CH01$0013

NEW HIGH DENSITY MTR FUEL
THE CEA-CERCA-COGEMA DEVELOPMENT PROGRAM

A. LANGUILLE

CEA/DEC Centre de Cadarache, 13108 St Paul Lez Durance CEDEX - FRANCE

J P. DURAND

CERCA, ZI les Bérauds BP1114 26104Romans FRANCE

A. GAY

COGEMA/BCR, 78182 1,rue des Hérons 78182 St Quentin en Yvelines CEDEX - FRANCE

ABSTRACT

The development of a new generation of LEU, high in density and with reprocessing capacities MTR fuel is a key issue to provide reactor operators with a smooth operation which is necessary for a long term development of Nuclear Energy.
In the RRFM'98 meeting, a joint contribution of CEA, CERCA and COGEMA presented a technical classification of the potential candidates uranium alloys. In this paper this MTR working group presents the development program of a new high density fuel.
This program is composed of three main steps : Basic Data analysis and collection, Plate Tests (Irradiation and Post Irradiation Examinations) and Lead Test Assemblies (Irradiation and Post Irradiation Examinations).
The goal to be reached is to make this new fuel available before the end of the present US return policy.

1. INTRODUCTION

The development of Low Enrichment Uranium (LEU) fuels has been set up by the Reduced Enrichment for Research and Test Reactors (RERTR) Program involving at an international level, operators, manufacturers and research laboratories [1]. Currently the highest density fuel qualified for research reactors is LEU silicide fuel U_3Si_2 at a density of 4.8 g/cm^3. It is obvious that this silicide type fuel is limited to 6.0 g/cm^3 [2] and the highest density is still under irradiation for qualification ; it must be noted that this density level is not high enough to satisfactorily convert certain HEU reactors.

By the end of 1996, a « MTR New Fuels » working group was set up in France, involving the three parties CEA, CERCA, COGEMA with the objective to define a relevant R&D program for the development of high density LEU MTR fuels. Various incentives (LEU conversion, back-end solutions,..) lead us to look at higher fuel densities such as 8.0 or 9.0 g/cm^3 ; Uranium-Molybdenum and Uranium-Niobium Zirconium alloys were considered last year [3]. This group has issued a feasibility report mid of 1998. Considering the whole fuel cycle, the three main components have been evaluated : CERCA has tested the manufacture feasibility, CEA has performed neutronics calculations and studied core performances and COGEMA the reprocessing feasibility.

After a brief recall of the potential candidates considered, the purpose of this paper is to present the preliminary results obtained for this evaluation and to give an overview of the programme of development committed in the next coming years.

2. PRELIMINARY STUDIES OF POTENTIAL CANDIDATES

2.1. Preliminary potential candidates

The high density LEU dispersion fuels proposed for development are mainly based on uranium compounds and uranium possibly with small amounts of other metals. Among the definite compounds very few seem interesting to be retained, in particular U_6X and U_3Si compounds have been discarded due to their poor irradiation behaviour (subjected to breakaway swelling at low burn ups [1]), U_2X compound types are not well defined on a metallurgical point of view. So only uranium nitride UN seems attractive in this family.

For the uranium alloys it is well known that the metastable γ phase is required due to its good irradiation behaviour. On this basis U-Ti is rejected because a large amount of Ti is needed to obtain the γ phase, and only two types of alloys are being investigated U-Mo and U-Nb-Zr alloy systems [4]. More precisely on the one hand U-Mo alloys ranging from 6% up to 10% Mo content are considered with an option of a third additional element such as Pt or Ru which increases the γ phase stability. On the other hand U-4Nb-2Zr and U-3Zr-9Nb are also considered [5].

2.2. Fuel manufacturing ability

Based on its manufacturing experience CERCA performed an investigation work mainly for the manufacture of γ phase alloys. After development of the casting and heat treatments, investigations on hot treated ingots showed the feasibility of such γ phase alloys. It is necessary to respect a fuel volume loading in the meat not greater than 55% in the process. The manufacturer was also able to master the manufacturing of fuel plates thanks to the use of the proprietary advanced processes used for U_3Si_2 at 6.0 g/cm^3 which have been adapted. The thermal stability of γ phase alloys was also tested on samples at 400 °C. U-Zr-Nb alloys seemed to be more difficult to fabricate than U-Mo alloys.
It has been demonstrated that the fuel manufacturing of high loaded plates with γ alloys is feasible, and that densities up to 9 g/cm^3 can be achieved.

2.3. Fuel plates inspection

Ultrasonic inspection is usually carried out on fuel plates in order to detect lack of bonding not detected by blister test. A comparative method is used with a standard defect in the meat area of the plate. The increase of density allowed by U-Mo produces a decrease in the US signal due to transitory effects between aluminium edges and the meat ; these effects make impossible an efficient inspection in the edge of meat area [5].

2.4. Fuel neutronics performances and irradiation behaviour

From a neutronic point of view , U-Mo and U-Zr-Nb are the best candidates for development and it is desirable to limit the concentrations of Mo and Nb to minimise cycle length losses from neutron absorption [3], [6]. Uranium nitride UN is not that interesting because nitrogen penalises at the end of cycle (due to the thermalization of the flux) and capture reaction on ^{14}N produces ^{14}C. On the top of that, the density is not so significantly higher than U_3Si_2.

Little is known about the irradiation behaviour of the proposed alloys dispersed in aluminium but the recent results obtained from microplates irradiated at 435 and 70 at%-U^{235} at low temperature (tests RERTR-1 and RERTR-2 in Advanced test reactor ATR at INEL [7]) showed clearly that U-Mo based alloys behave globally better than U-Zr-Nb alloys (where extensive reaction layers and signs of breakaway swelling have been observed).

Concerning the UN behaviour under irradiation, the results obtained from a series of test for FR applications showed a larger swelling rate than the UO_2 oxide one (1.0 %/at% compared to 0.6 a%/at%) in a temperature range (400 °C up to 1200 °C).

2.5. Reprocessing ability

For the evaluation of reprocessing ability of the potential candidates, COGEMA has performed an R&D program at the COGEMA/SEPA research laboratory based on U_3Si_2, U-Mo and U-Zr-Nb alloys in bulk pieces and powder [8]. The nitride UN was not included in this program as its dissolution in nitric acid is already proven (much higher dissolution kinetics than the oxide one).
Different reprocessing aspects have been checked : mainly dissolution, but also extraction and vitrification aspects have been evaluated.
The reprocessing ability led us to classify at the first rank U-Mo and UN fuels (similar to aluminades fuels), then U-Zr-Nb at the second rank and finally silicide U_3Si_2 fuels [6].

2.6. Selection of the reference candidates for the program

Regarding the whole fuel cycle (fabrication, neutronic performances, irradiation and reprocessing), taking into account the current knowledge, a compromise must be found between the neutronic performances and the fabrication/reprocessing ability. The final selection for the development of high density dispersed fuels for MTRs is the following : the reference is based on two candidates U-10Mo and U-8Mo, a back up solution could be a U-6Mo-X (X = Pt or Ru) [7].

3. DESCRIPTION OF THE QUALIFICATION PROGRAM

3.1. Overview of the program

A qualification program based on the candidates selected in the feasibility study has been planned under commitment of the French parties since beginning of 1999. An overview of the content of this program is presented hereafter. The objective of this program is clearly to demonstrate the fuel performance comparable to that of the current LEU silicide fuel used.

The selected U-Mo candidate alloys will be developed on the basis of full-sized plates fabrication, irradiation tests in reactors including post-irradiation examinations and reprocessing tests on irradiated plates (figure 1). A safety evaluation report will conclude the program.

3.2. R&D steps

3.2.1. Fuel manufacturing

A first step was taken in 1998 for the fabrication ability ; to date, CERCA is optimising the fabrication process for full-sized plates. A special procedure of casting and treatment has already been developed in order to get a homogeneous γ phase alloy. For the first tests the ingots were crushed and ground in order to get the powder but the mechanical behaviour of such alloys is completely different from U_3Si_2 and other methods of powder production are under investigation (cryogenic milling, hydride processing). High loaded plates were fabricated with a good uranium homogeneity controlled by X-ray inspection.
The manufacturing of full-sized plates for the first irradiation are under progress at CERCA and they are supposed to be ready for mid 99.

3.2.2. Irradiation program

Two series of irradiation experiments are foreseen to test the performances of high loaded plates. The first phase consists in irradiating a few full-sized plates in a special device which takes the place of a driver assembly in OSIRIS and/or HFR reactor(s). For example the IRIS device in the OSIRIS reactor allows the irradiation of 4 plates and an on-line measurement of the plate thickness is possible during shut down between two cycles. These experimental plates will be irradiated to approximately 60% burn up.
The second phase will focus on irradiation of complete lead test assemblies at least in two different experimental reactors (OSIRIS and HFR). Again the burn-up objective will be close to 80 %.

Fabrication of the device is under progress to-day. Neutronic and thermohydraulic calculations in the device are also under consideration.

So the logic is the following : in the first phase irradiation tests will include at least two candidates (U-10Mo and U-8Mo alloys) and perhaps a third one (U-6MoX). On the PIE results basis, a choice of only one uranium alloy (the final candidate) will be made and lead test assemblies will be tested in the second phase in various reactors (at least two reactors).

3.2.3. PIE

These experiments will be examined in Hot Cells (at CEA Cadarache) when the plates are available. The examinations to be performed will include non-destructive examinations such as visual inspection, dimensional characterisation, gamma spectrometry and also destructive examinations such as metallography, swelling measurements, electron microprobe analysis, burn up determination, FG release. For some controls, a technical method has to be studied, it is the case for example of the Fission Gas measurements in these cold fuels.

3.2.4. Reprocessing tests

In order to complete the reprocessing feasibility, firstly reprocessing tests are to be carried out on as-fabricated high loaded plates (with depleted uranium). and secondly when the first irradiation is completed, tests could be performed on irradiated plates if necessary.

4. MAIN FEATURES OF THE R&D PROGRAM

4.1. Planning

A planning has been forecasted with the objective to demonstrate the performances of a new high density fuel around 2005 (which fits the date of the end of non return US policy). This planning given in figure 1 is very tight with practically no margins. In particular, the second phase is a major challenge for a fuel qualification before 2006.

4.2. Cost evaluation

A first estimation of the global cost of such a program is based on very rough cost evaluation of different steps : fabrication, irradiation, PIE and reprocessing. It is expected this cost will be supported mostly by French partners but also by the international community for certain punctual action by reactors.

4.3. International collaboration

Taking into account the fact this program is first and foremost technically directly connected with the US-RERTR advanced fuel development, it is highly desirable that the results of this program launched by French partners should be discussed at an international level with other parties (R&D laboratories, reactor operators, European Community, ...).

5. CONCLUSION

A comprehensive program has just been committed by the French MTR working Group with the objective to develop a new generation of LEU high density fuel with reprocessing capabilities, it will provide a solution for MTR operators for a long term development of Nuclear Energy. The qualification will be performed on lead test assemblies in different reactors after a selection of the good candidate (objective of the first phase).
As there are strong links with current US-RERTR development program, a firm collaboration is desirable at European level and at an international.

References :

[1] J. L. Snelgrove , G. L. Hofman, C. L. Trybus, T . C. Wiencek (ANL)
« Development of very high density fuels by the RERTR program »
19th RERTR Seoul, Korea, October 1996.

[2] J. P . Durand, Y. Fanjas, A. Tissier (CERCA)
« MTR fuel inspection at CERCA »
15th RERTR Rotskilde, Denmark, October 1992.

[3] J. P. Durand (CERCA), B. Maugard (CEA), A. Gay (COGEMA)
« Technical ability of new MTR high density fuel alloys regarding the whole fuel cycle »
RRFM'98 meeting Bruges, Belgium, March 1998.

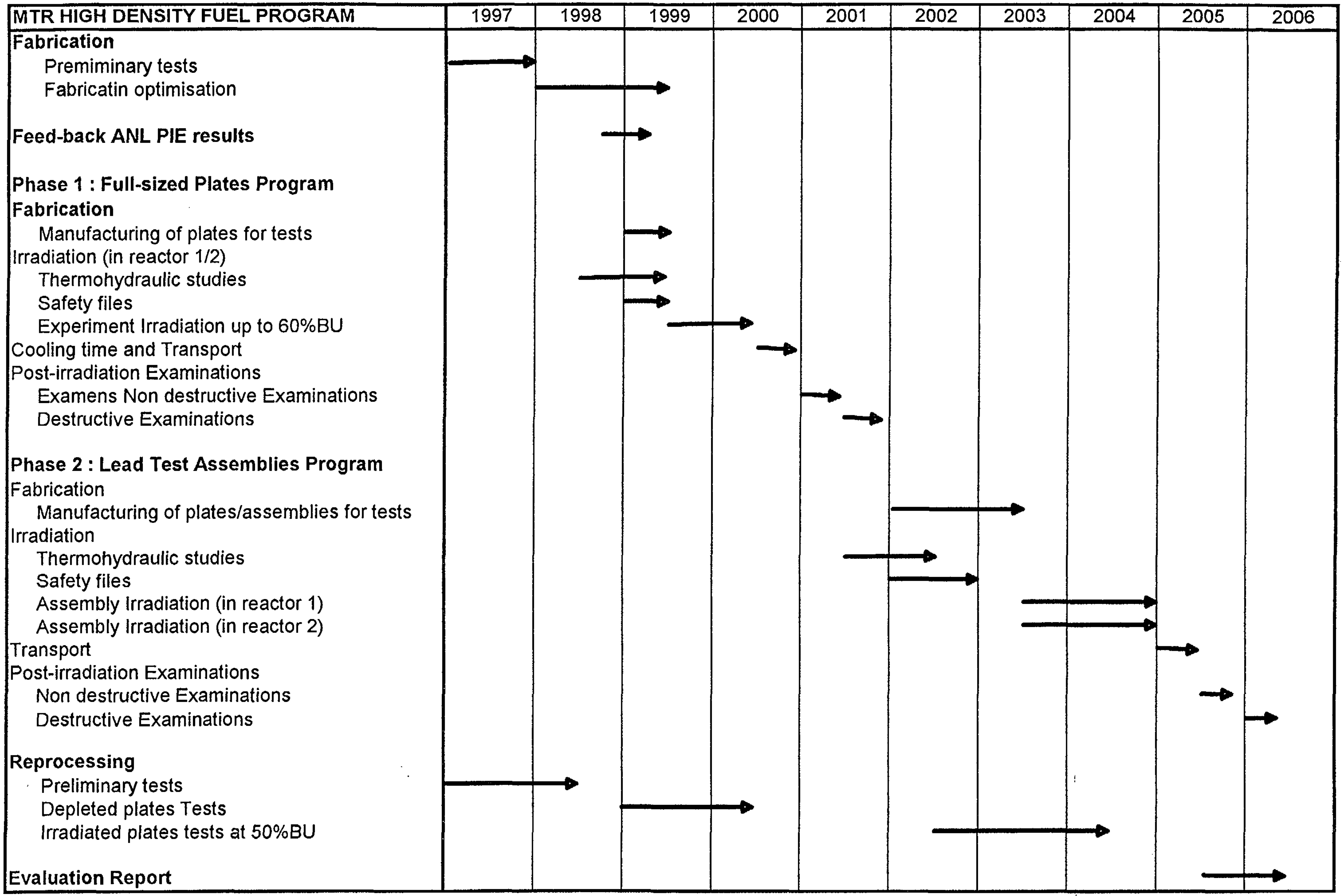
[4] M. K. Meyer, C. L. Trybus, G. L. Hofman, S. M. Frank, T . C. Wiencek (ANL)
« Selection and Microstructures of high density Uranium Alloys »
20th RERTR Jackson Hole, USA, October 1997.

[5] J. P. Durand, J. C. Ottone, M. Mahé and G. Ferraz (CERCA)
« LEU fuel development at CERCA - Status as October 1998 »
21st RERTR Sao Paulo, USA, October 1998.

[6] M. M. Bretscher, J. E. Matos and J. L. Snelgrove (ANL)
« Relative neutronic performance of proposed high density dispersion fuels in water-moderated and D_2O-Reflected Research Reactors »
19th RERTR Seoul, Korea, October 1996.

[7] S. L. Hayes, M.K. Keyer, G. L. Hofman and R. V. Strain (ANL)
« Post-irradiation Examination of high density Uranium Alloy Dispersion Fuels »
21st RERTR Sao Paulo, USA, October 1998.

[8] A. Gay, M. Belieres (COGEMA)
« Reprocessing ability of high density fuel for research and test reactors »
20th RERTR Jackson Hole, USA, October 1997.


MTR HIGH DENSITY FUEL PROGRAM
1997
1998
1999
2000
2001
2002
2003
2004
2005
2006
Fabrication
Premiminary tests
Fabricatin optimisation
Feed-back ANL PIE results
Phase 1 : Full-sized Plates Program
Fabrication
Manufacturing of plates for tests
Irradiation (in reactor 1/2)
Thermohydraulic studies
Safety files
Experiment Irradiation up to 60%BU
Cooling time and Transport
Post-irradiation Examinations
Examens Non destructive Examinations
Destructive Examinations
Phase 2 : Lead Test Assemblies Program
Fabrication
Manufacturing of plates/assemblies for tests
Irradiation
Thermohydraulic studies
Safety files
Assembly Irradiation (in reactor 1)
Assembly Irradiation (in reactor 2)
Transport
Post-irradiation Examinations
Non destructive Examinations
Destructive Examinations
Reprocessing
Preliminary tests
Depleted plates Tests
Irradiated plates tests at 50%BU
Evaluation Report