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Informal Report

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A Feasibility Study for a Postaccident Heat Removal Facility

Issued: January 1978

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of the University of California
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**E. W. Barts
C. E. Apperson, Jr.
W. E. Dunwoody
J. G. Bennett**

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A FEASIBILITY STUDY FOR A
POSTACCIDENT HEAT REMOVAL FACILITY

by

E. W. Barts, C. E. Apperson, Jr.,
W. E. Dunwoody, and J. G. Bennett

ABSTRACT

An initial feasibility investigation for PAHRTEF, a Postaccident Heat Removal Test Facility, is presented. The facility would provide an experimental capability for PAHR experiments beyond that available in any currently existing or proposed U. S. safety test facility. The facility design presented in this report is based upon the technology developed for the ROVER nuclear rocket propulsion program. The core is a graphite-moderated, helium-cooled, epithermal core with radial reflector control. The PAHR experiments are located just below the reactor containment vessel, very near the bottom of the core. The experiments (up to 55% enriched) are driven and controlled by neutrons leaking axially from the core such that the PAHRTEF core and the experiment form a coupled reactor system. The experiment can be designed so that it is extremely unlikely that the test fuel by itself could form a critical system. The investigation indicates that adequate fission heating of large PAHR experiments could be provided at low driver core power levels. Both the reactor and the experiment handling and examination equipment can use available technology and, whenever possible, existing equipment and buildings.

I. INTRODUCTION

The Los Alamos study of fast reactor safety test facilities¹ identified four separate types of facilities to satisfy a broad range of

experimental requirements. The characteristics of a Facility D, a facility to perform Postaccident Heat Removal (PAHR) experiments involving large masses of fuel and materials with relatively low internal heat generation, were identified.

A facility to perform Postaccident Heat Removal experiments has basically different requirements from one designed to study the initial and transition phases of a whole core-accident sequence. The differences are:

1. the PAHR facility is a steady-state facility providing decay power heat levels in relatively large masses of fuel, perhaps through low-power fission heating, rather than an adiabatic facility with burst capability, which heats the test fuel at full steady-state power levels and then simulates short-period reactor excursions (e.g., a Facility A or B);
2. the PAHR facility must provide for the uniform heating of large experiments of a squat geometry rather than shaped heating of a single subassembly (or larger) experiment containing long prototypic fuel elements; and
3. the PAHR facility must be capable of handling experiments of large diameter, small height, and low cooling requirements rather than experiments that provide prototypic thermal-hydraulic conditions for full-sized fuel elements and subassemblies.

Because of these differences, an initial investigation of a potentially low-cost and rapidly constructed PAHR facility has been performed. Such a facility would not provide all the capability of a Safety Test Facility,² but it would provide a test reactor that is better suited to and has greater capability for the performance of PAHR experiments. If a PAHR facility were constructed, the STF then could be reserved for tests better utilizing the STF's transient capability.

The purpose of a PAHR facility is to provide a test reactor for the performance of experiments to improve the understanding of the basic PAHR phenomenon and to demonstrate the postaccident containment capability of fast reactor designs. In-pile testing is necessary to provide volumetric heating of prototypic materials on a relatively large scale. Section II of this proposal identifies the required experimental capability of a PAHR facility and, in particular, identifies categories of in-pile tests that would use a PAHR facility. Section III describes briefly a facility at the Nuclear

Rocket Development Station (NRDS) at Jackass Flats, Nevada, which could be adapted for use as a PAHR facility. Section IV deals with the neutronic analyses of a helium-cooled, graphite-moderated driver reactor and an experimental configuration located below the reactor. Section V is concerned with the cooling requirements for the driver reactor. Section IV describes the experiment support requirements, including the vehicle handling and cooling requirements. Appendix A provides background information on the Rover Programs for which the facilities at NRDS were developed. Appendix B describes the details of the thermal-hydraulic model used to study cooling of the driver core.

II. REQUIRED EXPERIMENTAL CAPABILITY

The goal of PAHR experiments is to understand the long-term cooling and containment of accident-generated core debris. This implies a study of internally heated mixtures of fuel and steel in an appropriate postaccident environment. Effective retention of the core debris must be guaranteed so that there is no danger to the environment or the public. Adequate retention may occur because of the intrinsic design of the reactor, and if so, the related physical phenomena should be understood. If changes in design or additional retention systems are necessary, they should be based upon reliable engineering data. In any case, the retention system may have to be demonstrated to be adequate by proof tests.

A great deal of information can be obtained in out-of-pile experiments using simulant materials, electrically heated fuel, or thermite reactions. Other information can be obtained from small in-pile tests in existing facilities. In-pile testing capability is necessary to provide volumetric heating of prototypic materials. The STF will provide the capability of performing somewhat larger scale in-pile tests than possible in currently available facilities.

The STF is an adiabatic facility. It is designed primarily for performing severe overpower and transition-phase safety experiments without significant fuel preconditioning. PAHR tests in STF are limited to several minutes in length. This is true for any adiabatic facility that cannot operate at a steady-state power level adequate to simulate long-term fission product decay

heating. PAHR tests in STF are also limited in geometry. Tests with debris beds or molten pools of sufficient size and diameter-to-depth ratio to demonstrate containment cannot be performed.

Preliminary PAHR test requirements can be defined to provide a basis for the preliminary design of a special purpose PAHR facility. This facility would provide PAHR testing capability superior to that of STF and would relieve STF from performing PAHR tests.

Many specific PAHR phenomena and possible PAHR accident sequences are of interest. The phenomena and sequences depend upon the particular reactor and accident considered. In general, the problems of interest are:

1. decay fission product disposition and behavior, such as the fractionation of fission products from the fuel and subsequent distribution in fuel, steel, and sodium;
2. particulate/debris bed behavior, including the formation, interaction with sodium, settling, leveling, and heat transfer from, including dry-out and melting;
3. molten fuel melt-out, including in-place cooling, flow, and freezing below the core;
4. molten pool behavior with heat transfer to and the interactions with structures within the primary vessel; and
5. molten pool/core debris behavior including interactions with retention materials exterior to the primary vessel.

Not all of the above phenomena will require in-pile tests in a PAHR facility.^{3,4} Furthermore, specification of the precise experiments that will be required can be expected to change with changes in the understanding of the PAHR process and in the utilization of inherent and engineered retention concepts. A few assessments of experimental needs have been made.^{1,5,6,7} Many items are common to these assessments and it appears that in-pile tests can be divided into a few general categories.

A. Experiment Category I -- Containment Tests

Containment tests are proof tests of various containment barriers. In these tests debris beds or molten pools would be contained by steel, concrete, ceramics, or other materials. The interaction of the fuel, with simulated

decay heating, on the various containment barriers would be studied with a geometry and scale adequate to demonstrate containment.

For our preliminary reactor concept investigations a nominal test fuel heating rate of 5 kW/kg, simulating 2.5% of full-power decay heating of fuel with a steady-state power of 200 kW/kg, was chosen. A typical fuel layer thickness, with no porosity, of 50 mm was also selected. This quantity of material should provide a debris bed sufficient to study self-leveling and dry-out as well as to provide enough fuel thickness to obtain a molten pool. The selected test fuel diameter is 700 mm, based on obtaining a relatively flat radial power distribution with a 84-cm-diam driver core. The resulting diameter-to-depth ratio is 14 for zero porosity and 7 for 50% porosity, which should be satisfactory for tests requiring large diameter-to-depth ratios. The test parameters selected lead to a test fuel mass of about 167.4 kg and a test power of about 800 kW. These values were selected for preliminary design purpose and, as such, are subject to revision.

B. Experiment Category II -- Phenomenological Tests

Phenomenological tests are designed to enhance the understanding of those physical processes involved in the PAHR accident sequence which cannot be adequately studied out of pile. Such processes would include debris bed dry-out and interactions between molten fuel and ex-primary vessel retention materials. As previously indicated, tests in this category would be done when large-scale volumetric heating of prototypic materials is required to confirm other test results.

For preliminary design purposes the same test fuel mass, 167.4 kg; test fuel power density, 5 kW/kg; and total test power, 800 kW, is used as in Experiment Category I. The main difference is in test geometry; the test depth for zero-porosity fuel is 200 mm and the test diameter is 350 mm. This depth allows the study of deep-debris beds or molten pools. Note that the depth of the bed or pool during the PAHR accident sequence will be limited by the average enrichment of the core and the dimensions of the core prior to the accident, compared to the areas over which the debris bed or molten pool collects or acts. Again, these preliminary design values are subject to revision as experiments are better defined. Of course, the use of the PAHR facility for smaller PAHR tests is not precluded.

C. Experiment Category III -- Integral Tests

Integral tests follow the PAHR sequence from the initial core configuration to the final containment of the core debris. These tests involve the fuel in a bundle of seven or nineteen subassemblies (700 to 2000 kg) moving over long axial distances (several meters). Very few, if any, integral tests need to be performed. For desired tests, the mass of fuel involved suggests a self-driven experiment which could be operated long enough to build in fission products for actual decay heating. The motion of the fuel (and fission products) could then be followed in the proportioned reactor system. A specific PAHR facility would not be required for integral tests.

Test times ranging from several minutes to a few days may be required for Experiment Category I and Experiment Category II tests. Any experiment lasting over several minutes in time requires a steady-state facility. Thus, a requirement for total energy deposition or total experiment time does not have any meaning for a steady-state PAHR facility and a design value has not been given. Note that a steady-state power level would not be used to build in a significant fission product loading. The steady-state power level is that required to simulate decay heating by fission heating. Fission product fractionization can be assessed in other in-pile facilities or in out-of-pile experiments.

A necessary requirement placed on the proposed PAHR facility is that of safety. The enrichment of the test must be low enough to insure that the k_{eff} for the test alone is well below 1.0 and that the k_{eff} of the combined test and driver core is controllable. This must be true, regardless of any expected test configuration.

Provided the safety criteria are met, the performance of PAHR-type tests should be fairly straightforward. The coupling of the motion of the test fuel to the driver is much less than that in multi-subassembly tests in the STF, and PAHR tests do not involve the large-scale rapid movement of fuel which may occur in certain tests in STF.

The preceding experiment size requirements are similar to those desired in the Los Alamos safety test facility study.¹ This experiment size allows proof-in-principle testing of PAHR systems such as core catchers. The power level specified is that necessary to perform experiments without building in fission products. In addition, the experimental duration can be as long as desired.

III. PAHR FACILITY DESCRIPTION

The proposed location for the PAHR facility is at the test cell "C" complex in area 25 of the Nuclear Rocket Development Station at Jackass Flats, Nevada. Location and plot plans are shown in Figs. 1 and 2. This site was chosen to take advantage of existing facilities at a remote test location.

The PAHR facility will consist of a driver reactor, primary and secondary heat transport systems for the driver reactor, an experiment test vehicle, a test vehicle cooling system, and control facilities. The reactor and test vehicle control will be in a separate building from the reactor.

A. Reactor Building

The reactor building will contain the primary and secondary containment for the driver reactor, the test vehicle, and the primary coolant systems (including low-pressure and high-pressure helium receivers and gas cleanup systems). Outside of the secondary containment, the building will have gas sampling rooms, electrical equipment rooms, and mechanical equipment rooms. The gas sampling rooms will contain equipment for sampling the primary-coolant helium, the secondary-containment nitrogen, the test-vehicle argon, and the cooling air flow exiting from the high-efficiency particulate air filters (HEPA).

The primary containment of the driver reactor will be inerted by the helium coolant, the secondary containment will be inerted with nitrogen to maintain the oxygen level below 1%, and the test vehicle will be inerted with argon. The secondary containment will be designed for a maximum overpressure of two atmospheres.

The electrical equipment rooms will contain power switchgear, termination boxes, and signal conditioning equipment. The mechanical equipment rooms will contain secondary heat transport equipment, the building heating and ventilating equipment, hydraulic equipment, pneumatic equipment, and vacuum equipment.

A schematic cross-section of the reactor, test vehicle, and some of the equipment is shown in Fig. 3.

B. Reactor Systems

The core chosen is a short, one-meter-high, version of the PHOEBUS 1A nuclear reactor developed for and tested during the Rover Nuclear Rocket program.

Fig. 1. Nevada Test Site - Area 25 location plan.

Fig. 2. Test Cell "C" Complex - Nevada Test Site - Area 25

This reactor design was chosen to utilize, wherever possible, existing reactor components, facilities, and experience. The objective is to eliminate development time and reduce overall facility costs. The PHOEBUS 1A was cooled by hydrogen. The coolant chosen for this facility is helium. Helium coolant gives a harder neutron spectrum in the core and reduces the amount of moderating material between the reactor and the fissionable material in the experiment. A conventional heat transport system can be used, with primary and secondary coolant systems, to cool the reactor.

1. Reactor Vessel. The reactor vessel is shown in Fig. 4 and will be used both to support the reactor core, reflectors and control rods and to contain the helium coolant. The reactor vessel, primary heat transport system, and intermediate heat exchanger will be designed and fabricated according to the ASME Code, Section III, Class I rules for an internal pressure of 3.45 MPa and 700 K wall temperature. The vessel will be of welded construction approximately 1.6 m in diameter and 3.0 m high. The material considered at this time is Type SA-240 stainless steel. Internal radiation shielding is not required because of the relative short-term operation of the reactor.

Fig. 3.
PAHR reactor building cross section.

Fig. 4.
PAHRTEF core and PAHR experiment.

2. Reactor Core. The driver reactor core, 0.84-m-diam by 1-m-high, contains 1819 hexagonal Rover-type fuel elements. The fuel elements are 19.2 mm across flats with a round central flow channel of 12.2 mm diam. Unloaded graphite filler elements are used as required at the outside of the core to fill out the cylindrical geometry. The core is surrounded radially by an inner 91-mm-thick graphite reflector and an outer 114-mm-thick beryllium reflector. Twelve rotating control rods in the beryllium reflectors are 102-mm-diam beryllium cylinders, each covered with 120° of its surface with a $^{10}\text{B-Al}$ or $^{10}\text{B-Cu}$ plate. The control rods are driven by stepping motors with speed controlled by pulse repetition rate. Control rod position is measured by digital counting of the pulses driving the drums with a potentiometer analog

indication provided as backup. Scram is provided by loss of power on a magnetic clutch allowing (by a spring drive) the return of the control drum to the full-poison position. The top of the core is reflected by 50 mm of graphite and the bottom of the core is supported on a 304 stainless steel plate that also channels the coolant flow up through the core.

3. Instrumentation and Control. The control and operation of the PAHR facility can be based on the reactor's neutronic and physical resemblance to the Rover reactors operated at NRDS in Nevada. The Rover reactors were generally operated, after a rapid approach to design power, at steady operation for 15 to 45 minutes followed by a fast scrammed shutdown. The Rover reactors operated at high-power densities with large axial thermal gradients. The power levels were up to 5000 MW with coolant outlet temperatures up to 2500 K.

For the PAHR application, the reactor will be operated at modest power densities and thermal gradients. The proposed power levels are up to 30 MW with coolant outlet temperatures below 1100 K. Operation is at slowly varying power levels for the duration of the test (up to several days). Reactor physics calculations for the critical core loading and the control rod adequacy can be confirmed in the present PARKA critical assembly.

The reactor and the associated auxiliary systems will be operated from the control room in the Control Point Complex. To control and monitor the reactor standard nuclear instrumentation and controls would be used. Hard-wired safety systems would automatically shut down the reactor and take appropriate actions when out-of-limit conditions are detected by monitoring instruments. There will also be additional safety systems to detect malfunction or out-of-limit conditions in the reactor and test vehicle cooling systems and in other plant systems as needed. A computer located in the control room is available to store and display data, monitor systems operations, perform recurring operation sequences, and do data reduction.

IV. NEUTRONIC ANALYSIS

Coupled neutronic systems consisting of a driver reactor and an experiment have been studied to determine the required driver power and the sensitivity of the system to experiment variations. The driver reactor modeled in

these calculations is that described in Section III. The reactor has an intermediate neutron energy spectrum, which is advantageous for use in a FAHR test facility, and is capable of operating at powers in excess of those required by the experiments under consideration. Other preliminary studies⁸ have demonstrated that the system can be readily controlled and have defined the radial fuel enrichment distribution necessary to obtain an acceptable flat radial power profile.

Two experimental configurations have been investigated. Major attention has been devoted to an Experiment Category I, which has a test fuel region 700 mm in diam and 50 mm in depth containing 167.4 kg of UO_2 . This configuration was studied with stainless steel and zirconium oxide liner or container material, with homogeneous and heterogeneous fuel, and with various amounts of liner material melted into the fuel. The neutronic model for the driver reactor and experiment is shown in Fig. 5. Initial scoping studies have also been carried out for Experiment Category II with the same test fuel mass, 167.4 kg, but 350 mm in diameter and 200 mm in depth (see Fig. 6).

The calculations for these experiments were done with TWOTRAN-II,⁹ a two-dimensional transport code. Cross sections for this work were taken from the sixteen energy group set developed by Hansen and Roach¹⁰ for fast and intermediate critical assemblies. The cross sections are evaluated at 300 K but were considered adequate for these scoping calculations.

A. Experiment Category I

The basic Experiment Category I consists of a Rover-type driver and a sodium-cooled mass of UO_2 , 700 mm in diameter and 50 mm in depth. The calculations for this configuration assumed the UO_2 to have a density of $8.7 \times 10^3 \text{ kg/m}^3$, approximating that of molten UO_2 . No control material was present in the test vehicle.

A requirement for this experiment is that 5 kW/kg be deposited in the test UO_2 . With this constraint, the major design objective is to minimize the driver power without excessive ^{235}U enrichment in the test fuel. Thus, the system has been calculated with enrichment in the test fuel as a variable. The results of the calculations are illustrated in Figs. 7 and 8.

From these data (Figs. 7 and 8) and the consideration of criticality safety in the experiment, the system parameters can be chosen. To minimize driver

Fig. 5. Experiment Category I.

Fig. 6. Experiment Category II.

Fig. 7 Experiment Category I parameter study.

Fig. 8.
Optimization Study I.

power the test enrichment should be as high as possible. The requirement that the experiments should always be subcritical limits the enrichment. A rough calculation based on 167.4 kg of UO_2 in the experimental bed indicates that the test should contain in the range of 70 kg ^{235}U or less for it to be subcritical in all configurations. At a 55% enrichment, there are approximately 80 kg ^{235}U in the test. This implies that the test enrichment should be on the order of 55% or less. Under this condition, the driver power is 17 MW while the figure of merit is 1.4. The figure of merit is the ratio of the average power density in the test to the average power density in the driver. The test at 55% enrichment has a k_{eff} of only 0.55 when considered independently of the driver. As can be seen from Figs. 7 and 8, this system is not very sensitive to small perturbations in enrichment.

The preceding results apply to a ZrO_2 liner. If the liner is changed to stainless steel, the driver power requirement is increased by 33% to 22 MW and the figure of merit drops 25% to 1.08. The radial flux profiles for these two cases are shown in Fig. 9.

A mechanical design variable is the distance from the bottom of the driver reactor to the center line of the test. This distance can be greater than the nominal 445 mm used in the above calculations. The effect of increasing the separation is shown in Fig. 10 for a test with an enrichment of 55% and a ZrO_2 liner. An increase in separation of 150 mm necessitates a 64% increase in the driver power. This causes a corresponding 40% decrease in the figure

Fig. 9.
Radial fission profiles.

of merit to 0.8. This implies that for a separation increase of this magnitude the system will have to be optimized again with regard to enrichment. As Fig. 10 illustrates, changes on the order of 50 mm do not cause significant degradation of the optimized design.

The discussion up to this point has dealt with what might be defined as the initial conditions of the experiment. Since in the actual experiment the liner will be displaced or mixed with molten fuel, several experimental configurations have been studied, which are representative of typical molten fuel-liner configurations.

Assume the molten fuel begins to melt axially into a sacrificial liner of stainless steel. As the stainless steel melts, let us also assume that the fuel and stainless steel mix homogeneously. Figure 11 illustrates how the power of the driver must increase to maintain the deposition of 5 kW/kg in the test fuel. It is interesting to note that the rate of increase in the driver power is nearly linear as the melt depth increases. Another calculation showed that if the test fuel melts both 50 mm axially and radially, then the driver power must be increased to 30.5 MW from 22 MW to sustain the experiment power.

Another interesting variation of the initial experiment design is the addition of stainless steel into the experiment both in homogeneous and in heterogeneous combination with the fuel. If one assumes that the fuel and various amounts of stainless steel are homogeneously mixed in the molten state,

Fig. 10.
Power vs. separation.

Fig. 11.
Power vs meltdown depth.

the system behaves as illustrated by Fig. 12. In these calculations, the UO_2 inventory was held constant at 167.4 kg and the test depth was varied by adding stainless steel. In the homogeneous case, the steel brought some of the fuel closer to the driver and provided some moderation to help maintain the experiment power as the uranium density was decreased. When these calculations are performed with the stainless steel present as a crust floating on the molten fuel (see Fig. 12), the effect of the steel is to shield the test fuel from the driver. For the experiment power to be maintained the driver power must be increased.

Based upon the calculations performed for Experiment Category I, one can conclude that neutronically these experiments are feasible and that the experimental powers can be readily met without operating the driver reactor above 30 MW. However, since the experiment fuel is squat, maintaining a flat radial power profile is difficult.

B. Experiment Category II

The Experiment Category II consists of the same driver core as before but with sodium-cooled experiment 350 mm in diameter and 200 mm in depth. The experiment contains 167.4 kg of UO_2 at a density of $8.7 \times 10^3 \text{ kg/m}^3$ to simulate molten fuel. No control material is present in the test vehicle. The test liner is made of ZrO_2 .

Fig. 12.
Power vs addition of steel.

Calculations for this configuration have been limited to an initial parameter scoping study. This investigation was based on minimizing the driver power and test enrichment with the condition that 5 kW/kg be deposited in the test. The results of this study are illustrated in Figs. 13 and 14 as a function of test enrichment. At an enrichment of 45%, the driver power is at 9 MW and the figure of merit, the ratio of average power density of the test to that of the driver, is 3 (Fig. 14). The test centerline radial and axial power profiles are shown in Fig. 15 for the 45% enrichment case. The experiment at 45% enrichment has a k_{eff} of 0.9 when considered independently of the driver. Figures 13 and 14 imply that this configuration is more sensitive to perturbation than Experiment Category I.

Although it was indicated earlier that a 55% enrichment is allowable with respect to experiment criticality, a value of 45% was chosen to decouple the experiment and driver. This provides for better system control.

V. REACTOR HEAT TRANSPORT SYSTEM

A. Core Thermal-Hydraulics Study

A study of the steady-state driver core thermal-hydraulics has been carried out to ensure that the core can be cooled and to establish preliminary design criteria. Other core thermal-hydraulic studies of a similar reactor design¹¹ have been carried out for other coolants using similar fluid-thermal models.¹² The model described here was developed specifically for design of the PAHRTEF. The axial driver reactor power profiles are based upon the results of the neutronic calculations.

Flow through the reactor was divided into four flow regimes as follows:

1. adiabatic expansion to stagnation conditions,
2. adiabatic compression from stagnation conditions,
3. adiabatic flow between parallel plates, and
4. pipe flow with heat addition.

Figure 4 shows the reactor cross section and nine points in the flow that were used to specify the separate flow regimes. For these preliminary calculations two cooling loops were assumed. Thus, there are two inlet and two exit pipes for the gas coolant. Stagnation conditions were assumed to exist in the flow at points 2, 5, and 8. All heat transfer to the fluid was assumed to occur from

Fig. 13.
Experiment Category II parameter study.

points 6 to 7. Since the radial power profile is maintained relatively flat by the fuel loading, the flow and temperature distribution through the core was treated by considering a single circular channel surrounded by an adiabatic hexagonal boundary. The total flow was assumed to divide evenly among the 1819 coolant channels in the core.

The flow channel was discretized using finite element techniques (i.e., by expressing thermo-hydraulic functions of the length coordinate in terms of shape functions and nodal values). At each node, the energy, momentum, continuity, and equation-of-state requirements are satisfied. Details of the model are described in Appendix B.

Figure 16 shows the hot-channel axial power density profile for the 1-m length core as determined by the neutronic analysis for a 30.6-MW system. In this system, the driver power was 29.8-MW with the experiment supplying the balance. Three different flow rates were calculated using the thermal-hydraulic model. Figure 17 shows the hot channel fluid temperature profile for the lowest

Fig. 14.
Optimization Study II.

Fig. 15.
Axial and radial fission profiles.

Fig. 16.
Axial power density profile for the hot channel of the 30.64-MW system.

Fig. 17.
Predicted fluid temperature profile for the hot channel
of the 30.64-MW system with 10 kg/s mass flow rate.

flow rate 10 kg/s. Table I summarizes the calculations for the hottest and coldest channels in the core. As can be seen, the temperature gradient from the hottest to the coldest channel is not very large. Fuel element variations in temperature will be even less. Thus, the assumption of an adiabatic hexagonal boundary for each element is reasonable. Also, the pressure drop from the lower to upper plenums are very close indicating the assumption that the mass flow rate is evenly distributed over all channels is adequate. The average of the two exit temperatures indicates a driver power of 29.9 MW, which compares favorably with input driver power of 29.8 MW.

B. General Description of Design

Figure 18 shows a schematic of a single reactor cooling system. At full power, two such systems, each containing two closed-cycle helium loops, will remove 30 MW from the reactor core. The primary loops will remove the heat from the reactor core and transfer it to an intermediate heat exchanger located inside the secondary containment, thus forming closed helium cycles within the containment building. The secondary loops will take the heat to a heat dump and to the atmosphere located outside of the reactor building. Each cooling loop will circulate 10 kg/s of 3.45-MPa helium through the reactor, a total flow of 20 kg/s.

Fig. 18.
Driver reactor cooling schematic.

TABLE I
SUMMARY OF CALCULATIONS FOR THE
30.64-MW SYSTEM -HOT AND COLD CHANNELS

Mass Flow Rate kg/s (lbm/Hr)	Channel	Max. Fuel Temp. °C (°F)	Axial Location of Max. Fuel Relative to Core Mid-plane m(in.)	Max. Fuel Wall Temp. °C (°F)	Axial Location of Max. Wall Temp. Relative to Core Mid- plane m(in.)	Exit Gas Temp °C (°F)	Pressure Drop Lower Plenum to Upper Plenum kPa (psi)	Pressure Drop Inlet to Outlet kPa (psi)
10 (79 400)	HOT	1185 (2165)	0.15 (5.9)	1153 (2107)	0.16 (6.3)	806.9 (1489)	1.76 (0.26)	3.00 (0.44)
	COLD	1109 (2028)	0.14 (5.5)	1079 (1974)	0.15 (5.9)	751.6 (1385)	1.67 (0.24)	2.90 (0.42)
20 (159 000)	HOT	754.4 (1390)	0.12 (4.7)	721.2 (1330)	0.13 (5.1)	505.6 (942)	4.49 (0.65)	9.09 (1.31)
	COLD	712.6 (1315)	0.12 (4.7)	681.7 (1259)	0.13 (5.1)	478.0 (892)	4.33 (0.63)	8.90 (1.29)
30 (238 000)	HOT	601.2 (1114)	0.11 (4.3)	567.6 (1054)	0.12 (4.7)	405.2 (761)	8.25 (1.20)	18.2 (2.64)
	COLD	71.2 (1060)	0.11 (4.3)	539.9 (1004)	0.12 (4.7)	386.8 (728)	8.01 (1.16)	17.9 (2.60)

1. Helium Circulators. The helium circulators will be required to circulate 10 kg/s of 3.4-MPa nominal pressure helium at a temperature of 478 K. The blower will be vertically mounted and capable of circulating the flow against a 69-kPa design pressure head. The circulators for both the primary and secondary loops will be identical.

2. Filters. Ahead of the circulators in the helium stream will be a set of stainless steel woven wire mesh filters. The primary purpose of these filters is to protect the circulator internals from particulates of carbon and graphite in the helium stream. The filters will be designed to remove 100% of 6 μ m and larger particles and at least 98% of 2 μ m particles.

3. Intermediate Heat Exchanger. The intermediate heat exchanger will be designed to carry the 10 kg/s of 3.4-MPa nominal pressure helium, which will probably contain suspended graphite powder or particles. The primary loop inlet pressure is predicted to be 3.48-MPa at 766 K. The outlet temperature will be 478 K. The exchanger will be designed to allow a 24-kPa pressure drop in the primary loop including inlet and exit losses.

The shell and tube heat exchanger will be fabricated of type 316 stainless steel. Helium will enter the tube side through a 0.4-m-diam duct and circulate over approximately 500 m² of heat exchange area. The heat load imposed is 15 MW for each loop. The design will include the assumption that the exchanger must operate for 1000 thermal cycles over a period of 5 years of operation. In addition, the exchanger will be designed to survive a number of nonroutine events of an accidental nature.

The design will be balanced with the secondary loop transporting 10 kg/s helium at 3.7-MPa nominal pressure. A 344-kPa pressure differential will insure containment of fission products within the primary loop. Since the secondary loop serves as secondary containment in the event of a rupture, all components will be built to the same standards as those of the primary loop. The secondary helium will enter the heat exchanger shell side at 367 K and leave at 655 K.

4. Heat Dump. From the intermediate heat exchanger, the heat will be transported to an air-cooled heat dump located exterior to the reactor facility. This heat exchanger will drop the helium temperature from 655 K to 367 K. The heat dump will require approximately 1400 m² of heat exchange area for each 15-MW thermal load.

VI. EXPERIMENT SUPPORT

A. Facilities at the Nevada Test Site

The Nevada Test Site (NTS) was chosen to utilize the existing facilities and equipment and to provide a remote location to perform experiments without a large and expensive new facility. The Nuclear Rocket Development Station (NRDS) at NTS provides existing power, water, rail service, communications, hot and cold cell maintenance and disassembly facilities, fire protection, plant security protection, test site support personnel and contractors, and a resident architect-engineering contractor.

Major facilities that will be utilized are:

1. Test Cell "C" for location of the reactor facility and PAHR test;
2. R-MAD Reactor Maintenance and Disassembly facility for storage and preassembly of equipment and components;
3. E-MAD Engine Maintenance and Disassembly facility for final assembly, check out, disassembly, and examination of the test bed and experiment;
4. CP Complex - Control Point facility for reactor and test control and data acquisition;
5. Railroad System for transporting the test bed from E-MAD to test cell "C".

B. Experimental Vehicle Concept

Requirements from in-pile investigation of PAHR phenomena are for large diameter and relatively flat geometries with simulated decay-heat power level in the test fuel. For experimental convenience and safety reasons, the test vehicle is located below the driver reactor. A below core position for the test vehicle is preferred to facilitate the remote handling of the experiment. In most of the tests, there will be large amounts of test and containment structural materials around and below the test fuel, which would reduce the neutron flux too much in other arrangements. The test vehicle with the proper fixtures can be designed for ease of remote assembly and disassembly. The test vessel will act as the primary containment vessel for the test materials. The containment vessel would be designed per ASME pressure vessel code Section III for Class I components with the design parameters of 3.5 MPa and 370°C.

C. Experimental Vehicle Handling

The proposed experimental vehicle containing the postaccident heat removal experiment is mounted on a rail car along with most of the test cooling equipment. The initial assembly of the test vehicle with the instrumentation and cooling equipment would be done at the E-MAD (engine maintenance and disassembly) building in the cold cell. All equipment and instrumentation can be checked and calibrated before the rail car leaves E-MAD. Remote handling equipment, fixtures, and tools could be used later in the disassembly operation.

The test vehicle on the rail car would be moved on rails by the locomotive to test cell "C" and the reactor building, where the experiment is positioned under the reactor and the power and instrumentation lines connected.

After rechecking all systems the test would then be run and data collected by the data acquisition system. After the test is terminated the rail car is disconnected and the locomotive returns it to the E-MAD hot cell for remote disassembly and examination of the experiment.

D. Experimental Vehicle Cooling

The experiment vehicle cooling system uses sodium as the primary coolant and air as the secondary coolant. The experiment will have two or three separate coolant loops and the test vehicle will have a separate cooling loop. The general arrangement is shown in Fig. 19. Each loop will have its own flow control with pressure, temperature, and flow sensors. All loops will use common surge, supply tanks and oxygen control, and indication system. The oxygen control and indication system will be valved off during the test. The sodium system will be inerted with an argon gas system contained on the rail car. A schematic of the test vehicle cooling system as shown in Fig. 20 gives the general features of the proposed system. This system was chosen to eliminate the necessity for connecting and disconnecting sodium lines near the reactor. Instead, this would be done in the hot or cold cell at the E-MAD building.

Fig. 19.
Test vehicle on mounting platform.

Fig. 20.
Test vehicle cooling schematic.

APPENDIX A
ROVER BACKGROUND¹³

The Los Alamos Scientific Laboratory was an active participant in the nuclear rocket propulsion (Rover) program from 1955 through 1972. LASL's principal program activity was research and development pertaining to the nuclear reactor, which is the central part of a nuclear rocket engine. From 1961 to 1971, Aerojet General Corporation and Westinghouse Astronuclear Laboratory were the engine contractor and reactor subcontractor, respectively, for development of a nuclear rocket engine. In 1973, the nuclear rocket program was terminated because of the lack of a suitable mission for it in the planned national space program. During the Rover program, LASL designed, built, and tested thirteen reactors, gradually developing reactor fuel elements toward higher temperature and longer life capability. These were the two most important reactor performance parameters, however, high core power density (related to high engine-thrust-weight ratio) was also an important goal. In response to changes in potential missions and internal program objectives, reactors varying widely in size and power were tested. Design power, for example, varied from 50 to 500 MW. Phoebe 2, the largest reactor, with a design power of 5000 MW, was tested during the summer of 1968; Nuclear Furnace 1 was tested in 1971 at a maximum power of 50 MW.

The general design features of the Rover reactors included:

1. cylindrical cores of various diameters approximately 1.3 m long,
2. radial beryllium reflectors from 0.1 m to 0.2 m thick,
3. cooling by high-pressure hydrogen generally entering the reactor at near-cryogenic temperatures and leaving the reactor core at temperatures near 2500 K,
4. slender hexagonal fuel elements 1.9 cm across flats and 1.3 m long containing 19 axial coolant passages 0.25 cm in diameter,
5. graphite-based fuel elements containing either pyrocarbon-coated UC_2 bends or UC-ZrC solid solution,
6. NbC or (later) ZrC coating of all fuel element surfaces to help protect against corrosion by the hot hydrogen coolant, and
7. rotating control rods or drums in the reflector covered on 120° of their surface with boron-containing poison plates.

During the Rover program, substantial and sophisticated techniques were built up in many areas including fuel fabrication, property determinations, and testing. In addition, components from the several different reactor designs were accumulated. Some were parts from reactors built but not tested, others were spares from reactors that were tested, and others were parts usable after nuclear testing when their induced radioactivity had decayed sufficiently to permit the necessary personnel access.

The earliest Rover reactor cores were made by extruding a graphite mix containing the proper amount of enriched UO_2 . The UO_2 was converted to UC_2 during the high-temperature graphitization operation. Although the fuel elements were coated with NbC, problems related to the hydrolysis of UC_2 during storage led to its replacement by $\sim 150\text{-}\mu\text{m}$ pyrocarbon-coated UC_2 sphere "beads." By 1969, six reactor cores had been fabricated with the bead-loaded fuel elements.

The graphite matrix was still subject to considerable carbon loss due to leaching by the hydrogen propellant through cracks in the NbC during reactor operation. The reactor support system was also subject to these carbon losses, and a program of improvement of the support system was pursued. It was found that when a hot-pressed 46 vol% (NbC + TaC)-graphite part was coated with NbC or TaC, an extremely strong bond was formed between the matrix of the coating, which was adherent even after prolonged testing at high temperatures. High-temperature hydrogen leaching tests on the uncoated part showed that the integrity of the machined part was not lost even after all of the free carbon had been removed.

A program was started about 1966 to develop a procedure to fabricate (U, Zr)C-graphite fuel elements, which would incorporate the advantages of the support system carbide-graphite composite into the fuel elements. Extrusions of 30 and 35 vol% (U,Zr)C in graphite were found to consist of a connected network of carbide in a connected network of graphite. Either network was self-supporting when the other was removed.

Reactor cores were fabricated for Pewee II, NF-1, and NF-2 of 30 and 35 vol% carbides. The Rover program was cancelled before Pewee II or NF-2 could be tested; however, NF-1 was tested and postmortem examinations performed.

All of the fabricating equipment (extrusion press, ovens, and high-temperature heat-treating furnaces) are in operation in the reactor fuel fabrication facility of the Materials Technology Group at LASL. Most of the tooling

extrusion fixtures and heat treating fixtures are still available. Of great importance is that the expertise gained in the Rover program by some staff members and technicians is available. These people have continued to improve their capabilities over the past three years by developing coated particle fuels for the HTGR and extruding fuel for Argonne for the TREAT converter program. Over the past year and a half, carbide-graphite composites have been produced and supplied to Sandia as a candidate for the ACPR upgrading.

Briefly, the capabilities are available and operating to produce, evaluate, and qualify Rover-type carbide-graphite composite fuel elements for reactor use.

APPENDIX B

THERMAL-HYDRAULIC MODEL

The details of the thermal-hydraulic model developed to analyze steady-state heat removal from the driver core are presented in this appendix.

Outlet piping losses due to sudden expansion were assumed to be one velocity head, while losses due to sudden contraction (See Ref. 14) were assumed to be given by

$$\Delta p = \rho^* \left(\frac{1.5(1-\nu) V_2^2}{2g(3-\nu)} \right), \quad (1)$$

with p the pressure, ρ^* the average fluid density, V_2 the velocity in the smaller flow area, and ν the ratio of the flow areas, A_2/A_1 . Friction factors for transitional and turbulent flow were determined from Colebrook's relationship for smooth pipes,¹⁴

$$\frac{1}{\sqrt{f}} = -2 \log_{10} \left(\frac{2.51}{R_e \sqrt{f}} \right), \quad (2)$$

where f is the friction factor, and R_e is the Reynold's number. For laminar flow, the equation $f = 64/R_e$ was used.

The flow regime defined between 3 and 4 (Fig. 4) was assumed to be an adiabatic flow through a large radius annulus and thus was treated as flow between parallel plates. The flow area is approximately one-half the total flow area of the core cross-section. For turbulent flow in this region, the Darcy-Weisbach relation for frictional losses was used in the form

$$\Delta p = f \frac{L}{4m} \rho^* V^*{}^2, \quad (3)$$

where L is the length of flow, V^* is the average velocity over the section, and m is the hydraulic radius.

The fuel element used was a standard PARKA element of 1-m length and hexagonal flat-to-flat distance of 19.2 mm. The diameter of the single fuel-coolant

hole in the fuel was 12.2 mm. The fuel thermal conductivity, K_f , for UC-ZrC composite of 30% carbide volume fraction varies slightly with temperature,¹⁵ but for this study was taken to be

$$K_f = 30 \frac{W}{m-k} . \quad (4)$$

The upper limit on fuel temperature is about 2500°C.

The helium coolant was modeled as a perfect gas. For reactor conditions, the density in kg/m^3 , ρ , is given by the relationship

$$\rho = \frac{1661}{T} , \quad (5)$$

and the fluid heat capacity, in J/kg K, C_p , is a constant 5200. For other temperature-dependent properties, power law curves fitted to experimental data for the temperature and pressure range of interest were used. For the absolute viscosity in Pa - S, μ , the curve was

$$\mu = 2.54 \times 10^{-7} T^{0.752} , \quad (6)$$

and for the thermal conductivity in W/m K_{He} , the curve was

$$K_{He} = 4.82 \times 10^{-3} T^{0.611} . \quad (7)$$

In general, the coefficient of determination was 0.97 or better for these fits.

The film coefficient, h , was calculated at each nodal point from the turbulent flow relationship

$$\frac{h}{C_p G} P_r^{2/3} = \frac{0.023}{R_e^{0.2}} , \quad (8)$$

where G is the mass velocity, equal to the mass flow rate over the flow cross-sectional area, and P_r is the Prandtl number. In general, for the flow rates through the core considered, the film coefficient varied from 1000 to 25000 $\frac{W}{m^2-K}$.

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