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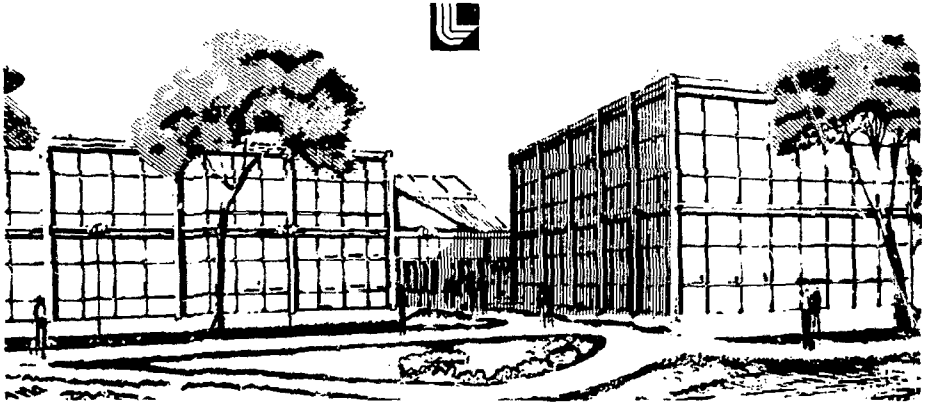

ACTINIDE CROSS SECTION DATA AND INERTIAL CONFINEMENT FUSION FOR LONG TERM WASTE DISPOSAL*

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ABSTRACT

Actinide cross section data at thermonuclear neutron energies are needed for the calculation of ICF pellet center burn-up of fission reactor waste, viz. 14 MeV neutron fission of the very long-lived actinides that pose storage problems. A major advantage of pellet center burn-up is safety: only milligrams of highly toxic and active material need to be present in the fusion chamber, whereas blanket burn-up requires the continued presence of tons of actinides in a small volume. The actinide data tables required for Monte Carlo calculations of the burn-up of ^{241}Am and ^{243}Am are discussed in connection with typical burn-up reactor fusion and fission spectra.

I. INTRODUCTION

There are two major categories of high level radioactive waste from fission reactors: fission products and actinides. All fission products that contribute significantly to the hazard potential have half lives of order 30 years or less, i.e., their contribution to the radioactivity drops on that relatively short time scale although they dominate during the first 200 years or so of storage because of their abundance in nuclear wastes [1]. The truly long term hazard potential is determined by actinides, some of which have half lives of over 10^6 years. The major actinides in question are: ^{244}Cm, ^{2348}Pu, ^{242}Cm, ^{241}Am, ^{243}Am, ^{240}Pu, ^{239}Pu, and ^{237}Np. The Cm and Pu isotopes

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have power source applications and will therefore normally be separated and (commercially) used. The fission cross sections for $^{241,243}Am$ and ^{237}Np are very low for typical fission spectrum neutrons. As a consequence, these isotopes are essentially useless and accumulate in abundance in fission reactors. They are therefore the standard test materials for the evaluation and comparison of potential long-term waste management concepts [1,2,3]. In this paper we describe a transmutation or "actinide burning" method which we had proposed several years ago [4] as a major alternative to blanket burn-up [2] methods.

We know of only one actual publication about Monte Carlo calculations of pellet center neutron reactions on heavy elements [5].

The (computer) design of reactor-size fusion pellets [6] has now been sufficiently finalized for quantitative transmutation studies. The relevant features of a "standard" fusion reactor pellet in its imploded configuration are described in Section 2. The corresponding neutron spectra are discussed in Section 3 along with the results for transmutation of the three isotopes in question using both the "standard" and a somewhat burn-up optimized pellet. We conclude in Section 4 with a brief discussion of the required actinide data.

2. FUSION REACTOR PELLETS

Several ultra-high gain reactor pellet designs have been published recently [7]. Gains of order 1000 are expected for double shell-type targets with split fuel. The present calculations were based on a relatively conservative, simple, one-shell design (by J. D. Lindl) with a gain of order 300. The imploded configuration of this standard pellet is reached after using about 6 MJ of laser energy, and the resulting thermonuclear yield is 1800 MJ. The same pellet configuration with its imploded density doubled served as an example of a design with improved transmutation.

3. MONTE CARLO CALCULATION OF NEUTRON REACTIONS IN THE EXPLOSION

Realistic computations of neutron spectra and reactions require careful consideration of the rapidly disassembling burning fuel. The broadening resulting from fast ions is, for example, quite significant [4,8]. A plasma of thermal electrons and ions, suprathermal electrons, x-rays and thermonuclear reaction products was evolved in time with Lagrangian thermonuclear burn and explosion calculations (see for example reference 9). Neutrons were Monte Carlo transported and the cross sections were taken from reference 10.

A transfer matrix formed from the 1978 evaluated-nuclear-data library (ENDL) neutron cross sections is used to calculate the neutron transport, reactions, and redistribution of neutrons among energy groups. Figure 1 shows the time-integrated neutron spectra for the fuel region of pellets in which the fuel density ρ times the fuel radius R equals 5.0 g/cm^2. The amount of downscattering is noticeably different from some published estimates [11] that did not involve the detailed explosion simulations and extensive Monte Carlo calculations performed here. The high-energy (> 15 MeV) neutrons contribute noticeably to the burn-up by fast fission.

Figure 2 depicts our burn-up results. The nonelastic reactions explicitly followed for each isotope in question are (n, n'), (n, 2n), (n, 3n), (n, 4n), (n,γ) and (n, fission). The actual loss per target seed is plotted in Fig. 2 as metric tons burned up for 10^8 shots. The curves turn over because increased admixture of high Z material to the fuel eventually squelches the thermonuclear bootstrapping by absorptive processes. The limits in the ratio of burned to admixed actinides as depicted in Fig. 2 indicate how a few milligrams of active material will accumulate from each shot. These quantities are, however, similar to the amounts of tritium and activated first wall/structure material produced per explosion in any ICF reactor. The complications of reprocessing unburned actinides therefore do not significantly change the operational hazard potential relative to a "pure fusion" plant. This is in marked contrast to any hybrid and/or blanket burn-up device. Continuous separation of fission products and heavy radioactive materials would seem to be straightfoward with the proposed remote systems of HYLIFE-type reactor concepts [12].

Although exact estimates are not possible without specifying the type of reactor and initial waste reprocessing involved, we conclude that our results demonstrate how just one ICF power plant could safely take care of the actinide waste of about 10 power equivalent fission reactors.

4. CONCLUSIONS

Table I lists the actinide neutron reaction data used in this work. Although the sensitivity to this input has not been calculated in any detail here, it appears that an accuracy of order ten percent would be required to make these feasibility studies more realistic. Reports on the state of the art of neutron data evaluation at this meeting would seem to indicate uncertainties up to factors of two on some of these quantities. In fact, to quote Brian Patrick: no significant total, elastic, non-elastic or capture cross section measurement is known for ^{241}Am....

In addition, more detailed information on fission product yield is of great interest here. The amount of certain long-lived products such as ^{129}I resulting from Am fission at thermonuclear neutron energies could make significant differences in the usefulness of this approach.

ACKNOWLEDGMENTS

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FIGURE CAPTIONS

Figure 1. Neutron spectra for a $\rho R = 5$ g/cm^2 pellet with admixture of 12 mg of ^{241}Am. The dashed line indicates the shape of the spectrum without fission contributions. (For further details cf. H. W. Meldner, ICF and Long Term Nuclear Waste Management, Nuclear Science and Eng., March 1979)

Figure 2. Metric tons of actinides burned up per year as function of their admixture to the DT fuel. The scale refers to 10^8 explosions/year roughly corresponding to a reactor power output of 0.5 to 1.5 GW.

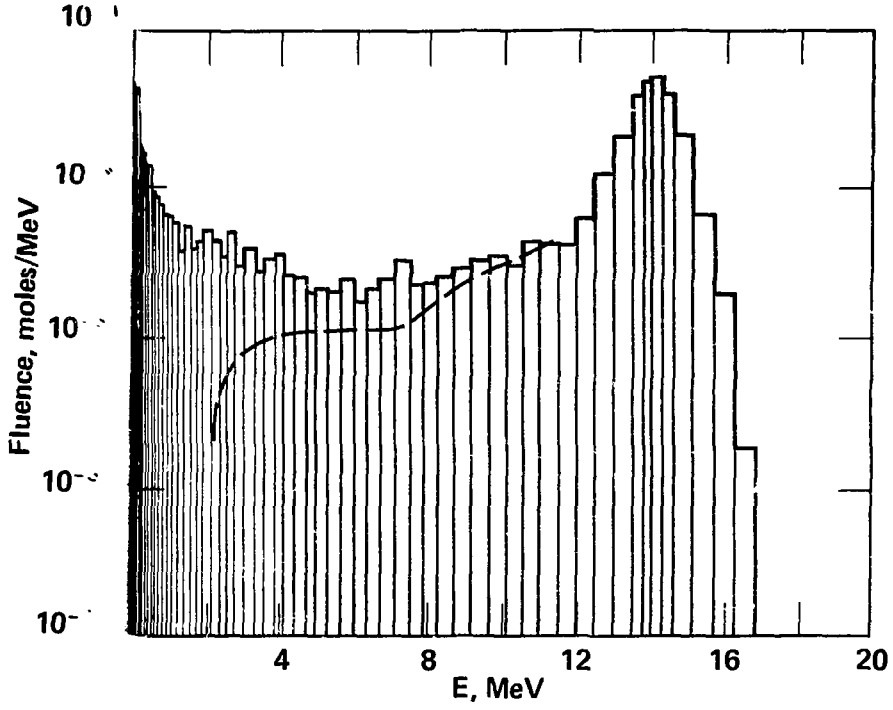

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Figure 1

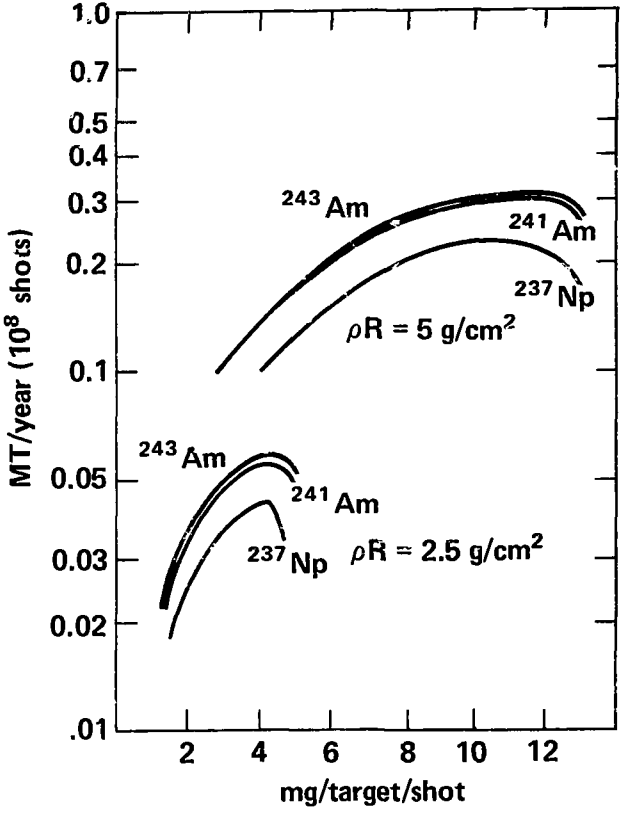
1.0
0.7
0.5
0.4
0.3
0.2
0.1
0.05
0.03
0.02
.01
MT/year (10^8 shots)
^{243}Am
^{241}Am
^{237}Np
$\rho R = 5$ g/cm^2
^{243}Am
^{241}Am
^{237}Np
$\rho R = 2.5$ g/cm^2
2
4
6
8
10
12
mg/target/shot

Figure 2