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Thermal Radiation in Gas Core Nuclear Reactors for Space Propulsion

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A diffusive model of the radial transport of thermal radiation out of a cylindrical core of fissioning plasma is presented. The diffusion approximation is appropriate because the opacity of uranium is very high at the temperatures of interest (>3000 K). We make one additional simplification of assuming constant opacity throughout the fuel. This allows the complete set of solutions to be expressed as a single function. This function is approximated analytically to facilitate parametric studies of the performance of a test module of the nuclear light bulb gas-core nuclear-rocket-engine concept, in the Annular Core Research Reactor at Sandia National Laboratories. Our findings indicate that radiation temperatures in range of 4000–6000 K are attainable, which is sufficient to test the high specific impulse potential (~2000 s) of this concept.

I. Introduction

GAS-CORE nuclear-reactor concepts offer advantages over the more traditional solid-core reactors that have been developed and used commercially. Maintaining the nuclear fuel in a gaseous state allows the fuel to be continuously reprocessed, thus avoiding the buildup of highly radioactive fission products and reducing the potential risk during an accident. Gas-core reactors can have much higher operating temperatures than solid-core reactors, which increases the electrical conversion efficiency of standard thermal cycles. Indeed, temperatures high enough to significantly ionize a working fluid are possible. This would allow the efficient use of MHD electrical generation. Furthermore, the high operating temperatures possible with a gas-core reactor make them particularly suitable for rocket propulsion.

Gas-core nuclear-rocket concepts[1,2] were developed at the United Technologies Research Center (UTRC) during the 1960s and 1970s under the Rover program. Although open-cycle systems have the potential for extremely high specific impulse, calculations indicated that a significant quantity of nuclear fuel would be lost. This problem was eliminated in the closed-cycle concept called the nuclear light bulb (NLB). The nuclear light bulb is based on heating tungsten-seeded hydrogen propellant to very high temperatures (>5000 K), using thermal radiation emanating from a fissioning plasma that is contained within an internally cooled transparent wall (see Fig. 1). An inward flowing vortex of a low Z buffer gas (neon or argon) separates the nuclear fuel from the transparent wall. This buffer gas is continously injected from the transparent wall and keeps the wall at sufficiently low temperatures to maintain structural integrity. The fuel is also continuously injected at a radius inside the buffer gas region. Eventually turbulence mixes these two gases. Thus a mixture of fuel and buffer gas is continuously removed from the center of the fuel containment region. The mixture is separated and then reinjected. The residence time of the buffer gas and fuel is approximately 4 s. Thermal radiation emitted from the fissioning uranium plasma passes radially outward through the relatively transparent buffer gas and confinement wall to the annular propellant heating region. Since the thermal radiation temperature ($T_{\rm rad}$ > 5000 K) can be significantly higher than the temperature of the transparent wall ($T_{\rm wall}$ ~ 1300 K), the propellant temperature is not limited by the maximum material temperature, as it is in solid-core concepts. Consequently, much higher engine performance levels are possible. Calculations indicate that very high specific impulse (~2000 s) are possible, while still obtaining high thrust (~400,000 N) and thrust-to-weight ratios greater than unity. These performance characteristics are enabling for a rapid round trip mission to Mars and for high energy missions to the outer planets. A recent review of the nuclear light bulb concept has been given by Mensing et al.[3]

When research on nuclear rockets was suspended in 1979, considerable research on the feasibility of the NLB concept had been accomplished by UTRC. In particular, a plasma heated to approximately 9500 K had been contained within water cooled fused silica walls by vortex confinement. Since the stability of the vortex confined plasma might be affected by the heating profile, the next logical step was to demonstrate confinement with nuclear heating. In the reference design of the NLB, selected to establish a set of performance characteristics for mission studies, seven modules are packed into an array with moderating material in between to obtain a critical assembly (see Fig. 1). However, a single module could be tested, if the appropriate neutron flux were provided by an auxiliary reactor. UTRC had made plans[4] to test such a module in the "nuclear furnace," which was a reactor designed for that purpose.

We want to demonstrate that high thermal radiation temperatures can be generated by a hydrodynamically confined fissioning plasma using the Annular Core Research Reactor (ACRR)[5] at Sandia National Laboratories. The work presented in this article is part of our evaluation of the suitability of the ACRR for performing single module testing of the NLB concept. The requirements for useful in-reactor testing are high thermal neutron flux (~10^{15} n/s/cm^2) to provide sufficient fission heating of the gaseous nuclear fuel within the test module, and testing times sufficiently long (5–20 s) to assess the uranium-plasma stability during startup and under

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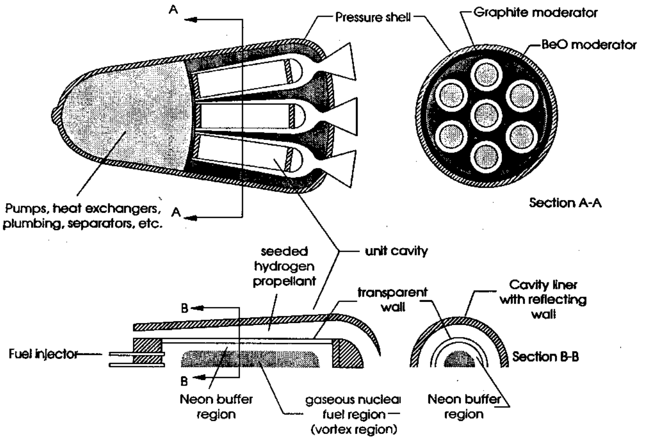

Fig. 1 Diagrams of the closed-cycle nuclear light bulb engine as developed by United Technology Research Center.

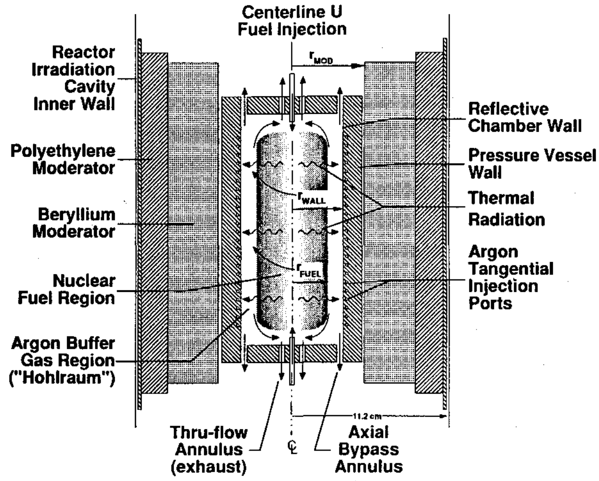

Fig. 2 Schematic of a single module of the nuclear light bulb for testing in the Annular Core Research Reactor at Sandia National Laboratories.

steady-state conditions. The ACRR is a pool-type research reactor capable of programmed transient operation, which is accomplished by a controlled withdrawal of three poison rods. The peak power and total energy yield of the reactor is limited by the negative reactivity feedback associated with heating of the reactor core fuel. The reactor is capable of generating an 80 MW pulse for approximately 5 s. The power shape may be tailored as desired by the appropriate withdrawal of the poison rods; however, the maximum reactor yield is limited to 400 MJ. Consequently, a 10-s reactor pulse would have an average power of 40 MW, etc. The ACRR has a 23-cm-diam dry central irradiation cavity within which experiments may be placed. The active core height is approximately 0.5 m, thus providing a test volume of approximately 0.02 m^3. The total neutron fluence in the central irradiation cavity is on the order of 8×10^{15} neutrons/cm^2; however, the energy spectrum of the fluence is relatively hard. When maximum thermal fission efficiency is desired, as is the case for testing a single module of the NLB, neutron moderating materials must be placed within the central irradiation cavity along with the experiment. Although the moderating material increases the thermal neutron flux, it reduces the available space for the test module, and consequently, the ratio of the volume-to-surface area of the radiating gaseous nuclear fuel. At constant thermal neutron flux, reducing the ratio of volume-to-surface area reduces the radiation temperature emanating from the fissioning plasma. A model of the radiation flow out of the fissioning plasma is necessary to determine the amount of moderating material that results in the highest radiation temperatures. We want to obtain high radiation temperatures because the propellant temperature, and thus, the specific impulse is determined by the radiation temperature emanating from the fuel.

This article is organized as follows. In Sec. II we develop a model of the thermal radiation transport in a cylindrical fissioning plasma. We demonstrate that all of the solutions of this model are contained within a single functional relating two dimensionless quantities. Section III is devoted to the presentation of calculations of the thermal neutron flux as a function of the thickness of the moderating material (module radius) within the ACRR. In Sec. IV, we show how the solution obtained in Sec. II can be used to obtain parametric variations of the performance of a single test module of the NLB within the ACRR. Our results are summarized in Sec. V.

II. Radiative Transport

Figure 2 is a diagram of the in-reactor test chamber to be used in the initial experiments. A reflecting rather than a transparent containment wall is used. This both simplifies the experiments, and leads to higher radiation and plasma temperatures for a given power level. We intend to use this simplified geometry to demonstrate the hydrodynamic confinement of a fissioning plasma at temperatures comparable to the designed operating point for the full rocket. Since the buffer gas is of low opacity, the annular region between the outer edge of the nuclear fuel and reflecting wall will act as a cavity containing thermal radiation. The thermal radiation within this region should be reasonably well described by a blackbody spectrum at the temperature of the fuel at the outer edge. We shall refer to the radiation temperature within this region as the drive temperature T_D, since it is this radiation that will heat the propellant when a transparent wall is used.

Theoretical studies of the opacity of the gaseous uranium[6–8] indicate that the mean free path of a photon in the fuel region will be much smaller than the fuel radius. Therefore, it is appropriate to calculate the radiant flux density F using the diffusion approximation[9]

$$F = -\frac{4\sigma}{3\sigma_{\mathrm{rad}} N_u} \nabla T^4 \tag{1}$$

where σ is the Stefan-Boltzmann constant, σ_{rad} is the Rosseland opacity of the fuel, N_u is the fuel number density, and T is the radiation temperature. Note that we are not studying effects directly dependent on details of the radiation spectrum. We are primarily interested in determining the drive temperature (radiation temperature between the edge of the fuel and the reflecting wall) which depends on the transport of heat out of the fuel. Therefore, the Rosseland opacity, which weighs regions of low opacity most heavily, is appropriate. Assuming steady state, the divergence of the radiant flux density must be balanced by the local rate of heating due to fission Q. Thus

$$\nabla \cdot F = -\frac{1}{r}\frac{\partial}{\partial r}\left(r \frac{4\sigma}{3\sigma_{\mathrm{rad}} N_u} \frac{\partial}{\partial r} T^4 \right) = N_u Q \tag{2}$$

where we have taken only the radial component of radiant flux and written the divergence in cylindrical coordinates. Neglecting the end losses is reasonable since the fuel modules typically have a length much greater than their radius, e.g., $r = 3$ cm and $l = 17$ cm. The fuel and the buffer gas are induced to flow in the azimuthal direction. However, the flow velocity is much lower than the sonic velocity. Therefore, we make the approximation that the fuel is at constant pressure. Furthermore, we assume that the fuel is in local thermal equilibrium (LTE) with the radiation, which is also a good approximation, since the mean free path of a photon in the fuel region is much smaller than the fuel radius.

The fuel number density can be obtained from the ideal gas law

$$N_u = (Pf_u/kT) \tag{3}$$

where f_u is the fraction of uranium in the fuel region by number, P is the total pressure, and k is the Boltzmann constant. Note that we make the assumption that the fuel is fully dissociated. This yields the smallest fuel density for a given pressure, and thus our results will be conservative. Equations (1–3) have been used previously to model the radiation flow in gas-core reactors.[10] In their study the fuel region was assumed to be bound directly by a containing wall, and therefore they set the radiation temperature at the wall equal to the wall temperature. Our situation is somewhat different since the outer boundary of fuel region at $r_{\rm fuel}$ is separated by the transparent buffer gas from a reflective wall at $r_{\rm wall}$. Within this region, $r_{\rm fuel} < r < r_{\rm wall}$, radiation flows freely due to the low opacity of the buffer gas and the diffusion equation is not appropriate. This region acts as a radiation-filled cavity, and consequently, the radiation temperature is nearly constant. Thus, we assume a constant radiation temperature T_D within this region. We obtain a boundary condition at the outer edge of the fuel by balancing the radiation flow out of the fuel against the radiant power absorbed by the partially reflecting wall. The radiant power leaving the fuel is the product of the area of the fuel surface $A_{\rm fuel}$, times the radiant flux $F(r_{\rm fuel})$ given by Eq. (1). The power absorbed by the reflecting wall is approximately $(1 - {\rm Ref})\sigma T_D^4 A_{\rm wall}$, where Ref is the reflectivity of the wall and $A_{\rm wall}$ is area of the wall. The balance of these powers yields the boundary condition

$$\left.\frac{\partial T}{\partial r}\right|_{R_{\rm fuel}} = -\frac{3}{16k}(1 - {\rm Ref})\frac{A_{\rm wall}}{A_{\rm fuel}}\sigma_{\rm rad}Pf_u \tag{4}$$

Since we are neglecting end effects in this analysis, the ratio of the wall and fuel surface areas is simply the ratio of the radii (see Fig. 2).

Note that we have assumed that the reflecting wall is cold, which will cause us to underestimate the drive temperature. We have also neglected transport of heat by conduction and convection since the radiation flow will dominate at the high temperatures expected within the core. A recent study[11] has shown that the inclusion of conduction and convection can have a substantial effect on the radiation temperature profile near the fuel edge where the radiation temperature is lowest. They assumed a wall temperature of 2000 K as compared to a typical edge of fuel temperature (drive temperature) of 4000 K in our study. This factor of 2 in radiation temperature corresponds to a factor of 16 in radiation flow. Furthermore, our primary goal is to calculate the drive temperature, which is determined by the total power generated in the fuel, the ratio of the surface area at the edge of fuel to the area of the reflecting wall, and the reflectivity of the outer wall. The temperature profile determines the fuel density, and therefore can affect the power generated within the fuel, but a change near the outer boundary will have only a small effect on the total amount of fuel. Thus, we feel this omission is appropriate.

Substitution of Eq. (3) into Eq. (2) yields a single equation in T. This equation along with Eq. (4) gives two equations which can be expressed in dimensionless form as

$$t^4[tt'' + 4t'^2 + (tt'/\rho)] = -\alpha_1 \tag{5}$$

$$t'|_{\rho=1} = -\alpha_2 \tag{6}$$

where

$$\alpha_1 = \frac{3Q\sigma_{\rm rad}}{16\sigma T_c^4} r_{\rm fuel}^2 \left(\frac{Pf_u}{kT_c}\right) \tag{7}$$

$$\alpha_2 = (1 - {\rm Ref})\frac{A_{\rm wall}}{A_{\rm fuel}}\frac{3r_{\rm fuel}\sigma_{\rm rad}}{16}\left(\frac{Pf_u}{kT_c}\right) \tag{8}$$

The dimensionless temperature $t = T/T_c$ is normalized to the temperature at the center of the fuel region, the dimensionless distance $\rho = r/r_{\rm fuel}$ is normalized to the radius of the fuel region and $t' = (dt/d\rho)$. To obtain these equations we have assumed that the Rosseland opacity is independent of both temperature and density. At the present time we have very little data on the opacity of uranium at the temperature and densities expected within the fuel region of the nuclear light bulb. As this information becomes available it will be relatively straightforward to generalize this model. For the present, keeping the model simple has the advantage that parameter studies can be done quickly to determine which parameters need to be known with greater accuracy.

For a given value of α_1, Eq. (5) may be integrated numerically from $\rho = 0$ to $\rho = 1$ to yield α_2 through Eq. (6). Repeating this process for different values of α_1 yields the functional relationship $\alpha_2(\alpha_1)$, which is a general solution to the problem. However, we are mainly interested in the drive temperature T_D, since the radiation temperature outside of the fuel will determine the maximum propellant temperature. Thus, we transform this relationship into a new function $\beta(\alpha)$ where

$$\alpha = \alpha_1\left(\frac{T_c}{T_D}\right)^6 = \frac{3Q\sigma_{\rm rad}}{16\sigma T_D^4} r_{\rm fuel}^2 \left(\frac{Pf_u}{kT_D}\right)^2 \tag{9}$$

$$\beta = \frac{2\alpha_2}{\alpha_1}\left(\frac{T_D}{T_c}\right)^5 = \frac{2(1 - {\rm Ref})A_w\sigma T_D^4}{r_{\rm fuel}A_{\rm fuel}Q\left(\frac{Pf_u}{kT_D}\right)} \tag{10}$$

Although this is a general solution to the problem at hand, the result is still not convenient, since T_D appears in both parameters α and β. Note that when the optical depth (size of the system/mean free path of a photon) of the plasma is very small, the radiation temperature is constant within the fuel region, and therefore, $\alpha = 0$ and $\beta = 1$. At large optical depths $\alpha \to \infty$, $T_c >> T_D$, and $\beta \to 0$. The simple analytic function

$$\beta = [1/(1 + \alpha)^{1/7}] \tag{11}$$

has these limits and has been found to fit the numerical results to within 5%. A similarly good fit to the numerically generated ratio of the drive temperature to the radiation temperature at the center of the fuel is given by the simple function

$$T_c = T_D(1 + \alpha)^{0.17} \tag{12}$$

Substitution of Eqs. (9) and (10) into Eq. (11) results in a polynomial. In this form the drive temperature can be found

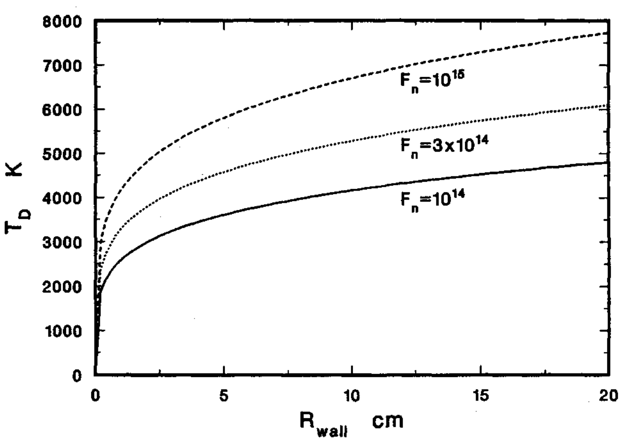

Fig. 3 Drive temperature is plotted as a function of the wall radius for several values of the neutron flux F_n in the units n/s/cm. The wall reflectivity is 0.9 and $\sigma_{\rm rad} = 2.5 \times 10^{-17}$ cm^2.

as a function of any of the other physical variables by simply finding the root of a polynomial. Figure 3 shows the resulting values of T_D as a function of the wall radius for different values of the thermal neutron flux. As expected, T_D increases as the module radius is increased. However, increasing the module radius can only be accomplished by removing neutron moderator from the ACRR, and thus lowering the thermal neutron flux. This dependence is investigated in the next section.

The parameter α is proportional to the number of photon mean free paths in the nuclear fuel, and is therefore typically large. Neglecting 1 as compared to α, Eqs. (9–11) can be solved explicitly for the drive temperature. The result is

$$T_D \cong 0.31 \left(\frac{r_{\text{fuel}}}{r_{\text{wall}}}\right)^{0.41} \frac{F_n^{0.21}(P_u r_{\text{wall}})^{0.17}}{(1 - \text{Ref})^{0.24}\sigma_{\text{rad}}^{0.034}} \tag{13}$$

where F_n is the thermal neutron flux (n/cm^2/s), P_u is the partial pressure of the uranium fuel in atmospheres, r_{wall} is the reflecting wall radius in cm, σ_{rad} is the Rosseland average total radiative cross section for the fuel in cm^2, and the drive temperature is in degrees Kelvin. This expression depends very weakly on opacity, therefore, we can confidently calculate the drive temperature even though the opacity of uranium is not well known.

III. Neutron Calculations

The coupling between the ACRR and the experiment was calculated with the Monte Carlo code MORSE[12] using nuclear data from the 218-group cross section library CSRL,[13] which had been collapsed to a 21-group structure. In these simulations, the experimental module was assumed to be a cylinder 17.8-cm high, containing U^{235} at a density of 7.1 × 10^{19} atoms/cm^3, surrounded by a 2.6-cm buffer gas region, and contained in a closed Zircaloy tube 1.8-cm thick. The fuel radius was varied from 2–6 cm, and in all cases the fuel, buffer gas, and Zircaloy tube were optically thin to the neutron flux. Surrounding the containment vessel was a layer of beryllium metal followed by a layer of polyethylene; both were included to moderate the neutron spectrum in the ACRR cavity. The outer radius of the apparatus was fixed at 11.4 cm (the diameter of the ACRR central cavity is 22.9 cm). As the radius of the fueled cylinder was increased, the thickness of the beryllium moderator was decreased going to zero at an inner moderator radius of 9.4 cm. For inner moderate radii beyond 9.4 cm, the thickness of the polyethylene was decreased. The dimensions of the experiment for various inner radii of the moderator r_{mod} (outer radius of the experimental module) are shown in Table 1.

The mass coupling factors, expressed in units of fission power per unit fuel mass per unit power in the reactor, are also shown in Table 1. The mass coupling factors decrease with increasing fuel radius due to the decrease in the thickness of the moderating materials. This dependence is approximately linear out to r_{mod} = 9.4 cm, dropping more sharply beyond 9.4 cm. The linear behavior for $r_{\text{mod}} < 9.4$ cm is caused by the decreased moderation that results from removing beryllium to accommodate the greater fuel volume, and by increasing axial loss of neutrons from the experimental region as the radius of the fueled region increases. The sharper drop for $r_{\text{mod}} > 9.4$ cm is due to the removal of some of the polyethylene, which is a better neutron moderator than beryllium. In the linear regime the mass coupling M_c is well approximated by the expression

$$M_c = 838[1 - (r_{\text{mod}}/15.4)] \tag{14}$$

where the units of M_c are the same as in the table. We shall use this expression in the next section to couple the results of the radiation transport and the neutronics calculations.

IV. Optimization Studies

Our goal in this section is to optimize the design of the fissioning plasma experiment within the ACRR so that we obtain the highest radiation temperatures consistent with engineering constraints of the ACRR. A major constraint is the wall thickness required to safely contain the high operating pressure of the experiment. The inner radius of the pressure vessel r_{wall} can be estimated using the following equation for thick walled cylinders[14]:

$$r_{\text{wall}} = r_{\text{mod}} \left(\frac{\sigma_t - 4P}{\sigma_t + 4P}\right)^{1/2} \tag{15}$$

where σ_t is the ultimate tensile strength of the material (4081 atm for Zircaloy), and a safety factor of 4 has been introduced into the equation.

Performance analyses were performed by solving Eqs. (9–11) using a root-finding technique. Test dimensions and conditions were introduced through the parameters α and β. The fuel radius was assumed to be a constant fraction of the inner radius of the pressure vessel ($r_{\text{fuel}} = 0.75 r_{\text{wall}}$). The value of Q, the fission heating term, was obtained from the expression $Q = M_c m_u P_{\text{ACRR}}$, where P_{ACRR} is power level of the ACRR, and m_u is the mass of a uranium atom. In all of our examples, the uranium number fraction f_u, which appears in both Eqs. (9) and (10), was taken to be 0.125. This corresponds to expected number density assuming fully dissociated UF_6 as the fuel. Significantly higher performance could be obtained by using pure uranium, but with a significant increase in the experimental complexity.

The calculated drive temperature (radiation temperature at the edge of the fuel) is plotted as a function of the reflecting

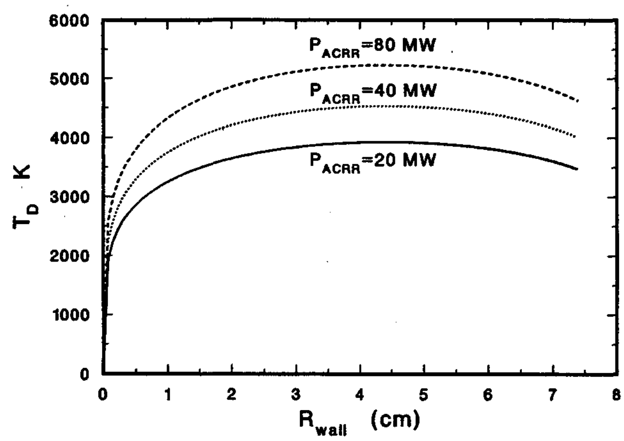

Fig. 4 Drive temperature is plotted as a function of the inner radius of the pressure vessel (radius of the reflective surface) for several values of the ACRR power level P_{ACRR}. The wall reflectivity was assumed to be 0.9, the pressure in the fuel region is 300 atm, σ_{rad} = 2.5 × 10^{-17} cm^2.

Table 1 Details of neutronics calculations

Beryllium inner radius, r_{mod}, cm	6.4	7.4	8.4	9.4	10.4
Beryllium thickness, cm	3.0	2.0	1.0	0.0	0.0
Polyethylene thickness, cm	2.0	2.0	2.0	2.0	1.0
Mass coupling, W/g per MW-ACRR	491	436	379	328	224

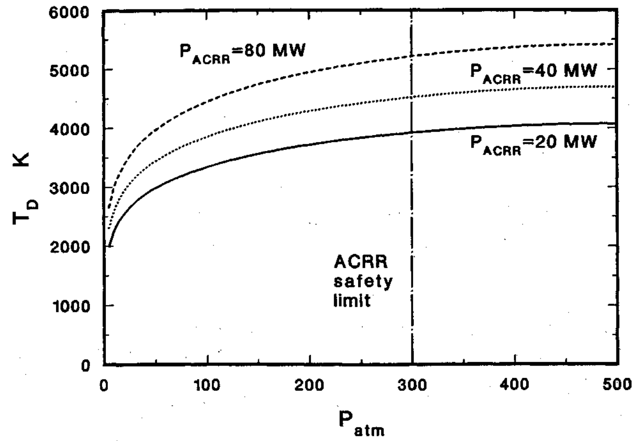

Fig. 5 Calculated drive temperature is plotted as a function of pressure within the fuel region for several values of the ACRR power. The Rosseland opacity was assumed to be $\sigma_{rad} = 2 \times 10^{-17}$ cm^2, the wall had a radius of 4 cm and a reflectivity was 0.9.

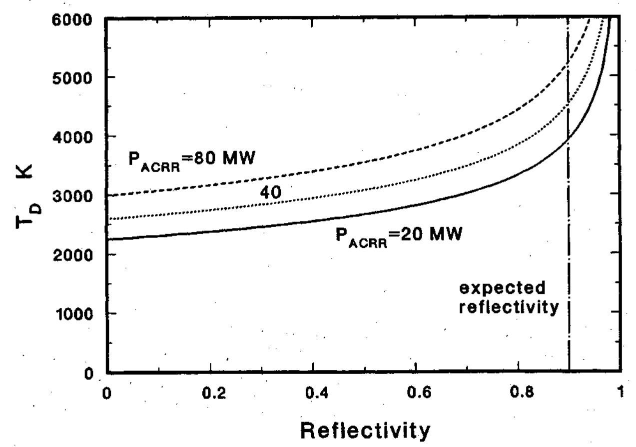

Fig. 6 Drive temperature T_D plotted as a function of wall reflectivity for several values of the ACRR power. The Rosseland opacity was assumed to be $\sigma_{rad} = 2 \times 10^{-17}$ cm^2, the wall had a radius of 4 cm and the pressure within the fuel region was 300 atm.

wall radius for several assumed ACRR power levels in Fig. 4. The maximum drive temperature occurs at a wall radius of roughly 4 cm. The maxima of these curves are broad and very little reduction in the drive temperature results from varying the wall radius by 2 cm in either direction. Furthermore, we have found that the optimum wall radius is insensitive to the variation of any of the physical quantities such as wall reflectivity, opacity, etc. This result gives us confidence in designing the appropriate size of the experimental module.

The variation of the drive temperature with the fuel pressure assuming a fixed reflecting wall radius is shown in Fig. 5. We expect to operate at approximately 300 atm. Thus, the ACRR should provide enough neutron flux to obtain drive temperatures of approximately 5000 K. The curves show that increasing the fuel pressure (or almost equivalently the uranium fraction above 0.125) would result in some improved performance. Studies to investigate alternative forms of fuel and the corresponding injection techniques are in progress.

Figure 6 shows drive temperature as a function of the wall reflectivity for several values of the ACRR power. We expect to have a wall reflectivity of approximately 0.9, which according to these calculations, results in a drive temperature of approximately 5000 K at an ACRR power level of 80 MW. The figure indicates that the drive temperature is a rather sensitive function of the wall reflectivity. However, even a perfectly absorbing wall results in drive temperatures of approximately 3000 K.

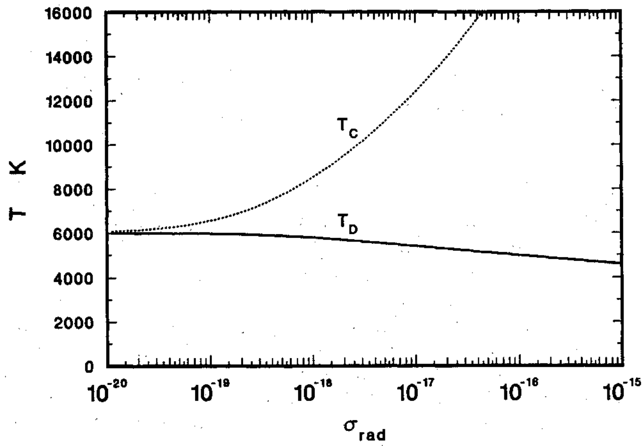

Fig. 7 Drive temperature T_D (solid curve) and the radiation temperature at the center of the fuel T_c (dotted curve) are plotted as a function of the Rosseland opacity. The wall had a radius of 4 cm with a reflectivity of 0.9, and the pressure within the fuel region was 300 atm. The ACRR power was assumed to be 80 MW.

The most complete calculations[7] of the opacity of uranium in the temperature range 5000–70,000 K indicate that $\sigma_{rad} \cong 1 \times 10^{-17}$–$5 \times 10^{-17}$ cm^2. Transmission measurements[8,15] of the absorption of uv radiation through cold UF_6 indicate $\sigma_{rad} \cong 1.0 \times 10^{-19}$–$2 \times 10^{-17}$ cm^2 for photon energies greater than 5 eV, which are responsible for most of the heat transport. Clearly, our knowledge of the opacity of uranium is rather uncertain. The drive temperature and the radiation temperature at the center of the fuel region are plotted as a function of the opacity of the uranium fuel (U^{235}) in Fig. 7. Although the radiation temperature at the center of the fuel increases strongly with opacity, the edge of fuel (drive) temperature decreases very slowly with increasing opacity. A factor of 10 increase in the opacity only reduces the drive temperature by about 500 K. This weak dependence on the opacity is fortunate and certainly adds to the likelihood of successful in-reactor experiments.

V. Summary

We have presented a model of the transport of thermal radiation out of a fissioning plasma core. The complete solution of this model is presented in a form that facilitates parametric studies of gas core nuclear reactors. Our results differ from previous studies in that we have included the effect of separating the fissioning fuel from the wall by a transparent buffer gas and allowed the wall to be partially reflecting. This results in higher radiation temperatures for a fixed specific fission power.

We have used this solution to study in-reactor tests of a single module of the nuclear light bulb, closed cycle gas core propulsion concept. We found that high thermal radiation temperatures (~5000 K) should be possible at the neutron flux levels (~10^{15} n/s/cm^2). We found that this drive temperature is rather insensitive to the opacity of the fuel. This is an encouraging result because the opacity of uranium at the temperatures of interest for a gas-core reactor are not well known and a considerable effort would be required to determine the opacity accurately. However, the drive temperature is strongly affected by the reflectivity of the wall. Therefore, it is important to determine if high reflectivity of a surface can be maintained in the environment of the reactor.

We found that, even for a perfectly absorbing wall, radiation temperatures of approximately 3000 K should be possible using ACRR to provide a neutron flux of (~10^{15} n/s/cm^2). This indicates that this reactor can be used to study the hydrodynamic stability of a fission heated vortex stabilize gas reactor core. This a logical next step toward developing the nuclear light bulb concept into a working propulsion system with the capability of traveling nearly five times faster than chemical rockets, and twice as fast as solid core nuclear rockets.

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