AN AMERICIUM-FUELED GAS CORE NUCLEAR ROCKET

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Abstract

A gas core fission reactor that utilizes americium in place of uranium is examined for potential utilization as a nuclear rocket for space propulsion. The isomer ^{242m}Am with a half life of 141 years is obtained from an (n, γ) capture reaction with ^{241}Am, and has the highest known thermal fission cross section. We consider a 7500 MW reactor, whose propulsion characteristics with ^{235}U have already been established, and re-examine it using americium. We find that the same performance can be achieved at a comparable fuel density, and a radial size reduction (of both core and moderator/reflector) of about 70%.

INTRODUCTION

The open cycle gas core (Ragsdale 1990) fission reactor (GCR) has been identified as a promising advanced propulsion scheme that could readily meet the objectives of the space exploration initiative (SEI) of sending a manned mission to Mars in the early part of the next century. The principle of operation in this system involves a critical fissile core in the form of a gaseous plasma that heats, through radiation, a hydrogen propellant which exits through a nozzle, thereby converting thermal energy into thrust as illustrated in Figure 1.

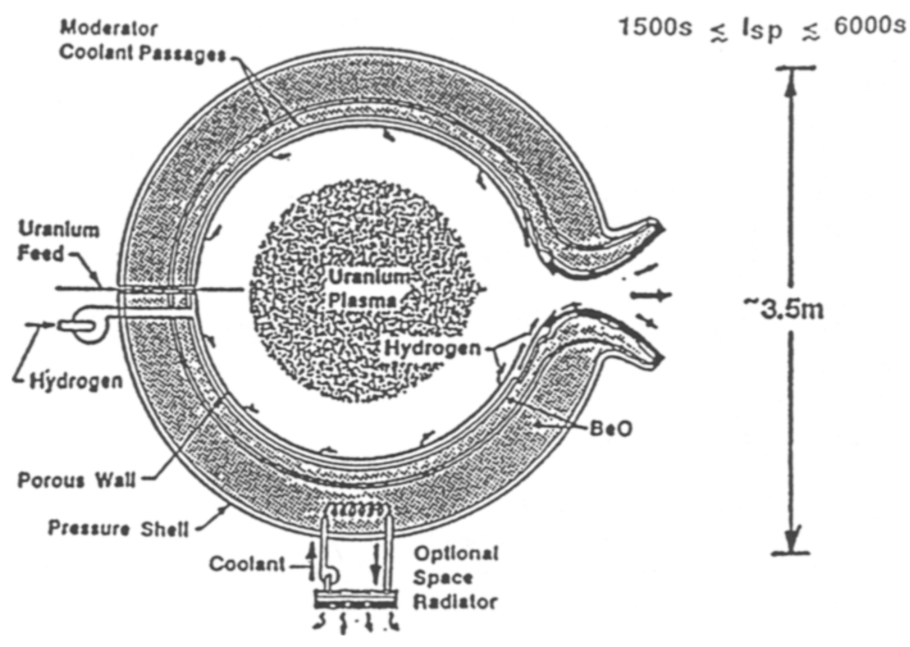

FIGURE 1. High Specific Impulse, Porous Wall Gas Core Engine (Courtesy of NASA, Lewis Research Center).

In contrast to solid core reactors where temperature limitations, imposed by material melting, place severe constraints on rocket performance, the gas core concept circumvents these limitations because the nuclear fuel is allowed to exist in a high temperature (10,000–100,000 K), partially ionized state referred to as the plasma. Nuclear heat released as thermal radiation from the surface is absorbed by a surrounding envelope of seeded hydrogen propellant which is then expanded through a nozzle to generate thrust. With this scheme, specific impulses of several thousand seconds appear to be feasible (Ragsdale 1990).

In a recent paper (Kammash and Galbraith 1992), we examined some of the physics issues associated with fuel confinement and stability in GCR. We found that steady state operation of the reactor is possible only for certain core profiles which may not always be compatible with the radiative aspect of the system. Moreover, we found that the system is susceptible to hydrodynamic and acoustic instabilities that could deplete the fuel in a short time if not properly addressed. In the absence of such problems, however, the propulsion characteristics of GCR can be assessed using a heat transfer model that utilizes a diffusion heat transfer analysis which takes into account the wall material temperature and heat flux limits (Poston and Kammash 1992). It is found that for a 7.5 GW reactor with a propellant flow rate of 5 kg/s, a specific impulse of 3300 s and a thrust of 200 kN can be obtained for a maximum heat flux of 100 MW/m^2.

None of the physics and engineering problems that face the development of the gas core nuclear rocket is perhaps more challenging than that associated with startup. One of the proposed solutions to this problem that might become feasible in the time period when GCR might become a propulsion contender, is the use of antiproton annihilation to generate the required number of neutrons (Jan and Kammash 1992). In calculating the amount of antihydrogen needed, a model consisting of a "cavity" reactor surrounded by a reflector-moderator is utilized, in which moderation of fast neutrons in the core is neglected, and thermal neutrons generated in the moderator-reflector enter the core to initiate the fission reactions. A two-group theory utilizing the time-dependent Fermi age and diffusion equations is used in which the fast neutron source is taken to consist of those resulting from the annihilation reactions and those resulting from the fission reactions taking place in the core. A D_2O moderator at room temperature is assumed, an effective multiplication factor, k_{eff} is calculated, and a power balance equation is utilized to calculate the neutron source strength needed to start the reactor. For the reactor described above, a source of about 10^{22} neutrons was found to be adequate for the startup.

Reactor Fueled with Americium

One of the isomers of americium-242 is ^{242m}Am; it has a half-life of 141 years and has the highest known thermal fission cross section. It is obtained from an (n, γ) capture reaction with ^{241}Am which itself has a relatively high thermal capture cross section (Ronen and Leibson 1988). The isotope ^{241}Am, with a half-life of 433 years, is obtained from the beta decay of ^{241}Pu which has a half-life of 14.4 years.

All these facts, namely high thermal fission cross section, relatively low capture cross section, relatively high ν (the number of neutrons produced per thermal fission), and long half-life make the isomer ^{242m}Am a very attractive nuclear fuel where lower fuel weight (reactor size) is important such as in space applications. The important thermal data for this isotope are given in Table 1.

This isotope decays mostly by internal conversion (99.52%) to ^{242}Am, and by alpha emission at 0.48% with an energy of 5.585 MeV. The isotope ^{242}Am in turn decays mostly (82.7%) by beta emission to $^{242}_{96}Cm$ which in turn decays by alpha emission (with 6.2158 MeV) at a half-life of 163 days. The other branch of decay of $^{242}_{95}Am$ is by electron capture (17.3%) to $^{242}_{94}Pu$ which in turn decays by alpha emission (with 4.983 MeV) at a half-life of 3.76×10^5 years.

TABLE 1. Thermal Data for ^{242m}Am.

	Property	Value
1.	Fission neutron yield per thermal neutron absorbed $\eta = \nu\sigma_f/\sigma_a$	2.693
2.	Number of Neutrons produced per thermal fission ν	3.264
3.	Thermal absorption cross-section; $\sigma_a = \sigma_\gamma + \sigma_f$	8000 b
4.	Fission Cross-section	6600 ± 300 b (or 7350 ± 500 b)[a]
5.	Radiative Cross-section, σ_γ	1400 ± 860 b (or 1650 ± 400b)[a]

[a] From Ronen and Leibson 1988.

It is useful to examine this information to see whether the decay scheme of ^{242m}Am leads to heating of the fuel so as to maintain it in a plasma form that is compatible with a gas core reactor. The range of an alpha particle of energy of 5.585 MeV in an americium gas of density ~10^{18} cm^{-3} is about 45 cm which, as we will note shortly, is approximately the desired size. With an ionization potential of 5.655 eV, we compare this value to the energy per atom associated with the alpha decay which we can readily calculate to be 8.700 x 10^{-4} eV/s. This means that, in the absence of radiation loss, it takes approximately two hours of decay to generate the ionization energy of 5.655 eV. However, the black body radiation from such a system is about 10^7 eV/atom-s, and the bremsstrahlung radiation (assuming instant ionization) is about 2.693 x 10^5 eV/atoms-s, thus the alpha decay is totally inadequate for ionizing the medium, and one must rely on the fission energy to achieve this objective.

As pointed out earlier, ^{242m}Am is obtained from an (n, γ) capture reaction with ^{241}Am, and since it has a high thermal fission cross-section itself, it would appear that due to the two competing processes, very little of ^{242m}Am would accumulate unless a steady supply ^{241}Am is provided. For a spherical reactor that produces 7.5 GW of power with ^{242m}Am fuel, at a density of 10^{18} cm^{-3} and a radius of 40 cm, a thermal neutron flux of 1.462 x 10^{17} cm^{-2}s^{-1} would be required. Noting however that each neutron captured by $^{241}_{95}Am$ gives rise to 5.5 MeV in gammas, each ^{242m}Am gives 190 MeV in capturable fission energy, and each ^{242m}Am neutron capture gives rise to 6.32 MeV in gammas, then, on the average, each neutron absorbed in ^{242m}Am gives:

$$\left(\frac{\sigma_f}{\sigma_f+\sigma_c}\right)(190)+\frac{\sigma_c}{\left(\sigma_f+\sigma_c\right)}(6.32) = 144 \ \ MeV \tag{1}$$

and in steady state the total energy released is 149.5 MeV/cm^3-s. If such a steady state reactor produces 7.5 GW of power then the rate of supply of ^{241}Am required is given by:

$$\frac{dN_{41}}{dt} = V\varphi\,\sigma_{41}N_{41}=V\varphi\sigma_{42}N_{42} \tag{2}$$

where V is the volume of the reactor, φ the flux, σ the absorption cross section and N is the density. With σ_{42} = 8000b, and N_{42} = 10^{18} cm^{-3} and the flux and size as noted above, we see that the rate of supply of ^{241}Am is 1.2599 x 10^{-1}g/s; or 10.885 kg/day. For a journey that takes 6 months on the basis of a continuous burn, acceleration/deceleration type of trajectory, approximately 2000 kg of ^{241}Am will be needed.

Returning to the comparison of a ^{242}Am-fueled GCR and one that uses ^{235}U, with the same moderator-reflector composition and reactor performance characteristics, we should note that the moderator neutron properties remain the same. These include the thermal age, the thermal transport mean free path, the thermal diffusion length, the thermal macroscopic absorption cross section, the thermal diffusion coefficient, and the thermal diffusion time. Clearly, the core properties will change, and these include the absorption cross section, the "interior greyness" factor, and both k_∞ and k_{eff}. In the uranium version (Jan and Kammash 1992) a k_{eff} of about 1.2 was obtained for a 7.5 GW reactor operating at 500 atm pressure and 65,000 K temperature. Figure 2, which shows the variation of k_{eff} vs. the reactor core radius, reveals that $k_{eff} \approx 1.2$ can be obtained at a radius of 40 cm in the case of ^{242m}Am, and that an optimum value of moderator thickness occurs at 0.6 of the core radius. This means that a total radial dimension of 64 cm will provide the same performance as a ^{235}U reactor with a total dimension of 200 cm, or a reduction in a radial size of about 70%. The reduction in volume is clearly more dramatic, and for space applications this could be significant if not critical.

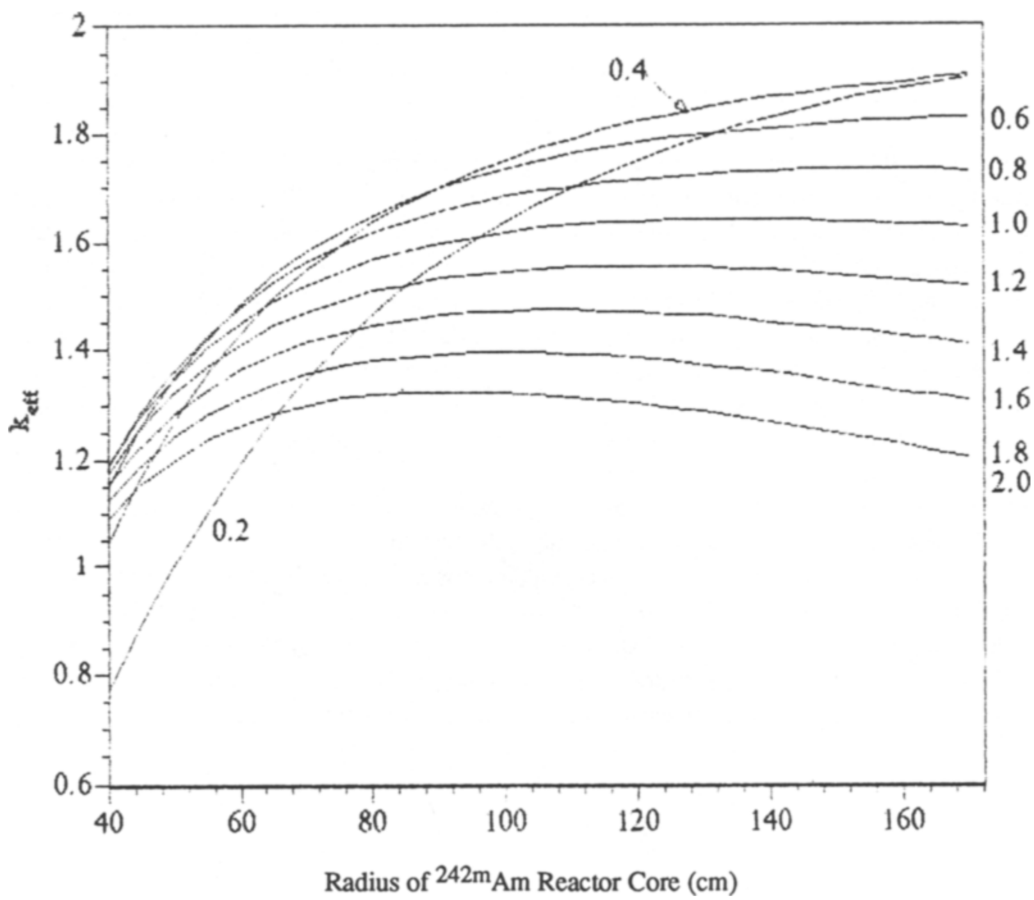

FIGURE 2. k_{eff} versus Radius of ^{242m}Am Gas Core for Different Values of α, where α is the Ratio of Moderator Thickness to Reactor Core Radius. ^{242m}Am Density is 1 x 10^{18} cm^{-3}.

CONCLUSION

We have examined in this paper the potential use of the americium isotope ^{242m}Am in a gas core nuclear rocket that could readily meet the objectives of SEI. Due to its large thermal fission cross section we find that a significantly smaller reactor could produce the same propulsion characteristics as a counterpart with ^{235}U. Since ^{242m}Am is formed by (n, γ) reaction on ^{241}Am, we have shown that approximately 2000 kg of the latter will be needed for a six month journey that utilizes a continuous burn, acceleration/deceleration type of trajectory.

Acknowledgments

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