EURATOM SIXTH FRAMEWORK PROGRAMME

Partitioning and Transmutation and Other Concepts to Produce Less Waste in Nuclear Energy Generation

LWR-DEPUTY

Light Water Reactor fuels for Deep Burning of Pu in Thermal Systems

DELIVERABLE 05

Scope definition of peformance and safety assessment for Th-Pu-MOX and metal-based Pu-IMF fuels in PWR and VVER

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Dissemination level : PU

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Summary

Within the LWR DEPUTY project performance and safety studies are being performed in order to quantify the transmutation potential of reactor cores fuelled with innovative fuels of the CERMET Pu-IMF (using Mo as inert matrix) and Th-Pu MOX types. Both PWR's and VVER's are being considered in this study. This document defines the tasks that will be carried out during the project. An initial definition of the assemblies for a PWR and a VVER using CERMET fuel is also presented. For a PWR reactor filled with Th-Pu MOX fuel, specifications defined within the Thorium cycle project of the 5th EURATOM framework program have been adopted.

Introduction

Within the LWR-DEPUTY project extensive studies are being performed to assess the potentials of reducing the amount of radioactive waste by deep burning Plutonium in light water reactors (LWR). Under investigation are innovative fuels which offer the prospect of burning Pu and at the same time minimize (or complete eliminate) the production of “new” Pu. The performance of both ceramic-in-metal (CERMET) fuel and Th-Pu MOX fuel are being assessed. Diverse aspects are being considered within the project: design, fabrication, licensing, in-pile irradiation, and safety.

Within work package 4, studies are planned with the aim of evaluating the performance and safety consequences to a LWR loaded with these types of fuels. The fuel design will be optimized to increase the transmutation rate of Pu, taking into account the safety margins allowed for LWR’s. Full core neutronic calculations will be performed for both PWR and VVER reactors loaded with 100% IMF assemblies, to determine reactivity coefficients, safety parameters, and the Pu transmutation rate. The safety of a VVER-440 reactor loaded with the same kind of fuel will also be evaluated by means of core transient analysis. Building on the results obtained during the Thorium cycle project of the 5th EURATOM Framework programme, core transient analysis for a PWR loaded with 100% Th-Pu MOX assemblies will be performed. In the coming chapters the scope of these studies will be defined.

1 Pu-IMF CERMET fuel

1.1 Steady state analysis

1.1.1 Definition of a PWR assembly

This sub-section includes the basic data proposed for a PWR assembly based on CERMET Pu-IMF, using depleted Molybdenum as matrix material. The proposed data will be used in the investigation of the burn-up behavior, transmutation performance, and main safety parameters, by means of cell, subassembly and full core simulations. No results will be presented here.

Assembly geometry

The basic geometry data are based on the specifications of a three-loop Westinghouse PWR, and considering a three-batch scheme, as proposed within the Thorium Project of the 5th EURATOM Framework Program [1]. Figure 1 shows schematically the mapping of the fuel assembly, where three types of fuel pins are used: (a) first zone: 84 pins, (2) second zone: 76 pins, and (3) third zone: 104 pins. At the moment all three zones use the same fissile content per pin. Fuel pins of zone #3 however, contain Gd_2O_3 homogeneously dispersed in the Mo matrix which works as burnable absorber, and decrease the initial reactivity to an acceptable level, taking into account the maximum allowed soluble boron concentration in the coolant. The main geometry and operational parameters are given in Table 1 below.

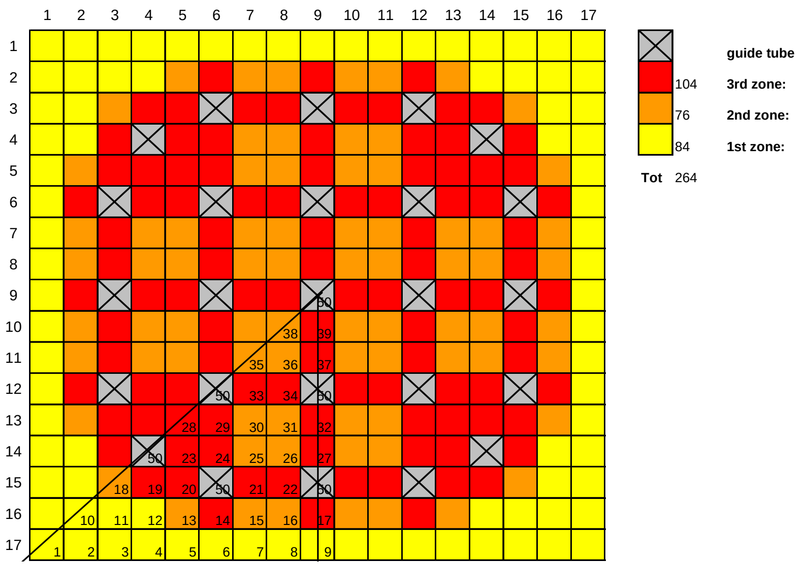

Figure 1 - Mapping of fuel assembly with definition of 3 fuel zones

Table 1 - Basic data for the Westinghouse 3-loop reactor core loaded with Pu-IMF

Core data		
Thermal power	MW	2800
Number of fuel assemblies	---	157
Active height	cm	365.0
Boron-10 composition	w/o	21.35
Boron-11 composition	w/o	78.65
Assembly data		
Assembly configuration, square	---	17
Assembly pitch	cm	21.5
Number of fuel rods	---	264
Guide tube positions	---	24
Instrument tube position	---	1
Fuel rod		
Clad material		Zirconium
Fuel diameter	cm	0.82
Gap diameter	cm	0.83
Clad diameter	cm	0.95
Pin pitch	cm	1.26
Guide and instrument tubes		
Material		Zirconium
Outer diameter	cm	1.25
Inner diameter	cm	1.15
Temperatures		
Fuel	K	930
Clad	K	586
Moderator	K	586
Densities		
Cladding density nominal	g/cm^3	6.555
Moderator density	g/cm^3	0.70
Theoretical PuO_2	g/cm^3	11.46
Theoretical Mo matrix	g/cm^3	10.2
Theoretical Gd_2O_3	g/cm^3	7.41
Matrix porosity	%	10

Fuel composition

The fuel pellets are composed of PuO_2 particles dispersed in a Mo matrix. The PuO_2 content is 12% of the total volume, and the Pu vector used is equivalent to 1st generation Pu discharged from current fleet of LWR's with a discharged burnup of 41 MWd/te (see Table 2). Depleted molybdenum is used as matrix, and its isotopic vector is also included in Table 2. Fuel pins of zone #3 contain 7.5 vol% of Gd_2O_3 (with natural Gd isotopic composition) dispersed homogeneously in the matrix. Table 4 includes the initial composition of the fuel in all three fuel zones. The sizes of the PuO_2 and Gd_2O_3 particles are disregarded, and therefore in the model they are homogenized with the matrix material.

Table 2 - Pu and Mo isotopic vector

Pu-vector		*Mo-vector*	
nuclide	wt.%	nuclide	at.%
Pu-238	2.58	Mo92	29.81
Pu-239	53.85	Mo94	2.88
Pu-240	23.66	Mo95	1.51
Pu-241	13.13	Mo96	1.72
Pu-242	6.78	Mo97	4.17
		Mo98	33.20
		Mo100	26.71

Table 3 - Fuel composition for all three fuel zones

	Fuel zone 1	***Fuel zone 2***	***Fuel zone 3***
	1/(barn*cm)	1/(barn*cm)	1/(barn*cm)
O-16	6.095E-03	6.095E-03	8.865E-03
Mo-92	1.508E-02	1.508E-02	1.380E-02
Mo-94	1.451E-03	1.451E-03	1.328E-03
Mo-95	7.610E-04	7.610E-04	6.962E-04
Mo-96	8.669E-04	8.669E-04	7.930E-04
Mo-97	2.102E-03	2.102E-03	1.923E-03
Mo-98	1.673E-02	1.673E-02	1.531E-02
Mo-100	1.346E-02	1.346E-02	1.231E-02
Pu-238	7.950E-05	7.950E-05	7.950E-05
Pu-239	1.646E-03	1.646E-03	1.646E-03
Pu-240	7.202E-04	7.202E-04	7.202E-04
Pu-241	3.980E-04	3.980E-04	3.980E-04
Pu-242	2.047E-04	2.047E-04	2.047E-04
Gd-152			3.693E-06
Gd-154			4.026E-05
Gd-155			2.733E-04
Gd-156			3.780E-04
Gd-157			2.890E-04
Gd-158			4.587E-04
Gd-160			4.037E-04

Based on this initial assembly design as proposed above by NRG, FZK will evaluate the safety parameters and Pu burning performance, and will suggest to other participants the possible improvements and/or alternative designs in order to satisfy the required constraints. In particular, according to some already performed scoping studies, the following parameters:

- Doppler coefficients
- Moderator temperature coefficients
- Void reactivity coefficients
- Boron worth
- Pu burning performance

will be evaluated. A benchmark will be proposed in order to evaluate the accuracy of the results.

NNL will also work on the optimization the fuel assembly design, focusing mainly on the distribution of the burnable poison pins, their weight percentage, and enrichment.

1.1.2 Transmutation, safety performance, and burnup behaviour of a PWR assembly

Based on the improved assembly design as result of the optimization studies (defined by NRG), the burnup, safety and transmutation of the chosen assembly design will be assessed by FZJ and FZK. The final target is to propose a fuel having a design as similar as possible to the one presently used in industry. FZJ and FZK will perform these assessments on pin cell and/or assembly level. The following parameters will be investigated as a function of burnup:

- Infinite medium multiplication factor (k_∞)
- Reactivity coefficients (fuel temperature and moderator temperature coefficients)
- Transmutation behavior (nuclide composition)

1.1.3 Full core evaluation of a PWR loaded with Pu-IMF

FZK and NNL will generate a whole core 3D model of 100% IMF and evaluate the global core performance, Pu destruction rates and safety parameters.

1.1.4 Definition of a VVER-440 assembly

This chapter includes the basic data proposed by VUJE for a VVER-440 assembly based on Pu-IMF, using molybdenum as matrix material. The proposed data will be used in the investigation of the transmutation performance, and main safety parameters, by means of fuel assembly and full core simulations. No results will be presented here.

Assembly geometry

The basic geometry data are based on the specifications of a VVER-440 reactor. Figure 2 shows schematically the mapping of the fuel assembly for VVER-440 used nowadays, Gd2 fuel assembly, which contains 126 pins of the fuel. The main geometry and operational parameters are given in Table 4 below.

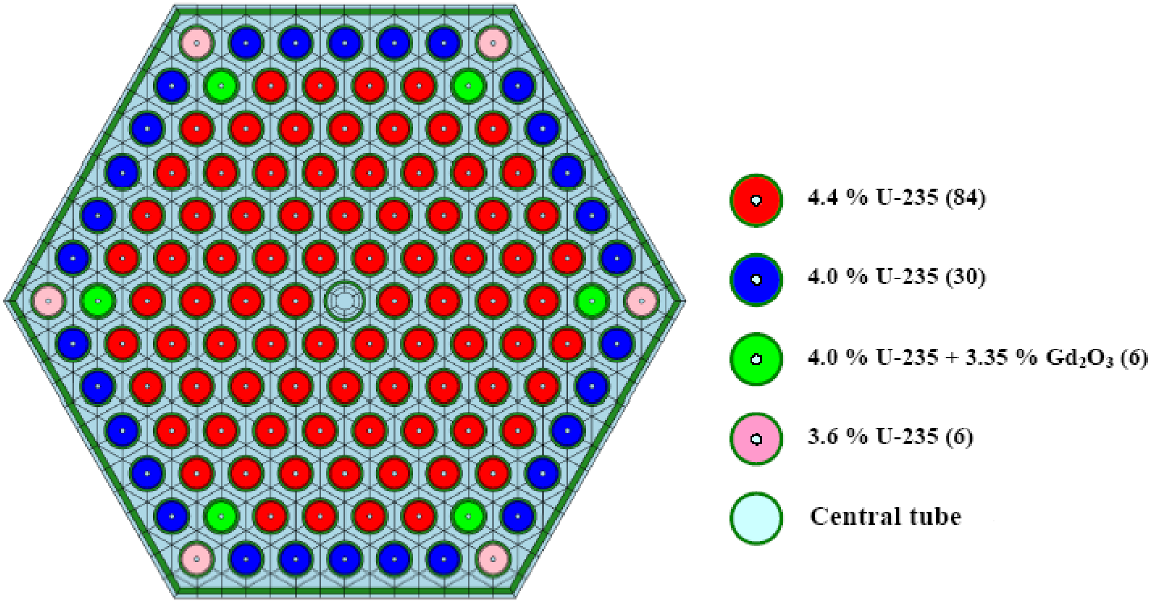

Figure 2 - Mapping of fuel assembly

Table 4 - Basic data for the VVER-440 reactor core loaded wth Pu-IMF

Core data		
Thermal power	MW	1375
Number of fuel assemblies	---	349
Active height	cm	250
Boron-10 composition	w/o	19.8
Boron-11 composition	w/o	80.2
Boron concentration	ppm	cca 500
Assembly data		
Assembly configuration	---	hexagonal
Assembly pitch	cm	14.7
Shroud outer dimension	cm	14.5
Shroud thickness	cm	0.15
Assembly shroud material	---	Zirconium + 2.5% Niob
Number of fuel rods	---	126
Instrumentation tube position	---	central
Fuel rod		
Clad material		Zirconium + 1% Niob
Fuel diameter	cm	0.7600
Fuel inner hole diameter	cm	0.1200
Clad diameter	cm	0.9101
Clad thickness	cm	0.0686
Pin pitch	cm	1.23
Instrumentation tube		
Material		Zirconium + 1% Niob
Outer diameter	cm	1.0328
Inner diameter	cm	0.88

Temperatures		
Fuel	K	778
Clad	K	587
Moderator	K	557
Densities		
Cladding density nominal	g/cm^3	6.555
Moderator density	g/cm^3	0.76
Theoretical UO_2	g/cm^3	10.07

Fuel composition

The fuel assembly will be assembled of two different fuel types (CFA - Composite Fuel Assembly). UOX pins will be for calculation purposes made of pure 4.2% U-235 oxide (mean value of Gd2 fuel assembly enrichment), no impurities will be taken into account. Pu-IMF fuel will be distributed in selected pins, the mapping of the fuel assembly will be optimized on safety requests of operation of VVER core. The first task is to perform this optimization of Pu-IMF pins distribution in the VVER-440 assembly. The main aim is to reach as high Pu content in the assembly as possible.

The inert matrix fuel is composed of Pu from the spent UOX fuel dispersed in molybdenum matrix. The Pu vector used in IM fuel will be equivalent to 1st generation Pu discharged from the equilibrium UOX loading as proposed in Table 2. The Pu content in IMF will be tuned for reaching multiplication properties similar as reached by UOX fuel. Molybdenum will be used as matrix; the vector of molybdenum is the same as used for PWR – see Table 5. To fulfill power peaking factors limitations Gd absorbers shall be added in the assembly.

Table 5 - Pu and Mo isotopic vector for VVER-440 Pu-IMF

Pu-vector		*Mo-vector*	
nuclide	wt.%	nuclide	at.%
Pu-238	2.58	Mo92	29.81
Pu-239	53.85	Mo94	2.88
Pu-240	23.66	Mo95	1.51
Pu-241	13.13	Mo96	1.72
Pu-242	6.78	Mo97	4.17
		Mo98	33.20
		Mo100	26.71

1.1.5 Transmutation, safety performance, and burnup behaviour of a VVER-440 CFA

The burnup behavior of VVER composite assembly will be studied. Work will be focused on:

- multiplication factor,

- nuclide composition changes,
- reactivity coefficients (fuel temperature coefficient, moderator temperature coefficient, reactor power coefficient)
- beta effective
- boron worth
- control rod worth

as a function of burnup. Peaking factors will be studied as well. The aim of optimization of the assembly is to reach as good safety indicators as possible, which means that the Pu-IMF fuel shall be operated in the VVER-440 core without any major changes to the present reactor design.

Transmutation properties will be studied by calculating the nuclide composition changes during the burnup and by calculating fuel cycle indicators.

Burnup behavior, safety and transmutation performance will be calculated by the spectral code HELIOS 1.10 [2].

The result of this work, model of optimized CFA for the VVER-440 core with Pu-IMF fuel, will be studied in the following chapters.

1.1.6 Full core evaluation of a VVER loaded with Pu-IMF

Full VVER-440 core evaluation will be performed using the macrocode BIPR7 with its library based on HELIOS calculation and parameterized with the OKA code. The core will be assembled of CFA assemblies exclusively. The start of IMF full core investigation will be the introduction of CFA assemblies into a UOX core, with cycle-by-cycle discharging of the UOX assemblies. An equilibrium cycle with full CFA loading will be proposed and evaluated.

1.2 Transient analysis

This chapter describes the transient behavior that will be analyzed for VVER-440 with CFA assemblies.

1.2.1 Safety assessment of a VVER-440

Transient analysis of a full CFA VVER-440 core will be performed. A control rod ejection scenario will be studied, where the most effective CR is ejected from the core.

Data for calculation will be prepared by spectral code HELIOS 1.10 and transformed into a library to be used by the macrocode. The equilibrium cycle as mentioned in the section 1.1.6 will be used for the selection of the transient initial condition; the calculations will be performed with the DYN3D code [3].

2 Th-Pu MOX fuel

2.1 Core transient analysis of a Westinghouse 3-loop PWR

Within the Thorium Cycle Project of the 5th EURATOM Framework Program a 3-loop Westinghouse PWR was analysed, where the core had a 100% loading of Th-Pu MOX fuel. A fuel shuffling scheme with four batches had been proposed, and an equilibrium cycle was simulated where cycle 10 was defined as the equilibrium cycle. Considering the aim of the studies at that time, the controls rods were not implemented and thermal-hydraulics feedback was not taken into account.
The following parameters were evaluated:

- Cycle length
- Accumulated burnup at each position
- Nuclide densities
- Relative power distribution across the core
- Critical boron concentration, and
- Feedback coefficients (fuel temperature coefficient, moderator temperature coefficient, boron worth, and void coefficient)

As a natural follow up of this study a core transient analysis for the same reactor type and same reference core will performed within the LWR-DEPUTY project. A control rod ejection scenario will be considered. Starting from the equilibrium core, the reactor will be brought to the initial condition before the initiation of the transient scenario. Hot zero power (HZP) at beginning of cycle is chosen as the initial state, with 10^{-4} % of the full power.

The following sequence of events will be considered: (1) reactor is at HZP; (2) one single CR (with the highest worth) gets stuck into the core; (3) reactor is made critical adjusting the boron concentration; (4) due to a failure of the CR shaft, the CR is ejected from the core within 100 ms.

The following parameters and their time evolution will be evaluated:

- Multiplication factor
- Power
- Maximum fuel temperature
- Average fuel temperature

- Peaking factor
- Cladding temperature and heat flux

NRG will calculate this transient scenario using the code PANTHER . The necessary nuclear library containing the diffusion and kinetic parameters in two energy groups will be generated using the lattice codes WIMS8.

2.2 Safety assessment and transient analysis for a Siemens-KWU PWR

In Germany, burning reactor-grade plutonium in Siemens-KWU PWR's has been common practice for several years. There is much experience in utilizing uranium-plutonium (U-Pu) MOX fuel assemblies. For example, one third of the PWR fuel assemblies are made of MOX, the rest contains conventional low-enriched-uranium oxide fuel.

Replacing U-Pu MOX by Th-Pu MOX is expected to cause only minor changes in the reactor core behavior; given the fact that thorium has similar (resonance) absorption properties as uranium-238. Starting with rather small modifications in real core configurations of operating NPP may facilitate the introduction of Th-Pu MOX, and make licensing of this innovative fuel easier. However, the preservation of PWR safety features has to be demonstrated, clearly and in detail, prior to real introduction of Th-Pu MOX.

Respective fuel-assembly and core calculations are to be performed, using the multi-group lattice transport code HELIOS and the 2-group nodal diffusion code DYN3D.

First, HELIOS is to be validated for the innovative PWR fuel against a benchmark problem based on measurements of a ThPu-MOX rodlet that was irradiated during four fuel cycles in the PWR of NPP Obrigheim (Germany). This validation is to be performed under task 4.3 of work package 4. Next, a real Siemens-KWU PWR core loading pattern (equilibrium cycle) will be considered, with one third of the fuel assemblies consisting of uranium-plutonium MOX. This pattern suggests itself to be modified by simply replacing the conventional MOX by thorium-plutonium MOX, preserving the whole geometry of fuel rods and assemblies. A respective two-group library of diffusion and kinetics parameters will be generated by HELIOS to be used as input for the reactor dynamics code DYN3D.

DYN3D can be applied to calculate safety-relevant parameters for the modified core, such as assembly-power peaking factors, maximum and average fuel temperatures, control-rod and boron worth, and, most important, the fuel and moderator temperature coefficients (FTC and MTC). A comparison should be made to the respective parameters in real (recent) core configurations.

FZD will calculate the transient scenario described in the previous section by using the code DYN3D, but for a Siemens-KWU PWR. Furthermore, the behavior of the modified core is to be studied for an anticipated transient without scram (ATWS) caused by a loss of feed water supply. According to the guidelines of the German Reactor Safety Commission (RSK), postulated ATWS events, though very unlikely, have to be analyzed with regard to their consequences on the safety of nuclear power plants. Such analyses are carried out in order to show that at any time

- the mechanical integrity of the primary circuit is guaranteed, and
- the cooling of the reactor core is ensured.

Since the course of ATWS transients is determined by a strong interaction of the neutron kinetics with the thermal hydraulics of the system, coupled 3D-neutron-kinetic/thermal-hydraulic code systems are adequate tools for the analysis of such transients. The coupled code system DYN3D/ATHLET [4]can be applied, ATHLET modeling the thermal hydraulics of the PWR circuits, which influence the core neutron kinetics.

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