LA-UR -91-1592

LA-UR--91-1592

DE91 013307

Los Alamos National Laboratory is operated by the University of California for the United States Department of Energy under contract W-7405-ENG-36

TITLE DESIGN STUDIES OF THE MODERATED HEAT PIPE THERMIONIC REACTOR (MOHTR) CONCEPT

AUTHOR(S) William A. Ranken, N-DO, MS-E552
John A. Turner, N-12, MS-E551
Los Alamos National Laboratory
P. O. Box 1663
Los Alamos, NM 87545

SUBMITTED TO 26th Intersociety Energy Conversion Engineering Conference

DESIGN STUDIES OF THE MODERATED THERMIONIC HEAT PIPE REACTOR (MOHTR) CONCEPT

William A. Ranken
John A. Turner

Los Alamos National Laboratory
P. O. Box 1663
Los Alamos, NM 87545

ABSTRACT

Design studies, based primarily on neutronics analysis, have been conducted on a thermionic reactor concept that uses a combined beryllium and zirconium hydride moderator to facilitate the incorporation of heat pipe cooling into compact thermionic fuel element (TFE) based designs useful in the tens of kilowatts electrical power regime. The goal of the design approach is to achieve a single point failure free system with technologies such as TFEs, high-temperature heat pipes, and ZrH moderation, which have extensive test data bases and have been shown to be capable of long lifetimes. Beryllium is used to thermally couple redundant heat pipes to TFEs and ZrH is added to reduce critical size. Neutronic analysis undertaken to investigate this design approach shows that greater reactivity can be achieved for a given geometry with a combination of the two moderator materials than with ZrH alone and that the combined moderator is much less sensitive to hydrogen loss than more traditional ZrH-moderated thermionic reactor designs. These and other analytical approaches have demonstrated the credibility of a heat pipe cooled thermionic reactor concept that has a reactor height and diameter of 60 cm and a reactor mass of 400 kg for 30-kWe power output.

INTRODUCTION

Thermionic reactor power systems are under investigation for a variety of potential space applications, including US Air Force observational satellites, crew cabin power for manned interplanetary missions, nuclear electric transport of cargo for these missions, and planetary surface power systems. Perhaps the most important of the emerging design criteria for these potential applications, especially the first two, is that the systems be immune from single-point failure mechanisms that could result in loss of mission. By far the most important consequence of rigorous application of this criterion to system design is on the reactor cooling loop. Circulating liquid metal (or gas) cooling systems must either be made redundant or be replaced by other methods of heat removal. The initial design of the current SP-100 space power system attempted to achieve redundancy with multiple independent pumped liquid metal cooling loops, but this approach proved infeasible and was dropped in favor of a single main coolant loop. This experience suggests that direct thermal radiation heat transfer and heat pipe cooling are the main contenders for realizing single-point failure avoidance.

The direct thermal radiation approach to thermionic reactor heat transfer is best exemplified by a design initiated by investigators at the General Electric Company in the early 1960s [1], later described by Rasor et. al. [2], and now commonly referred to as STAR-C [3]. In this concept the reactor core consists of a cylindrical stack of UC_2 plates encased in graphite. It is operated at a sufficiently high temperature (~2000 K) to heat the emitters of an array of plane parallel thermionic diodes that encompasses its cylindrical surface. The emitter design is a tungsten fin arrangement that achieves a threefold heat flux amplification so as to reduce the required core temperature as much as possible. The emitter fin and the core radiating surface must have high emissivity for the same reason. This design approach results in a relatively simple and compact power supply in the range of 5 to 15 kWe. It tends to become increasingly cumbersome as the design power level is raised above about 20 kWe. Modifications of the STAR-C approach have been suggested that reduce the size of a direct radiation coupled system at higher power levels (100 kWe and beyond) [4]. This reduction is achieved primarily by reducing the emitter fin area and increasing the areal power density in the thermionic diodes. These changes increase the heat flux that must be radiated from the reactor core by a factor of about four and require a core surface temperature increase to 2300 K.

It is the very high reactor core operating temperatures that constitute the major development concern, and, indeed, technical feasibility concern, for radiation coupled thermionic systems. Fuel irradiation data for uranium carbide fuels in this temperature range, though sparse, nevertheless tend to justify this concern in that fuel swelling rates are very high and fission gas retention may exceed 50% despite the high temperature. The latter result translates into high fuel pressure on the graphite, or any other material, used to contain the fuel. Other areas where data are not only sparse but tend to have a negative slant are long-term movement of fission products and their interaction with core and conversion components, diffusion and evaporation of carbon, effectiveness of coating materials, and maintaining high emissivity surfaces at high temperatures for long operating times (on the order of 7 to 10 years). These and other factors tend to make the development of radiation coupled thermionic reactors a high technical risk proposition.

It was the concern about high development risk for radiation coupled thermionic reactor designs that led to the genesis of the MOHTR (Moderated Heatpipe Thermionic Reactor) concept for meeting the single-point failure avoidance criterion for space power system design with low risk technology. The MOHTR concept relies on heat pipes to remove reactor waste heat in a redundant fashion. This is not, of itself, a new approach in thermionic design, because the high-temperature heat pipe was invented for the purpose of cooling an unusual thermionic reactor design [5], and numerous attempts have been made to marry heat pipe and thermionic conversion technology. This has been done for both fast and thermal spectrum reactor configurations [6,7], but for power levels in the tens of kilowatts, a moderated thermionic system is preferable because its tolerance for low fuel volume fraction in the core offers a substantial reduction in critical size and mass. The novel aspect of the MOHTR design is that it uses both beryllium and zirconium hydride as moderator materials in a reactor based on in-core TFEs. The purpose of the beryllium is to provide direct thermal conduction bonding between heat pipes and TFEs while the zirconium hydride is added to the matrix in a heterogeneous fashion to increase reactivity and produce a significantly more compact core than could be obtained with beryllium alone.

MOHTR DESIGN APPROACH

One version of the MOHTR design approach is shown in Fig. 1. In this version the beryllium moderator is used to form the matrix of single TFE modules that each contain three stainless steel/potassium heat pipes. Zirconium hydride rods are either nested between adjacent modules, to which they are thermally coupled only by radiation, or are incorporated into the individual modules with sodium bonding, as in the alternative module geometry shown in Fig. 1. Any two of the heat pipes in a given module can remove the heat generated in the TFE with no loss of performance of the TFE.

The circular geometry of the MOHTR configuration in Fig. 1 was selected to facilitate power flattening by varying the fuel-to-moderator ratio. This method has been shown by USSR investigators to be very effective with very little loss of reactivity [8]. The central hole will contain a B_4C safety rod during launch and beryllium during reactor operation. For very low powers it may contain a driver rod made of a shell of UO_2 with a BeO central core cooled by a separate set of high-temperature heat pipes. This arrangement would permit a more compact reactor design. (By operating the driver rod heat pipes at a high temperature (~1500 K), the heat generated in the driver can be removed with very little extra radiator area requirement.)

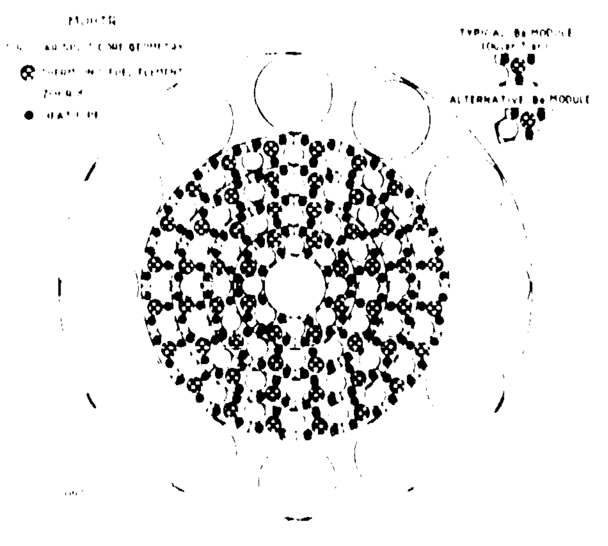

Figure 1: *Illustration of the Combined Beryllium/Zirconium Hydride MOHTR Design Approach.*

Several considerations led to the selection of a moderated, heat pipe cooled, TFE-based reactor as a low-risk approach to a single-point failure free space power supply. The most important of these is the large amount of development and testing work that has been done on TFEs and TFE reactor systems. Most of the thermionic reactor development activity in this country and abroad has been concentrated on the in-core thermionic reactor with either fast or thermal neutron spectra. As a consequence, many TFEs have been tested in in-core irradiation tests and a program of life testing of TFE components is currently under way in the USA [9]. In the USSR the development of two designs of moderated TFE-based thermionic reactors, TOPAZ-I and TOPAZ-II, has been carried to the point where two units of the former have been flight tested (for 6 and 11 months) and several electrically heated and fission heated tests of the latter design have been done. From both the USA and USSR work it is known that the feasibility of constructing TFEs with multiyear ($\geq$3-year) lifetimes is a certainty and lifetimes of 10 years appear attainable.

Another consideration in the selection of the MOHTR design approach has been the extensive testing that has been done on high-temperature heat pipes. One example is the successful test of a nickel/potassium heat pipe for over 4 years, with evidence of much longer lifetime capability. Another is the in-core irradiation testing of over two dozen sodium and potassium filled stainless steel heat pipes at temperatures required for the MOHTR design without any failures for test times of up to 3 years in a very high fast neutron flux.

With regard to the moderator, it was recognized that a very large technology base has been established for zirconium hydride configurations based on irradiation and other testing carried out both in the USA and the USSR [10,11,12]. The irradiation behavior of beryllium has not received the kind of effort spent on the other components of the MOHTR core. Recent developments in beryllium metallurgy and fabrication methods have, however, brought this material to the point where it is routinely used for structural components in satellite design and even drawn into tubing for heat pipes and heat pipe pinch-off tubes. The sparsity of beryllium irradiation data is somewhat counterbalanced by the observation that extrapolation of the most recent irradiation data [13] shows a several-fold neutron exposure margin before fast neutron damage becomes a problem at the required MOHTR operating temperatures.

NEUTRONIC STUDIES

Because of the unusual combination of materials and structures used in the MOHTR design approach, a considerable amount of work has been devoted to neutronic analyses that delineate the characteristics and capabilities of this type of system. In the main the analysis has been based on a hexagonal type of array, a 60° sector of which is shown in Fig. 2. This represents a system containing 91 fuel elements that produce a nominal total power output of 30 kWe. The fuel element modeling is based on the TOPAZ II single converter TFE [14] with a somewhat smaller emitter diameter of 15 mm and a total fueled length of 40 cm. The fuel is modeled as a homogenized combination of molybdenum and 95% dense UO_2 (93% enriched) in the volume ratio of 17 to 83. (The actual fuel configuration consists of discs of molybdenum and UO_2 forming an

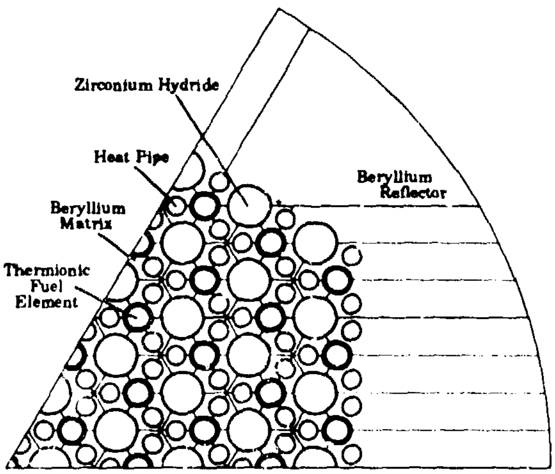

Figure 2: *Reference Calculational Model for MCNP Neutronic Analysis of the MOHTR Design Approach* ($k_{eff} = 1.026 \pm 0.003$).

alternating stack designed to minimize emitter swelling caused by residual fission gas in the UO_2.) The fueled emitter cladding is 0.95-mm-thick molybdenum coated with a 0.05-mm-thick layer of 184W. The TFE is assumed to be built into a beryllium module which, with a 0.2-mm nickel lining, serves as the collector. There are three stainless steel heat pipes per TFE, each having a diameter of 11 mm with a wall thickness of 0.60 mm. The ZrH moderator elements that are nested between the modules are contained in glass-lined stainless steel tubes that have a diameter of 24 mm with a wall thickness of 0.75 mm.

The average core diameter of the reactor modeled in Fig. 2 is 40 cm. The overall reactor diameter is 59 cm and the overall height is 60 cm. The axial elevation of the reactor is modeled in three zones - the 40-cm fueled zone and top and bottom reflectors that are each 10 cm thick. Its overall mass is 400 kg, half of which is core mass. The effective multiplication factor of this base design case is: $k_{eff} = 1.026$.

The approach used in the MOHTR neutronic design scoping calculations was to vary materials and some component dimensions in the model shown in Fig. 2 and determine the effect on k_{eff}. Some of the design parameters that have been investigated are: (1) stainless steel fraction, (2) heat pipe void fraction, (3) beryllium/ZrH ratio, (4) effect of hydrogen loss, (5) neutron streaming through heat pipe and TFE penetrations in the end reflectors, (6) UO_2 enrichment, (7) TFE diameter, (8) side reflector worth, and (9) effect of splitting the core at midplane. In addition, calculations have been made to determine fast neutron flux distribution and energy deposition distribution.

Because stainless steel or stainless steel-like alloys were the materials used in obtaining much of the irradiation data base on heat pipes and ZrH rods, these materials were incorporated into the baseline MOHTR calculational model. However, in a moderated reactor their use adversely affects the criticality so that minimal usage is desirable. Similarly, the addition of heat pipe void into the core suppresses the effective multiplication factor so that liberal usage of heat pipes is not advisable. Figure 3 shows the increase in k_{eff} as the stainless steel in first the heat pipes, then the ZrH rods, and then both together are removed and replaced with void. It can be seen that the change in reactivity with stainless steel removal percentage is roughly linear and that removal of all the stainless steel increases the reactivity 7%. It is possible, if not particularly conservative, to remove about 40% of the stainless steel in the core by thinning the walls of the heat pipes and ZrH containers. This would increase the value of k_{eff} by about 3%.

Figure 4 shows the results of a series of calculations that were made to quantify the effects of heat pipe void fraction. The void fraction dependence was obtained by filling the void space of first one, then two, and then all three of each set of heat pipes associated with a given TFE, first with beryllium and then with ZrH. It can be seen that replacement with ZrH is somewhat more effective than replacement with

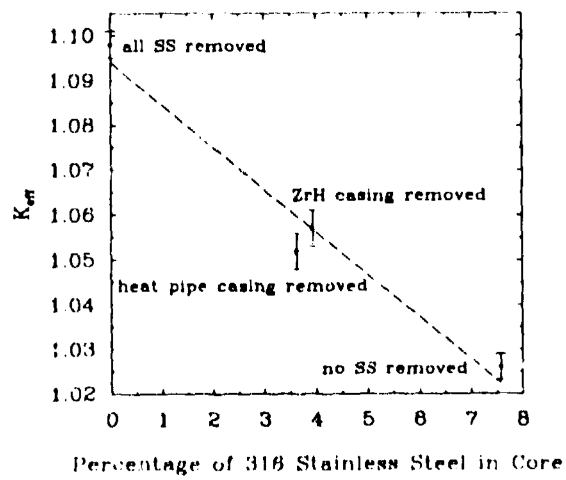

Figure 3. *Effect of Type 316 Stainless Steel on Reactivity of MOHTR Reference Model.*

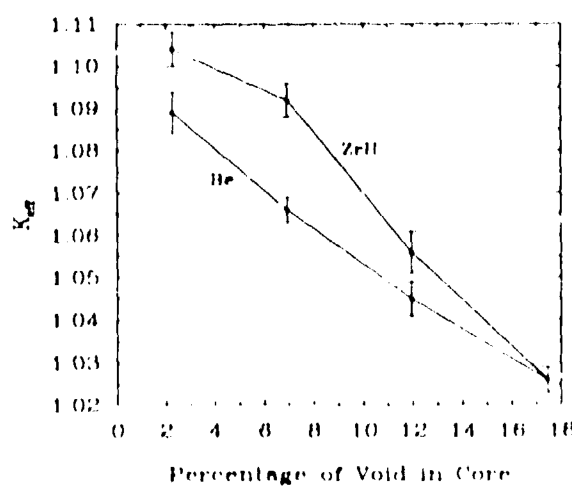

Figure 4. *Effect of Heat Pipe Void on k_{eff} of MOHTR Reference Model*

beryllium. Reducing the heat pipe void fraction from 17.5% to 12% of the core volume would increase k_{eff} by 3%. Additional calculations have shown that this effect and that of removing stainless steel are additive so that a potential gain of 6% in k_{eff} is available from these changes.

One of the key questions about the MOHTR design was how much penalty in k_{eff} resulted from the replacement of ZrH with beryllium. The first calculation to address this question was done by simply replacing all the beryllium in the model with ZrH and then replacing all the ZrH with beryllium. The surprising result of this calculation was that the combination beryllium/ZrH moderator gave higher reactivity than the all-ZrH moderated configuration. This is shown by the bottom curve in Fig. 5 where k_{eff} is plotted against the percentage of beryllium in the non-TFE portion of the core volume. This result was checked by filling the heat pipe voids with either ZrH or beryllium in various combinations in addition to interchanging the material in the moderator rods and matrix of the model sector. The result was a range of beryllium-to-ZrH ratios varying from zero to 100% for the case of no heat pipe void in the reactor core. The behavior of k_{eff} under these conditions is shown by the middle curve in Fig. 5. To determine whether the stainless steel heat pipe walls and ZrH rod walls were influencing the effect of the moderator contained within them, the calculations were repeated with the stainless steel removed. This gave the top curve in Fig. 5. For all three curves the maximum reactivity occurs for the case of roughly half beryllium and half ZrH.

Among the important features of the curves in Fig. 5 is the pronounced effect of heterogeneity of the ZrH placement on reactivity. The difference between the points labeled 47% and 53% is that in former case, the ZrH is in the form of rods (the true ZrH rods and the ZrH added in the heat pipe voids) in a beryllium matrix, and in the latter case, the ZrH and beryllium are interchanged so that the ZrH forms the matrix. Presumably the 4% higher reactivity for the ZrH in rod form results from the neutrons being substantially slowed down in the ZrH but tending to escape into the beryllium matrix before being subjected to capture by hydrogen. Another important feature revealed in Fig. 5 is the dramatic increase in reactivity produced by changing only 19% of the beryllium of an all-beryllium moderated core to ZrH. For the case of stainless steel being included, but no heat pipe void, the value of k_{eff} with only 19% replacement of beryllium by ZrH already exceeds the value for an all-ZrH moderated core. A third notable feature of Fig. 5 is that the dependence of reactivity on ZrH percentage in the moderator is relatively slowly varying over the range of 20% to 100%. This suggests that the loss of hydrogen from the glass-lined stainless steel tubes that contain the ZrH, a factor tending to limit the permissible operating temperature for long life ZrH moderated systems, is less critical for ZrH/beryllium moderated systems.

The reduced effect of hydrogen loss on reactivity suggested in Fig. 5 was directly checked in another set of calculations. Figure 6 shows the change of reactivity of the reference core calculational model as a function of hydrogen loss. Even with a 15% loss of hydrogen (changing from $ZrH_{1.7}$ to $ZrH_{1.45}$), the loss of reactivity is limited to about 2.5%. For an all-ZrH moderated system, the expected change for the same percentage hydrogen loss would be about three times this value.

One of the concerns with heat pipe reactor designs is that the relatively large number of holes in the end reflectors will result in enhanced neutron leakage caused by streaming effects. This concern was addressed by determining the change in criticality when the heat pipe void spaces in the end reflectors were filled with beryllium. This produced a change in reactivity from 1.026±0.003 to 1.033±0.003, indicating that neutron loss by streaming is not an important effect.

Other neutronics investigations have shown that: (1) increasing the ^{235}U enrichment from 93% to 97% did not change k_{eff} within the standard deviation of the calculation of ±0.003; (2) changing the fueled emitter cladding from molybdenum to ^{184}W decreased k_{eff} by about 0.5%; (3) increasing the diameter of the fueled emitters from 15 to 18 mm gave a 10% increase in reactivity; and (4) the reactivity decrease resulting from removing the side reflector was 13%. This last calculation was a first step in determining the best control method for the MOHTR design approach.

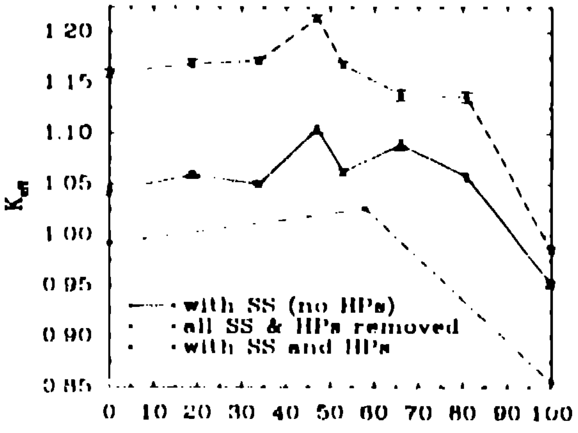

Figure 5: Dependence of MOHTR Reference Model k_{eff} on Beryllium to Zirconium Hydride Ratio

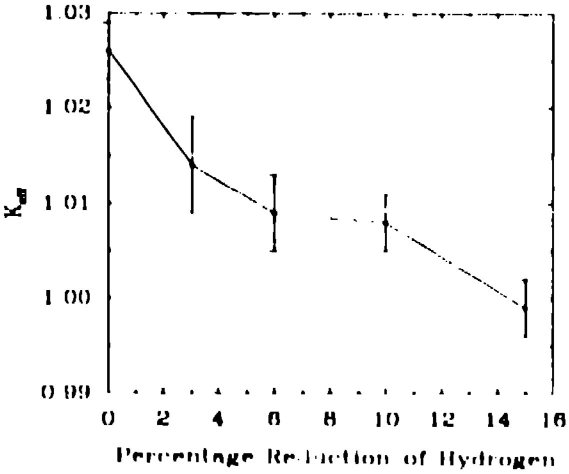

Figure 6: Change of k_{eff} in MOHTR Resulting from Loss of Hydrogen from Zirconium Hydride

The type of TFE used in the calculational model, one long thermionic converter per TFE, makes pre-launch testing of the TFEs by electrical heating feasible and greatly simplifies fission gas venting from the fueled emitters. It also eliminates the need for a thermally conductive sheath insulator in the TFE and results in a larger prompt negative fuel expansion contribution to the reactor temperature coefficient of reactivity relative to a multi-converter TFE with its axially segmented fuel column. However, the long converter length results in a very significant decrease in converter voltage output at higher TFE power levels. For this reason there is a definite advantage to splitting the core at midplane to retain the single converter TFE advantages while reducing the length of a converter (and the TFE) by 50%. In this arrangement the midplane region separating the two core sections will consist primarily of a beryllium support section, beryllium electrical connectors, and BeO insulation. Calculations were made of the loss of reactivity caused by the splitting process. The results indicated this loss can be kept below 2% if the separation is restricted to 5 cm, the average beryllium density is maintained at 60% of theoretical, and the total fuel column length remains unchanged. Because heat pipes emerge from both ends of the split core design, it is advantageous to rotate the reactor 90° to the shield-payload axis. The heat pipes make a bend of almost 90° as they exit from the end reflectors and form a radiator consisting of two diverging fan configurations.

In order to obtain input numbers for the thermal analysis of MOHTR core configurations, to determine a starting point for the design process of flattening the radial and axial input power in the TFEs, and to make an assessment of neutron damage potential, a number of calculations were made of the distribution of neutron and gamma induced power density in the TFEs and the moderator and also of the fast neutron flux distribution in the various components of the MOHTR core. The calculations were made for the case of a total heat generation rate of 330 kWt in the 91 TFEs. Figure 7 shows the power density in the fuel as a function of radius at core midplane and at planes 4 cm in from the ends of the core. The peak-to-average ratio of power density for this configuration is 1.5. The results for power density in the beryllium matrix and ZrH rods have very nearly the same shape as these fuel power density curves. Power deposition density in the ZrH is 2.6% of the fuel power density, with 44% caused by neutrons and the rest by gamma radiation. The power deposition density in the beryllium matrix is 1.2% of the fuel value, with 52% caused by neutrons. Figure 8 shows the fast neutron flux (neutron energy > 1.0 MeV) in the various core constituents at core midplane as a function of core radius. Additional calculations have shown that the ratio of E> 0.1 MeV neutron flux to E>1 MeV neutron flux is in the range 2.3 to 2.5. The values are low enough that no difficulty is expected from fast neutron damage to either the polycrystalline alumina seal insulators at the ends of the TFEs or the beryllium matrix at temperatures up to 1100 K for a 7 year lifetime.

CONCLUSIONS

The neutronics analyses conducted on the MOHTR design approach have indicated that beryllium can be used to thermally couple TFEs and high-temperature heat pipes in a moderated reactor with no penalty in critical size, provided rods of zirconium hydride are also included. Thus it is possible to assemble a compact thermionic power system that has no inherent single-point failure sources and yet uses technologies that have been shown to be capable of long lifetime. The results of MCNP analyses of variations of the base case calculational model (containing approximately 70 volume per cent moderator) reveal that the replacement of as little as 20% of the beryllium by ZrH gives a higher value of k_{eff} than an all-ZrH moderated core. A corollary feature of the combined moderator approach is that the reactivity of the core is significantly less sensitive to a given percentage loss of hydrogen than traditional all-ZrH moderated designs such as the TOPAZ-I and TOPAZ-II reactors developed and tested in the USSR. Other analytical results indicate that k_{eff} is rather sensitive to stainless steel and heat pipe void fraction, but insensitive to neutron streaming through the numerous heat pipe penetrations in the reactor end reflectors. It is also relatively insensitive to ^{235}U enrichment (for enrichments exceeding 90%), but can be substantially increased by

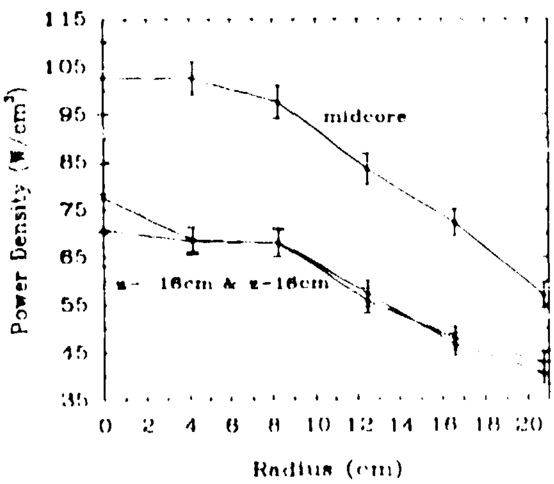

Figure 7: Unflattened Radial Power Density Profile in TFE Fuel in MOHTR Reference Model

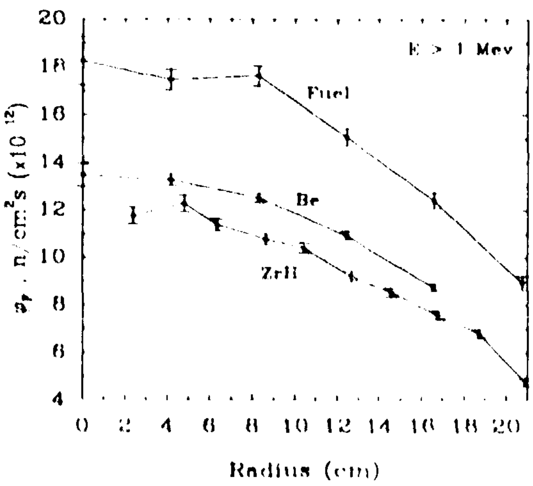

Figure 8: Radial Profiles at Core Midplane of Fast Neutron Flux in Moderator Components and TFE Fuel in MOHTR Reference Model.

increased TFE fueled emitter diameter. The preferred configuration for power levels in the range of a few tens of kilowatts is a split core arrangement that utilizes single converter TFEs of the type developed for TOPAZ-II and has heat pipes emerging from both ends of the reactor. This approach gives the system pre-launch testing capability using electrical heating while minimizing electrical losses in the TFEs (by reducing the single converter length) and retaining the usual advantages of heat pipe cooled reactor systems. In addition to redundancy, the latter include the elimination of electromagnetic pumps, heat exchangers, coolant thermal expansion compensators, and any concerns over startup with a frozen coolant as well as the removal of decay heat. The design studies conducted to date indicate that the MOHTR approach is not only feasible, but can result in a compact reactor that, for the 30-kWe design point, would be 60 cm in height and diameter and have a mass of 400 kg.

ACKNOWEDGMENTS

The authors wish to express their appreciation to the US Air Force Phillips Laboratory for its support of this work.

REFERENCES

[1] R. A. Becker et al., "Thermionic Converter Cells for Nuclear Reactor," U. S. Patent 3,590,286, filed November 1968, issued June 1971.

[2] N. S. Rasor et al., "Thermionic Reactor Based on Radiant Heat Transfer and Demonstrated Components," Thermionic Conversion Specialist Conference, Palo Alto, California, 1967.

[3] H. J. Snyder, "STAR-C Thermionic Reactor for Low Power Space Applications," Proceedings of the 22nd Intersoc. En. Conv. Eng. Conf., 1987.

[4] M. Jacox, "Small Ex-Core Heat Pipe Thermionic Reactor (SEHPTR)," Proceedings of the 26th Intersoc. En. Conv. Eng. Conf., 1991.

[5] G. M. Grover, "Pancake Reactor," U. S. Patent No. 3,243,613, filed January 1964, issued March 1966.

[6] J. J. Ward, R. Breitwieser, and R. M. Williams, "Conceptual Design of a 150 kWe Out-of-Core Nuclear Thermionic Converter System," Conference Record of the 1970 Thermionic Conversion Specialist Conference.

[7] D. R. Koenig, "Heat Pipe Nuclear Reactor for Space Power," Proceedings of the 11th Intersoc. En. Conv. Eng. Conf., 1976.

[8] G. Ya. Artyukhov et al., "K_{eff} and Power Profile Optimization in a 'TOPAZ' Type Reactor," Transactions of Anniversary Specialist Conf. on Nuclear Power Engineering in Space, Obninsk, USSR, May 15-19, 1990.

[9] W. A. Ranken and R. C. Dahlberg, "Thermionic Reactor Investigations in the US," Transactions of Anniversary Specialist Conf. on Nuclear Power Engineering in Space, Obninsk, USSR, May 15-19, 1990.

[10] J. H. Van Osdol et al., "Design and Integration of Zirconium Hydride Reactor Power Systems," Report No. AI-AEC-13072, Atomics International Division, Rockwell International, 1973.

[11] I. A. Izhvanov et al., "Hydride Moderator Element," Transactions of Anniversary Specialist Conf. on Nuclear Power Engineering in Space, Obninsk, USSR, May 15-19, 1990.

[12] S. P. Volkov et al., "Effect of Reactor Irradiation on Zirconium Hydride Moderator," Transactions of Anniversary Specialist Conf. on Nuclear Power Engineering in Space, Obninsk, USSR, May 15-19, 1990.

[13] J. M. Beeston et al., "Comparison of Compression Properties and Swelling of Beryllium Irradiated at Various Temperatures," J. Nuc Mat. **122** and **123**, 802-809, 1984.

[14] V. P. Nickitin et al., "Perspectives of the Development of the Thermionic Space NPS with the Existing Single-Cell TFEs," Transactions of Anniversary Specialist Conf. on Nuclear Power Engineering in Space, Obninsk, USSR, May 15-19, 1990.