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Analytical Design Studies of In-Reactor
Tests of a Nuclear Light Bulb Unit Cell

NASA Contract No. SNPC-70

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FOREWORD

An exploratory experimental and theoretical investigation of gaseous nuclear rocket technology is being conducted by the United Aircraft Research Laboratories under Contract SNPC-70 with the joint AEC-NASA Space Nuclear Systems Office. The Technical Supervisor of the Contract for NASA was Captain C. E. Franklin (USAF) for the first portion of the contract performance period and was Dr. Karlheinz Thom for the last portion of the contract performance period. Results obtained during the period September 16, 1970 and September 15, 1971 are described in the following seven reports (including the present report) which comprise the required second Interim Summary Technical Report under the Contract:

1. Roman, W. C. and J. F. Jaminet: Experimental Investigations to Simulate the Thermal Environment and Fuel Region in Nuclear Light Bulb Reactors Using an R-F Radiant Energy Source. United Aircraft Research Laboratories Report K-910900-7, September 1971.
2. Klein, J. F.: Experiments to Simulate Heating of the Propellant in a Nuclear Light Bulb Engine Using Thermal Radiation from a D-C Arc Radiant Energy Source. United Aircraft Research Laboratories Report K-910900-8, September 1971.
3. Bauer, H. E.: Initial Experiments to Investigate Condensation of Flowing Metal-Vapor/Heated-Gas Mixtures in a Duct. United Aircraft Research Laboratories Report K-910900-9, September 1971.
4. Rodgers, R. J., T. S. Latham and H. E. Bauer: Analytical Studies of Nuclear Light Bulb Engine Radiant Heat Transfer and Performance Characteristics. United Aircraft Research Laboratories Report K-910900-10, September 1971.
5. Latham, T. S. and H. E. Bauer: Analytical Design Studies of In-Reactor Tests of a Nuclear Light Bulb Unit Cell. United Aircraft Research Laboratories Report K-910900-11, September 1971. (present report)
6. Krascella, N. L.: Spectral Absorption Coefficients of Helium and Neon Buffer Gases and Nitric Oxide-Oxygen Seed Gas Mixture. United Aircraft Research Laboratories Report K-910904-2, September 1971.
7. Palma, G. E. and R. M. Gagosz: Effect of 1.5 Mev Electron Irradiation on the Transmission of Optical Materials. United Aircraft Research Laboratories Report K-990929-2, September 1971.

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Analytical Design Studies of In-Reactor Tests of a
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Analytical Design Studies of In-Reactor Tests of a
Nuclear Light Bulb Unit Cell

SUMMARY

Analytical design studies were continued to determine the design characteristics and performance of a small-scale model of a nuclear light bulb unit cell suitable for testing in a nuclear reactor. The test reactor considered was the Nuclear Furnace, a solid-core nuclear rocket fuel element test reactor. The Nuclear Furnace can provide a specific fission power level in the test region ranging from 25 to 40 kw/gm of U-235 fuel.

The results of the study indicate that a meaningful series of in-reactor demonstration tests of fuel containment, transparent-wall performance, and propellant heating could be conducted using the Nuclear Furnace reactor. Tests conducted with a configuration having reflecting walls and operating at a pressure of 500 atm could result in thermal radiation fluxes corresponding to black-body radiating temperatures ranging from 5800 to 6300 K. Tests could also be conducted to demonstrate that internally cooled transparent walls are capable of withstanding both the nuclear radiation and thermal environments anticipated for a nuclear light bulb engine and that seeded propellant can be heated to exhaust temperatures in the range of 3300 to 3700 K.

Simulation tests of models similar in geometry to those anticipated for in-reactor demonstration tests have been performed using the UARL r-f induction heater and d-c arc facilities. It is recommended that thorough development and testing of in-reactor test models at power levels and operating conditions anticipated for in-reactor tests be performed using these facilities prior to nuclear testing.

RESULTS AND CONCLUSIONS

1. The Nuclear Furnace reactor can be modified to accommodate a small-scale nuclear light bulb unit cell. The specific fission power level in the test region would be in the range from 25 to 40 kw/gm of U-235 fuel. Several tests of 10 to 15 min duration each could be run using a single core loading with approximately one month between tests.

2. A reference or basic test configuration with the following principal characteristics was selected: (a) test region diameter equal to 6.6 cm; (b) test region length equal to 17.78 cm; (c) argon used as the buffer gas to provide a radial inflow vortex flow; (d) U-235 fuel injected in the form of uranium hexafluoride during test start-up and in the form of a mixture of submicron particles and either uranium hexafluoride or gaseous argon during steady-state operation; (e) mass of U-235 contained in fuel region equal to 6.2 gm; (f) average partial pressure of U-235 fuel within the fuel region equal to 195 atm; (g) test region total pressure equal to 500 atm; and (h) ratio of fuel region diameter to test region diameter equal to 0.6.

3. The calculated performance of the basic test configuration for a specific fission power level of 30.5 kw/gm of U-235 fuel was as follows: (a) total test region power equal to 190 kw; (b) net radiated heat flux equal to 0.71 kw/cm^2 ; (c) equivalent black-body radiating temperature equal to 3330 K; (d) outward radiating thermal heat flux equal to 7.1 kw/cm^2 and inward reflected thermal heat flux equal to 6.39 kw/cm^2 (difference equals net radiated heat flux of 0.71 kw/cm^2); and (e) fuel surface radiating temperature equal to 5940 K.

4. It would be feasible to place internally-cooled, fused silica, transparent walls between the fuel-containment test region and a propellant duct such that propellant heating demonstration tests and tests exposing the transparent-wall structure to both the nuclear radiation and thermal environments of the test region could be performed. The addition of a propellant duct and transparent-wall structure would require a reduction in the test region diameter from 6.6 cm to approximately 5.08 cm.

5. Transparent-wall tests could be performed in which the total thermal radiation flux (incident plus reflected) passing through a transparent-wall structure would be 6.20 kw/cm^2 compared with 23.4 kw/cm^2 for the reference nuclear light bulb engine.

6. It is estimated that seeded hydrogen propellant could be heated to bulk exhaust temperatures in the range of 3300 to 3700 K in propellant heating demonstration tests in the Nuclear Furnace.

7. Handling of the nuclear fuel and fission products from the nuclear light bulb demonstration tests would be accomplished using the same effluent scrubber system designed for current Nuclear Furnace tests.

8. Simulation tests of models similar in geometry to those anticipated for in-reactor demonstration tests have been performed using the UARL r-f induction heater and d-c arc facilities. Thorough development, using these facilities, of in-reactor test models and diagnostic instruments to observe test results at the power levels and operating conditions anticipated for in-reactor tests should be performed prior to nuclear tests.

INTRODUCTION

Major emphasis in the nuclear light bulb (Ref. 1) feasibility research being conducted under Contract SNPC-70 has been placed on (1) simulation of the thermal environment in the fuel region and surrounding transparent-wall structure using small-scale models with r-f heated vortexes and (2) simulation of propellant heating in the nuclear light bulb engine by heating a seeded stream of gas with thermal radiation from a d-c arc discharge. The next possible major step in the development of a nuclear light bulb engine, building on the results from the simulation experiments, is a series of demonstration tests in which the arc- and r-f-heated plasmas are replaced by a fissioning uranium plasma as the energy source for thermal radiation.

A driving reactor suitable for use in an in-reactor test of a small-scale nuclear light bulb unit cell should have a high power density and an available test site on the order of 7.5 cm (3.0 in.) in diameter and several cm long. The Nuclear Furnace, a solid-core nuclear rocket fuel element test reactor, is well suited for such tests, having a core configuration in which an 8.38-cm (3.30-in.) diameter cylindrical hole can be provided on the core axis with sufficient thermal neutron flux to sustain a fissioning plasma in the test cell.

The studies described herein are a continuation of the in-reactor test feasibility analyses of Ref. 2. Specific emphasis in the current studies was placed on the design and performance characteristics of demonstration tests based on the use of the Nuclear Furnace as the driving reactor. Test operations, test instrumentation, test safety considerations, and results of preliminary reactor physics experiments conducted at the Los Alamos Scientific Laboratory are also discussed in the following section of the report.

DESCRIPTION OF NUCLEAR FURNACE TEST REACTOR

To obtain the desired performance from in-reactor demonstration tests of a nuclear light bulb unit cell requires a high power density nuclear test reactor which can provide a specific fission power sufficient to create the fissioning plasma in the gaseous fuel region. In addition, the space available for the test configuration should be a cylindrical hole 7.62 cm (3.0 in.) or more in diameter. Finally, the frequency of operation of the test reactor should be high enough to allow several tests per year to be conducted. The Nuclear Furnace, a solid-core nuclear rocket fuel element test reactor designed by Los Alamos Scientific Laboratory (LASL), satisfies these requirements. The Nuclear Furnace core is approximately 32.0 cm (13.0 in.) in diameter and 134.5-cm (53.0-in.) long. The 49 fuel elements are contained in a tube bundle assembly with interstitial light-water moderation. The core is self-contained in an aluminum vessel and can be removed through the top of the reactor.

The power level of the Nuclear Furnace is rated at 44 megw, but can be extended to about 66 megw with additional coolant flow. The specific fission power levels in the test region at the center of the core shown in Fig. 1 for the range of power levels from 44 to 66 megw would be from 25 to 40 kw/gm of U-235 fuel.

Discussions have been held with LASL staff members to explore the feasibility of inserting a small-scale nuclear light bulb unit cell in the Nuclear Furnace. It was suggested by LASL personnel that it might be feasible to remove four fuel elements at the center of the core and insert an aluminum casing along the core centerline which would accommodate an 8.38-cm (3.30-in.) dia unit cell model. Measurements made on the Nuclear Furnace zero-power critical assembly at Los Alamos indicate reactivity lost by removal of the four central fuel elements could be restored by increasing the fuel loading of the remaining 45 fuel elements.

A schematic diagram of the Nuclear Furnace reactor with the four central fuel elements replaced by an aluminum-cased test hole is shown in Fig. 1. A test assembly containing a small-scale model of a nuclear light bulb unit cell would be inserted through the top of the core lid. Details of the test assembly are discussed in the following section. All plumbing and instrumentation wiring would be passed through the top of the core assembly as shown in Fig. 2. With such wiring and plumbing connections, the complete test assembly could be removed remotely in the assembly-disassembly bay and replaced without removing the driving reactor core. According to LASL personnel, test turnaround times of approximately one month might be feasible with this design.

Effluent from the test region would be mixed with that from the Nuclear Furnace and would flow into the scrubber system at the bottom of the core. All effluent would be stored in a containment vessel and run through ion exchange columns to remove unspent nuclear fuel and fission products.

DESCRIPTION OF TEST CELL CONFIGURATIONS

The small-scale unit cells used in the in-reactor tests are designed to simulate many of the conditions expected to exist in a full-scale nuclear light bulb engine unit cavity. Each of the test cell configurations described would contain a gaseous nuclear fuel cloud suspended in a vortex flow field driven by tangentially injected argon buffer gas. The peripheral walls of the test region would be highly reflective to the thermal radiation from the fuel region.

Basic Configuration

The test region of the basic test cell configuration is a cylindrical cavity with an inside diameter of 6.60 cm (2.60 in.) and a length of 17.78 cm (7.00 in.). The cylindrical peripheral wall is made of aluminum and has a highly reflecting inner surface. Flow channels to provide for the injection of buffer gas, fuel, and required coolant for the components are provided. The entire assembly is contained in an aluminum-lined fiberglass pressure vessel. A sketch of the central portion of the test assembly is shown in Fig. 3. The outside diameter of the pressure vessel is 8.13 cm (3.20 in.) so that the complete assembly can be inserted in the 8.38-cm (3.30-in.)-dia test hole available in the Nuclear Furnace.

Dimensions and material specifications for the components shown in Fig. 3 are given in Table I. The end-wall coolant and fuel-injection ducts are formed by a series of concentric tubes which form the following flow passages: (1) the fuel-injection duct; (2) the inner bypass flow annulus; (3) the outflow annulus; (4) the outer bypass flow annulus; (5) the end-wall coolant outlet passage; (6) the end-wall coolant inlet passage; and (7) the liner coolant annulus. The region between the end-wall coolant inlet and outlet passages is not used for coolant flow. This region would be used for instrumentation purposes.

The total length of the in-reactor test assembly required to extend from the lid of the Nuclear Furnace core to the core bottom is approximately 188.0 cm (74.0 in.), including about 30.5 cm (12.0 in.) of effluent duct between the outlet of the test cell pressure vessel and the core bottom.

The pressure vessel is a filament-wound fiberglass cylinder approximately 157.5-cm (62.0-in.) long with an inside diameter of 7.36 cm (2.90 in.) and an outside diameter of 8.13 cm (3.20 in.). The wall thickness of 0.385 cm (0.15 in.) is required for operation at an internal pressure of 500 atm. A design stress level of 75,000 psi in the fiberglass was used. The maximum allowable uniaxial tensile stress for glass fibers is on the order of 300,000 psi. If it is assumed that the pressure vessel is wound so that it has similar tensile stress properties

in both axial and hoop stress (alternate circumferential and axial fiber directions for 45 deg angle winding), the maximum allowable stress would be reduced to 150,000 psi. This value was further reduced to the design level stress of 75,000 psi to provide a safety factor of 2.0.

The pressure vessel is cooled on both sides so that the maximum allowable temperature of 390 K (700 R) will not be exceeded at the values of neutron and gamma ray heating anticipated at full-power operation of the test reactor. Cooling on the outside surface of the pressure vessel would be provided by flowing hydrogen coolant between the annulus bounded by the pressure vessel and the aluminum test hole casing. Control of the coolant flow rate would be provided by orifices located at the inlet end of the annular passage.

The strength of the resin used in the fabrication of fiberglass components is degraded by prolonged exposure to radiation. Heat resistant phenolic resin binder has been shown (see Ref. 3) to resist radiation deterioration up to dosages of 10^{10} rads. The peak neutron and gamma heating rates on the core centerline are estimated to be 25 to 35 w/gm for the fiberglass materials. It is estimated that a pressure vessel could be used in the Nuclear Furnace for at least 30 min of full-power operation before the radiation dose reaches 5×10^9 rads (approximately 50 percent of the dose at which the tensile strength begins to deteriorate). The most desirable mode of operation for in-reactor tests is one in which test runs of approximately 10 to 15 min duration are performed, after which the test reactor is taken by railcar to the remote assembly-disassembly bay for removal of the test assembly and possible insertion of a new assembly for the next test series. If test turn-around times are on the order of 30 days, insertion of the subsequent test assemblies could take place after fission product radiation dose rates from the fuel elements have decayed to levels such that the new test configuration would be exposed to doses on the order 5×10^7 rads or one percent of the allowable pressure vessel dose limit. Based on the fission product decay curves of Ref. 4, insertion of the subsequent test assemblies could take place 20 days after shut-down of the Nuclear Furnace with no possibility that the total dosage to the fiberglass during the time until the next test will exceed 5×10^7 rads.

Schematic diagrams of the pressure vessel end walls are shown in Fig. 4. The inlet end of the pressure vessel would be secured to the top of the reactor core lid as shown in Fig. 1. The capped (upper) end wall is similar in design to that currently employed in the UARL r-f simulation experiments described in Ref. 5. The capped end walls contain a winding ring and retainer flange as shown in Fig. 4(a). Several penetrations through the top of the pressure vessel are provided to supply the required buffer gas, fuel and coolant flow. The outlet end of the pressure vessel is wound over a dome-shaped end wall to allow the test assembly to be inserted into the aluminum test-hole casing. The pressure vessel is wound on a 0.075-cm (0.03-in.)-thick aluminum pressure vessel liner. The liner is welded to the

end-wall dome at the outlet end and to the upper end-wall plate on the inlet end. The filament-wound pressure vessel is bonded to the thin aluminum liner.

Modifications to Basic Configuration

Several modifications to the basic test cell configuration were considered in the studies. These include the use of an aluminum pressure vessel instead of a fiber wound pressure vessel and use of fuel injector locations other than along the centerline of the test cell. A schematic diagram of a test cell with an aluminum pressure vessel and off-axis fuel injectors is shown in Fig. 5. The possible use of an aluminum pressure vessel was included because aluminum is not as susceptible to radiation damage effects as fiberglass. However, the use of aluminum would reduce the available inside diameter of the test region to 6.35 cm (2.5 in.) because the aluminum pressure vessel would have to be 0.584 cm (0.23 in.) in thickness to contain the 500 atm pressure within its stress limitations. The basic design of flow annuli and end walls would be relatively unchanged by the use of an aluminum pressure vessel. However, the use of an all-aluminum pressure vessel would reduce the test performance levels due to greater absorption of thermal neutrons than occurs when the fiberglass pressure vessel is used.

The fuel injector location in the basic configuration was placed on the centerline to be consistent with the preferred injection locations employed in simulation experiments in the 1.2-megw r-f induction heater. However, it may be possible to inject the nuclear fuel off the centerline as shown in Fig. 5. This injection scheme would result in a simpler thru-flow port and bypass flow annulus configuration. The off-axis fuel injectors would also provide more space for instrumentation leads in the end-wall region.

Propellant Heating Configuration

The unit cell for in-reactor tests of transparent walls and propellant heating is shown in Fig. 6. In this configuration, provision is made for insertion of transparent walls across a segment of the test region liner. Behind the transparent wall array is a propellant heating channel similar in geometry to that employed in the UARL d-c arc propellant heating experiments described in Ref. 6. Reflective aluminum liners are employed around the test region and around the propellant heating channel. The transparent wall array is clamped into position by a split manifold as shown in Fig. 6. Transparent-wall coolant would be routed through the split manifolds. The inclusion of the transparent wall and propellant heating channel reduces the test region diameter to 5.08 cm (2.00 in.). The

propellant must be seeded with submicron particles to absorb the radiant energy. It is also necessary to use unseeded buffer flows at the edge of the propellant heating channels to keep the transparent walls and reflecting liners clean. The porous foam and seeded hydrogen inlet used to tailor the propellant channel flow are also shown in Fig. 6.

TEST PERFORMANCE LEVELS

A detailed description of the methods used to calculate the performance levels expected in the demonstration tests is contained in Ref. 2. The values selected for those parameters which affect the test performance are shown in Table II. Both the range of values investigated and the values chosen for a reference test performance level are shown in Table II. The specific fission power of the driver reactor is the key performance parameter with regard to the power levels obtainable from the driving reactor. Other important parameters used in the analysis of performance are (1) operating pressure, (2) the fuel-to-cavity radius ratio, (3) the density containment factor (the ratio of average fuel density within the fuel region to the buffer-gas density at the edge of the fuel region), (4) the ratio of the average fuel residence time to buffer-gas residence time in the vortex, (5) the reflectivity of the aluminum liner on the inner surface of the vortex cavity, and (6) the partial pressure ratio of fuel to total pressure (the ratio of average fuel partial pressure in the fuel cloud to the total pressure in the test region). The range of containment parameters used were within the range of measured values in the heated and unheated two-component gas vortex tests described in Refs. 7 through 9.

Performance Levels with Basic Configuration

The calculated ranges of equivalent black-body radiating temperatures and edge-of-fuel temperatures obtainable in tests of the basic configuration (see Fig. 3) are shown in Fig. 7. Results are shown for several values of the two containment factors: fuel-to-cavity radius ratio, r_F/r_T , and the density containment factor, K_F . The range of specific fission power for U-235 in the Nuclear Furnace is also shown on Fig. 7. Figure 8 shows the range of radiated power obtainable for different values of r_F/r_T and K_F . A reference performance level was chosen by selecting the specific values of the containment parameters shown in Table II and on Fig. 9. It can be seen from Fig. 9 that for the range of specific fission power obtainable in the Nuclear Furnace (25 to 40 kw/gm of U-235), values of radiated power from 150 to 230 kw (142 to 218 Btu/sec), with corresponding equivalent black-body radiating temperatures of 3300 to 3600 K (5940 to 6500 R) and edge-of-fuel temperatures from 5700 to 6300 K (10,300 to 11,350 R) could be obtained.

The increase in edge-of-fuel temperature over the equivalent black-body radiating temperature results from the presence of the polished aluminum reflective liner. With a reflective liner, the fuel region surface radiating temperature would be increased by the factor $(1/1-R)^{1/4} = 1.78$ for R equal to 0.9 (i.e., to provide a net outflow of radiant energy of one unit, the true gross outflow must be ten units to make up for the reflection of nine units back toward the fuel). The

equivalent fuel-region black-body radiating temperature, T^* , corresponds then to a net radiant heat flux which is a tenth of the gross radiated heat flux. Note, however (see Ref. 10), that the true edge-of-fuel temperature for zero reflectivity is slightly less than T^* , and, for finite reflectivity, is slightly less than the edge-of-fuel temperature, T_6 . A reference reflectivity of $R = 0.9$ was chosen on the basis of the results of Ref. 1 in which it was shown that the spectral reflectivity of polished aluminum when averaged over the wavelength range encompassing the photons emitted from the hot fuel cloud would be equal to or greater than 0.90.

To establish specific reference performance levels for in-reactor tests for determination of coolant flow requirements and fuel and buffer-gas injection rates, a reference specific fission power was chosen of 30.5 kw/gm of U-235. The performance levels for in-reactor tests at this specific fission power are shown in Table III. All other containment parameters for this performance level are the same as those chosen for the reference performance levels shown in Fig. 9. When U-235 is used as the nuclear fuel, the fuel loading is $M_F = 6.2$ gm (0.0137 lb), the total power is 189.8 kw (180 Btu/sec), and the radiated power is 173.4 kw (164.3 Btu/sec). This results in a surface radiating temperature, $T_6 = 5940$ K (10,690 R), and an equivalent black-body radiating temperature of $T^* = 3300$ K (5940 R). Pu-239 has a higher average fission cross section than U-235. Also shown on Table III is an estimate of performance using Pu-239 for which the specific fission power due to the higher fission cross section was chosen as 64.1 kw/gm of Pu-239. The corresponding performance levels with Pu-239 are approximately twice those achievable with U-235. However, Pu-239 is a more difficult nuclear fuel to handle because of its shorter half-life and because of its hazard as an airborne contaminant. Principal interest should be focused on the performance of tests using U-235 since that is the nuclear fuel currently handled in the Nuclear Furnace core and scrubber system.

The ranges of nuclear fuel partial pressure in the fuel-containment region for the reference in-reactor test are shown in Fig. 10. For the reference performance level, the average fuel partial pressure in the fuel region is 195 atm, or approximately 0.39 times the total operating pressure of 500 atm.

An important concern in evaluating the performance level achievable in a given test is that of determining whether there is the possibility of condensation of liquid metal droplets in the fuel region due to the vapor pressure of the fuel cloud becoming too high at the local temperature conditions. By extrapolating the curves of Ref. 11, in which fuel-region temperature distributions and centerline temperatures were calculated relative to the radiated heat flux per unit length of plasma, it was determined that for the in-reactor test the centerline fuel temperature would be approximately 10,000 K (18,000 R). Extrapolation of the uranium-metal vapor pressure curve calculated from the data of Ref. 12 to pressures on the order of 200 atm indicate that at this temperature and partial pressure the local temperatures are well above the condensation point for the vaporized metal.

The effect on calculated performance levels due to variations in the fuel-to-cavity radius ratio, r_F/r_T , the density containment factor, K_F , the ratio of fuel-to-buffer-gas residence time, τ_F/τ_B , and the aluminum liner reflectivity, R , are shown in Fig. 11. The greatest variations in performance result from changes in r_F/r_T and K_F (see Figs. 11(a) and 11(b)), principally because these parameters have a direct effect on the amount of nuclear fuel stored in the test cell. Note that in all cases, the cell length and diameter is held constant. Variation of the aluminum liner reflectivity (Fig. 11(d)) has a large effect on the gross outward photon flux which is proportional to T_6^4 . The slight rise in radiated power as the aluminum liner reflectivity decreases results from an increase in stored nuclear fuel because the buffer-gas density at the edge of the fuel cloud, ρ_{B_6} , increases (note that the density containment factor, K_F , is held constant in this case). The effect of variations in the ratio of fuel-to-buffer-gas residence time is felt principally through the amount of energy convected from the fuel region, and only has a significant effect on the test performance for values of the ratio of 0.2 or less. Generally, the amount of energy convected from the test region is dependent upon the buffer-gas flow rate and its average temperature. As the ratio of fuel-to-buffer-gas residence time increases, the convected energy from the system becomes more and more dominated by the enthalpy contained in the buffer flow. Thus, the variation of radiated power and radiating temperatures with increasing residence time ratio is very slight.

Investigations were also conducted to determine the effect of varying the operating pressure of the unit cell test on performance with all of the containment parameters held at the reference levels. The pressure vessel thickness was varied with increasing pressure to keep the hoop stress and the axial stresses in the system constant. The results of these studies of the ranges of radiating temperatures and power levels for in-reactor tests at different operating pressures are shown in Fig. 12. It can be seen that, by increasing the pressure from 500 to 3000 atm, radiating power levels on the order of 500 kw (475 Btu/sec) can be achieved for the reference specific fission power level. The corresponding equivalent black-body radiating temperature and edge-of-fuel temperature are 4300 K (7730 R) and 7500 K (13,500 R), respectively. At the highest pressure considered, $P = 3000$ atm, the edge-of-fuel temperature, T_6 , is 7500 K (13,500 R) as compared to an equivalent black-body radiating temperature of 8300 K (15,000 R) in the reference engine.

Performance Levels for Propellant Heating Configuration

Since propellant heating demonstration tests and tests to demonstrate the feasibility of using the transparent-wall structure in the radiation and thermal environments of a nuclear reactor are of importance, analyses were also performed to determine the ranges of radiating temperatures and power levels for in-reactor tests of a cell having a propellant heating channel. The results of these analyses

are shown in Fig. 13. The geometry and dimensions of the unit cell with a propellant heating channel were shown in Fig. 6. For the reference specific fission power level of 30.5 kw/gm of U-235, the propellant heating configuration would have a radiating power of 120 kw (114 Btu/sec). The corresponding equivalent black-body radiating temperature and edge-of-fuel temperature would be 3300 K and 4800 K (5940 R and 8620 R), respectively. The temperatures were calculated by assuming that the effective surface reflectivity of the aluminum liner was reduced from $R = 0.9$ to $R = 0.8$ due to the presence of absorbent propellant. On the basis of propellant heating results in the UARL d-c arc heater facility (see Ref. 6), this power level and radiating temperature should result in propellant heating demonstration tests which would deliver bulk propellant exit temperatures on the order of 3300 to 3700 K (5940 to 6650 R).

Transparent-wall tests could also be performed with no seeded propellant flowing through the propellant heating channel. The total thermal radiation flux (incident plus reflected) passing through the transparent wall structure would be 6.2 kw/cm^2 (5470 Btu/sec-ft²) compared with 23.4 kw/cm^2 (20,600 Btu/sec-ft²) for the reference nuclear light bulb engine. These radiation fluxes were calculated assuming the combined surface reflectivity of the aluminum liner and propellant channel liner was $R = 0.9$.

OPERATING PROCEDURES

The Nuclear Furnace reactor is transportable by rail from the test site to the remote assembly-disassembly bay. If a Nuclear Furnace core were devoted to the tests of a nuclear light bulb unit cell, the mode of operation would be to test the unit cell in a run of approximately 10 to 15 min duration and then disconnect the reactor supply systems and move the entire system by rail to the assembly-disassembly bay. The in-reactor test assembly could be withdrawn from the Nuclear Furnace core and replaced with a new test assembly in the remote handling bay. The minimum turn-around time between the tests was estimated by LASL personnel to be on the order of 30 days.

Effect of In-Reactor Tests on the Nuclear Furnace Operation

The geometric and physical effects of the insertion of a test assembly in the Nuclear Furnace have been discussed previously. The effects on the operation of the Nuclear Furnace are related principally to the adjustments necessary to insure that the core lifetime is extended by running the solid-core nuclear rocket fuel elements cooler than usual. In addition, it would be desirable to run the Nuclear Furnace at the highest possible power level (66 megw) in order to obtain the highest possible performance levels in the tests. Additional remote connection booms would have to be fabricated to penetrate the bell shield of the Nuclear Furnace (see Fig. 1) so that the coolant, fuel injection, and buffer-gas flow systems and instrumentation leads could be remotely connected and disconnected in a manner similar to that used in present Nuclear Furnace tests. Use of a mode of operation which allows removal of one test assembly and insertion of a new assembly implies that all access to the test region must be through the top of the test assembly. This is considered to be desirable, because access to the test region through the core or the scrubber section beneath the core would be extremely difficult.

The effects of introducing the test-region effluent gases into the scrubber system must be explored in greater detail. The quantities of UF_6 and argon which will be added to the effluent are relatively small, but the formation of HF molecules in the effluent may cause some corrosion of the scrubber system components. However, flushing the scrubber after test runs might be an adequate procedure to remove acids from the system's walls and thereby minimize corrosion. Also, the effect of the fuel loading in the test region on the total reactivity of the Nuclear Furnace requires measurement. Preliminary measurements on the Nuclear Furnace zero power critical assembly have indicated that the presence of 6.2 gm (0.0137 lb) of fuel in the test region would not have significant enough reactivity to be detectable as a perturbation to the control drum positioning. This factor should, however, be thoroughly evaluated in the zero-power critical assembly tests.

Test Cell Flow Requirements

Flow circuits for buffer-gas injection, bypass flow injection and cooling of the pressure vessel, end walls and liner are required for the test assembly. A schematic diagram of a possible flow system, using argon for all cooling requirements, is shown in Fig. 4. At the inlet end of the test assembly (Fig. 4(a)), argon enters through two annuli: (1) the inner argon bypass flow annulus and (2) the outermost annulus shown in Fig. 4(a), which is connected to the argon plenum. The argon is routed to four separate circuits from the plenum: (1) the outer bypass flow annulus, (2) the buffer-gas injector tubes, (3) the pressure vessel and liner coolant annulus, and (4) the end-wall coolant annulus inlet. Bypass flow is injected into the exhaust annulus at the end wall as shown in Fig. 4(a). The buffer gas is contained in tubes as shown in Fig. 3 and is injected through the liner walls to drive the vortex. The pressure vessel and liner coolant passes through the annulus shown on Fig. 4(a) to the lower end region. The end-wall coolant passes behind the end wall and then is routed to the inlet-end exhaust port as shown. At the outlet end, the liner and pressure vessel coolant enters a lower plenum region as shown in Fig. 4(b). Since the temperature rise in the liner and pressure vessel coolant is small (approximately 28 K (50 R)), a portion of this flow is routed from the lower plenum for lower-end-wall cooling and for bypass flow in the lower end as shown. The inner bypass flow passes through a series of struts in the exhaust duct which connect the inner and outer bypass annuli. The piping shown at the outlet end of the test assembly leads to the exhaust scrubber of the Nuclear Furnace.

The flow rates in the in-reactor test unit and the corresponding steady-state operating temperature levels are shown in Tables IV and V. The required end-wall and liner coolant flow rates were determined by calculating the required flow which would result in a maximum end-wall and liner temperature of approximately 555 K (1000 R). The bypass flow rate was determined by calculating the amount of cold argon (278 K (500 R)) which must be added to the cavity exhaust to reduce the mixed-mean temperature to approximately 1300 K (2840 R) so that the fuel will be condensed to solid form.

The orifice sizes required to establish the flow distribution, assuming a 50 atm pressure loss in the orifices, are shown in Table VI.

Figure 4 shows the fuel injectors entering from both ends of the test region. If it is desirable to keep the fuel lines inside the test cell pressure vessel, the fuel injector shown in Fig. 4(b) can be supplied from a series of tubes which pass between the liner and the pressure vessel walls to the fuel injectors at the inlet end of the test cell.

In addition to the argon-cooled system shown in Fig. 4, a two-gas system using hydrogen as a liner and end-wall coolant was also considered. The two-gas system is more complex since it does not allow the liner coolant to be used as a buffer gas in the outlet region. The required flow rates for a hydrogen-cooled liner and end walls are shown in Table V and the required orifice sizes are shown in Table VI. It may also be possible to use an all hydrogen-cooled system if hydrogen can be used as a bypass gas.

Sequence of Test Events

The sequence of events for a reference in-reactor test is shown in Table VII as a function of time after the Nuclear Furnace reaches steady-state power level. The first minute after the Nuclear Furnace reaches its steady-state power level would be used to bring the test configuration to thermal equilibrium. Subsequent to that, a period of 1.5 min would be used to bring the test region to the full 500-atm operating pressure. Since the ambient pressure of the Nuclear Furnace is approximately 40 atm, the high pressure buffer-gas and coolant flow systems would have to be activated to supply the test regions as soon as the test region is pressurized. At the end of this period, the pressure would be up to 500 atm and the configuration should again be in thermal equilibrium.

After the test region is pressurized, the injection of liquid UF_6 can commence. Liquid UF_6 was chosen as the form of nuclear fuel for starting the test because it will vaporize very rapidly upon entry into the cavity and, hence, would not have a tendency to centrifuge to the walls. The critical temperature of UF_6 is approximately 337 K (605 R) and the range of temperature over which UF_6 , at the operating pressures of the system, would remain as a liquid is from 337 K to 503 K (605 R to 906 R) (see Ref. 13). The heat of vaporization of liquid UF_6 at a temperature of 350 K (630 R) is 790 joules/gm (430 Btu/lb), and the total enthalpy rise in liquid UF_6 flowing in the injector leading into the test region can be controlled by selecting the thickness of a neutron absorbing sleeve around the injector such that the UF_6 is allowed to vaporize at the time of injection. If the fuel partial pressure in the fuel region is limited to 195 atm, the corresponding power level for pure UF_6 injected into the vortex would be about 90 kw (85.5 Btu/sec). This would be the power level of the system after liquid UF_6 had been injected for approximately 1.0 min.

In order to increase the power level to the desired steady-state value of 190 kw (180 Btu/sec), it would be necessary to add submicron uranium particles to the fuel injection flow during steady-state operation. There are two options for injection of fuel during this part of the test cycle. One option would be to add the submicron particles to the liquid UF_6 such that the mass flow ratio of uranium particles to UF_6 would be 3.4 (necessary to provide the required containment). A second option would be to use an argon/uranium-particle mixture in which the mass

flow rate ratio of uranium particles to argon carrier gas would be about 6. Both of these options would result in equal numbers of uranium and carrier-gas atoms in the injected flow. The use of UF_6 throughout the cycle would appear to be more attractive because it would prevent the condensation of uranium metal on any of the test region surfaces or thru-flow duct surfaces. The addition of metallic uranium in the form of submicron particles might cause the formation of lower fluorides in the effluent. The deposition of this material on the thru-flow port walls could be avoided by injecting fluorine-rich bypass gas into the system and thereby causing rapid fluoridation of the lower uranium fluorides in the thru-flow duct. The effect of enriching a gas mixture in a plasma or a fissioning gas with fluorine was shown to be successful in maintaining a UF_6 vapor in the experiments reported in Ref. 14. After the uranium particles are added to the fuel flow, full-power steady-state operation can continue for approximately 3 min, allowing ample time to measure heat balance data and to observe the spectral emission from the fissioning plasma.

Shutdown of the test would be accomplished by terminating fuel injection flow and purging the fuel injection lines with high-pressure argon. This shutdown process would take approximately 1 min. Following the shutdown, depressurization of the system would begin. After the pressure is lowered to an ambient pressure of approximately 40 atm, the Nuclear Furnace hydrogen flow would be introduced to cool the test assembly. Once Nuclear Furnace hydrogen is flowing through the test assembly, the shutdown procedure for the reactor itself can proceed. The total weights of various coolants, buffer gases, and fuels required for a 10 min test are shown in Table VIII.

EXTERNAL SUBSYSTEMS REQUIRED FOR TESTS

The principal components external to the test assembly required to supply fuel, coolant, and buffer gas, and to handle the effluent from the test region are shown on Fig. 14. Also shown are an alternate coolant system employing high-pressure hydrogen and an alternate fuel injection system employing a uranium-particle/argon-carrier-gas system (the basic design uses argon coolant and a uranium-particle/UF₆-carrier-fluid for fuel injection). Components of the alternate systems are enclosed in dashed lines on Fig. 14.

Fuel Injection System

The UF₆ supply used during start-up of the tests is contained in a pressure vessel within an oven. The purpose of the oven is to maintain the UF₆ supply at an appropriate temperature such that it will remain as a liquid. The UF₆ is forced out of the supply vessel by high-pressure argon injected through the top of the supply vessel. The lines leading from the UF₆ supply vessel to the test location are wrapped with electrical heater tape to provide temperature control of the flowing fuel. Uranium particles are added to the fuel injection flow to achieve full-power, steady-state operation by increasing the ratio of uranium atoms to fluorine atoms injected into the fuel region. A uranium particle canister and mixing chamber for this purpose is depicted on Fig. 14. An alternative method for supplying fuel to reach full-power, steady-state operation would be to use a uranium-particle/argon-carrier-gas mixture. Therefore, the high-pressure argon lines (see Fig. 14) are shown connected to the particle canister and mixing chamber. In either system, particle flow would be metered such that equal numbers of argon or fluorine and uranium atoms are injected into the fuel region. The fuel injection system would supply a fuel/carrier-gas mixture which would ultimately result in a carrier-gas partial pressure equal to approximately one half the fuel partial pressure; the assumption being that the nuclear fuel, on the average, is singly ionized and has twice the partial pressure as the carrier gas, which tends to have a higher ionization potential and therefore remains neutral.

Coolant Supply Systems

The argon and hydrogen coolant supply systems are shown in Fig. 14. These systems would be employed to inject coolant for purposes of cooling the liner of the test region and for cooling the end walls. Either system would be activated when the test region is pressurized above the Nuclear Furnace ambient pressure. The capacity of the supply systems is shown in Table VIII. When the pressure of the test region drops below that of the Nuclear Furnace, approximately 40 atm, the

Nuclear Furnace hydrogen coolant supply would be regulated to initiate flow of coolant throughout the test assembly.

Buffer-Gas Supply System

The buffer gas would be supplied by the same manifold that supplies the primary argon coolant. The buffer-gas flow is regulated by orifices located at the entrance to the test region. The orifice sizes required to provide the desired flow of argon buffer gas are shown on Table VI. The four buffer-gas supply lines pass directly down the outside of the inner aluminum liner. Small holes are drilled in the reflective aluminum liner to provide a tangential flow of argon into the vortex region. The size and number of the injector holes were chosen on the basis of experimental two-component gas vortex tests described in Ref. 8. The injection system consists of four rows of 0.064-cm (0.025-in.)-dia injection holes with 48 holes in each row. The injection velocity on the buffer gas is approximately 293 cm/sec (9.6 ft/sec).

Effluent Handling System

The Nuclear Furnace is equipped with an effluent scrubber system as shown in Fig. 1. This scrubber system was built to handle fuel and fission fragments which might be emitted from fuel elements whose cladding had eroded. This same system is adaptable for use in handling the effluent of the test region. In addition to the effluent from the outflow end of the test region, there will also be effluent from the top of the test region. This effluent would be sampled as depicted in Fig. 14. The objective of the sampling manifold would be to allow effluent samples to be taken at various time intervals during operation and later examined for their content of unspent nuclear fuel and fission fragments. During start-up, the reaction of the effluent fluorides with the water injected into the scrubber will lead to the formation of hydrofluoric acid. This acid may become corrosive to the scrubber components. However, the total mass of fluorine passed through the test region during a typical operation would be from 30 to 300 gm (0.066 to 0.66 lb). The scrubber water flow rate is 10.9 kg/sec (24 lb/sec) and the mixture would result in a very weak acid solution. Post-test flushing of the scrubber components should minimize any corrosive effects.

INSTRUMENTATION REQUIRED FOR TESTS

The principal method of determining the performance of an in-reactor test will be through measurement of the heat balance in the various components of the test assembly. These components will be heated by the neutron and gamma fluxes from the driving reactor before any nuclear fuel is introduced into the vortex. The coolant flow requirements and temperatures in the system before introduction of nuclear fuel will serve as tare measurements for subsequent measurements of test performance. Once the test region is operating at a full-power, steady-state condition, temperature measurements of the coolant flow past the reflective liner and along the end walls will indicate the total amount of energy incident on the test region periphery. If these temperatures remain steady, it will be an indication that there is no deposition of nuclear fuel occurring on these surfaces. Control of flow rate through the test cell will be accomplished by orificing the various coolant, buffer-gas, and fuel injection supply lines. The orifice sizes are indicated in Table VI for both the argon and hydrogen coolant systems. Measurement of temperatures would be accomplished by the use of thermocouples. The position of the various thermocouples required to determine the heat balance in the test assembly are shown in Fig. 15. The instrumentation used would be the same type as that employed to measure coolant temperatures and surface temperatures in present Nuclear Furnace tests. Instrumentation leads from the thermocouples would be passed through the top of the test assembly as shown in Fig. 2 and then to the control console depicted schematically in Fig. 14.

One of the most important quantities to be observed in the tests is the spectral emission from the fissioning uranium plasma. Methods for observing the spectral emission from the test region would be thoroughly developed in the UARL r-f simulation tests. One method being considered employs an optical tube inserted through the top of the test configuration, down through the annular region between the end-wall coolant channels, and then through the end wall of the test cavity. It would also be desirable to insert a similar optical tube through the liner coolant passage to observe the spectral flux emitted radially. These could be tubes which have a low flow of neon bled through them allowing them to be transparent to the radiation emitted from the nuclear fuel cloud. Corner mirrors would be employed to conduct the emitted light to a detector system packaged within a cooled shield just beneath the bell shield which covers the entire Nuclear Furnace assembly (see Fig. 14). The detector signals would be transmitted to the control console from this instrumentation package. Tests to verify that such an instrumentation package can be housed under the bell shield, cooled, and shielded to the extent that the instrumentation will function properly should be performed, if possible, in early Nuclear Furnace tests.

In addition to the observation of the emitted spectrum of photon energy, it would be desirable to sample the buffer-gas, fuel, and bypass effluent as it comes out of the top of the test region. This sampling can be achieved by employing the gas sampling manifold depicted in Fig. 14 discussed previously. A number of sample tubes could be filled by drawing off samples from the effluent during the operation of the test. Analysis of the contents of the sample tubes could be performed at the same time that post-test examination and measurements are being made on the test configuration withdrawn in the remote assembly-disassembly bay.

TEST SAFETY CONSIDERATIONS

Pressure Vessel Failure Modes

The pressure vessel of the in-reactor test assembly should be designed such that if a failure occurs, a downward expulsion of both materials and gases results. This can be designed into the system by having a compression support at the top of the test assembly which bears against the bell shield. The probable failure mode would then be the expulsion of gases downward into the scrubber. Should the pressure vessel fail such that the lower portion of the test assembly were propelled downward, the aluminum casing in which the test region is inserted could be tapered such that motion of the lower pressure vessel dome would be restricted by the aluminum casing. This should prevent the downward-moving dome from striking the bottom of the scrubber and possibly penetrating the containment vessel of the scrubber should this type of failure occur. The volume of gases in the high-pressure test region relative to the total volume of gases in the Nuclear Furnace is approximately 2.5 percent. Immediate expulsion of the high-pressure gases from the test region into the remainder of the Nuclear Furnace pressure vessel would result in an increase of pressure in the entire system of approximately 25 percent (40 atm to 50 atm). Quick closing valves would be incorporated in all high-pressure systems such that a drop in pressure on the order of 10 or 20 atm would activate these valves and cut off high-pressure gas supplies. It should also be noted that the high-pressure gas supply lines are of small capacity and therefore would not add greatly to the total pressure pulse while the quick acting shut-off valves are being activated.

Heating Due to Fuel Deposition on Test Cell Walls

Deposition of fuel on the surfaces of the vortex chamber or on the surfaces of the thru-flow ducts through which the test effluent passes into the scrubber chamber will cause a steady rise in the temperature of the coolant fluids flowing through these particular components. Analyses were performed to determine which of these regions were most sensitive to the deposition of nuclear fuel. Three regions considered were the test region end walls, the reflective aluminum liner, and the thru-flow duct. The results of these calculations are shown in Fig. 16, from which it can be seen that the deposition of nuclear fuel on the thru-flow duct walls results in the largest rate of increase of the temperature. For example, if the rate of deposition of nuclear fuel were 0.1 gm/sec on the walls of the thru-flow ducts, after 5.0 sec the mixture of fuel, buffer gas, and bypass gas would increase in temperature by approximately 420 K. If this temperature rise were observed in an actual test, it could be used as a signal to terminate the injection of nuclear fuel and thereby limit an excursion in temperature, or

it could be used as a signal to increase the bypass flow in the thru-flow ducts to compensate for the energy added to the stream due to the presence of nuclear fuel on the walls. An additional measure which might be used to prevent further deposition on the walls of the thru-flow ducts would be the addition of fluorine gas to the bypass argon flow. It has been shown in experiments of Ref. 14 that the addition of fluorine to a uranium-hexafluoride mixture, which is dissociating either due to thermal energy or due to fission, tends to reverse chemical reactions such that the UF_6 content is restored.

Effluent Radioactivity Level

The total amount of nuclear fuel employed in a given test is approximately 300 gm, of which approximately 5×10^{-4} gm will undergo fission and create fission fragments. If contained and shielded, these fission products would not be considered a significant radiation hazard. The primary radiation hazard would occur in the case of accidental release of these fission products immediately after or during a test. Release of these fission products could result if a break occurred in the scrubber containment vessel or the pressure vessel of the Nuclear Furnace. Measures to prevent occurrence of a penetration of the containment envelope of the Nuclear Furnace were discussed above.

Emergency Shutdown Procedures

Rapid shutdown of the in-reactor test system would be required if an excessively high thermocouple reading indicated a high local temperature somewhere in the system or if a rapid drop in pressure indicated a penetration or leak in the test assembly pressure vessel. The temperature measurements throughout the system would indicate if there is a local or distributed buildup of uranium fuel on surfaces of the vortex chamber or thru-flow ducts. The effects of this buildup on temperature rise was discussed previously. Should a temperature rise occur which is in excess of a predetermined level, the emergency shutdown procedure would be to close off the fuel injection immediately. This shutdown would be automated and coupled to the temperature readout system. In the event of a rapid pressure drop in the system, the procedure should be to turn off both the fuel injection and the high-pressure gases. These systems could be activated by a limiting change in pressure in the system on the order of 10 to 20 atm. Since the phasing-in of the Nuclear Furnace hydrogen coolant as soon as the pressure approaches the ambient pressure of the system is already included in the system, the rapid shutdown of high-pressure gases would not interrupt the cooling of the test assembly.

R-F SIMULATION AND REACTOR PHYSICS EXPERIMENTS

Prior to conducting in-reactor tests, it is necessary to thoroughly check the performance characteristics of the entire test assembly. These tests could be conducted in the UARL 1.2-megw r-f heater facility. The principal components to be tested are the pressure vessel, the flow distribution system, the fuel injection system, and the emergency shutdown system. The pressure vessel is the component which will require the greatest number of tests in order to prove its integrity and to determine failure modes should the pressure vessel leak or fail. After a pressure vessel has been proven to be reliably strong and has also been shown to have the proper failure modes, if failure should occur, it would then be desirable to insert the flow-distribution end walls and test configuration inside the pressure vessel and cold-flow test the entire system.

The fuel injection system involves the use of liquid UF_6 and a system of uranium-particle/carrier-fluid injection. There are likely to be complicated chemical reactions taking place in these systems as well as complex flow rate control requirements. A prototype fuel injection system should be fabricated early in the program and tested to insure that fuel can be successfully injected into the test region with the necessary flow rate control. The complete test assembly should be tested with all of its components in the r-f tests at the power and pressure levels anticipated in the nuclear tests. The test assembly used should include all of the features of the in-reactor tests, especially the instrumentation required to measure test performance, spectral emission, and the location of the fissioning plasma within the test region.

R-F Simulation Experiments

A considerable number of preliminary non-nuclear tests of configurations similar in geometry to the in-reactor test unit have been performed using the 1.2-megw r-f heater to produce a radiant energy source. The calculated performance levels of the in-reactor tests described in this report and the performance levels obtained in these r-f tests are shown in Table IX. At present, the major difference between the two configurations arises from the difference in the operating pressure. In addition, tests in which uranium partial pressure ratios equal to 0.4 are achieved in the fuel-containment region must be performed to simulate conditions in the in-reactor test unit. The values of radiant heat flux per unit area which have been obtained in the r-f heater encompass the anticipated levels for the in-reactor tests, but these values were obtained with discharge diameters which were less than half of the test region diameter.

The dimensions of the test configuration used in r-f tests could be increased to equal those of the basic test configuration; however, to date the dimensions have been limited to a diameter of 2.24 in. and a length of 5.5 in. due to the existing work coils used in the r-f heater. Flow characteristics such as buffer-gas injection area and buffer-gas injection velocities have been used over a range encompassing those values anticipated for in-reactor tests. In addition, injection of UF_6 and WF_6 simulated fuel has been accomplished in a few tests in the r-f heater. These tests will be continued as the operating pressure level of the system within the r-f induction heater is increased. Thus, it appears possible that the conditions of performance and thermal environment and duplication of the gaseous and fuel materials can be simulated in r-f simulation tests. Plans for the FY 1972 r-f test program include the development of a 500-atm test section to be as compatible as possible with the design for an in-reactor test. Cold-flow experiments on this configuration are planned for the FY 1972 program so that the configuration will be ready for tests in the r-f induction heater during FY 1973.

Reactor Physics Experiments

There are several experiments which should be run in the Nuclear Furnace zero-power critical assembly to assess the effect of various components in the test configuration on the Nuclear Furnace reactor and on the thermal neutron flux profile in the test region. The most important test would be to assess the influence of the neutron-absorbing shield which is required in the fuel injection system. This shielded tube must run down the axis of the test assembly from the top end and must also have a branch duct which passes between the reflective aluminum liner and the pressure vessel to supply fuel to the injector at the bottom end of the test region. The influence of the shielded injector should be measured to evaluate its effect on test performance and also to evaluate the efficiency of the neutron-absorbing sheath in reducing the specific fission power level inside the fuel injector. Another test which should be performed in the zero-power critical assembly is to evaluate the worth of the approximately 6.0 gm of nuclear fuel in the test region at full-power conditions. In particular, it should be determined whether the reactivity worth of that fuel loading is great enough to be a detectable perturbation of control drum position.

One of the key instrumentation packages required for an in-reactor test is a system to observe the spectral emission from the uranium plasma. The best location for this instrumentation package is currently believed to be in the open region between the core lid and the bell shield (see Fig. 14). A reactor experiment should be performed to determine whether this instrument package can be adequately protected from the effects of the nuclear radiation environment by a cooled shield. This test should be run with a light source to simulate the uranium plasma emission and with the instrument package in place under the bell shield to determine the effectiveness of the cooled shield in preventing radiation interference with the detector.

If successful tests of a small-scale nuclear light bulb unit cell are run, the next step in proving the feasibility and operating characteristics of the nuclear light bulb engine would be to develop a self-critical cavity reactor. In the future, conceptual designs of a self-critical cavity which would have adequate performance to demonstrate the operating characteristics of the nuclear light bulb engine should be developed and zero-power critical assemblies should be built up to verify the physics calculations of this configuration.

ESTIMATED TEST COSTS

Detailed costs of an in-reactor test of the nuclear light bulb unit cell are difficult to estimate at this time because of uncertainties in the details of the design characteristics of the test cell, the coolant flow systems, and the fuel injection system. Therefore, the best estimates of cost at this time can be determined principally by selecting the time scale for development of a prototype test in the UARL r-f induction heater and adding these costs to the approximate cost of designing and building a Nuclear Furnace adapted for nuclear light bulb tests. It is estimated that the development of a prototype and thorough testing of the prototype at UARL will require approximately 30 months. The funding level to accomplish these tasks in this period of time would be on the order of 1.5 to 3 million dollars. Estimates of the cost of constructing a Nuclear Furnace for in-reactor tests range from 3 to 5 million dollars. Therefore, the cost to conduct a few demonstration tests as discussed in this report would be on the order of 4.5 to 8 million dollars.

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LIST OF SYMBOLS

A	Area, cm^2 or ft^2
C_T	Ratio of fuel-to-buffer-gas residence time, dimensionless
D_T	Test diameter, cm or ft
$\frac{dT}{dt}$	Rate of change of temperature, deg K/sec
g	Gravitational acceleration constant, cm/sec^2 or ft/sec^2
K_F	Density containment factor, dimensionless
k	Specific heat ratio, dimensionless
L_T	Unit cavity length, cm or ft
M_F	Fuel loading, gm or lb
P	Pressure, atm
P_O	Pressure downstream of orifice, atm
$\bar{P}_{F6}$	Average fuel partial pressure for volume inside r_6 , atm
$\bar{P}_{F6}/P$	Fuel partial pressure ratio, dimensionless
Q_F	Specific fission power, kw/gm of U-235
Q_R	Power radiated, kw or Btu/sec
Q_T	Total power, kw or Btu/sec
R	Reflectivity, dimensionless
r_F/r_T	Fuel to cavity radius ratio, dimensionless
r_T	Cavity radius, cm or ft

LIST OF SYMBOLS (Continued)

r_6	Fuel region radius, cm or ft
T	Temperature, deg K or deg R
T^*	Equivalent black-body radiating temperature, deg K or deg R
t_a	Operating time of in-reactor test unit, min
T_6	Temperature at edge of fuel, deg K or deg R
v	Specific volume, cm^3/gm or ft^3/lb
W	Mass flow rate, gm/sec or lb/sec
ρ_{B6}	Buffer-gas density at edge of fuel, gm/cm^3 or lb/ft^3
$\bar{\rho}_{F6}$	Average fuel density, gm/cm^3 or lb/ft^3
τ_F/τ_B	Ratio of average fuel residence time to buffer-gas residence time, dimensionless

TABLE I

UNIT CELL DIMENSIONS AND MATERIAL SPECIFICATIONS

Schematic Diagram of Unit Cell Shown in Fig. 3

Midplane Cross Section				
Region	Material	ID, cm (in.)	OD, cm (in.)	Thickness, cm (in.)
Fuel	U-235 or Pu-239 and Argon	--	3.96 (1.56)	--
Buffer Gas	Argon	3.96 (1.56)	6.60 (2.60)	1.32 (0.57)
Reflective Aluminum Liner	Aluminum	6.60 (2.60)	6.85 (2.70)	0.125 (0.05)
Liner Coolant	Argon or Hydrogen	6.85 (2.70)	7.21 (2.84)	0.18 (0.07)
Pressure Vessel Liner	Aluminum	7.21 (2.84)	7.36 (2.90)	0.075 (0.03)
Fiberglass Pressure Vessel	Fiberglass Composite	7.36 (2.90)	8.13 (3.20)	0.385 (0.15)
Cross Section Midway Between End Wall and Pressure Cap				
Fuel Injector	Boral	0.031 (0.012)	0.092 (0.036)	0.030 (0.012)
Inner Bypass Flow	Argon	0.092 (0.036)	0.762 (0.300)	0.325 (0.132)
Thru-Flow-Duct Inner Wall	Stainless Steel	0.762 (0.300)	1.020 (0.400)	0.127 (0.050)
Annular Thru-Flow Duct	U-235 or Pu-239 and Argon	1.020 (0.400)	1.530 (0.600)	0.254 (0.100)
Thru-Flow-Duct Outer Wall	Stainless Steel	1.530 (0.600)	1.780 (0.700)	0.127 (0.050)
Outer Bypass Flow	Argon	1.780 (0.700)	2.284 (0.900)	0.254 (0.100)
Bypass Flow Annulus Outer Wall	Aluminum	2.284 (0.900)	2.540 (1.000)	0.127 (0.050)
End-Wall-Coolant Inner Annulus	Argon or Hydrogen	2.540 (1.000)	3.050 (1.200)	0.254 (0.100)
End-Wall-Coolant Inner Annulus Wall	Aluminum	3.050 (1.200)	3.300 (1.300)	0.127 (0.050)
Stagnant Region	Argon or Hydrogen	3.300 (1.300)	5.840 (2.300)	1.270 (0.500)
Stagnant-Region Outer Wall	Aluminum	5.840 (2.300)	6.100 (2.400)	0.127 (0.050)
End-Wall-Coolant Outer Annulus	Argon or Hydrogen	6.100 (2.400)	6.600 (2.600)	0.254 (0.100)

Note: Remaining dimensions from reflective aluminum liner out to pressure vessel OD same as in midplane cross section.

TABLE II

VALUES SELECTED FOR PARAMETERS AFFECTING
IN-REACTOR TEST PERFORMANCE

Specifications for Unit Cell Given in Fig. 3 and Table I

Parameter	Range of Values	Reference Value
Operating Pressure, P-atm	500-3000	500
Specific Fission Power, Q_F -kw/gm of U-235	0-100	30.5
Fuel-to-Cavity Radius Ratio, r_F/r_T	0.6-0.9	0.6
Density Containment Factor, $\bar{\rho}_{F6}/\rho_{B6} = K_F$	0.3-0.7	0.7
Ratio of Average Fuel Residence Time to Buffer-Gas Residence Time, τ_F/τ_B	0.1-1.0	1.0
Aluminum Liner Reflectivity, R	0.5-0.9	0.9
Fuel Partial Pressure Ratio, $\bar{P}_{F6}/P$	0.3-0.45	0.4

TABLE III

PERFORMANCE LEVELS FOR IN-REACTOR TESTS

Specifications for Unit Cell Given in Table I and Fig. 3

Operating Pressure, $P = 500$ atm

$$\bar{p}_{F_6}/\rho_{B_6} = 0.7, r_F/r_T = 0.6, \tau_F/\tau_B = 1.0, R = 0.9$$

Argon Buffer Gas

	Reference Design Performance	Performance Using Pu-239 Fuel
Fuel	U-235	Pu-239
Specific Fission Power, Q_F -kw/gm of Fuel (Btu/sec-lb)	30.5 (1310)	64.1 (2755)
Fuel Loading, M_F -gm (lb)	6.22 (1.37×10^{-2})	5.33 (1.18×10^{-2})
Total Power, Q_T -kw (Btu/sec)	189.8 (180.0)	341.8 (324.0)
Power Radiated, Q_R -kw (Btu/sec)	173.4 (164.3)	321.5 (304.5)
Surface Radiating Temperature, T_6 -deg K	5940 (10,690)	6920 (12,460)
Equivalent Black-Body Radiating Temperature, T^* -deg K (deg R)	3333 (6000)	3890 (7000)
Average Fuel Partial Pressure for Volume Inside r_6 , $\bar{p}_{F_6}$ -atm	195	190
Fuel Partial Pressure Ratio, $\bar{p}_{F_6}/P$	0.39	0.38

TABLE IV

FUEL AND BUFFER-GAS FLOW CONDITIONS FOR IN-REACTOR TESTS

Specifications for Unit Cell Given in Table I and Fig. 3

Mass of Fuel in Cavity, gm (lb)	6.220 (1.37×10^{-2})
Fuel Flow Rate Per Injector Tube, gm/sec (lb/sec)	0.800 (1.77×10^{-3})
Argon Carrier-Gas Flow Rate Per Injector Tube, gm/sec (lb/sec)	0.133 (2.92×10^{-4})
Fission Heating Rate in Fuel, kw/gm	30.5
Fuel Injection Temperature, deg K (deg R)	833 (1500)*
Argon Buffer-Gas Flow Rate, gm/sec (lb/sec)	25.3 (0.056)
Argon Bypass Flow Rate, gm/sec (lb/sec)	45.4 (0.100)
Temperature of Fuel and Buffer-Gas Mixture Exiting Cavity, deg K (deg R)	1755 (3160)
Inlet Temperature of Bypass Flow, deg K (deg R)	278 (500)
Temperature of Inner Bypass Flow at Injection, deg K (deg R)	684 (1229)
Temperature of Outer Bypass Flow at Injection, deg K (deg R)	1000 (1800)
Mixed Mean Temperature of Fuel, Buffer and Bypass Flow, deg K (deg R)	1170 (2105)
Temperature of Fuel, Buffer and Bypass Flow at Outlet of Active Core Region, deg K (deg R)	830 (1495)

* Heat generation in fuel injectors to 2 kw/gm by addition of neutron absorbing material in tube walls.

TABLE V

COOLANT FLOW CONDITIONS FOR IN-REACTOR TESTS

Specifications for Unit Cell Given in Table I and Fig. 3

	Argon Coolant	Hydrogen Coolant
Liner Coolant Flow Rate, gm/sec (lb/sec)	9360 (20.6)	213 (0.47)
Liner Coolant Annular Width, cm (in.)	0.190 (0.075)	0.190 (0.075)
Total Heat Load to Liner, kw (Btu/sec)	147 (139.3)	147 (139.3)
Coolant Inlet Temperature, deg K (deg R)	278 (500)	278 (500)
Temperature Rise in Liner Coolant, deg K (deg R)	29.5 (53)	46.5 (84)
Film Temperature Difference in Liner, deg K (deg R)	218 (392)	182 (328)
Wall Temperature Difference Across Liner, deg K (deg R)	58 (104)	58 (104)
Maximum Temperature in Liner, deg K (deg R)	583.5 (1049)	564.5 (1016)
Coolant Flow Rate in Each End Wall, gm/sec (lb/sec)	635 (1.4)	11.8 (0.026)
Radiant Heat Flow to Each End Wall, kw (Btu/sec)	15 (14.2)	15 (14.2)
Temperature Rise in End-Wall Coolant, deg K (deg R)	45.5 (82)	83 (149)
Film Temperature Difference in End-Wall Region, deg K (deg R)	164 (295)	164 (295)
Temperature Difference Across End Wall, deg K (deg R)	32 (58)	32 (58)
Maximum Temperature in End Wall, deg K (deg R)	519.5 (935)	557 (1002)

TABLE VI

ORIFICE SIZES FOR FLOW CONTROL
FOR IN-REACTOR TESTS OF A UNIT CELL

Specifications for Unit Cell Given in Table I and Fig. 3

Flow Rate Requirements Given in Tables IV and V

Orifice Flow Per Unit Area Calculated Using

$$\frac{W}{A} = \left[\frac{2 \text{ kg}}{k-1} \left(\frac{P}{v} \right) \left\{ \left(\frac{P_0}{P} \right)^{2/k} - \left(\frac{P_0}{P} \right)^{\frac{k+1}{k}} \right\} \right]^{1/2}$$

Coolant Inlet Properties at 278 K (500 R) and 550 atm

Component	Total Flow Rate gm/sec (lb/sec)	Number of Orifices	Flow Rate Per Orifice, gm/sec (lb/sec)	Orifice Area cm ² (in. ²)	Orifice Diameter, cm (in.)
Liner, Argon Coolant	9360 (20.6)	8	1170 (2.58)	0.136 (2.11 x 10 ⁻²)	0.416 (0.164)
Liner, Hydrogen Coolant	213 (0.47)	8	26.6 (0.0588)	3.67 x 10 ⁻³ (5.69 x 10 ⁻⁴)	0.066 (0.026)
End-Wall, Argon Coolant	1370 (2.8)	4	317 (0.7)	3.68 x 10 ⁻² (5.7 x 10 ⁻³)	0.216 (0.0852)
End-Wall, Hydrogen Coolant	23.6 (0.052)	4	5.9 (0.013)	8.14 x 10 ⁻³ (1.26 x 10 ⁻⁴)	0.032 (0.0126)
Argon Bypass	45.4 (0.10)	8	5.67 (0.0125)	6.56 x 10 ⁻⁴ (3.0 x 10 ⁻⁴)	0.0289 (0.0196)
Buffer Gas	25.4 (0.056)	8	3.18 (0.007)	3.68 x 10 ⁻⁴ (5.7 x 10 ⁻⁵)	2.19 x 10 ⁻² (8.55 x 10 ⁻³)

TABLE VII

SEQUENCE OF EVENTS FOR REFERENCE IN-REACTOR TEST

t_A = Time in Minutes After Nuclear Furnace Reaches Steady-State Power Level

$t_A = 0$ to $t_A = 1$ min
1. Flow hydrogen from nuclear furnace supply system through test region to bring components to thermal equilibrium.
$t_A = 1$ min to $t_A = 2.5$ min
1. Pressurize test region to operating level (500 atm).
2. Start high-pressure buffer-gas and coolant flow.
$t_A = 2.5$ min to $t_A = 3.5$ min
1. Start injection of liquid UF_6 .
$t_A = 3.5$ min to $t_A = 4$ min
1. Add particles of uranium to UF_6 to increase uranium partial pressure.
$t_A = 4$ min to $t_A = 7$ min
1. Full-power, steady-state operation.
2. Measure flows, temperatures and record spectral emission data.
$t_A = 7$ min to $t_A = 8$ min
1. Shut off fuel injection flow.
2. Purge fuel injection lines with argon.
$t_A = 8$ min to $t_A = 9$ min
1. Depressurize unit to nuclear furnace level.
2. Switch over to cooling with hydrogen from nuclear furnace supply system.
$t_A = 9$ min to $t_A = 10$ min
1. Shut down nuclear furnace.

TABLE VIII

TOTAL WEIGHT OF MATERIAL EXPENDED DURING EACH TEST RUN

Test Time = 10 min

Test Sequence Described in Table VII

Flow Circuit	Total Weight, gm (lb)
Nuclear Furnace Hydrogen*	900 (1.99)
Argon Buffer Gas	1.06×10^4 (23.5)
Liner and End-Wall Coolant, Argon	2.67×10^6 (5900)
Liner and End-Wall Coolant, Hydrogen	5.34×10^4 (118)
U-235 Fuel (Uranium In UF_6 and Particles)	390 (0.86)
Argon Carrier Gas for UF_6 -Argon System	571 (1.26)
Fluorine Contained In UF_6 for UF_6 -Argon System	26 (0.06)
Fluorine Contained In UF_6 for All- UF_6 System	298 (0.67)

*Coolant required to remove neutron and gamma ray heating when test cell not in operation.

TABLE IX
COMPARISON OF CHARACTERISTICS OF IN-REACTOR
TESTS AND TESTS IN THE 1.2-MEGW R-F HEATER

	In-Reactor Test	R-F Simulation Tests (To Date)*
Cavity Inside Diameter, cm (in.)	6.60 (2.60)	5.7 (2.24)
Cavity Length, cm (in.)	17.78 (7.00)	14.0 (5.5)
Radiant Heat Flux Per Unit Area, kw/cm ² (Btu/sec-ft ²)	0.71-1.31 (626-1155)	0.09-2.52 (80-2220)
Equivalent Black-Body Radiating Temperature, deg K (deg R)	3330-3910 (6000-7000)	2000-4670 (3600-8400)
Operating Pressure, atm	500	40
Fuel	UF ₆ , U-Particles	WF ₆ , UF ₆
Buffer Gas	Argon	Argon
Buffer-Gas Injection Velocity, m/sec (ft/sec)	2.9 (9.6)	1.52-45.7 (0.50-150.0)
Buffer-Gas Weight Flow, gm/sec (lb/sec)	29 (0.064)	2.27-13.6 (0.005-0.03)
Fuel-to-Cavity Radius Ratio	0.6	0.44
Total Power Level, kw (Btu/sec)	189.8-341.8 (180.0-324.0)	Up to 258 (245)

*Values indicate range of levels achieved in any given test configuration to date.

SCHEMATIC DIAGRAM OF NUCLEAR FURNACE

POWER LEVEL = 44 - 66 MEGW

SCHEMATIC DIAGRAM OF PIPING AND ELECTRICAL CONNECTIONS TO TEST CELL

IN-REACTOR TEST UNIT CELL

SEE TABLE I FOR MATERIALS
SPECIFICATIONS AND DETAILED DIMENSIONS

FIG. 3

SCHEMATIC DIAGRAMS OF PRESSURE VESSEL END WALLS

SEE TABLE VI FOR ORIFICE SIZES FOR CONTROL OF FLOW

SEE TABLES IV AND V FOR REFERENCE FLOW CONDITIONS

(a) CAPPED END WALL (INLET END)

(b) DOMED END WALL (OUTLET END)

SCHEMATIC DIAGRAM OF UNIT CELL WITH ALUMINUM PRESSURE VESSEL AND OFF-CENTER FUEL INJECTORS

USE OF ALUMINUM PRESSURE VESSEL REDUCES
LINER ID TO 6.35 CM (2.50 IN.)

FLOW ANNULI AND WALL THICKNESS SAME AS
SHOWN FOR UNIT CELL IN FIG. 3

UNIT CELL FOR IN-REACTOR TESTS OF TRANSPARENT WALLS AND PROPELLANT HEATING

RANGES OF EQUIVALENT BLACK BODY RADIATING TEMPERATURES AND EDGE-OF-FUEL TEMPERATURES FOR IN-REACTOR TESTS OF A UNIT CELL

SPECIFICATIONS FOR UNIT CELL GIVEN IN TABLE I AND FIG. 3

OPERATING PRESSURE = 500 ATM

U-235 NUCLEAR FUEL

r_F / r_T = FUEL-TO-CAVITY RADIUS RATIO

$$K_F = \bar{\rho}_{F6} / \rho_{B6}$$

$$T_6 = 1.78 T^*$$

RANGES OF RADIATED POWER FOR IN-REACTOR TESTS OF A UNIT CELL

SPECIFICATIONS FOR UNIT CELL GIVEN IN TABLE I AND FIG. 3

OPERATING PRESSURE = 500 ATM

U-235 NUCLEAR FUEL

r_F/r_T = FUEL-TO-CAVITY RADIUS RATIO

$$K_F = \bar{\rho}_{F6}/\rho_{B6}$$

RANGE OF RADIATING TEMPERATURES AND POWER LEVELS FOR IN-REACTOR TESTS OF A UNIT CELL FOR REFERENCE VALUES OF PARAMETERS AFFECTING PERFORMANCE

SPECIFICATIONS FOR UNIT CELL GIVEN IN TABLE I AND FIG. 3

VALUES OF REFERENCE DESIGN PARAMETERS

$$P = 500 \text{ ATM}$$

$$C_T = 1.0$$

$$R = 0.9$$

$$r_F / r_T = 0.6$$

$$K_F = 0.7$$

$$r_T = 3.30 \text{ CM (1.30 IN.)}$$

$$L_T = 17.78 \text{ CM (7.00 IN.)}$$

RANGES OF NUCLEAR FUEL PARTIAL PRESSURE IN FUEL-CONTAINMENT REGION FOR IN-REACTOR TESTS OF A UNIT CELL

SPECIFICATIONS FOR UNIT CELL GIVEN IN TABLE I AND FIG. 3

$$R = 0.9$$

$$C_T = 1.0$$

$$K_F = 0.7$$

$$r_F/r_T = 0.6$$

EFFECT OF VARIATIONS OF FLOW AND DESIGN PARAMETERS ON IN-REACTOR TEST PERFORMANCE

K910900-11

ALL PARAMETERS HELD AT REFERENCE DESIGN
VALUES EXCEPT FOR INDICATED VARIABLE

VALUES OF REFERENCE DESIGN PARAMETERS

$P = 500 \text{ ATM}$ $R = 0.9$ $K_F = 0.7$
 $C_T = 1.0$ $r_T = 3.30 \text{ CM (1.30 IN.)}$ $L_T = 17.78 \text{ CM (7.00 IN.)}$ $r_F/r_T = 0.6$ $Q_F = 30 \text{ KW/GM}$

○ DENOTES REFERENCE DESIGN VALUE

(a) EFFECT OF VARIATION IN FUEL-TO-CAVITY
RADIUS RATIO

(b) EFFECT OF VARIATION IN DENSITY CONTAINMENT FACTOR

(c) EFFECT OF VARIATION IN FUEL-TO-BUFFER
GAS RESIDENCE TIME

(d) EFFECT OF VARIATION IN ALUMINUM LINER REFLECTIVITY

FIG. 11

RANGES OF RADIATING TEMPERATURES AND POWER LEVELS FOR IN-REACTOR TESTS OF A UNIT CELL AT DIFFERENT OPERATING PRESSURES

SPECIFICATIONS FOR UNIT CELL GIVEN IN TABLE I AND FIG. 3

$$R = 0.9$$

$$C_T = 1.0$$

$$K_F = 0.7$$

$$r_F/r_T = 0.6$$

U-235 NUCLEAR FUEL

RANGES OF RADIATING TEMPERATURES AND POWER LEVELS FOR IN-REACTOR TESTS OF A UNIT CELL WITH A PROPELLANT HEATING CHANNEL

GEOMETRY AND DIMENSIONS OF UNIT CELL WITH PROPELLANT HEATING CHANNEL SHOWN IN FIG. 6

$$P = 500 \text{ ATM}$$

$$C_T = 1.0$$

$$R = 0.8$$

$$r_F / r_T = 0.6$$

$$K_F = 0.7$$

$$r_T = 2.54 \text{ CM (1.00 IN.)}$$

$$L_T = 17.78 \text{ CM (7.00 IN.)}$$

SCHEMATIC FLOW DIAGRAM OF NUCLEAR LIGHT BULB IN - REACTOR TEST

INSTRUMENTATION LOCATIONS FOR IN-REACTOR TESTS OF A UNIT CELL

SPECIFICATIONS FOR UNIT CELL GIVEN IN TABLE I AND FIG. 3

FLOW CONDITIONS AND CALCULATED TEMPERATURES IN UNIT CELL GIVEN IN TABLES IV AND V

FIG. 15

TEMPERATURE VARIATIONS DUE TO POSSIBLE DEPOSITION OF NUCLEAR FUEL ON END WALLS, REFLECTIVE ALUMINUM LINER AND THRU-FLOW DUCT WALLS

TOTAL FUEL FLOW RATE, $W_F = 1.6 \text{ GM/SEC}$

THERMOCOUPLE LOCATIONS SHOWN IN FIG. 15

RATES OF CHANGE OF TEMPERATURE SHOWN DUE TO FISSIONING OF DEPOSITED FUEL ONLY

United Aircraft Research Laboratories UNITED AIRCRAFT CORPORATION

U
A

October 15, 1971

U. S. Atomic Energy Commission
Division of Technical Information Extension
P. O. Box 62
Oak Ridge, Tennessee

Subject: Transmittal of reports comprising the second Interim Summary Technical Report under Contract SNPC-70, "Investigation of Gaseous Nuclear Rocket Technology"

Enclosures: (A) Report K-910900-7 (E) Report K-910900-11
(B) Report K-910900-8 (F) Report K-910904-2
(C) Report K-910900-9 (G) Report K-990929-2
(D) Report K-910900-10

Gentlemen:

Copies of seven reports, Enclosures (A) through (G), which comprise the second Interim Summary Technical Report under Contract SNPC-70 between the joint AEC-NASA Space Nuclear Systems Office and the Research Laboratories of United Aircraft Corporation, are being sent to you as part of the general distribution of reports under this Contract.

Very truly yours,

UNITED AIRCRAFT RESEARCH LABORATORIES

George H. McLafferty
George H. McLafferty
Senior Program Manager

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