obtain versions of existing calculational programs developed by national laboratories and other contractors and to integrate them into the analysis of the thermionic reactor system. New programs are written where necessary.

Previous to this reporting period, several computer programs had been written or modified to perform analyses in the above areas. Some of the programs were in need of revision to reflect changes in reactor and system designs.

5.2.1 System Optimization Programs

Updating of the TIRP (ThermIonic Reactor Parameters) program continued. The hydraulics portion of the program was enlarged to calculate the total length of piping required outside the core for several arrangements of core, power conditioner, and radiator. The shielding portion of the code was updated. A subroutine (HTPIPE) was added to calculate the size, number, and mass of the heat pipes needed in the radiator. Calculational techniques for the heat pipes are based on work in Reference 18). An updated description of the program and instructions for preparation of input were prepared.

Work on a new computer program to calculate I-V characteristics was terminated. Calculations uncovered a basic theoretical error in the model of Hansen and Warner which was being used in the program.

5.2.2 Reliability Analysis

The reliability analysis of the NEP reference design was continued. A component functional listing was compiled and a preliminary assessment of the criticality of each of these functions was made. A partial failure mechanisms list was also compiled.

5.3. TECHNOLOGY EVALUATION AND DEVELOPMENT

The objectives of this subtask are to determine where technology which has been or is being developed can be applied to the thermionic reactor power system and to determine requirements for new technology. In future years, the development of new technology and adaptation of technology for the thermionic reactor power system will be an objective.

The approach will be to review literature and existing programs which may be applicable and to prepare a technology evaluation report and preliminary technology development plan which are revised and updated annually.

The preparation of the first technology evaluation report was begun in the latter part of FY 71. The completion of this report has been delayed due to a reduction in personnel and a reordering of priorities.

During the last four months, two sections of the technology evaluation report were written. The section on radiation shielding included lithium hydride neutron shields and gamma shields of tungsten, uranium, lead and alloys of tungsten and uranium. The section on radiators included heat pipes, heat pipe panels, finned tubes, emissive coatings, and brazed joints for bonding heat pipes to coolant ducts. These sections of the report and the previously written section on reactor components were consolidated in the close-out files.

5.4 APPLICATIONS ANALYSIS

The objectives of this subtask are the definition of power requirements and the evaluation of performance capabilities of thermionic nuclear electric systems for space applications.

Information exchange and liaison with potential users in government and aerospace organizations will be maintained. Modifications in the design of thermionic reactor systems will be implemented as necessary to meet integration requirements.

Previous effort in this area has established basic guidelines on power system selection for first generation nuclear electric propulsion missions and assessed the effect of reactor lifetime constraints on mission capability (Reference 19).

Mission analysis studies indicate that a nominal 100 kWe reactor power system with a Titan/Centaur or a Shuttle/Centaur is appropriate for most solar system missions with payloads under 2000 kg. Required thrust times for typical missions are in the range of 10,000 to 20,000 hours, excluding power use during coast period or at the target.

Other recent activities in this task have been related to identifying potential non-propulsion applications for thermionics. The following table lists those which had suggested further investigation:

<u>Application</u>	<u>Power Requirement</u>
DOD Satellite	5-10 kWe
Undersea Propulsion	30-50 kWe
Undersea Stationary Power	5-100 kWe

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APPENDIX A

TFE AND CELL TEST RESULTS

TFE AND CELL TEST RESULTS

This appendix consists of:

- Device, design, and processing parameter tables (Tables A-1 to A-3)
- Operational history tables, which give test details for all recent cell, capsule, and TFE tests (Tables A-4 and A-5)
- Summaries of individual tests, test history graphs, and neutron radio-graph results for in-pile tests, in the following order:

- TFE Tests

TFE 6F2	TFE 2F2	TFE 1F3
TFE 6F3	TFE 2F3	TFE 1F4
TFE 6F4	TFE 1F1	TFE 2E2
TFE 6F5		
TFE 2F1		

- Cell Tests

Mark VIIA IC-D3
Mark VI IC-I6

- Capsule Test

FC-3

- Out-of-Pile Test

Mark VIIA OC-A4

Gulf-GA-A--12608 #040106154

[illegible]

NOTES: PROPOSED ITEMS IN PARENTHESES
TBS - TO BE SELECTED
ALL DEVICES ARE POSITIVE POTENTIAL UNLESS NOTED
ALL DEVICES HAVE LIQUID CESIUM RESERVOIR UNLESS NOTED
ALL DEVICES USE SPRINGS FOR FUEL HOLDDOWN UNLESS NOTED

* - IC = IN-PILE CELL:
OC = OUT-OF-PILE CELL:
TFE IS DESIGNATED BY: NUMBER OF CELLS, CELL DESIGN
LETTER, SERIAL NUMBER (i.e.: 6F

** - TO OBTAIN FUEL VOLUME FRACTION AND OTHER QUANTITIES, USE A THEORETICAL DENSITY OF 13.0 g/cc FOR CARBIDE FUEL AND 10.9 g/cc FOR OXIDE FUEL.

Table A-1. Device, Design and Processing Parameters

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GWlf-GA-A--12608 #040106154

[illegible]

NOTES: PROPOSED ITEMS IN PARENTHESES
TBS - TO BE SELECTED
ALL TFEs ARE POSITIVE POTENTIAL UNLESS NOTED
ALL TFEs HAVE LIQUID CESIUM RESERVOIR UNLESS NOTED
AXIAL REACTOR PITCH = 2.90 inches/73.7 mm
ALL TFEs USE SPRINGS FOR FUEL HOLDDOWN UNLESS NOTED

* - IC = IN-PILE CELL;
OC = OUT-OF-PILE CELL;
TFE IS DESIGNATED BY: NUMBER OF CELLS, CELL DESIGN LETTER,
SERIAL NUMBER (i.e.: 6F1)

** - TO OBTAIN FUEL VOLUME FRACTION AND OTHER QUANTITIES, USE A THEORETICAL
DENSITY OF 13.0 g/cc FOR CARBIDE FUEL AND 10.9 g/cc FOR OXIDE FUEL.

Table A-2 Device Design and Processing

Table A-2. Device, Design, and Processing Parameters

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Gulf-GA-A--12608 # 040106154

[illegible]

NOTES:

PROPOSED ITEMS IN PARENTHESES
TBS - TO BE SELECTED

ALL TFEs ARE POSITIVE POTENTIAL UNLESS NOTED

ALL TFEs HAVE LIQUID CESIUM RESERVOIR UNLESS NOTED

AXIAL REACTOR PITCH = 2.50 INCHES (73.7 mm)

ALL TFE'S WITH SPRINGS FOR FUEL HOLDDOWN UNLESS NOTED

* - IC = IN-PILE CELL
IC = OUT-OF-PILE CELL

TFE IS DESIGNATED BY: NUMBER OF CELLS, CELL DESIGN LETTER,
SERIAL NUMBER (1.e.: 6F1)

** - TO OBTAIN FUEL VOLUME FRACTION AND OTHER QUANTITIES,
USE A THEORETICAL DENSITY OF 10.0 g/cc FOR CARBIDE
FUEL AND 10.9 g/cc FOR OXIDE FUEL

*** BASED ON A FABR. DENSITY OF 45% THEORETICAL.

Table A-3. Device, Design, and Processing Parameters

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Gulf-GA-A--12408 # 040106154

DEVICE*	SIGNIFICANT FEATURES	F E A T U R E S					EMITTER T.C.s	TEST ENVIRONMENT	TEST BEGIN	TEST ENDED	OPERATING HOURS	BURNUP		FAST FLUENCE ≥0.1 MeV x 10 ²⁰	AVERAGE TESTING CONDITIONS AND PERFORMANCE DATA										ELECTRODE POWER DENSITY W/cm ²			RELATIVE POWER****			RELATIVE EFFICIENCY			PRESSURE IN FISSION TRAP torr	NEUTRON RADIOGRAPH OR HOT CELL RESULTS					FAILURE MODE	CAUSE OF FAILURE	NUMBER OF REACTOR SCRAMS/ SHUTDOWNS	REFERENCE	DEVICE*																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																								
		CELL ORDER	EMITTER*/COLLECTOR	AREA (HOT) cm ²	GAP (HOT) mTls/mm	FUEL						f/cc** x 10 ⁻⁹	AT % U		EMITTER TEMPERATURE °K	COLLECTOR TEMPERATURE °K	SHEATH TEMPERATURE °K	CESIUM RESERVOIR TEMPERATURE °K	POWER INPUT WATTS	FUEL POWER DENSITY*** W/cc	LEAD POWER OUTPUT WATTS	ELECTRODE EFFICIENCY %	CURRENT DENSITY amp/cm ²	ELECTRODE VOLTAGE VOLT/CELL	AVERAGE	INITIAL	FINAL OR CURRENT	AVERAGE	INITIAL	FINAL OR CURRENT	AVERAGE	INITIAL	FINAL		HOURS	MAXIMUM EMITTER DIA INCREASE mTls/mm	MAXIMUM CLAD REACTED mTls/mm	MAXIMUM DIA FUEL SINTERING %	FUEL BEHAVIOR						OTHER																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																							
MARK VI EXPERIMENTAL CELLS																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																				
IC - 14	VACUUM VENTED UO ₂ FUEL	14	W(WF ₆)/Mo	12.5	8/0.20	UO ₂	2 IN FUEL	STATIC He, TITR	1-29-69	8-01-71	9754	31.	1.6	4	1840	989	-	611	770	284	66	13	9.8	0.61	6.0	6.0	5.1	1.22	1.20	1.26									ENVELOPE LEAK	NOT FOUND	16	10	GULF-GA-A11061	IC-14																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																								
IC - 15	CHLORIDE TUNGSTEN - DUPLEX EMITTER	15	W(WCl ₆)/Nb	12.5	9/0.23	90UC-102rC-4W	2 IN FUEL	STATIC He, TITR	10-10-69	1-17-72	8-02714-3791	7.3	0.38	3	~1830	884	-	624	500	168	40	11	7.8	0.48	3.6	5.0	3.5	0.77	1.05	0.77									ENVELOPE LEAK	NOT FOUND	5	4		IC-15																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																								
IC - 16	WELDED FINAL ASS'Y; DUPLEX EMITTER	16	W(WCl ₆)/Nb	11.6	8/0.20	UO ₂	-	STATIC He, TITR	4-08-72	1-22-73	6334	12.6	0.55	3.7	1820	814	-	603	494	177	56	15	7.9	0.62	4.9	5.8	4.5	1.01	1.15	0.87									NONE**		6	6		IC-16																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																								
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IC - B1	FIRST PROTOTYPE CELL-THERMAL CYCLE TEST	B1	W(WF ₆)/Mo	26	8/0.20	90UC-102rC	2 IN FUEL	STATIC He, TITR	9-25-65	10-06-65	164	-	-	-	2020	1000	-	598	-	-	120	-	6.0	0.83	5.0	-	-	-	-	-	-											NONE		11	1	GA-7499	IC-B1																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																					
IC - B2	FIRST LONG TERM PROTOTYPE	B2	W(WF ₆)/Mo	25.6	8/0.20	90UC-102rC	2 IN FUEL	STATIC Ar, TITR	9-12-66 12-06-66	12-06-66 8-10-67	0-1870 1870-4543	14	0.61	3	2070	973 955	-	-	1681	275	75 0	6 0	7.0 4.1	0.47 0	3.3 0	6.9 0	2.4 0	0.48 --	1.12 -	0.42 -											ENVELOPE LEAK	LEAK NOT FOUND	13	7	GA-7660	IC-B2																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																						
IC - D1	FUEL WITH TUNGSTEN ADDED	D1	W(WF ₆)/Nb	25.6	8/0.20	90UC-102rC-4W	2 IN FUEL	STATIC He, TITR	1-04-68	2-08-68	728	6.7	0.07	0.4	1960	967	-	658	1240	198	151	16	10.8	0.68	7.4	7.4	6.6	1.05	1.05	1.00											ENVELOPE LEAK	SUSPECT Cu/Nb WELD	9	1	GA-8901	IC-D1																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																						
IC - D2	TEST UO ₂	D2	W(WF ₆)/Nb	25.6	8/0.20	UO ₂	2 IN FUEL	STATIC He, TITR	6-29-67	9-24-67	2006	1.6	0.31	2	2030	943	-	668	1664	297	125	11	11.6	0.54	6.2	7.8	4.4	0.91	0.90	0.91											INTERELECTRODE SHORT	EMITTER SWELLING	11	1	GA-8901	IC-D2																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																						
IC - D3	REPEAT D2 TEST W/O T/C CLIPS	D3	W(WF ₆)/Nb	26.0	8/0.20	UO ₂	2 IN FUEL	STATIC He, TITR	9-8-72	11-13-72	0-281 281-1370	2.1	0.11	0.9	1820 1250	1020 1000	-	1003 1007	1130	169	85 0	11.6 0	7.5 0	0.52 0.08	4.1 0.0	4.4 0	-1 -	-1 -	-											INTERELECTRODE SHORT	EMITTER SWELLING	0	2		IC-D3																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																							
OC - A4	GRAPHITE SORPTION RESERVOIR; MAP MARK VIIA (OUT-OF-PILE)	A4	W(WF ₆)/Nb	26.7	8/0.20	-	4	VACUUM	12-05-68	12-1-72	814-24536 24536-26236	-	-	-	2053 ~1650	974 ~790	-	1091 1053	1536 840	-	132 50	12	9.6 5.4	0.61 0.42	5.9 2.3	7.0 -	-	8.0 -	1.00 -	0.53 -														OC-A4																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																								
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IC - C1	FIRST LARGE DIAMETER CELL	C1	W(WF ₆)/Nb	45.5	8/0.20	UO ₂	2 IN FUEL	STATIC He, TITR	2-15-68	5-08-68	1496	2.6	0.12	1	2080	966	-	671	2257	156	270	16	11.0	0.64	7.0	5.3	8.5	1.00	1.02	0.94												INTERELECTRODE SHORT	EMITTER SWELLING	3	3	GA-8924	IC-C1																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																					
IC - C2	BACKUP TO C1	C2	W(WF ₆)/Nb	45.5	8/0.20	UO ₂	2 IN FUEL	STATIC He, TITR	11-01-68	11-01-68	< 1	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-												ENVELOPE LEAK	EMITTER CRACK [FUEL CHIP]	0	1	GA-8985	IC-C2																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																					
IC - C3	TEST GRAPHITE SORPTION RESERVOIR	C3	W(WF ₆)/Nb	45.5	8/0.20	90UC-102rC-4W	2 IN FUEL	STATIC He, TITR	1-29-69	10-06-69	5445	4.8	0.22	3	~1970	978	-	1143	1832	79	124	9	8.5	0.39	3.2	2.0	2.3	0.55	0.39	0.43											ENVELOPE LEAK	EMITTER CRACK AT BOTTOM	12	3	GACD-11000	IC-C3																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																						
IC - C11	TEST CARBIDE FUEL	C11	W(WF ₆)/Nb	45.5	10.5/0.27	90UC-102rC-4W	2 IN FUEL	STATIC He, TITR	6-13-69	10-18-70	7881	6.8	0.32	3	1840	875	-	643	1760	76	134	10	7.0	0.49	3.4	4.3	3.7	0.94	0.94	0.97												ENVELOPE LEAK	EMITTER CRACK AT BOTTOM	8	8	GULF-GA-B11062	IC-C11																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																					
IC - C13	FUEL VENT TUBE. NEW PEDESTAL	C13	W(WF ₆)/Nb	46.0	10.5/0.27	UO ₂	-	STATIC He, TITR	8-28-71 9-03-71	0-138 4-03-72	138-3512	1.6	0.07	2	1790 1860	835 747	-	631	1351 742	70 38	141 0	15 0	6.0 0	0.66 2.18	3.9 0	- 0	- 0	- -	1.09 -	-											OPEN CIRCUIT	B/S BAR BREAK CONTAINMENT LEAK	5	3		IC-C13																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																						
IC - C14	DIMENSIONAL STABILITY OF CARBIDE FUELED MARK VIIB. FIRST RUPTURE DISC	C14	W(WCl ₆)/Nb	46.0	8.8/0.22	90UC-102rC-4W	-	STATIC He, TITR	9-05-70	**	1073	0.98	0.05	0.4	~1770	955	-	632	1860	82	132	9	6.9	0.49	3.4	4.5	3.0	0.92	0.94	0.88													IC-C14																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																									
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2E1	FIRST FLASHLIGHT CONFIGURATION	E2 E1	W(WCl ₆)/Nb	26.3	10/0.25	UO ₂	-	STATIC He, TITR	12-05-69	1-17-72	12535	24	1.1	9	1820	~1030	-	604	1990	173	119	12	8.4	0.46	4.0	6.1	0.7	0.78	0.96	0.18												INTERELECTRODE SHORT	EMITTER SWELLING	11	14		2E1																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																					
2E2	SAME AS 2E1	E4 E3	W(WCl ₆)/Nb	26.3	10/0.25	UO ₂	-	STATIC He, TITR	8-28-71	1-22-73	11084	22.2	1.36	8.7	1820	990	-	596	2070	178	203	17	8.2	0.69	5.6	4.8	5.8	1.05	1.06	0.94													NONE**		11	10		2E2																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																				
2E3	SAME AS 2E1 BUT CARBIDE FUELED	E6 E5	W(WCl ₆)/Nb	26.3	10/0.25	90UC-102rC-4W	-	STATIC He, TITR	10-22-70	3-30-71	3661	7.4	0.35	2	~1670	~1110	-	613	2200	182	79	7	5.9	0.40	2.5	4.1	2.2	-	0.87	-												ENVELOPE LEAK	IN CESIUM RESERVOIR AREA	1	3	GULF-GA-B11062	2E3																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																					
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LC-9	FIRST DUPLEX CLAD EMITTER	LC-9	W(WCl ₆)/Nb	16.1	9/0.23	-	4	VACUUM	12-20-66	1-22-73	46647	-	-	-	1970	1072	-	625	934	-	118	14	10.5	0.76	8.0	7.5	8.2	0.95	0.89	0.97													NONE**		-	58		LC-9																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																				
LC-10	CARBIDE FUEL SLABS IN EMITTER WALL	LC-10	W(WCl ₆)/Nb	16.5	8/0.20	90UC-102rC-4W	2	VACUUM	8-21-70	10-04-71	8244	-	-	-	1870	1072	-	627	876	-	74	12	10.6	0.50	5.2	5.9	5.3	0.69	0.83	0.76												ENVELOPE LEAK	COPPER TO NIOBIUM WELD	-	15		LC-10																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																					
LC-11	CARBIDE FUEL SLABS IN EMITTER WALL	LC-11	W(WCl ₆ -CAST)	18.6	8/0.20	90UC-102rC-4W	2	VACUUM	7-16-70	12-11-72	18569	-	-	-	1870	1072	-	626	966	-	106	13	10.6	0.61	6.5	6.5	6.2	0.92	0.92	0.89												NONE**		-	28		LC-11																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																					
LC-12	CARBIDE FUEL SLABS IN EMITTER WALL	LC-12	W(WCl ₆)/Nb	10.6	8/0.20	90UC-102rC-4W	2	VACUUM	1-09-73	1-22-73	313	-	-	-	1870	1073	-	623	814	-	54	8	10.6	0.55	5.8	6.2	5.8	0.84	0.88	0.84												NONE**		-	1		LC-12																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																					
FUEL-CLAD CAPSULES																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																				
V2E	CARBIDE CAPSULE (PLUMBROOK)	V2E	W(WF ₆)/INCONEL	25.9	27/0.68	90UC-102rC-4W	4 IN FUEL	STATIC He, PLUMBROOK	9-29-71	1-05-73	6300	14.9	0.74		1550	730	-	-		245																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																</

NOTES: TBS - TO BE SELECTED

* - IC = IN-PILE CELL; OC = OUT-OF-PILE CELL; TFE IS DESIGNATED BY:
NUMBER OF CELLS, CELL DESIGN LETTER, SERIAL NUMBER (I.E.: 6F1)

+ - TUNGSTEN FROM WCl_6 IS DEPOSITED OVER BASE OF TUNGSTEN FROM WF_6 (DUPLEX EMITTER).

++ - COLD, MACROSCOPIC FUEL VOLUME USED

+++ - W(WC1₆) EMITTER - RELATIVE TO CPOP; W(WF₆) EMITTER - RELATIVE TO OC-A4

★★ - OPERABLE 1/22/73

Table A-4. In-Pile and Out-of-Pile Operational History Table

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Gulf-6A-A--12608 #040106154

[illegible]

NOTES: TBS - TO BE SELECTED

* - IC = IN-PILE CELL; OC = OUT-OF-PILE CELL; TFE IS DESIGNATED BY:
NUMBER OF CELLS, CELL DESIGN LETTER, SERIAL NUMBER (I.E.: 6F1)

+ - TUNGSTEN FROM WCl_6 IS DEPOSITED OVER BASE OF TUNGSTEN FROM WF_6 (DUPLEX EMITTER).

++ - COLD, MACROSCOPIC FUEL VOLUME USED

+++ - W(WC1₆) EMITTER - RELATIVE TO CPOP; W(WF₆) EMITTER - RELATIVE TO OC-44

0
** - OPERABLE 1-22-73

Table A-5. In-Pile and Out-of-Pile
Operational History Table

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PRESENT STATUS

Hot cell examination in progress.

CHRONOLOGICAL TEST HISTORY

The TFE experienced six shutdowns at 1241, 2579, 4307, 5067, and 7685 hours and nine scrams at 116, 359, 1575, 1594, 4328, 4898, 4906, 5713, and 6756 hours.

At 765 hours the current density was raised to lower the emitter temperatures. Operating condition changes following startups at 1241, 1594, and 2579 hours are visible (Figs. A-1,2). From 2866 to 3226 hours a high emitter temperature was maintained for diagnostic purposes. The performance loss at 4307 hours followed a shutdown. At 4767 and 5713 hours the current density was reduced in order to maintain the emitter temperature with fission gas in the inter-electrode space.

PERFORMANCE HISTORY (Figures A-1 to A-4)

The TFE was initially operated for a period of 70 hours with emitter temperatures near 1700°K during which its performance was observed to fall, apparently as a result of a dropping emitter substrate work function. Subsequent operation near 1925°K resulted in a slow improvement in performance, diagnosed to be the result on an increase in the emitter substrate work function. Consequently, the TFE was operated with an average emitter temperature of 2000°K from 2866 to 3226 hours in an attempt to clean the emitter surface. A small improvement in performance resulted, substantiating the diagnosis. Subsequent operation at 1875°K was stable until the shutdown at 4307 hours at an electrode performance approximately two thirds that predicted by CPOP under the estimated operating conditions of the TFE. The source of the contaminant which is apparently affecting the electrode work functions in the TFE has not been identified.

The power output dropped at 4307 hours following a shutdown for neutron radiography. This drop is attributed to a lowering of the emitter temperature due to the presence of fission gas in the interelectrode space. The drop beginning at ~5200 hours has been attributed to the same cause. The TFE's performance fell again at 6400 hours, probably due to a continued increase in the interelectrode space fission gas pressure.

Intermittent shorting behavior has been observed beginning at ~4700 hours, probably as the result of interelectrode shorting in the top cell. More severe shorting to the sheath tube began at 7000 hours, possibly through the top cell's sheath insulator.

NEUTRON RADIOGRAPHY RESULTS (The 7685 hour data is shown in Fig. A-5)

Radiographs were taken at 1241, 2579, 4307, 5713, and 7685 hours.

Emitter Dimensional Stability: No swelling was observed in any emitter (≤ 0.10 mm [4 mils] on the diameter) prior to 5713 hours. At that time 0.53 mm (21 mils), 0.09 (4 mils), and 0.08 mm (3 mils) of diametral swelling was visible in emitters 1, 2, and 6 respectively. None (≤ 0.08 mm [3 mils]) was detectable in emitter 3, 4, and 5. At 7685 hours the observed diametral swellings in emitters 1 to 6 were 0.68 mm (28 mils), 0.07 mm (3 mils), 0.11 mm (4 mils), 0.40 mm (16 mils), 0.32 mm (12 mils), and 0.50 mm (20 mils).

Fuel-Clad Reaction Layers: In emitters 1 (86% enrichment) and 6 (78%) a maximum of 0.13 mm (5 mils) of clad had reacted at 1241 hours. No significant changes were noted in later radiographs.

In emitters 2 (34%) and 5 (34%) no reaction was detected (≤ 0.08 mm [3 mils]) at 1241 hours but reactions were detected in #5 at 2579 hours and #2 at 4307 hours (~0.13 mm [5 mils] of clad reacted). No changes were found at 5713 hours. At 7685 hours a maximum of 0.13 mm of clad had reacted in both emitters.

No reaction was detected in emitters 3 (20%) and 4 (20%) until 5713 hours. At 7685 hours a maximum of 0.13 mm of clad had reacted in both.

Fuel Behavior: Fuel sintering was observed in all emitters, with a maximum of 4% and 5% (on the diameter) in the central two emitters (3 and 4, enrichments 20%), 1% and 3% on emitters 2 and 5 (enrichments 34%), and 2% and 1% on emitters 1 and 6 (enrichments 86% and 78%). At 7685 hours the pellets in all emitters had returned to full diameter and were fuzing together. In addition, in many of the emitters the fuel was expanding into the central cavity and into the stem area.

Other: The titanium backup plate for the rupture disc in emitter 4 sagged. Those in emitters 3, 5, and 6 bowed up slightly.

At 5713 hours contaminants in the interelectrode space were found in cells 1, 4, and 6, and at 7685 hours contamination was visible in the interelectrode spaces of all cells.

Cracks were visible in the emitter clads of cell 1 at 5713 hours and cells 1 and 6 at 7685 hours. The seal convolution in cell 6 deformed around the emitter stem and may be cracked.

SUMMARY AND CONCLUSIONS

The thermionic performance of TFE 6F2 was low but fairly stable throughout the test due to the presence of an electrode contaminant of uncertain origin. A leak between the fission gas and interelectrode spaces apparently occurred in association with a shutdown at 4307 hours and the TFE's current density was adjusted to maintain its emitter temperature as the gas pressure built up. Probable intermittent shorting of the top cell was observed from ~4700 hours on. At 7000 hours a short to the sheath tube, possibly through the top cell's trilayers, occurred.

Neutron radiograph results indicated that fuel sintering occurred to various degrees in all emitters. Emitter dimensional changes generally followed the return of the fuel pellets to full diameter after sintering. Cracks were visible in the clads of those emitters that changed the most (1 and 6) at 7685 hours. A contaminant was visible in the interelectrode space of all cells at this time.

Fig. A-1. TFE 6F2 Operating History

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Fig. A-2. TFE 6F2 Operating History

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Fig. A-3. TFE 6F2 RELATIVE PERFORMANCE HISTORY

Fig. A-4. TFE 6F2 IV Characteristics

#1

#2

#3

#4

#5

#6

Neutron Radiographs of TFE 6F2 at 7685 Hours (0°)

Fig. A-5

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TFE 6F3

STATUS AS OF SHUTDOWN ON JANUARY 22, 1973

Hours: 8062 Relative Power: 0.92

Average Emitter Temperature: 1740°K

Performance: Dropping slowly with fission gas present in the
 interelectrode space

CHRONOLOGICAL TEST HISTORY

TFE 6F3 experienced seven shutdowns at 1728, 2488, 5106, 5438, 6683, 6985, and 8062 hours, and six scrams at 1733, 2319, 2328, 3134, 4177, and 6792 hours.

Reactor control rods near the TFE were raised at 1922 hours to preferentially raise the temperature of the TFE's upper cells, in an effort to achieve a better emitter temperature profile and to reduce performance cycling. The operating current density was reduced at 2170 hours to raise the emitter temperatures.

Testing was terminated at 8062 hours for programmatic reasons.

PERFORMANCE HISTORY (Fig. A-6 to A-9)

The performance of the TFE dropped during the initial 30 hours of testing before stabilizing. At ~120 hours the TFE's operating correlations were measured. They indicated the average collector temperatures were nearly optimum and there was no evidence that the collectors were running significantly hotter than the TFE sheath. Individual correlations between emitter and sheath temperatures were obtained.

The TFE's initial relative power was slightly better than expected considering the device's wide spacing (14 mils, $P_R \sim 1.05$). This, plus the fact that TFE 6F3's optimum cesium reservoir temperature was $\sim 10^\circ\text{K}$ lower than that observed in TFE 1F1 for similar operating conditions indicates an oxygen reservoir of some sort may have been present in the TFE.

Beginning at 829 hours the TFE experienced cyclic performance decreases during which its output power dropped $\sim 40\%$ for $1/2$ hour periods. These cycles have been attributed to the existence of a leak between the TFE's cesium and fission gas spaces which allows the accumulation of cesium in the cooler portions of the fission gas spaces in the TFE's lead. This cesium contacts hot regions of the lead at irregular intervals resulting in brief periods of high cesium pressure operation with concurrent low TFE power output. A total of 134 periods of cycling had been experienced as of 8062 hours. Operation with a cesium reservoir temperature below optimum has been found effective in reducing the performance cycles repetition rate, and the TFE has been operated slightly below optimum from ~ 1000 hours. Operating low in the core to increase the power to the TFE's upper cells has also proved helpful. The TFE design was modified to correct this deficiency.

Following a shutdown at 1728 hours evidence of fission gas in the inter-electrode gap was found. This is probably gas vented from the fuel at the time of the shutdown. The fission gas pressure acts to reduce emitter temperature in the TFE, but is not high enough to significantly affect thermionic performance.

Following the shutdown at 6683 hours, an internal short to the sheath tube, probably occurring through a sheath insulator in one of the TFE's upper cells, was observed. The resistance of the short appeared to increase slowly after the 6985 hour shutdown.

NEUTRON RADIOGRAPHY RESULTS

Radiographs were taken at 0, 1728, 3134, 5106, 6683, and 8062 hours. The last three sets are shown in Figs. A-10 to A-13.

Emitter Dimensional Stability:

Maximum Emitter Dimensional Change (on Diameter)

Cell	1728 hrs	3134 hrs	5106 hrs	6683 hrs	8062 hrs
1 (mm)	≤0.08	≤0.05	0.06	0.14	0.17
(mils)	≤3	≤2	2	6	7
2 (mm)	≤0.08	≤0.05	0.09	0.08	0.24
(mils)	≤3	≤2	4	3	9
3 (mm)	≤0.08	0.08	0.11	0.17	0.25
(mils)	≤3	3	4	7	10
4 (mm)	≤0.08	0.08	0.13	0.20	0.18
(mils)	≤3	3	5	8	7
5 (mm)	≤0.08	0.13	0.18	0.24	0.27
(mils)	≤3	5	7	9	11
6 (mm)	≤0.08	0.12	0.19	0.14	0.20
(mils)	≤3	5	7	5	8

Distortion was observed in the emitter bottoms of cells 1, 2, and 4 at 3134 hours and thereafter. The maximum distortion consisted of ~ 0.6 mm (25 mils) deflection downward at the center at 8062 hours (cell 2). No distortion was observed in cells 3 or 6 (≤ 0.1 mm [5 mils]). A downward deflection of ~ 0.3 mm (10 mils) was visible in cell 5 at 8062 hours.

Fuel-Clad Reaction Layers: No evidence of fuel-clad reactions has been observed.

Fuel Behavior: The fuel had fully redeposited by 1728 hours except possibly at the top and bottom corners. Isothermal cavities are visible in all emitters. The fuel cover plates in most emitters were distorted slightly (~ 0.5 mm [20 mils]) by the fuel pushing upward at 3134 hours and thereafter.

Other: The titanium backup plates in emitters 2, 4, and 6 were pushed up 3-4 mm at 1728 hours, indicating that the titanium rupture disc had bonded to the plate and that fission gas pressure had then pushed the assembly up.

SUMMARY AND CONCLUSIONS

Despite the presence of a leak between the fission gas and cesium spaces in the TFE, its thermionic performance prior to 6683 hours was high and stable. The TFE is subject to power loss cycles of $\sim 1/2$ hour duration at irregular intervals due to cesium behavior in the fission gas spaces. This condition occurs as a result of capsule heat transfer characteristics and would not occur in a thermionic reactor. There is evidence that a partial sheath insulator short after the 6683 hour shutdown has reduced the TFE's performance, and the short resistance seems to be increasing.

Neutron radiographs indicate the presence of swelling in all emitters, with the maximum 0.27 mm (11 mils) in emitter 5 at 8062 hours.

Fig. A-6. TFE 6F3 Operating History

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Fig. A-7. FTE 6F3 Operating History

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Fig. A-8. TFE 6F3 RELATIVE PERFORMANCE HISTORY

Fig. A-9. TFE 6F3 IV Characteristics

#1

#2

#3

#4

#5

#6

Neutron Radiographs of TFE 6F3 at 5106 Hours (120°)

Fig. A-10

#1

#2

#3

#4

#5

#6

Neutron Radiographs of TFE 6F3 at 6683 Hours (120°)

Fig. A-11

#1

#2

#3

0°

120°

240°

Neutron Radiographs of TFE 6F3 at 8062 Hours

Fig. A-12

A-37

#4

#5

#6

0° 120° 240°
Neutron Radiographs of TFE 6F3 at 8062 Hours

Fig. A-13
A-38

TFE 6F4

STATUS AS OF SHUTDOWN ON JANUARY 22, 1973

Hours: 2956 Relative Power: 1.00
Average Emitter Temperature: 1700°K
Performance: Stable

CHRONOLOGICAL TEST HISTORY

This test has experienced four shutdowns at 332, 1577, 1879, and 2956 hours and one scram at 1686 hours. Operating correlation data was obtained during the first 250 hours of testing. The reactor power was raised at 238 hours. After 311 hours the current was adjusted to provide the lowest emitter temperature consistent with a lead power output of 700 watts. An emitter temperature of 1700°K was maintained after 668 hours.

The TFE's input power was temporarily low following the 1577, 1686, and 1879 hour startups.

Testing was terminated at 2956 hours as a result of the program closeout.

PERFORMANCE HISTORY (Fig. A-14 to A-17)

The TFE's performance initially was quite high with a $P_R > 1.1$, dropping during the first one thousand hours to a $P_R = .95$. During this period the efficiency of the second cell down in the TFE decreased to a degree which explains much of the performance loss. Its optimum cesium reservoir temperature is high indicating an emitter contaminant may be present. The high performance of the rest of the TFE indicates an oxygen source may also be present. Performance has been stable from 1000 hours, with lead power output of ~700 watts with an average emitter temperature of 1700°K.

NEUTRON RADIOGRAPHY RESULTS (Figs. A-18 to A-23)

Radiographs were taken at 0, 1577, and 2956 hours.

Emitter Dimensional Stability: No emitter dimensional changes were observed at 2956 hours (≤ 0.05 mm [2 mils] on the diameter).

Fuel-Clad Reaction Layers: No evidence of fuel-clad reactions was observed at 1577 hours. Four mils (0.10 mm) of cladding had reacted at 2956 hours in cells 1 and 6.

Fuel-Behavior: No fuel sintering has been observed ($\leq 0.5\%$)

SUMMARY AND CONCLUSIONS

Following an initial performance drop apparently associated with the second cell down in the TFE, the test has operated stably with at $P_R \approx 1.0$.

No emitter swelling or fuel sintering has been observed.

Fig. A-14. TFE 6F4 Operating History

Fig. A-15. TFE 6F4 Operating History

Fig. A-16. TFE 6F4 Relative Performance History

Fig. A-17. TFE 6F4 IV Characteristics

#1

#2

#3

0° 120° 240°
Neutron Radiographs of TFE 6F4 at 0 Hours

#4

#5

#6

0°

120°

240°

Neutron Radiographs of TFE 6F4 at 0 Hours

Fig. A-19

#1

#2

#3

0°

120°

240°

Neutron Radiographs of TFE 6F4 at 1577 Hours

Fig. A-20

A-47

0°

120°

240°

Neutron Radiographs of TFE 6F4 at 1577 Hours

Fig. A-21
A-48

#1

#2

#3

0°

120°

240°

Neutron Radiographs of TFE 6F₄ at 2956 Hours

#4

#5

#6

0°

120°

240°

Neutron Radiographs of TFE 6F4 at 2956 Hours

Fig. A-23

A-50

TFE 6F5

STATUS AS OF SHUTDOWN ON JANUARY 22, 1973

Hours: 1379 Relative Power: 0.98
Average Emitter Temperature: 1800°K
Performance: Stable

CHRONOLOGICAL TEST HISTORY

This test has experienced two shutdowns, at 302 and 1379 hours, and one scram at 109 hours. Between startup and 109 hours, it was operated in reactor hole C-10. Between 109 and 302 hours, it was operated in F-5, and since 302 hours in E-19. Operating correlations data was taken at 400 hours.

Testing was terminated at 1379 hours as a result of the program closeout.

PERFORMANCE HISTORY (Fig. A-24 to A-27)

The TFE's performance has been stable with a $P_R \approx 1.0$ since its operation was optimized at 400 hours.

NEUTRON RADIOGRAPHY RESULTS (Fig. A-28 to A-33)

Radiographs were taken at 0 and 1379 hours.

EMITTER DIMENSIONAL STABILITY

Maximum Emitter Dimensional Changes (On Diameter)

Cell	<u>1379 hrs</u>
1	0.11 (mm) 4 (mils)
2	≤0.05 (mm) ≤2 (mils)
3	0.20 (mm) 8 (mils)
4	≤0.05 (mm) ≤2 (mils)
5	0.07 (mm) 3 (mils)
6	0.14 (mm) 5 (mils)

The four fuel pellets in emitters 1, 2, and 5 are slightly cocked and the emitter clad has swollen to partially conform to the resulting staggered structure. Some swelling of these emitters was anticipated due to the designed interference fit between fuel and clad at operating temperatures. The slightly staggered nature of the fuel stack has apparently accentuated this swelling. There was less swelling in emitters 2 and 4, which was probably due to the specially shaped fuel pellets which were used to reduce the stress on the clad.

Fuel-Clad Reaction Layers: Three mils (0.08 mm) of clad had reacted in cells 1 and 6 at 1379 hours.

Fuel Behavior: The fuel in emitters 2 and 3 has sintered ~ 1/2%.

Other: One open fission gas vent hole is visible in both cells 5 and 6.

SUMMARY AND CONCLUSIONS

The performance of TFE 6F5 has been stable with a $P_R \sim 1.0$ since operating conditions were stabilized at 400 hours. The use of fuel pellets designed for interference fit at temperature in combination with stacking irregularities had generated some emitter swelling at 1379 hours.

Fig. A-24. TFE 6F5 Operating History

Fig. A-25. TFE 6F5 Operating History

Fig. A-26. TFE 6F5 Relative Performance History

Fig. A-27. TFE 6F5 IV Characteristics

#1

#2

#3

0°

120°

240°

Neutron Radiographs of TFE 6F5 at 0 Hours

Fig. A-28

A-58

#4

#5

#6

0°

120°

240°

Neutron Radiographs of TFE 6F5 at 0 Hours

Fig. A-29

A-59

#1

#2

#3

0°

120°

240°

Neutron Radiographs of TFE 6F5 at 1379 Hours

Fig. A-30
A-60

#4

#5

#6

0°

120°

240°

Neutron Radiographs of TFE 6F5 at 1379 Hours

Fig. A-31

A-61

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TFE 2F1

STATUS AS OF SHUTDOWN ON JANUARY 22, 1973

Awaiting hot cell examination.

CHRONOLOGICAL TEST HISTORY

TFE 2F1 has experienced six shutdowns at 1338, 3066, 3826, 6444, 6776 and 8021 hours, and seven scrams at 334, 352, 3086, 3657, 3663, 4422, and 5515 hours. Following the shutdown at 1338 hours the TFE was moved in core to allow testing at higher current densities. It was moved again at the 3066 hour shutdown to slightly reduce the power input. High emitter temperature tests for diagnostic purposes were performed at 250, 290, 1176 hours, and from 3976 to 4120 hours. At 5451 hours the current density was reduced to allow higher emitter temperature operation. Testing was terminated at 8021 hours due to emitter temperature uncertainties and low thermionic performance.

PERFORMANCE HISTORY (Figures A-32 to A-35)

During the initial 70 hours of testing with an average emitter temperature of $\sim 1400^{\circ}\text{K}$ the TFE's performance dropped rapidly and simultaneously the optimum cesium reservoir temperature rose, indicating a change in electrode work functions. The rate decreased following a raise in emitter temperature. The TFE was then twice operated open circuit with emitter temperatures of $\sim 1930^{\circ}\text{K}$ each time, resulting in improved performance and a lower optimum cesium reservoir temperature. The second test resulted in a relative performance compared to CPOP of 95%, a 20% improvement over the TFE's pre-heat treatment performance. These tests strongly indicated the presence of an emitter contaminant.

The performance improved further with another raise in operating emitter temperature ($\sim 1850^{\circ}$ avg.) to 105% of CPOP and remained at that level until

1176 hours when a third high temperature test to 1970° was performed. The third test resulted in a performance drop, associated with indications of cesium instabilities and possible shorting during the open circuit period of operation. The precise reason for the performance drop has not been ascertained. The post-test relative power of 90% improved until the reactor was shutdown for neutron radiographs at 1338 hours.

Operation with higher input power resulted in stable performance with a calculated $Pr \sim 1.15$. The emitter temperatures may actually have been hotter than calculated during this period, further explaining the high stable performance.

Following the 3066 hour shutdown, the TFE's performance began dropping again. Once again a period of operating at higher emitter temperatures, 1960°K , resulted in improved performance. Resumption of testing at $\bar{T}_E = 1830^{\circ}\text{K}$ resulted in further performance drops. At 5150 hours the emitter temperature was raised in an attempt to improve the performance. An immediate improvement in performance followed by a steady decline resulted. Commensurate with the decline a cyclic performance change with a period of ~one day was experienced. The cycles have been attributed to cesium refluxing in fission gas spaces in the TFE's electric lead, indicating a leak between the fission gas and cesium spaces in the TFE had occurred.

A comparison of IV curves taken during the test (Figure A-35) indicates that an emitter work function change, not a gas effect, was producing the performance loss between 3000 and 5150 hours, but fission gas in the interelectrode space may have been contributing to the performance loss after 5150 hours. The IV curves and the TFE's power output characteristic indicate an interelectrode short occurred in the top cell at ~6700 hours.

NEUTRON RADIOGRAPHY RESULTS (The last two sets are shown in Figs. A-36, 37)

Radiographs were taken at 0, 1338, 3066, 4472, 6444, and 8021 hours.

Emitter Dimensional Stability: No swelling was observed in either emitter until 8021 hours, when ~0.51mm (20 mils) was observed on the diameter of the top emitter. No swelling was observed in the bottom emitter (≤ 0.05 mm) at that time.

Fuel-Clad Reaction Layers: A maximum of 0.08mm (3 mils) of cladding had reacted in both emitters at 1338 hours (81% enrichment and 80% enrichment). No change was observed at 3066 or 4472 hours. A maximum of 0.13mm (5 mils) of clad had reacted at 6444 and 8021 hours.

Fuel Behavior: Both sets of fuel pellets were sintered a maximum of 2% at 1338 hours; the bottom cell pellets had returned essentially to full diameter at 3066 hours, but some of those in the top remained sintered at 4472 hours. At 6444 hours all pellets had returned to full diameter. The central fuel cavity was almost closed in both emitters at 6444 and 8021 hours, and the fuel had risen into the stem area 2mm in the top and 3mm in the bottom emitter at 8021 hours.

Other: The spring locators in both emitters were broken at startup. (The design was changed as a result of this observation). After 4472 hours a contaminant was visible in the interelectrode space of both cells. The quantity of contaminant visible at 8021 hours makes it very probable that both emitters are cracked. The collector bus bar tabs broke off in the first 1338 hours of operation. The TFE and bus bar configuration has remained stable since that time, however.

SUMMARY AND CONCLUSIONS

The performance of TFE 2F1, initially high, has suffered throughout testing due to the presence of an emitter temperature sensitive contaminant. Some

of this contaminant is visible in the neutron radiographs. There is evidence that a leak between the cesium and fission gas spaces occurred at 5150 hours, and an interelectrode short in the top cell at 6700 hours. No swelling was observed in the bottom emitter during the test. While the top emitter showed no swelling prior to 6444 it swelled ~20 mils between 6444 and 8021 hours. A maximum of 0.13mm (5 mils) of cladding had reacted in both emitters by 8021 hours. The collector bus bar tabs broke early in the test, probably due to the high temperatures that these leads were operated at, but the lead structure retained its current carrying capability.

Fig. A-32. FTE 2F1 Operating History

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Fig. A-33. FTE 2F1 Operating History

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Fig. A-34. TFE 2F1 Relative Performance History

Fig. A-35. TFE 2F1 IV Characteristics

0°

120°

240°

Neutron Radiographs of TFE 2F1 at 6444 Hours

Fig. A-36

A-73

0°

120°

240°

Neutron Radiographs of TFE 2F1 at 8021 Hours

Fig. A-37

A-74

TFE 2F2

STATUS AS OF SHUTDOWN ON JANUARY 22, 1973

Hours: 5574 Relative Power: 0
Average Emitter Temperature: 1820°K
Performance: Unstable due to gas and electrode
 contamination

CHRONOLOGICAL TEST HISTORY

The test has experienced five shutdowns at 2618, 2950, 4195, 4497, and 5574 hours and three scrams at 646, 1689, and 4304 hours. Following an initial period of reactor and TFE stabilization, the TFE's current was adjusted to provide a lead power of 320 watts. This resulted in an average TFE lead power production equivalent to that required in a 6F TFE producing a lead power of 1000 watts.

Evidence of a fission gas leak to the interelectrode space was observed after the 4195 hour shutdown, and the current density was dropped to maximize output power. At the 4497 hour shutdown the TFE was moved in core.

PERFORMANCE HISTORY (Figs. A-38 to A-41)

The relative performance of this TFE was high initially but had stabilized at ~ 1.0 by 100 hours. Collector optimum data taken at 190 hours indicated the TFE had an optimum collector temperature higher than OC-A4 and it was operating with a collector temperature below optimum. For calculating purposes, the optimum has been assumed to be 130° higher than that found in OC-A4.

Prior to the gas leak at 4195 hours, the performance of the TFE was stable within $\pm 10\%$. Following the gas leak, its performance became more erratic with performance cycles with periods of the cycle of 1 1/2 days occurring near the end of the test. Higher optimum cesium reservoir temperatures were observed during the lower portion of the cycle than during the higher. Whether these cycles and the general loss of performance is due to high gas pressures or electrode contamination has not been ascertained.

The cycles and performance loss are not entirely due to cesium instabilities, though they may be a contributing factor.

NEUTRON RADIOGRAPHY RESULTS (Final three sets, Figs. A-42 to A-44)

Radiographs were taken at 0, 646, 2618, 4195, and 5574 hours.

Emitter Dimensional Stability: No swelling was detected at 5574 hours in the top emitter ($\leq 0.05\text{mm}$ [2 mils] on the diameter). Swelling amounting to 0.08 mm (3 mils) on the diameter in the bottom emitter was noted at 646, 2618, and 5574 hours, but was not detected at 4195 hours.

Fuel-Clad Reaction Layers: No reaction layer was detected in either emitter at 646 hours but $\sim 0.08\text{ mm}$ (3 mils) of clad had reacted at 2618, 4195, and 5574 hours.

Fuel Behavior: Fuel sintering was observed in both emitters at 2618 and 4195 hours, and amounted to $\sim 1\%$ on the diameter in both. At 5574 hours the fuel had swollen back in both cells. The fuel was bulging upward $\sim 1.8\text{ mm}$ (80 mils) in the top cell and $\sim 1.0\text{ mm}$ (40 mils) in the lower at this time.

Others: The fuel holddown devices appear to be distorting slightly due to the upward swelling of the fuel.

SUMMARY AND CONCLUSIONS

The performance of TFE 2F2 was stable and as predicted for 4195 hours. It produced power equivalent to a lead power of 1000 watts in a 6F TFE during that period. Following a gas leak after a shutdown at 4195 hours its performance rapidly fell as the result of gas and electrode contamination and testing was terminated at 5574 hours.

No swelling of the top emitter was observed during the test and the bottom emitter swelled ~ 0.08 mm during the first 646 hours of operation and stabilized there. The fuel in both emitters sintered and had swollen back by 5574 hours.

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Fig. A-38. FTE 2F2 Operating History

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Fig. A-39. TFE 2F2 Operating History

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Fig. A-40. TFE 2F2 Relative Performance History

Fig. A-41. TFE 2F2 IV Characteristics

0°

120°

240°

Neutron Radiographs of TFE 2F2 at 2618 Hours

Fig. A-42

0°

120°

240°

Neutron Radiographs of TFE 2F2 at 4195 Hours

Fig. A-43

0°

120°

240°

Neutron Radiographs of TFE 2F2 at 5574 Hours

Fig. A-44

TFE 2F3

STATUS AS OF SHUTDOWN ON JANUARY 22, 1973

Hours: 2956 Relative Power: 0.8
Average Emitter Temperature: 1750-1850°K
Performance: Dropping slowly. Fission gas in interelectrode space.
Partial lower cell short.

CHRONOLOGICAL TEST HISTORY

This test has experienced four shutdowns at 332, 1577, 1879, and 2956 hours and one scram at 1686 hours. It was moved to a new reactor hole at 1879 hours.

Testing was terminated at 2956 hours as a result of program closeout.

PERFORMANCE HISTORY (Figs. A-45 to A-48)

Based on ignition current temperatures, the performance of TFE 2F3 was initially high with $P_R \sim 1.3$ and dropped smoothly prior to stabilizing at ~ 1100 hours. Sheath temperature-heat flux correlations and ignition current temperature estimates indicate the performance loss was due to fission gas in the interelectrode space. Slow response to cesium temperature changes confirms this diagnosis. The equilibrium thermal conductance in the gas was $k = 3.2 \times 10^{-4}$ watts/cm²°K. For purposes of calculating emitter temperature and P_R , the conductance of this gas was assumed to have increased linearly in time between 0 and 1070 hours.

Following the 1879 hour shutdown, collector bus currents indicated a partial short in the bottom cell of the TFE. This has been confirmed by neutron radiographs taken at 2956 hours. The resistance of the short appears to have decreased further around 2500 hours.

The rapid swelling visible in the bottom emitter of the TFE is not consistent with the low temperatures at which ignition currents indicate this device has operated. It is quite possible that the presence of the gas has affected the ignition current correlation. The high calculated relative power also indicates emitter temperature estimates may be low. An emitter temperature 75-100°K hotter than those calculated would result in relative powers nearer unity. If that is the case the average emitter temperature during the test may be more nearly 1850°K than the 1760°K calculated.

NEUTRON RADIOGRAPHY RESULTS (Figs. A-49 to A-51)

Radiographs were taken at 0, 1577, and 2956 hours.

Emitter Dimensional Stability:

	<u>Maximum Emitter Dimensional Change (on Diameter)</u>	
Cell	1577	2956
1 mm	≤0.05	≤0.05
mils	≤2	≤2
2 mm	0.10	0.33
mils	4	13

Fuel-Clad Reaction Layers: No evidence of fuel clad reactions has been observed.

Fuel Behavior: The fuel in both emitters had completely redeposited by 1577 hours. It was bulging upward ~ 1 mm in the top emitter and 0.8 mm in the bottom emitter at 2956 hours.

Other: The titanium backup plate for the rupture disc has risen up slightly in the top emitter at 1577 hours and was up 1.1 mm at 2956 hours. It had sagged 4 mm in the bottom emitter at 1577 hours and 4.8 mm at 2956 hours.

SUMMARY AND CONCLUSIONS

This test has suffered from interelectrode gas throughout testing, making estimates of the emitter temperature difficult. An average based on ignition current calculations is 1760°K . TFE performance would be more consistent with an 1850°K average.

The two emitters behaved quite differently. The top emitter, with a 2 mm clad, showed no signs of swelling although the fuel had bulged up ~ 1 mm and the rupture disk backup plate had swollen upward by 2956 hours, indicating possible fission gas containment. The lower emitter, with 1 mm clad swelled and partially shorted at 1879 hours. The fuel had bulged upward less than that in the top emitter at 2956 hours and a sagging rupture disk backup plate indicates the fission gases have probably been released from the emitter structure. Thus, despite the uncertainties in the average operating temperature, one can conclude that the 2 mm clad has significantly less clad swelling.

Fig. A-45. TFE 2F3 Operating History

Fig. A-46. TFE 2F3 Operating History

A-93

Fig. A-47. TFE 2F3 Relative Performance History

Fig. A-48. TFE 2F3 IV Characteristics

Neutron Radiographs of TFE 2F3 at 0 Hours

Fig. A-49

0°

120°

240°

Neutron Radiographs of TFE 2F3 at 1577 Hours

Fig. A-50

A-96

0°

120°

240°

Neutron Radiographs of TFE 2F3 at 2956 Hours

Fig. A-51

TFE 1F1

STATUS AS OF SHUTDOWN ON JANUARY 22, 1973

Awaiting hot cell examination.

CHRONOLOGICAL TEST HISTORY

The TFE's operating current has been adjusted through the test to maintain an emitter temperature of $\sim 1800^{\circ}\text{K}$ despite input power changes as required by other test devices. Seven shutdowns have been experienced at 543, 1784, 3122, 4850, 5610, 8228 and 8560 hours. Ten scrams have occurred at 138, 660, 902, 2118, 2135, 4870, 5450, 5456, 6256, and 7299 hours. Periods of changing operating conditions primarily due to reactor power stabilization are visible following most startups. The reactor startup following the 543 hour shutdown extended over several days, as required for test initiation of TFE 6F2.

PERFORMANCE HISTORY (Figs. A-52 to A-55)

The initial operating conditions of the TFE were established at 117 hours. The emitter temperature was estimated at that time on the basis of ignition current data, and the collector temperature was estimated on the basis of OC-A4 data and collector optimum data taken with the TFE at 1160 hours. This data indicated the collector was running 200° hotter than the sheath tube. The electrode power performance of TFE 1F1 based on these estimates was 7% better than that predicted by CPOP. Between 4000 and 8228 hours, the thermionic performance of 1F1 decreased slowly at the rate of 6%/10,000 hours. The decrease may have been the result of nonuniform interelectrode spacings resulting from emitter swelling. During the 8228 hour shutdown the TFE developed a partial interelectrode short which did not clear on start up. Testing was consequently terminated at 8560 hours.

SHEATH INSULATOR TEST HISTORY (Figs. A-56 to A-57)

At 1352 hours testing of the TFE's sheath insulator was initiated with the application of a 5.0 volt potential between the emitter and sheath tube. The sheath current rose at that potential from an initial value of approximately 8 ma to 40 ma, where it appeared to stabilize. At 5300 hours the voltage was raised to 7.5 with a resultant current of 180 ma. At 5900 hours the voltage was raised to 10.0, resulting in operation with some ignited currents at 600 ma. At 6400 hours the current ignited further and stabilized with a current near one ampere. The test history and samples of the sheath tube-emitter IV characteristics at different times during the test are shown in the figures. The source or sources of the discharge currents (which dissipate a total of ~10 watts) was not ascertained.

NEUTRON RADIOGRAPHY RESULTS (Final 2 sets, Fig. A-58)

Radiographs were taken at 0, 1784, 3122, 4850, 6256, 8228, and 8560 hours.

Emitter Dimensional Stability: No emitter swelling was observed ($\leq 0.10\text{mm}$ [4 mils] on the diameter at 1784 or 3122 hours, but the clad had swollen a maximum of 0.14mm (6 mils) at 4850 hours, 0.27mm (11 mils) at 6256 hours, 0.32mm (13 mils) at 8228 hours, and 0.37mm (14 mils) at 8560 hours.

Fuel-Clad Reaction Layers: No evidence of fuel clad reactions was observed.

Fuel Behavior: The fuel had fully redeposited by 1784 hours. Cracks along the fuel edges have been visible since 4850 hours, but no venting cracks have been seen.

Other: The titanium backup plate for the rupture disc appears to have fallen on one side between 4850 and 6256 hours and it was as far down as the upper radiation shields at 8560 hours.

SUMMARY AND CONCLUSIONS

TFE 1F1 performed stably at 7% above its predicted level until 4000 hours, at which time it started decreasing at a rate of 6%/10,000 hours. This decrease may be due to the observed emitter swelling. Such swelling could result in both off cesium optimum operation of portions of the emitter and increased conduction cooling by interelectrode cesium vapor, resulting in reduced efficiency. Testing was terminated at 8560 hours following an inter-electrode short which occurred during a shutdown at 8228 hours.

Sheath insulator testing was conducted from 1352 hours and at test termination the TFE was supporting a stable discharge current of ~1 amp at 10 volts.

Fig. A-52. TFE 1F1 Operating History

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Fig. A-53. TFE 1F1 Operating History

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Fig. A-54. TFE 1F1 RELATIVE PERFORMANCE HISTORY

1F1 TFE IV CHARACTERISTICS

Fig. A-55

TFE 1Fl Parasitic Discharge Test History

Fig. A-56

1.	28 October 1971	1352 Hours
2.	1 December 1971	2044 Hours
3.	5 January 1972	2837 Hours
4.	18 February 1972	2837 Hours
5.	3 April 1972	4848 Hours
6.	23 May 1972	6256 Hours

● Operating Point 23 May 1972

Fig. A-57. TFE 1F1 Parasitic Discharge IV Characteristics

0°
8228 Hours

120°
8228 Hours

240°
8228 Hours

0°
8560 Hours

120°
8560 Hours

240°
8560 Hours

Neutron Radiographs of TFE 1F1

Fig. A-58

TFE 1F3

PRESENT STATUS

Awaiting hot cell examination.

CHRONOLOGICAL TEST HISTORY

TFE 1F3 experienced two shutdowns at 760 and 3378 hours. It has also experienced five scrams, at 20, 591, 598, 1406 and 2449 hours. The operating current density was lowered at 140 hours from 7.0 to 6.0 A/cm² to raise the emitter temperature. The sheath temperature was lowered at 615 hours from 1150°K to 1090°K to allow operation nearer the optimum collector temperature. From 910 hours to 1055 hours the emitter temperature was raised from 1860°K to 1930°K in order to ascertain its affect on performance. At 2290 hours the TFE was lowered in the core in an attempt to stop cesium refluxing. Testing was terminated at 3378 hours.

PERFORMANCE HISTORY (Fig. A-59 to A-62)

TFE 1F3 is unique in that it was fabricated with a hole between the fission gas and cesium spaces. Fission gas effects must therefore be a prime suspect for any performance changes seen.

The performance of the TFE has varied in time from the beginning of testing. While it initially had a relative power of ~ 1 , it experienced a large loss of performance between 250 and 450 hours. The prime cause of the performance loss appears to be a change in collector work function. Emitter work function changes would be reflected in optimum cesium reservoir changes which have not been observed. Emitter temperature loss due to gas would result in a loss in open circuit voltage, as would a raise in collector temperature due to trilayer unbonding. The changes in V_{oc} which were observed could be attributed almost entirely to changes in sheath temperatures. In addition, the horizontal shift nature of the IV curves is indicative of ϕ_c changes as opposed to emitter temperature or work function changes. Changes have also been observed during the test in the TFE's response to collector temperature variations, suggesting changes either in collector-sheath tube temperature differences or collector work function. The presence of fission gas must

remain suspect since the interelectrode and fission gas spaces in this TFE are connected. It should be noted that TFE 1F2 had fuel similar to TFE 1F3, and preliminary hot cell results indicate a very low fission gas release rate for that device.

Cesium refluxing in the vicinity of the TFE lead seal was observed beginning at 2000 hours. The refluxing resulted in intermittent shorts lasting typically for tens of hours. At 2290 hours the TFE was lowered in the core and the current density increased to maintain the emitter temperature. The resultant increased heating in the stem area cleared the shorting. Intermittent shorting, possibly through the sheath insulator, began at 3200 hours, however, and the TFE was partially but stably shorted when the test was terminated at 3378 hours.

SHEATH INSULATOR TEST HISTORY (Figures A-63, 64)

Discharge testing of TFE 1F3 was initiated at 64 hours with a voltage of 5.0 volts and a current of 275 ± 25 ma. This current remained steady until 615 hours. At that time the collector temperature was lowered 160°K and the sheath current fell from 260 ma to as low as 150 ma. At 621 hours the current rose rapidly as a result of a collector-sheath tube short, and testing was terminated.

NEUTRON RADIOGRAPH RESULTS (The last set is shown in Fig. A-65)

Radiographs were taken at 0, 1406, and 3378 hours.

Emitter Dimensional Stability: No detectable swelling (≤ 0.05 mm [2 mils] on the diameter) was observed at 1406 hours. At 3378 hours the emitter diameter had increased a maximum of 0.07 mm (3 mils).

Fuel Clad Reaction Layers: Approximately 0.08 mm (3 mils) of clad had reacted by 1406 hours. The amount reacted had not increased by 3378 hours.

Fuel Behavior: There is some fuel deposition on the clad I.D. visible opposite all four fuel pellets. The fuel was sintered a maximum of 1% on the diameter at 1406 and 3378 hours.

SUMMARY AND CONCLUSIONS

The performance of TFE 1F3 has been unstable and poor due at least in part to collector work function changes. No definite evidence of gas in the interelectrode space has been found. Sheath insulator testing was terminated at 621 hours because of shorting, primarily to the collector. The test was terminated at 3378 hours with possible shorting through the sheath insulators.

The emitter diameter had increased a maximum of 0.07 mm (3 mils) at 3378 hours.

Fig. A-59. TFE 1F3 Operating History

Fig. A-60. 1F3 Operating History

Fig. A-61. TFE 1F3 Relative Performance History

Fig. A-62. TFE 1F3 IV Characteristics

TFE 1F3 Emitter-Sheath Potential Test

Fig. A-63. TFE 1F3 Parasitic Discharge Test History

11 April 72 64 Hours

● Operating Point Hours

TFE 1F3 Sheath-Emitter IV Characteristics

Fig. A-64. TFE 1F3 Parasitic Discharge I-V Characteristics

0°
3378 Hours

120°
3378 Hours

240°
3378 Hours

Neutron Radiographs of TFE 1F3 at 3378 Hours

Fig. A-65

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TFE 1F4

STATUS AS OF SHUTDOWN ON JANUARY 22, 1973

Hours: 4928 Relative Power: 0.40
Average Emitter Temperature: 1840°K
Performance: Dropping

CHRONOLOGICAL TEST HISTORY

Testing of TFE 1F4 was initiated on June 15, 1972. The secondary containment gas was changed at 140 hours to lower the collector temperature. The TFE has experienced five shutdowns at 1972, 2304, 3549, 3851, and 4928 hours, and two scrams at 1043 and 3658 hours.

Testing was terminated at 4928 hours as a result of program closeout.

PERFORMANCE HISTORY (Fig. A-66 to A-70)

The performance of this TFE was stable from startup to ~ 3500 hours. The apparent change in emitter temperature, input power, and relative performance at 140 hours are the result of changing operating correlations concurrent with the secondary gas change.

The initial relative power and optimum cesium reservoir temperature of this TFE were comparable to those expected of a nonoxygenated device. The low optimum collector temperature suggested additive performance. Collector temperatures were varied in an attempt to observe an oxygen additive effect on the IV curves as shown in the figures. No effect was found, and consequently, the presence of additive performance remained unconfirmed.

Following the 3549 hour shutdown, the performance dropped and it continued to fall until test termination at 4928 hours. Analysis of the device's IV curves indicate the probable reason for the performance loss was an emitter work function change. A less likely alternative is fission gas in the interelectrode space, which cannot be completely ruled out.

It is interesting to note that the performance loss occurred in approximately the same time some of the fuel pellets returned to full diameter.

SHEATH INSULATOR TEST HISTORY (Fig. A-71, 72)

Discharge testing of TFE 1F4 was initiated at 193 hours with a voltage of 5.0 volts. The current remained stable at 1 ma. At 1700 hours the voltage was raised to 10.0 volts with a stable current resulting of 4 ma. At 1980 hours the polarity was reversed and the sheath tube made positive with respect to the emitter. The insulator resistance fell, and at 2320 hours it developed a bistable mode as shown in Fig. A-72. Currents could be driven stably for short periods on either IV curve. The insulator would switch from one mode to another for reasons which have not been ascertained. Tests indicate the current flow was through the collector sheath insulator. After 2400 hours testing was terminated and current was driven by the TFE potential alone.

NEUTRON RADIOGRAPHY RESULTS (Figs. A-73, 74)

Radiographs were taken at 0, 1972, 3549, and 4928 hours.

Emitter Dimensional Stability: No detectable change in emitter diameter (≤ 0.05 mm [2 mils]) was measured at 1972, 3549, or 4928 hours.

Fuel Clad Reaction Layers: Approximately 0.08 mm (3 mils) of clad had reacted by 1972 hours. No change was noted at 3549 hours or 4928 hours.

Fuel Behavior: The fuel was sintered a maximum of 1/2% on the diameter at 1972 hours, and 1% on the diameter at 3549 hours. The top two pellets remained sintered ~ 1% at 4928 hours, but the bottom two had swollen back to approximately the clad I.D.

SUMMARY AND CONCLUSIONS

TFE 1F4 operated stably with performance comparable to a similar but non-oxygenated TFE for ~3500 hours. It is possible that additive performance was masked by electrode contaminants in the device. It is also possible the processing through the 1520°K sheath tube bonding cycle during fabrication had a detrimental effect on the oxygenated molybdenum layer on the collector. The performance of the TFE dropped after 3500 hours, probably due to electrode contamination effects.

Sintering of the carbide fuel was a maximum of 1/2% at 1972 hours and 1% at 3549 hours. The bottom two pellets had swollen back by 4928 hours.

Fig. A-66. TFE 1F4 Operating History

Fig. A-67. TFE 1F4 Operating History

Fig. A-68. TFE 1F4 RELATIVE PERFORMANCE HISTORY

Fig. A-69. TFE 1F4 IV CHARACTERISTICS

Fig. A-70. TFE 1F4 IV Characteristics

Fig. A-71. TFE 1F4 Emitter Sheath
Potential Test

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Fig. A-72. TFE 1F4 Parasitic Discharge IV Characteristics

0°
0 Hours

120°
0 Hours

240°
0 Hours

0°
1972 Hours

120°
1972 Hours

240°
1972 Hours

Neutron Radiographs of TFE 1F4

Fig. A-73

0°
3549 Hours

120°
3549 Hours

240°
3549 Hours

0°
4928 Hours

120°
4928 Hours

240°
4928 Hours

Neutron Radiographs of TFE 1F4

Fig. A-74

A-133

TFE 2E2

STATUS AS OF SHUTDOWN ON JANUARY 22, 1973

Hours: 11084 Relative Power: 0.94

Average Emitter Temperature: 1840°K

Performance: Dropping slowly

CHRONOLOGICAL TEST HISTORY

The TFE's operating current has been adjusted throughout the test to maintain an emitter temperature average of 1850°K. Ten shutdowns have been experienced, at 543, 1784, 3122, 4850, 5610, 8228, 8560, 9805, 10107, and 11084 hours. Eleven scrams have occurred at 138, 660, 902, 2118, 1235, 4870, 5450, 5456, 6256, 7299, and 9914 hours. Periods of operating condition change, primarily due to reactor power stabilization, are visible following most startups. The TFE was moved in core following shutdown at 543 hours.

Testing was terminated at 11084 hours as a result of the program closeout.

PERFORMANCE HISTORY (Fig. A-75 to A-78)

The TFE's initial conditions were established at 116 hours. Emitter temperatures calculated from ignition current measurements indicate the two emitters in TFE 2E2 are operating at two temperatures ~130°K apart with the bottom probably hotter. The reported temperatures are the averages of these two emitter temperatures. Its collector temperature was estimated using collector optimum data taken at 1150 hours in conjunction with OC-A4 data. On the basis of these measurements the collectors of TFE 2E2 are estimated to be running about 130°K hotter than the collector heat rejector. Based on these estimates at 116 hours the TFE 2E2 had an electrode performance 6% better than predicted by CPOP.

The TFE's performance was stable within $\pm 10\%$ throughout the test, but since 8000 hours there were indications of a slow performance drop possibly due to emitter swelling.

NEUTRON RADIOGRAPHY RESULTS (Last three sets, Figs. A-79 to A-81)

Radiographs were taken at 0, 1784, 3122, 4850, 6256, 8228, 9805, and 11084 hours.

Emitter Dimensional Stability:

	<u>Maximum Emitter Dimensional Changes (on Diameter)</u>						
	1784	3122	4850	6256	8228	9805	11084
Top Emitter (mm)	≤ 0.10	≤ 0.10	≤ 0.10	0.11	0.14	0.17	0.22
(mils)	≤ 4	≤ 4	≤ 4	4	6	7	9
Bottom Emitter (mm)	≤ 0.10	0.10	0.18	0.31	0.33	0.36	0.46
(mils)	≤ 4	4	7	12	13	14	18

Fuel-Clad Reaction Layers: No evidence of fuel-clad reactions has been observed.

Fuel Behavior: The fuel was fully redeposited at 1784 hours. None had moved above the radiation shields in either emitter or below the pedestal in the bottom emitter, but a small amount of fuel had moved below the pedestal in the top emitter at 3122 hours. The fuel in the top emitter was cracked through in all radiographs after startup except those at 11084 hours (presumably due to differential thermal expansion on cooling). Cracks across the fuel have also been observed in the bottom cell. Cracks appear at the upper top corners of the fuel in the top emitter.

Other: The fission gas vent holes are visible in both cells and appear open.

SUMMARY AND CONCLUSIONS

The performance of the TFE has been stable within $\pm 10\%$ throughout 11084 hours of testing, although there is evidence of a slow drop after 8000 hours. Swelling is visible in both emitters, but is significantly greater in the bottom emitter which operated at a higher temperature. The swelling may be contributing to the slow performance loss. No evidence of shorting behavior was observed.

Fig. A-75. TFE 2E2 Operating History

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Fig. A-76. TFE 2E2 Operating History

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Fig. A-77. TFE 2E2 RELATIVE PERFORMANCE HISTORY

Fig. A-78. TFE 2E2 IV Characteristics

0°

120°

240°

Neutron Radiographs of TFE 2E2 at 8228 Hours

Fig. A-79

0°

120°

240°

Neutron Radiographs of TFE 2E2 at 9805 Hours

Fig. A-80

0°

120°

240°

Neutron Radiographs of TFE 2E2 at 11084 Hours

Fig. A-81

MARK VIIIA IC-D3

STATUS AS OF SHUTDOWN ON JANUARY 22, 1973

Awaiting hot cell examination.

CHRONOLOGICAL TEST HISTORY

Following an initial period of operation at 9 A/cm^2 , the cell was taken open circuit at 280 hours. The reactor was shut down briefly at 332 hours. Between 337 and 341 hours, the emitter temperature fell several hundred degrees. Testing was terminated at 1577 hours.

PERFORMANCE HISTORY (Fig. A-82, 83)

The initial performance of IC-D3 was difficult to determine accurately because of uncertainties in ignition current determined emitter temperature resulting from uncertainties in cesium pressure. These uncertainties arise from the fact that the cesium reservoir is graphite-cesium lamellar compound. The best estimate is a value of $P_R = 1.0 \pm 0.2$. These estimates of the emitter temperature based on ignition currents indicate it was running approximately 150°K colder than the thermocouple temperatures.

At 280 hours the cell was taken open circuit and the cesium reservoir temperature reduced in order to maximize the emitter temperature rise. The emitter thermocouple temperature rose to $\sim 2450^\circ\text{K}$, indicating the emitter surface temperature was near 2300°K . The emitter thermocouple temperature dropped slowly until 337 hours, probably as the result of degradation. Between 337 and 341 hours the temperature fell to $\sim 1200^\circ\text{K}$. The cesium reservoir temperature was raised between 500 and 1000 hours to determine if interelectrode gas might have been the source of the cooling. No such evidence was found. The probably reason for the drop, substantiated by neutron radiograph pictures, was contact between the emitter and collector as a result of emitter swelling.

NEUTRON RADIOGRAPHY RESULTS (Fig. A-84)

Radiographs were taken at 0 and 1577 hours.

Emitter Dimensional Stability: The emitter diameter had increased a maximum of 0.30 mm (12 mils) at 1577 hours.

Fuel-Clad Reaction Layers: No evidence of fuel clad reaction layers was observed.

Fuel Behavior: The fuel was fully redeposited at 1577 hours.

SUMMARY AND CONCLUSION

Following 280 hours of normal operation IC-D3 was taken open circuit to determine the effect of high emitter temperature operation. Such operation, with an emitter temperature near 2300°K, resulted in emitter swelling sufficient to short out and cool the emitter in 57 hours. The pressure exerted on the emitter came both from the UO_2 fuel and the 1 atmosphere helium pressure in the fuel cavity.

Fig. A-82. Mark VIIA IC-D3 Operating History

Fig. A-83. Mark VIIA IC-D3 Operating History

0°
1577 Hours

120°
1577 Hours

240°
1577 Hours

0°
0 Hours

120°
0 Hours

240°
0 Hours

Neutron Radiograph of IC-D3

Fig. A- 84

Mark VI IC-I6

STATUS AS OF SHUTDOWN ON JANUARY 22, 1973

Hours: 6334 Relative Power: 0.87

Average Emitter Temperature: 1830°K

Performance: Dropping slowly

CHRONOLOGICAL TEST HISTORY

IC-I6 has experienced six shutdowns at 760, 3378, 3710, 4955, 5257, and 6334 hours. It has also experienced six scrams at 20, 591, 598, 1406, 2449, and 5064 hours. Its performance characteristics were mapped between 3270 and 3320 hours. The emitter temperature was cycled once a week to a temperature 100°K above the operating point for a period of ~ 4 hours beginning at 3378 hours.

Testing was terminated at 6334 hours for programmatic reasons.

PERFORMANCE HISTORY (Figs. A-85 to A-88)

The initial performance of the cell was high with a P_R of ~ 1.2, but it dropped and stabilized at a P_R ~ 1.0 during the first 400 hours of testing. A slow downward trend is evident after ~3000 hours. This may be due to interelectrode gap changes caused by emitter swelling. The scatter in the calculated data is the result of scatter in collector heat rejector power data which is used in the calculations, and does not represent real variations in the cell's operating conditions.

NEUTRON RADIOGRAPHY (Last three sets, Figs. A-89 to A-91)

Radiographs were taken at 0, 1406, 3378, 4955, and 6334 hours.

Emitter Dimensional Stability: No swelling was observed until 4955 hours, at which time the emitter diameter had increased by 0.11 mm (4 mils) relative to the emitter bottom. At 6334 hours it measured 0.08 mm (3 mils) relative to the bottom, but there was evidence the bottom had swollen 3-4 mils, making a total of 6-7 mils. The emitter temperature was cycled to a value 100°K higher than nominal for ~ 4 hours once a week. Eight such cycles were performed between 3378 and 4955 hours and seven between 4955 and 6334 hours. Therefore, these temperature cycles increased the emitter diameter about 0.5 mils (0.12 mm) per cycle.

Fuel-Clad Reaction Layers: No evidence of fuel clad reactions has been observed.

Fuel Behavior: As of 1406 hours the fuel had fully redeposited and had risen upward ~0.3 mm (10 mils) on one side, cocking the radiation shields slightly. at 3378 hours the radiation shields had been tilted up ~ 0.6 mm (25 mils).

SUMMARY AND CONCLUSIONS

Following an initial period of high performance, IC-I6 stabilized with relative power nearly that predicted. The performance has dropped slowly following the initiation of emitter temperature cycling, probably as the result of concurrent gap size variations.

Neutron radiograph measurements indicate the emitter temperature cycling has resulted in a faster swelling rate than observed in IC-T4, a test operated under similar conditions.

Fig. A-85. Mark VI IC-I6 Operating History

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Fig. A-86. Mark VI IC-I6 Operating History

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Fig. A-87. I6 RELATIVE PERFORMANCE HISTORY

Fig. A-88. IC-I6 IV Characteristics

0°
3378 Hours

120°
3378 Hours

240°
3378 Hours

Neutron Radiographs of Mark VI IC-I6

Fig. A-89

0°
4955 Hours

120°
4955 Hours

240°
4955 Hours

Neutron Radiographs of Mark VI IC-I6

Fig. A-90

Neutron Radiographs of Mark VI IC-I6

Fig. A-91

FUEL CLAD CAPSULE FC-3

STATUS AS OF SHUTDOWN ON JANUARY 22, 1973

Testing was terminated at 1379 hours for programmatic reasons.

CHRONOLOGICAL TEST HISTORY (Fig. A-92 to A-95)

This test experienced a scram at 109 hours, shutdowns at 302, and 1379 hours.

The initial 108 hours of testing were conducted with helium as the secondary containment gas. Neon gas was introduced into the secondary containment at this time in order to bring the emitter temperatures closer to their design values (1800°K). The data taken during the next 150 hours was used to derive the correlation between the emitter thermocouples and the average of the two thermocouples in the inconel sleeve (collector) surrounding each emitter. A correlation was also derived between the emitter thermocouples and the thermocouples placed on each tantalum emitter cap. In both of these correlations, it was assumed that the average emitter surface temperature was 50°K hotter than the emitter thermocouple temperature (based on 2D calculations). All measured and calculated temperatures are shown on the performance plots.

Most of the thermocouple changes observed in the first 200 hours of testing were probably the result of redistribution of the UO_2 fuel. At 236 hours the emitter thermocouple in the top emitter dropped 80°K, probably due to some flaw in the thermocouple itself. While the slow drop in the emitter thermocouple temperature visible in all the emitters later in the test may be evidence of fuel redeposition, it is more likely that it is the result of thermocouple degradation.

The input power values shown were derived from the correlation presented in Section 1 and the emitter temperature values.

NEUTRON RADIOGRAPH RESULTS (Figs. A-96 to A-99)

Radiographs were taken at 0 and 1379 hours.

Emitter Dimensional Stability

Maximum Emitter Dimensional Change (on Diameter)

Emitter	1379 Hours
1	≤ 0.08 (mm) ≤ 3 (mils)
2	≤ 0.05 (mm) ≤ 2 (mils)
3	0.05 (mm) 2 (mils)
4	0.10 (mm) 4 (mils)

Fuel-Clad Reaction Layers: No evidence of fuel-clad reactions were observed.

Fuel Behavior: The fuel had extensively redeposited in the top two emitters and appeared to be completely redeposited in the bottom two emitters at 1379 hours. Some top corner cracks were visible in the fuel in the top emitter.

Other: The fuel voids visible in the top two emitters are quite complex and the actual fuel internal geometry following redepositions is not clear.

SUMMARY AND CONCLUSIONS

There is evidence that some emitter swelling may be occurring in the bottom two emitters at 1379 hours. There is not enough data yet to confirm that the different fuel geometries in the top two emitters are reducing the swelling rate.

All four emitter thermocouples were working at 1379 hours, although there is evidence they may be suffering degradation.

Fig. A-92. FC-3-1 Operating History

Fig. A-93. FC-3-2 Operating History

Fig. A-94. FC-3-3 Operating History

Fig. A-95. FC-3-4 Operating History

#1

#2

0°
0 Hours

90°
0 Hours

135°
0 Hours

Neutron Radiographs of FC-3 Top Emitters

Fig. A-96

A-168

#3

#4

0°
0 Hours

120°
0 Hours

240°
0 Hours

Neutron Radiographs of FC-3 Bottom Emitters

#1

#2

0°

90°

225°

1379 Hours

1379 Hours

1379 Hours

Neutron Radiographs of FC-3 Top Emitters

A-98
A-170

#3

#4

0°

120°

240°

1379 Hours

1379 Hours

1379 Hours

Neutron Radiographs of FC-3 Bottom Emitters

Fig. A-99

A-171

MARK VIIA OC-A4

The test of Mark VIIA OC-A4 included performance mapping with a liquid cesium reservoir, performance mapping with a graphite sorption reservoir-cell combination, and life testing. Testing was terminated after a total of 26,296 hours of operation and a post-operative examination was completed.

CHRONOLOGICAL TEST HISTORY

The performance mapping of OC-A4, with both a liquid and a graphite sorption reservoir, was completed on March 13, 1969 after 814 hours of operation and life testing was initiated with an input power of 1535 watts and a current density of 10 A/cm². Degradation of the emitter thermocouples occurred during the final mapping of the cell, and the initial emitter temperature (2050°K) was estimated on the basis of the initial cell mapping. After 24536 hours of operation a remotely programmable driving supply was attached and the cell operated to test that unit. The testing was terminated after a total of 26296 hours of operation and a post-operative examination conducted.

PERFORMANCE HISTORY (Figs. A-100, A-101)

The initial electrode power density was 6.8 W/cm². The small abrupt changes in output power which occurred during the test are attributed to filament aging with concurrent changes in the emitter temperature profile. The cell's output appears to have slowly dropped a few per cent between initiation of the testing and 15,229 hours. At that time some low power diagnostic measurements were taken and the cell returned to an output power about 15% lower than previously obtained. The power output stabilized at a value 35% lower than the steady-state value. The optimum cesium reservoir temperature increased 25°C prior to the performance change at 15,229 hours. From that time to 24,056 hours, it increased another 20°.

The slow drop in performance during the first 15,229 hours of life testing has been attributed primarily to emissivity changes due to carbon on the collector. Changes in emitter and collector work function probably also contributed to the loss. The rise in optimum cesium reservoir temperature may have been the result of emitter work function changes, changes in the graphite reservoir sorption characteristics, or losses of cesium. A loss of ~ 110 mg of cesium is necessary to explain the full optimum temperature rise if it were the sole mechanism. In reality a combination of these mechanisms probably occurred.

The sudden loss of power at 15,229 hours was due to emitter contamination and a consequent work function change. The work function change following startup was sensitive to the cesium pressure history during the startup, indicating a very small amount of contaminant was present on the emitter and its loss rate to the collector was affected by cesium pressure. The most probable contaminant which explains the performance loss is titanium (from the Cu-Ti final closure braze).

Post-Operational Examination (Figs. A-102 to A-105)

A cross sectional view of OC-A4 is shown in Fig. A-102. Following mapping with the liquid reservoir, the cesium was transferred into the graphite and the liquid reservoir assembly was pinched off in the secondary pinch off area.

During disassembly the cell easily separated at the final closure braze joint where the collector meets the insulator seal. Much of the braze material appears to have been lost from the joint.

The collector and the I.D. of the insulator seal were found to be coated with a layer of carbon or carbide (Fig. A-103, 104). The coating over the collector appeared to be uniform and $\leq 5\mu$ thick, suggesting carbon transport probably occurred in a gaseous phase perhaps in a CO-CO₂ cycle rather than by surface migration. No coating was found on the emitter (Fig. A-103). An emission spectrographic analysis of the emitter surface showed the presence of niobium ($0.1\mu\text{g}/\text{cm}^2$), aluminum ($0.06\mu\text{g}/\text{cm}^2$) and silicon in some spots ($0.1\mu\text{g}/\text{cm}^2$).

A similar analysis of the collector revealed copper ($4.0\mu\text{g}/\text{cm}^2$), silicon ($0.4\mu\text{g}/\text{cm}^2$), silver ($0.2\mu\text{g}/\text{cm}^2$), titanium ($0.1\mu\text{g}/\text{cm}^2$), iron ($0.08\mu\text{g}/\text{cm}^2$), and aluminum ($0.08\mu\text{g}/\text{cm}^2$). The presence of copper and titanium confirm the final closure braze as a source of contamination. The small amounts of other impurities, particularly silicon, were probably introduced with the cesium and graphite. Microhardness measurements indicate no carburization of the emitter or collector deeper than 50μ had occurred. Similar results were found for the niobium pedestal which supported the graphite reservoir, although a carbide layer $\sim 3\mu$ thick is visible (Fig. A-105).

The graphite (Fig. A-104) sorption reservoir appeared to be in good condition, although it had partially delaminated along the layers. Such delaminations have been observed in the past shortly following cesium loading to levels equivalent to those in A4's reservoir, so delamination was probably not responsible for the changes during the test life.

SUMMARY AND CONCLUSIONS

Mark VIIA OC-A4 was operated a total of 26,296 hours, demonstrating the integrity of the envelope for that period. The use of a graphite sorption reservoir was demonstrated. A full performance map of the cell was obtained.

Post operational examination and test diagnostics show carbon was transported to the collector from the sorption reservoir, probably in a gaseous phase. Performance losses due to this were probably less than 5%. Carbide layer formation on the collector was $\leq 5\mu$ thick. The niobium pedestal used to support the graphite reservoir had a carbide layer $\sim 3\mu$ thick. The significant thermionic performance loss observed after 15,229 hours of life testing appears to have been associated with emitter contamination by the final closure braze material rather than the graphite reservoir.

Thus, the weakest portion of the cell was the final closure braze. TFE designs are of all welded construction. The use of a graphite sorption reservoir for periods as long as 20,000 hours would appear feasible, and a test of its affect on performance in an all welded system would be useful.

Fig. A-100. Mark VIIA OC-A4 Life Test History

Fig. A-101. Mark VIIA OC-A⁴ Life Test History

Fig. A-102. Mark VIIA OC-A4

Emitter

Insulator Seal

Fig. A-103. Mark VIIA OC-A4 Post-Test Appearance

Sorption Reservoir Section

Collector

Graphite-Cesium Sorption Reservoir

Fig. 104. Mark VIIA Post-Test Appearance

Collector X1000

Emitter X1000

Sorption Reservoir Niobium Support Surface X1000

Fig. 105. Mark VIIA OC-A4 Post-Operational Metallographic Examination

APPENDIX B

QUALITY ASSURANCE PLAN

APPENDIX B

QUALITY ASSURANCE PLAN

1.0 INTRODUCTION AND SCOPE

At the time of project termination, the Thermionic Reactor Power System program activities were in a "pre-reactor experiment" contractual phase. The quality program plan reflects that program status and shows how quality assurance activities were to have been expanded to meet program needs.

This program plan applied to all activities affecting Thermionic Fuel Element (TFE) quality under Project 0018, Contract AT(04-3)-840, and is presented here as it would have been had the project continued.

2.0 MANAGEMENT AND PLANNING

- 2.1 The purpose of this quality program plan is to describe the work elements to be accomplished by GEES in assuring accomplishment of the Company policy regarding quality. That policy is: "The products created by Gulf Energy and Environmental Systems for its customers shall be of high quality in materials, workmanship, and performance. Inherent in this policy is the concept that creation of a product of high quality requires that features essential to product function and appearance be equal to or superior to similar features in the products of others available to a customer."

2.2 This plan includes requirements and procedures that are currently needed and in use. Additional items will be included as the program evolves in a manner that will provide a comprehensive Quality Assurance Program for use on components manufactured for the Thermionic Reactor Experiment (TREX). The milestone chart (Figure B-1) portrays how the Quality Assurance Program is expected to evolve.

The quality assurance and reliability milestones on Figure B-1 are derived from schedule information contained in the "Thermionic Reactor Power System Development - Fuel Element Plan", Gulf-GA-C12307 and the "Thermionic Reactor Power System Development, Volume IV, Reactor Plan", Gulf-GA-C12308. The attached Quality Program Plan Implementation Schedule for TFE Qualification Testing (Table B-1) provides a schedule to prepare for passing a Quality Assurance Program audit covering manufacturing of thermionic fuel elements for 5,000 hour TFE qualification testing. Quality Assurance Plan revision will be accomplished on a yearly schedule to provide a document satisfactory for meeting Thermionic Reactor Space Power Systems milestones.

2.3 The organization chart (Figure B-2) illustrates organization alignments. Even though the prime responsibility for this Quality Program document lies with the GGA Quality Assurance Division, many groups such as Engineering, Testing, Material Control and Production are called upon to perform tasks related to quality. The Quality Assurance Division is responsible for performing audits to assure that all groups are performing their tasks as intended. The Quality Assurance Division also performs the Program Audits shown on the milestone chart (Figure B-1).

THERMIONIC REACTOR SPACE POWER SYSTEM

PROGRAM MILESTONES

FIGURE B-1. PROGRAM MILESTONES

TABLE B-1

QUALITY PROGRAM PLANIMPLEMENTATION SCHEDULEFOR TFE QUALIFICATION TESTING (5,000 HOURS)

Q. A. PROGRAM TASK (INDIVIDUAL RESPONSIBLE)	Document Draft Submittal	Document Approved	Full Implementation
1. Single individual designated responsible for Q. A. Program. (Holland)	---	---	Complete 10/01/72
2. Single individual designated responsible for Q. A. Program (individual independent of Fabrication and Engineering). (Holland)	---	---	9/01/73
3. Add another Q. C. inspector (process inspection and Q. A. Program task implementation such as calibration program, certification program and data package completeness). 1/4 time. (Horner)	---	---	4/01/73
4. Add Q. A. engineer 1/4 time (accomplish technical document reviews, writes procedures, act on CCB and MRB, implement Q. A. progress tasks). (Holland)	---	---	6/01/73
5. Add Q. A. engineer 1/2 time. (Holland)	---	---	9/01/73
6. Prepare list of items requiring calibration in Fab. and Test areas. (Horner)	11/15/72	12/15/73	1/15/73
7. Prepare document describing <u>operator</u> qualification and certification program including training program lecture outlines. (Manning)	2/01/73	2/15/73	3/01/73
8. Prepare document describing <u>process</u> qualification and certification program for critical process. (Manning)	2/01/73	2/15/73	3/01/73
9. Prepare statistical quality control charts for critical processes. (Matt)	2/25/73	3/25/73	4/25/73

TABLE B-I (Cont.)

Q. A. PROGRAM TASK (INDIVIDUAL RESPONSIBLE)	Document Draft Submittal	Document Approved	Full Implementation
10. Prepare document describing traveler system. (Appleby)	11/15/72	12/01/72	2/25/73
11. Prepare document describing design review system. (Appleby)	12/01/72	2/25/73	3/25/73
12. Prepare detailed test procedures including encapsulation, hook-up, monitoring and recording. (Yates)			
Qualification Testing	9/01/73	1/01/74	3/01/74
Experimental Testing	2/01/73	2/15/73	3/01/73
13. Prepare detailed inspection instructions. (Matt)	7/01/73	8/01/73	9/01/73
14. Compile procedures into Q. A. manual. (Matt)	8/01/73	9/01/73	10/01/73
15. Prepare document describing audit program including audit schedule, audit checklists and audit findings and corrective action methods. (Matt)	9/01/73	10/01/73	11/01/73

FIGURE B-2 ORGANIZATION CHART

2.4 There are examples of other AEC quality programs active within GEES. The HTGR reactor programs utilize 10CFR50 Appendix B "Quality Assurance Criteria for Nuclear Power Plants" and the HEDL program utilizes RDT Standard F2-2T, "Quality Assurance Program Requirements". Procedures available from these programs shall be used for the TFE program when applicable.

2.5 Significant developments and the status of the Quality Assurance Program shall be reported to GEES management and to SNS in the thirdly reports.

3.0 QUALITY REVIEWS OF TECHNICAL DOCUMENTS

3.1 Technical documents used shall be drawings, process specifications, material specifications, fabrication process specification (travelers), and cell rosters.

3.2 Reviews conducted shall be as follows:

<u>Reviewing Organization</u>	<u>Drawings</u>	<u>Process Specs.</u>	<u>Mat'ls. Specs.</u>	<u>Travelers</u>	<u>Cell Rosters</u>
Engineering	x	x	x		x
Fabrication	x	x	x	x	
Q. A. Engineering		x	x		
Configuration Control Board	x		x		
Testing	x				x
Materials		x	x		

4.0 TRACEABILITY AND IDENTIFICATION

4.1 Each Thermionic Fuel Element shall be given an identifying number and all fabrication and test results shall be identifiable to this number. The numbers shall be those appearing in the cell roster.

- 4.2 Fabrication traceability from raw materials, detail parts through completed TFE's shall be maintained by the use of purchase order numbers, heat numbers where required, and traveler numbers.

5.0 PROCUREMENT

- 5.1 Purchasing of all materials and services shall be accomplished by means of a Procurement Requisition (PR), GEES Form 8, and a Purchase Order (PO), GEES Form 4. Technical documents shall be referenced on the PR and PO. Parts and materials shall be inspected upon receipt and deviations from PO requirements reported on an Inspection Report (GGA Form 663). Travelers shall reference PO number (and heat number, if any) for materials and parts.
- 5.2 PR's shall be reviewed by Engineering before transmittal to Purchasing. PO's shall be reviewed by Production Control to assure compliance with PR.

6.0 FABRICATION AND ASSEMBLY

- 6.1 Work shall be accomplished as prescribed on Fabrication Process Specifications (travelers). The travelers shall reference pertinent drawings and specifications.
- 6.2 Work shall be accomplished by technicians qualified by a documented training and certification program.
- 6.3 Special processes shall be qualified by a documented certification program.
- 6.4 Completed end items shall be final inspected, including a final check of the data package, prior to delivery to test.

7.0

INSPECTION AND TESTS

- 7.1 Receiving inspections shall be conducted in accordance with requirements in PO's.
- 7.2 In-process inspections shall be accomplished in accordance with inspection points called out on travelers.
- 7.3 Final inspection shall be accomplished in accordance with instructions on travelers.
- 7.4 Testing shall be accomplished as specified on the cell roster. Significant test results shall be issued with the monthly contract progress report and in a test booklet after each reactor shutdown for neutron radiographs. Documentation of testing procedures is provided for in the Implementation Schedule.

8.0

NONCONFORMING ARTICLES AND MATERIALS

- 8.1 Deviating parts and materials shall be reported on an Inspection Report (IR).
- 8.2 The IR shall be processed as described in Quality Procedure QP-001, except that the Configuration Control Board (CCB) makes final "use as is" decisions.
- 8.3 Deviating parts and materials shall be conspicuously tagged and segregated until disposition is obtained on the IR. Reject materials shall be conspicuously marked and segregated to prevent inadvertent use on a production item. Rework shall be accomplished on a rework traveler.

9.0 CALIBRATION AND STANDARDS

- 9.1 Electronic items shall be calibrated in accordance with Quality Procedure QP-016, "Maintenance and Calibration of Electronic Equipment".
- 9.2 Mechanical items shall be calibrated in accordance with Quality Procedure QP-017, "Calibration of Mechanical Measuring and Test Equipment".

10.0 STAMPS

- 10.1 Upon completion of inspections, the Q. A. Inspector shall stamp the traveler or PO with his stamp.
- 10.2 Inspection stamps, types used, issuance and control are described in Quality Assurance Directive QAD-006, "Issue and General Use of Inspection Stamp".

11.0 HANDLING, PRESERVATION, PACKAGING, STORAGE AND SHIPPING

- 11.1 Parts and materials requiring special handling shall be so noted on the travelers.
- 11.2 Clean rooms shall be operated as described in Procedures HCR-1, HCR-2, HCR-3, HCR-4, HCR-5, and HCR-6. *
- 11.3 Completed TFE's shall be shipped to testing in special containers in cooperation with the Office of Nuclear Material Control (ONMC).

* These exist in draft form only at the present time.

12.0

STATISTICAL QUALITY CONTROL

- 12.1 Currently no sampling plans are used, however, in the event the need arises MIL-STD 105 or MIL-STD 414 shall be used if possible. If these two plans cannot be used, a special plan shall be prepared in writing and approved by the CCB.
- 12.2 Control charts and process capability studies will be used as pertinent.

13.0

DATA PACKAGE

- 13.1 The data package for completed TFE's shall be as follows:
- 13.1.1 Completed, stamped-off travelers.
 - 13.1.2 A complete set of drawings used.
 - 13.1.3 Completed IR's.
- 13.2 Additional TFE data shall be contained in:
- Thirdly Reports
 - Cell Rosters
 - Test Summaries
 - Special Topical Reports
 - "H" Series Material Specifications
 - Fabrication and Engineering Documents (FED's)
 - Process Specifications
 - Production Control Notebook
 - Purchase Order files (material test reports and certifications)

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