SOME IDEAS ABOUT HYBRID SYSTEM CONCEPTS

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ABSTRACT

Hybrid systems were studied in the past for fissile material production. Subsequently these ideas were reconsidered for minor actinide and long-lived fission product destruction as alternative to the traditional final disposal of nuclear waste. Now there are attemps to extend the use of the concepts developed for minor actinide incineration to plutonium burning.

The most promising hybrid system concept considers fuel and target both as liquids.

From the results obtained, it is possible to discard solid and sodium targets. The solutions adopting composite targets seem, at the present stage, the most promising, but still there remains the problem of Pu production at a level not acceptable in a burning system.

The most suitable solvent is heavy water for minor actinide annihilation in the blanket of a hybrid system.

Due to the criticality conditions and the necessity of electric energy production, the blanket using plutonium dissolved in molten salts is the most convenient one.

1. INTRODUCTION

A hybrid system is composed of a particles accelerator, a target, for neutron production, a blanket, for transmutation or fissile material production, and their respective interfaces (i.e. the window between accelerator and target).

These systems were studied in the past for fissile material production. In the last years these ideas were reconsidered for minor actinide (MA) and long-lived fission product destruction as alternative to the traditional final disposal of nuclear waste.

As plutonium is the most important of the actinides (about 95% in mass), the incineration of minor actinides is nonsense if the plutonium problem is not solved .

Plutonium being an excellent nuclear fuel, the most convenient solution should be its burning in a nuclear reactor to produce electric energy. But in the case of very big stockpiles, it seems interesting to burn a part of it in less efficient, but less time consuming systems, such as thermal hybrid systems as demonstrated by the Academy of Science of the United States.

In the present paper different solvents, moderators and target materials are analyzed for MA and plutonium burning.

The most promising hybrid system concept considers both fuel and target as liquids.

2. MINOR ACTNIDES

^{237}Np, ^{238}Np, ^{241}Am, ^{242m}Am, ^{243}Am , ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm and ^{247}Cm.are considered MA taking into account the concentration existing in PWR standard spent fuel reprocessed after 10 years cooling.

3. PLUTONIUM

In the present paper, plutonium means all the actinides present in the discharged PWR standard spent fuel except the uranium isotopes. Standard PWR spent fuel is defined as UO_2 fuel enriched at 3.3% irradiated at 33000 MWd/t

4. TARGET

The target can be solid or liquid. In the first case, it is possible to consider two accident scenarios: loss of coolant and loss of beam. In the first case if the beam is not switched off quite soon the target will melt: in the second one,the fast quenching can provoke a strong thermal shock.

The molecular dynamic calculations seem to demonstrate that long-range order disappears in the areas travelled through by very energetic particles; this means that in these zones the material of the target becomes similar to glass. This result should be confirmed by the behaviour of the tungsten target at the Paul Scherrer Institute (PSI) [1] after a thermal shock it broke into very small parts, practically like powder.

The previous considerations seem to indicate the convenience of liquid targets.

In the case of a heavy metal flowing target, the protons do not penetrate greatly and the majority of the neutrons are produced in the first 20 cm of the target. The most interesting manner to flatten the neutron flux shape is to use composite targets: a inner liquid low mass target, such as lithium or sodium (primary target), surrounded by a solid secondary target made of heavy metal (such as lead or uranium).

A set of calculations for pure materials was carried out considering uranium, tungsten, lead, tin, sodium and beryllium. Uranium targets produce some amounts of ^{234}U, ^{235}U and ^{236}U as spallation product and ^{239}Pu due to epithermal neutron absorption; lead produces the long-lived ^{206}Pb; low mass number targets have low neutron yields and a very high fraction of neutrons are in the high energy range ($E > 20$ MeV). Due to the low neutron yield, it is possible to discard sodium as spallation target and tungsten because it is strong neutron absorver.

A second set of calculations was performed for composite targets (outer dimensions 20 cm diameter and 60 cm high), U-Be and Pb-Be, with the central beryllium zone with different diameters. Low energy neutron ($E < 20$ MeV) distribution in the target is significantly improved. The neutron yield is drastically reduced and the production of U isotopes remain high in the case of uranium targets. The neutron yield for U-Be target with a 2 cm diameter for the beryllium zone (22.94) is quite similar to the yield corresponding to lead (23.35).

A third set of calculations was performed for a target containing thefuel. This is plutonium dissolved in ^{7}LiF. The same Pu concentrations of the fuel (10, 50, 100 g/l) were considered. The maximum neutron yield achieved is 2.32 with a high energy neutron fraction about of 0.33 at 100 g/l; this means that direct spallation on fuel is not convenient.

Taking into account all the parameters considered it is most convenient to use a lead target.

5. BLANKET

For a neutron flux greater than 1.0 10^{15} n/(cm^2 s) it is impossible to use the traditional system: solid fuel and cooling water. In fact some blanket proposals consider liquid fuel: actinide dissolved or dispersed in heavy water. The most recent blanket concepts are based on plutonium or minor actinides dissolved in molten salts, particularly ^{7}LiF - ^{9}BeF, used in the Molten Salt Breeder Reactor Experience at the Oak Ridge National Laboratory.

A series of ANISN cell calculations were carried out considering Pu and MA:

1 - dissolved in heavy water and moderated by heavy water;

2 - dissolved in ^{7}LiF and moderated by heavy water;

3 - dissolved in ^{7}LiF and moderated by graphite.

Pu is at the above mentioned concentrations and MA at 2.5, 5, 10, 50 and 100 g/l as initial concentrations, the maximum neutron multiplication factor (K_{inf}) were verified with MCNP:

5.1 K_{inf} results for MA

The highest K_{inf} values correspond to the case of MA moderated and dissolved in D_2O; the minimum one for MA dissolved in ^{7}LiF and moderated by graphite. In this last case to reach K_{inf} values close to 1.0 it is necessary to consider MA concentrations higher than 500 g/l while the same figure is achieved with a MA concentration about 10 g/l for heavy water. This means that burners using D_2O as solvent and moderator can run with much smaller MA inventory than the corresponding ones having ^{7}LiF as solvent and graphite as moderator.

Taking into account the potential danger of the MA it should be possible to use a less efficient thermal cycle but using D_2O improves the safety conditions of the blanket.

On the other hand for burning plutonium it should be necessary to obtain the maximum efficiency as regards neutron economy therefore burning simultaneously the long-lived fission products should be not convenient. Nevertheless heavy water has better absorption properties than the molten salts.

It seems more convenient to burn simultaneously MA and long-lived fission products using heavy water as solvent and moderator. The long-lived fission products can be placed in the reflector.

5.2 K_{inf} results for Plutonium

The highest K_{inf} values correspond to the case of Pu moderated and dissolved in D_2O; the minimum one for Pu dissolved in ^7LiF and moderated by graphite.

In any case the minimum K_{inf} value is 1.45550; this means that the three sets of solvent-moderator studied are equivalent.

The outlet temperature of the molten salts can reach values of about 640 °C without increasing the pressure, for D_2O is 300 °C at 10 MPa. The efficiency of the thermal cycle for molten salts can reach 42%, against 33% for D_2O.

The increase of efficiency means that the same amount of electric energy is produced reducing the mass of fission products, particularly the long-lived ones.

In [2] another important advantage of the molten salts blanket is analysed that concern its reprocessing.

Hybrid systems having lead target with an effective radius of the fissile zone of 250 cm and a height of 500 cm reach the criticality for a Pu concentration of 7.5 g/l and the corresponding neutron flux is 1.1 10^{15} n/(cm^2 s). With characteristics preliminary these calculations demonstrate that it is possible to burn about 552 kg of Pu per year.

6. CONCLUSION

From the results obtained, it is possible to discard solid and sodium targets. The most convenient solution for target material is, at the present stage, liquid lead, but it is necessary to consider research work concerning composite targets, because these still remains the problem of Pu production, which is not acceptable in MA or Pu burning systems.

It seems more convenient to burn simultaneously MA and long-lived fission products using heavy water as solvent and moderator.

For energy production, the blanket using plutonium dissolved in molten salt allows higher thermal efficiency.

The considered hybrid concept is suitable for plutonium burning.

7. REFERENCES

[1] M.Victoria, private communication.

[2] K. Furukawa, et al., Thorium Molten Salt Nuclear Synergetics, Journal of Nucl. Sci. and Technol. 27 1157-1178 (1990)