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Latest Calculations for a Gas Cooled Fast Reactor with a Vanadium Clad Pin Core

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Latest Calculations for a Gas Cooled Fast Reactor with a Vanadium Clad Pin Core

by

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Abstract

The paper gives the results of the latest nuclear and thermal calculations performed for a GCFR with vanadium alloy (V-3Ti-1Si) clad pins and oxide and carbide fuel. The helium temperature at reactor outlet is slightly above 700°C, in case of oxide fuel, which should give a reasonable plant efficiency with a direct cycle gas turbine and/or reduced capital costs. The core fissile inventory and the doubling time are 2800 and 1600 kg, and 18 and 8 years for oxide and carbide fuel respectively. The problems connected with the choice of the cladding material (creep strength, high temperature embrittlement, swelling, compatibility with fuel and coolant) are shortly discussed.

Zusammenfassung

Der Bericht enthält die Ergebnisse der neuesten nuklearen und thermischen Rechnungen für einen gasgekühlten schnellen Reaktor mit Brennstäben in einer Hülle aus Vanadinlegierung (V-3Ti-1Si) sowie oxidischen und karbidischen Brennstoff. Die Heliumtemperatur am Reaktorausgang liegt beim oxidischen Brennstoff knapp über 700°C, was mit einer Gasturbine im direkten Kreislauf zu einem annehmbaren Anlagenwirkungsgrad bzw. zu verringerten Anlagekosten führen müßte. Der Spaltstoffeinsatz im Core und die Verdopplungszeit betragen 2800 und 1600 kg bzw. 18 und 8 Jahre für oxidischen bzw. karbidischen Brennstoff. Die Probleme im Zusammenhang mit der Wahl des Hüllwerkstoffs (Kriechfestigkeit, Hochtemperaturversprödung, Schwellen, Verträglichkeit mit Brennstoff und Kühlmittel) werden kurz gestreift.

Introduction

The reasons that make gas cooling attractive for fast breeders have been dixcussed elsewhere. [1,2,3,4,5] The main advantages relative to sodium-cooled breeders are the probable lower capital cost, especially with direct cycle gas turbines, the expected higher availability, easier maintenance and repair (although these two last points are difficult to quantify without a prototype) and the better breeding. Furthermore, when really high coolant temperature are available the heat sink for the gas turbines could be air or a heating system. Indeed a water reactor of 1000 Mwe has a waste of heat of about 1900 Mwth, enough to heat 400,000 homes if available at say 60^{o}C, which is not practicle with water reactors or sodium reactors with steam turbines.

The disadvantages are the slightly higher plutonium inventory and the much higher coolant pressure with the consequent safety problems (loss of coolant accident, and inadequacy of helium natural convection to remove afterheat from the core). These safety problems can, however, be overcome by the use of a concrete pressure vessel and highly reliable and redundant coolant circulators. On the other hand, GCFR's do not suffer from safety problems of sodium-cooled fast reactors, such as big positive void coefficients and fuel-coolant reactions.

We would like here to stress again, as said before, that there are still many open questions regarding the fuel for a GCFR. The main effort should therefore be directed to fuel development and in-pile tests.

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Vanadium Cladding

In Karlsruhe we strive as has been said, ultimately, to have helium gas outlet temperatures from the core equal or greater than 700°C, with an eye to gas turbine application or steam turbine application with rather small heat exchangers. Steels are not strong enough for such temperatures, although they could be used first at lower temperatures with a steam turbine, especially if a vented fuel concept is adopted.

A series of alloys based on vanadium, titanium, niobium, and silicon are being developed by Böhm of the Materials Laboratory of the Karlsruhe Center in collaboration with the Metallgesellschaft.[6,7] Of these we choose the one containing 96% V, 3% Ti, 1% Si because, after 20,000 hours at 850°C, it has practically the same stresse rupture strength as niobium-containing alloys and the highly neutron absorbing elements have been eliminated (10% niobium in the cladding would produce a decrease of the breeding ratio of 0.13)[3]

Out-of-pile creep tests performed at the Karlsruhe Materials Laboratory have shown that this alloy can stand the stresses given by the coolant gas outer pressure, the gaseous fission products inner pressure, and the pressure given by the swelling of the oxide fuel for a period of two years and temperatures up to 850°C, provided that the out-of-pile mechanical properties of this alloy are not considerably worsened by the very high fast neutron doses to which the fuel elements of a fast reactor are subjected. However, there are indications[7,8] that, like other vanadium alloys, this one tends to maintain ductility when irradiated at high temperatures in a fast flux, or at least is much less affected than steels or nickel-basis alloys.

Irradiation experiments carried out by Karlsruhe on different V-alloys indicated that no high-temperature-embrittlement occurs for fast doses ($E \geqslant 0.1$ Mev) up to 1.4×10^{21} and temperatures up to $750^{o}C$.[7] These results have been confirmed by results obtained by ANL for fast doses up to 3×10^{22} and the same temperatures. Recent results at the Karlsruhe cyclotron show that a helium concentration of 1×10^{-6} atoms/metal atom, equivalent to a fast dose of about 1×10^{22} ($E \geqslant 0.1$ Mev), does not produce any high-temperature-embrittlement in the V-3Ti-1Si alloy up to $900^{o}C$, although other vanadium alloys embrittle in the temperature range $850-950^{o}C$. [8]

The information on swelling of vanadium and vanadium-based alloys under a high fast flux is spare. From the data of Wiffen and Stiegler[9] it is possible to deduce that pure vanadium under a dose of 1.7×10^{22} ($E \geqslant 0.1$ Mev) at $600^{o}C$ shows a volume increase of 0.1%.Harkness[10] did not find any pores in V-20 Ti after a dose of 2×10^{22} at $550^{o}C$. Both these results tend to indicate that the swelling rate is smaller than that of stainless steel.

Vanadium alloys embrittle very rapidly in the presence of oxygen and nitrogen. We have therefore looked into the compatibility problems. The main considerations are given below.

Assuming that the fuel elements remain in the reactor for two years, that helium leakage from the core is equal to 0.1% per day and that every two years the reactor is emptied of helium and filled again with commercially available helium (containing the impurities O_2 = 10 vpm, H_2O =10 vpm, N_2 = 25 vpm), then, during their life, the pins come in contact with 1.7 reactor fillings, that is 26 vpm of O_2 and 43 vpm of N_2.

Assuming that all these impurities react only with the cladding, the weight percentage of O and N impurities in the vanadium alloy at the end of the pin life will be 200 ppm, which is well below the limit where these impurities start to affect the vanadium alloy mechanical properties (3000-5000 ppm). It is clear that the amount of water coming from leakages of the heat exchangers could be considerably higher than the amount mentioned above. We recommend therefore the use of this cladding only in connection with gas turbine cycles, or with steam turbine cycles, where the steam pressure is lower than the helium pressure, for instance 130 and 120 amts respectively.

The compatibility of vanadium and vanadium based alloys containing small quantities of titanium, is good, respectively, with UO_2 and UC for temperature up to 1000 °C and 900 °C; it appears that the compatibility of uranium carbide with vanadium is better than with steels[11,12]. The effect of fission products is still nuclear but we hope to obtain information from our irradiations in FR2. Hofmann[13] has found recently with out-of-pile compatibility experiments that the worst fission products are iodine, selenium, tellurium and cesium. The results of these tests with oxide fuel and these fission products in concentrations corresponding to burn-up of 50 to 100% are given in Table 1.

Table 1 - Extent of reaction zones. UO_2+ J + Se + Te + Cs

Alloy	600 °C/1000 h	800 °C/1000 h
V-Zr2-Cr15	10 µm	50 µm
X8CrNiMoVNb1613(4988)	100 µm	150 µm
AISI 321 SS	150 µm	300 µm
Inconel 625	30 µm	2 mm
Incoloy 800	400 µm	2 mm

All the here simulated fission products react preferably with chromium, we think therefore that the V-3Ti-1Si alloy should be even better than the best investigated alloy (V-Zr2-Cr15).

Performance

Table 2 shows the results of the thermal and nuclear calculations with V-3Ti-1Si clad ful pins.

The nuclear calculations were performed with the latest cross section set of Karlsruhe, the so called MOXTOT set. Compared with the previous sets, this one is improved as follows:

a. the latest evaluated values for α(Pu) are incorporated (values of Gwin)
b. the capture cross section of U^{238} have been lowered (MOXON data)
c. the cross sections of the higher Pu isotopes have been improved.

The details of this set have been presented at the BNES conference in June 1969 in London by E. Kiefhaber et al.[14]

Furthermore, the Pu equilibrium composition has been modified. In the previous calculations[3] this was assumed to be a mixture between plutonium coming from LWR's and GCFR's, while in the present calculation the Pu isotopic composition is the equilibrium composition from GCFR's only. This decreases the breeding ratio for oxide fuel by 0.044, while the new set of cross sections produce a decrease of 0.143. An additional 0.015 is due to the elimination of the 0.4% U^{235} in the blankets, assumed in the previous calculations. Thus the relatively low breeding ratio of 1.315 for a GCFR does not come from the V-3Ti-1Si cladding, which has as low a neutron absorbtion as steel, but from the use of this new set of data and from the new Pu composition.

For the oxide the helium temperature at core outlet is 720^{o}C, while the maximum hot spot temperature at clad mid-wall is 850^{o}C. For the carbide this temperature should be about the same, but it is not sure that the cladding can withstand the higher swelling pressure of the carbide and the considerable thermal stresses across the clad wall. However, if the vanadium alloy maintains its out-of-pile ductility after irradiation these stresses are relatively low. In the oxide case the pressure drop in reactor has been calculated with a constant artificial roughness on the pin walls, while in the carbide case three different roughnesses have been assumed. For continously variable roughness, the pressure drop in the reactor would be 3.5 and 8.7 kg/cm^2, respectively. The roughness data are from Wilkie[15] and Dalle Donne and Meerwald.[16]

The out-of-pile inventories and the system doubling time have been calculated with the assumptions used in the ENEA study[4], i.e., with an out-of-pile time for reprocessing and refabrication of 0.75 years. The resulting out-of-pile inventories are very large, especially in the carbide case, and show that there is a big incentive for trying to reduce this out-of-pile time and/or increase the mean fuel discharge burnup.

Table 2. Main Parameters of 1000 Mwe GCFR

Vanadium Clad (96% V, 3% Ti, 1% Si)	Oxide	Carbide
Core volume (liters)	8470	4181
Fuel volume fraction	0.298	0.2845
Coolant volume fraction	0.552	0.5697
Structure mat. (16/13 SS) volume fraction	0.073	0.070
Cladding volume fraction	0.077	0.0758
Fuel pin diameter (cm)	0.74	0.835
Clad thickness (mm)	0.4	0.45
Reactor inlet coolant temperature (°C)	410	390
Reactor outlet coolant remperature (°C)	720	700
Max.nom. surface temperature (°C)	756	767
Max.fuel pin linear power (W/cm)	440	1271.7
Coolant pressure at reactor inlet (kg/cm^2)	100	120
Pressure drop in reactor (kg/cm^2)	4.8	10.2
Total thermal output (incl.heat produced in blankets) (Mwth)	2449	2778
Plant net efficiency with gas turbine cycle	40.8%	36%
Core power density (kw/liter)	267	581
Pu isotopic composition	0.8306,01431,0.0211,0.0052	
Maximum total neutron flux (n/cm^2 sec)	0.78×10^{16}	1.32×10^{16}
Maximum fast neutron flux ($E \geq 0.1$ Mev) (n/cm^2sec)	0.46×10^{16}	1.06×10^{16}
Maximum fast dose ($E \geq 0.1$ Mev) (n/cm^2)	1.9×10^{23}	2.5×10^{23}
Average core enrichment	13.7%	13.2%

Table 2 Continued

	Oxide	Carbide
Total fissile mass in core (kg of Pu 239 + Pu 241)	2818	1586
Average rating (Mwth/(kg Pu 239 + kg Pu 241)	0.869	1.752
Mean discharge burnup (Mwd/t)	55,000	55,000
Fuel smear density (% of theoretical)	83%	80%
Internal conversion ratio = $\frac{\text{neutrons captured in core fert.mat.}}{\text{neutrons absorbed in core fiss.mat.}}$	0.766	0.776
Total breeding = $\frac{\text{neutrons captured in reac.fert.mat.}}{\text{neutrons absorbed in reac.fiss.mat.}}$	1.315	1.476
Breeding gain = $\frac{\text{excess Pu atoms produced}}{\text{total atoms fissioned}}$	0.330	0.469
Doppler constant $- T \frac{dk}{dT}$ (T in ^oK)	0.731×10^{-2}	0.572×10^{-2}
Total reactivity in coolant (\$)	0.69	0.76
Out-of-pile fissile inventory (kg)+	1415	1507
System linear doubling time (years)+	17.7	7.9

+ Calculated with: plant load factor = 0.8; out-of-pile time for reprocessing and refabrication = 0.75 years; reprocessing Pu losses = 2%.

The net plant efficiency have been calculated according to the gas turbine cycle suggested by Bammert.[15] Other authors[16] suggest that the optimum efficiencies would be lower, with consequent reduction in component size and capital costs.

Table 3 shows the assumptions made to obtain the efficiency indicated in Table 1. The Table shows also a solution with reduced capital costs and lower plant efficiency (33.5%). At this stage we are not able to say which is the better of these solutions, because we have no data on the capital costs.

Table 3. Gas turbine cycle data

	High capital cost solution	Low capital cost solution
Reactor inlet helium temperature (°C)	410.5	343.9
Reactor outlet helium temperature (°C)	720	706
Pressure drop in reactor	4.8%	4.5%
Pressure drop in rest of the circuit	5.5%	5.5%
Turbine internal efficiency	90%	91%
Turbine expansion ratio	2.585	3
Recuperator temp.difference (°C)	27	45
Number of coolers and compressors	3	2
Internal efficiency of compressors (LP, MP, HP)	88%,87%,86%	88%,87%
Helium temperature at compressor inlet (°C)	20	30
Plant net efficiency	40.8%	33.5%
Plant net output (Mwe)	1000	962

The helium temperature at reactor outlet in the low capital cost solution is only 706°C because a recent cladding statistical hot spot analysis based on a method developed by Amendola[19] has shown that the mixed mean helium temperature at reactor outlet is 706°C, and not 720°C, mainly due to the strong power radial gradient in the outermost fuel subassemblies and in the radial blanket. The calculation takes into account of the coolant mixing as well, which is based on experiments of Baumann and Möller[20].

In the case of the low capital cost solution the amount of heat available for heating (water at 60°C) would be 1450 Mwth, i.e. 1245 Gcal/h.

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