COMPARISON BETWEEN FEMAXI-6 AND TRANSURANUS PREDICTIONS REGARDING MOX FUELS BEHAVIOUR

COMPARAȚIE ÎNTRE PREDICȚIILE CODURILOR FEMAXI-6 ȘI TRANSURANUS PRIVIND COMPORTAREA COMBUSTIBILULUI MOX

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Abstract: *The purpose of the present paper is to asses fuel performances codes capability to predict mixed oxide (MOX) fuel behaviour. In this respect, two computer tools were used, TRANSURANUS and FEMAXI-6 for the simulation of two experimental rods loaded with MOX fuel. Codes predictions were compared with experimental measurements. The results presented in the current paper are: fuel centerline temperature, volume of fission gas release, fuel rod internal pressure and pellet-clad diameter gap, and the values obtained are plotted only for those models which agree fairly well with the measurements. As a general conclusion, MOX fuel behaviour is reasonably predicted with TRANSURANUS and FEMAXI-6.*

Keywords: MOX fuel, FEMAXI-6, TRANSURANUS, key models.

Rezumat: *Scopul lucrării este de a evalua capabilitatea codurilor de analizăa performanțelor combustibilului de a prezice comportarea combustibilului de tip oxid mixt de uraniu și plutoniu (MOX). În acest context, două instrumente de calcul, TRANSURANUS și FEMAXI-6,au fost utilizate pentru simularea comportării la iradiere a două elemente combustibile exprimentale încărcate cu MOX. Predicțiile celor două coduri au fost comparate cu măsurători experimentale. Rezultatele prezentate în lucrare se referă la: temperatura din centrul combustibilului, volumul de gaze de fisiune eliberate, presiunea internă și dimensiunea interstițiului dintre pastilă și teacă. În concluzie, comportarea combustibilului MOX este adecvat prezisă .*

Cuvinte cheie: combustibil MOX, FEMAXI-6, TRANSURANUS, modele cheie

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1. Introduction

In general, the modelling activities are required in order to gain a better understanding of nuclear fuel behaviour under irradiation, especially in the context of the Generation IV activities and the utilisation of new materials for fuel and cladding. However, it is difficult to predict the complex behaviour of a fuel or cladding in a new situation or the behaviour of a new material. Though, some guidance can be obtained by using specific fuel performances codes.

FEMAXI-6 [1] and TRANSURANUS [2] are very complex tools that can be applicable to a wide range of designs and conditions. In order to verify their capability to predict MOX fuel behaviour, data from IFA-597 experiment were used. The experiment consisted in the irradiation of two rods loaded with MOX fuel, named in this study as ROD1 and ROD2 (one solid and one hollow). Fabrication data and power histories are provided in this paper. During the experiment, fuel central temperature and internal pressure were measured. Results of the gas puncturing and gas analysis were also obtained [3].

Both fuel performances codes consider a single fuel rod and the surrounding structure. The coolant temperature and pressure are given as input data. The heat transfer coefficient between fuel rod and coolant was calculated. The thermal and mechanical analysis are interrelated and coupled through an iterative procedure.

The paper starts with a brief discussion about models and parameters used in the input files for FEMAXI-6 [4] and TRANSURANUS simulations. Based on comparison of codes results with experimental values, we identified different models which are able to predict more accurate MOX fuel behaviour under given specific experiment conditions. Therefore, most of the input options and models were kept the same in the analysis of the two experimental rods. Each fuel rod was divided into a number of four segments and the results are graphically represented only for the segment corresponding to fuel thermocouple position. The input parameters with axial variation are related to fabrication data (different geometry for the fuel pellet) and operating conditions (linear rod power and fast flux neutron).

2. Models and parameters

In our study, a peculiar attention was given to pellet thermal conductivity (the most important parameter), for which different models have been used. Both TRANSURANUS and FEMAXI-6 codes have implemented several models for the evaluation of thermal conductivity of MOX pellets, such as Baron and Halden models used in FEMAXI-6 calculations and TRANSURANUS simulations were carried out using as well Halden model and TRANSURANUS standard model - best estimated correlation of ITU for MOX fuel [5].

Another important aspect in predicting fuel rod behaviour is related to fuel relocation. In FEMAXI-6, the initial relocation is given as input data, which is an increment in pellet diameter; in our case this value is assumed to be 20% of the initial diameter gap size for ROD 1 and 18% for ROD2. TRANSURANUS code has implemented several relocation models, selected by an input option called *ireloc*. In the present analysis, we used an option where radial, tangential and axial strain increment due to relocation is calculated according to modified FRAPCON – 3 model.

Fission gas release and fuel swelling represents also a delicate matter which has to be covered in fuel rod analyses because has an significant impact on fuel temperatures, internal pressure and may lead to enhanced pellet-cladding mechanical interaction (PCMI).

Pellet swelling model from FEMAXI-6 used in this study is the Studsvik swelling model, where the maximum pellet volume reduction by densification is assumed to be 1.0 vol %. In TRANSURANUS calculation, Preusser correlation was used, where swelling is constant 2% at start-up, 1.2 % if gap is open and 0.65% in case of contact between fuel and clad [2].

Both codes have an input parameter at which irradiation induced densification is completed, in our case the value was taken 3000 MWd/tU (72 MWh/kgU).

FEMAXI-6 uses White – Tucker model for predicting fission gas release of MOX pellets with a diffusion coefficient of Turnbull. Regarding TRANSURANUS option, we selected a simplified model (standard option) that threats fission gas release and gaseous swelling with two separate models. We used the same effective diffusion coefficient of Turnbull.

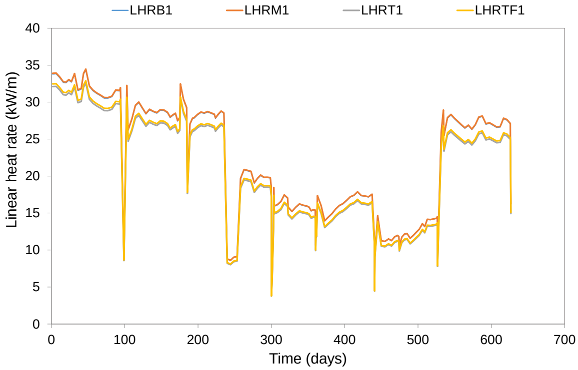
For the gap thermal conductance evaluation, the modified Ross and Stoute model was used in FEMAXI-6 and URGAP model was selected in TRANSURANUS code, this last mentioned model is applicable for different gas mixtures, therefore the user can select four options for the thermal conductivity of gas mixtures. In our case we use TRANSURANUS standard option.

3. Fuel rods data

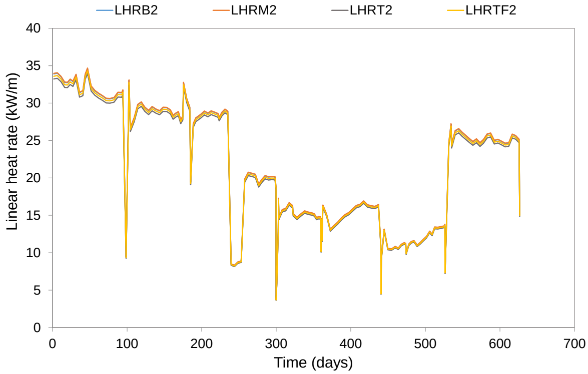
Fabrication data of the two fuel rods irradiated in Halden reactor are summarised in Table 1. All the two fuel rods were instrumented with fuel thermocouple and pressure transducers. Fuel stack length of ROD1 was 224 mm and consisted of 17 solid pellets and 4 hollow pellets at the top to allow thermocouple instrumentation. The entire fuel stack for ROD2 consisted only of pellets with central hole and the total length was 220 mm.

The linear heat rates (LHR) of each fuel rod are illustrated in Fig. 1 at different axial positions, including the local heat rate at the axial elevation of the thermocouple (LHRTF1, LHRTF2) [3].

Experimental fuel rods were divided in four axial segments, corresponding to axial power distribution provided in the documentation from IFA-597 experiment. As it can be seen, the fluctuations are negligible, however we use all these power histories (Fig. 1) in our analyses. As it was mentioned above, code results are provided for the segment corresponding to the thermocouple axial elevation.



a) ROD1



b) ROD2

Figure 1. Power histories for ROD1 and ROD2

Table 1. Specifications and test conditions

	ROD1	**ROD2**
Pellet		
Pellet type	MOX	MOX
Enrichment fissile Pu/Pu total/U^{235} (%)	6.07	6.07
O/M ratio	1.999	1.999
Fuel density (g/cm^3)	10.54	10.54
Pellet radius (mm)	4.02	4.02

	ROD1	**ROD2**
Central hole radius (mm)	0.9 / 0.0	0.9
Pellet lenght (mm)	10.7	10.5
Dish depth (mm)	0.26	0.26
Land width (mm)	5.3	5.3
Chamfer height (mm)	0.15	0.15
Chamfer width (mm)	0.3	0.3
Grain size (μm)	4.4	4.4
Pellet surface roughness (μm)	1.4	1.4
Clad		
Inner radius (mm)	4.11	4.11
Outer radius (mm)	4.75	4.75
Cladsurface roughness (μm)	< 0.15	< 0.15
Fuel rod		
Pellet number	21 (4 pellets with hole)	21 (all hollow)
Fuel Rod lenght (mm)	224	220
Filling gas	He	He
Initial gas pressure (MPa)	0.5	0.5
Termocouple position (mm)	184	180
Operating conditions		
Fast flux level (n/cm^{2}s)	1.61* E11 *LHR	1.61* E11 *LHR
Coolant pressure (MPa)	3.36	3.36
Coolant temperature (°C)	240	240

4. Results

The results obtained with FEMAXI-6 and TRANUSRANUS codes are represented only for the axial segment corresponding to thermocouple position and compared with experimental data. Measurements contained information about fuel centreline temperature, plenum pressure and fission gas release. Additionally, pellet-clad diameter gap was calculated with FEMAXI-6 and TRANSURANUS and its evolution with irradiation is presented below.

In this study a limited sensitivity analysis was carried out and different models for thermal conductivity of MOX pellets were used.

In all studied situations, the calculated pellet centreline temperature for ROD1 agrees fairly well with measurements. In the case of the experimental fuel rod with hollow pellets, ROD2, FEMAXI-6 calculations sliglthly underestimate the measurements at the begging of the irradiation period and then the plotted values are situated above the experiment. The maximum differences reach about 100 °C.

Regarding the simulation of an experiment it is known that cannot always predict the experiment perfectly. What it is important to know is if the differences between measured and calculated values are considered acceptable. In our case, the maximum differences obtained for the fuel temperature are below 10% deviation from the experiment. Both codes over predict the fuel temperature measurements in last part of the irradiation period. The differences between the models predictions used in the sensitivity study are small, so are the differences between the two codes predictions, especially for ROD1 (Fig. 2, Fig. 3).

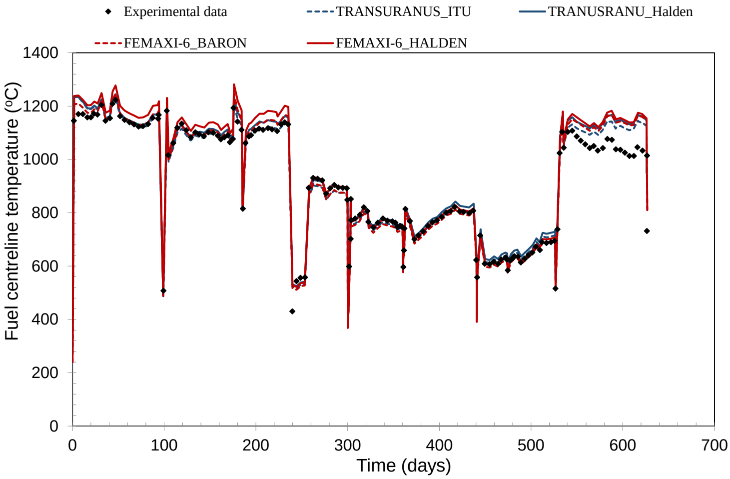


Figure 2. Measured and calculated pellet centreline temperatures for ROD1

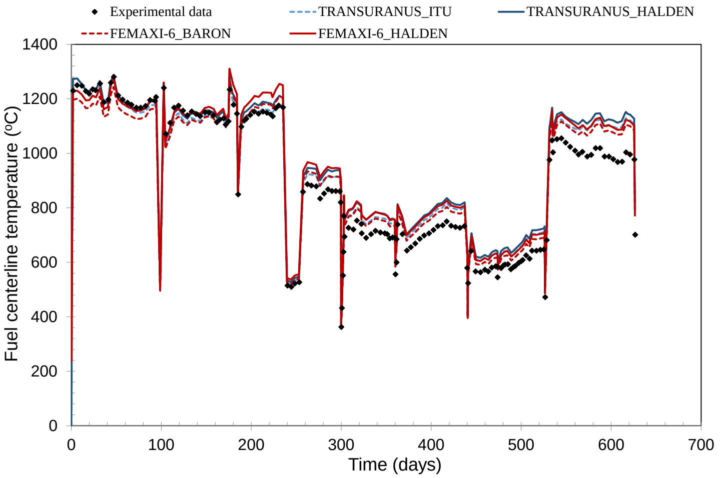


Figure 3. Measured and calculated pellet centreline temperatures for ROD2

In Fig. 4 one can see an increase in internal pressure compared with experimental data, in the first two stages of the irradiation, except the values obtained with FEMAXI-6 and Baron model obtained for the second cycle (between 250 – 550 days). Both codes underestimate the experiment at the end of the irradiation period and their predictions look similar. In the case of ROD2, where we obtained higher differences between the estimated fuel temperatures compared with experimental data, these differences are reflected in the evolution of the internal pressure and fission gas release. As can be seen from Fig. 5, FEMAXI-6 results obtained with BARON model are very similar with the experimental data for all the irradiation period, while TRANSURANUS predictions are more closer with the measurements in the later part.

It should be noted that in Fig. 6 and Fig. 7 is represented the total amount of Xe and Kr release from fuel and not the total amount of fission gas release. Taking into account the power increase in the latter stages of irradiation, notable gas release was registered. The interdependencies between fuel temperature, fission gas release and internal pressure can be observed in the graphs. For ROD1, the calculated volume of fission gas release obtained with both codes is underestimated, which is normal if we take into account the under prediction of the internal pressure. In the case of ROD2, where the internal pressure at the end of the irradiation is more closer to experimental data (Fig. 5), the values calculated for the fission gas release agree very well with the measurements (Fig. 7), especially when the TRANSURANUS standard model for MOX thermal conductivity was used.

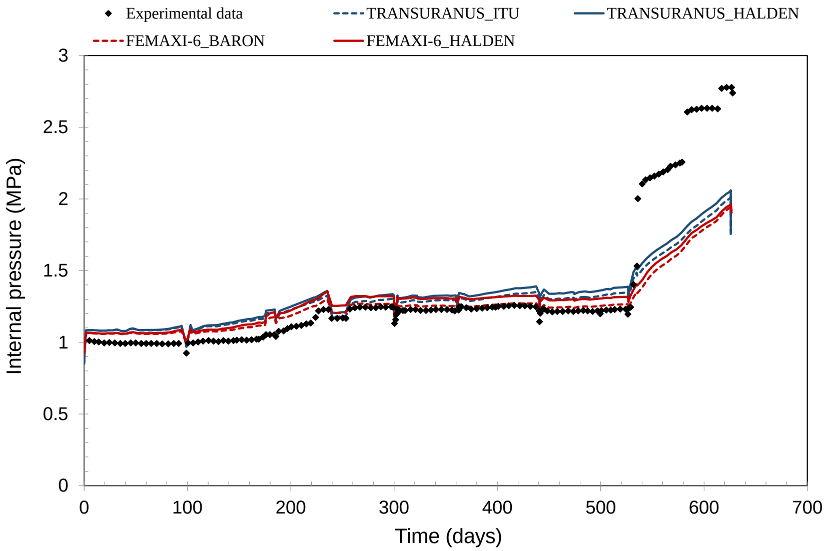


Figure 4. Measured and calculated internal pressure for ROD1

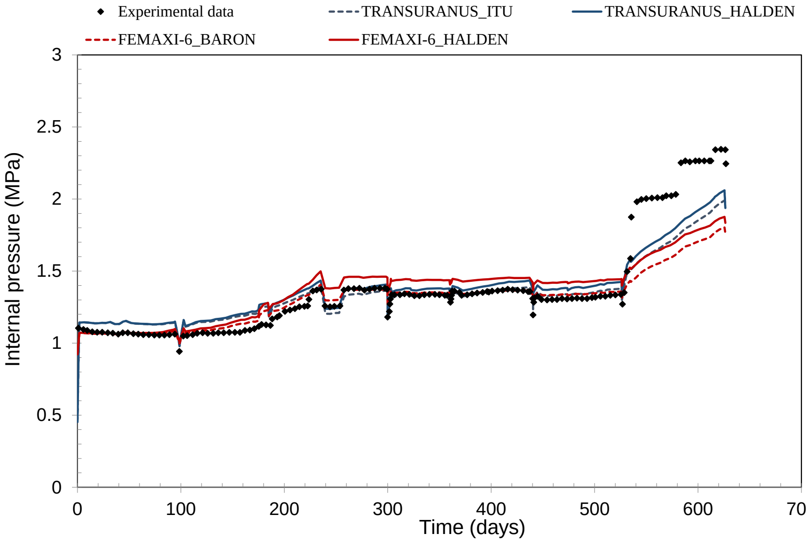


Figure 5. Measured and calculated internal pressure for ROD2

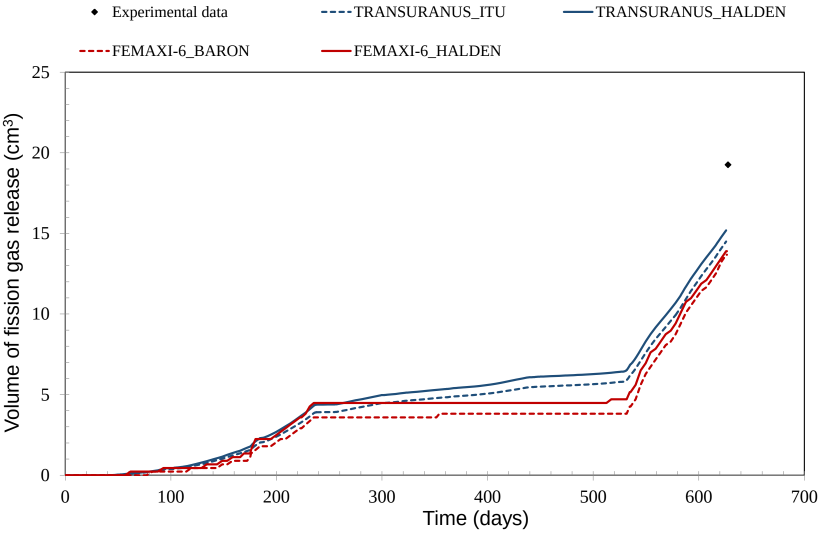


Figure 6. Measured and calculated volume of fission gas release for ROD1

Deformations of fuel and cladding determine gap dimension, parameter for which its evolution is depicted in Fig. 8 and Fig. 9. The values for pellet-clad diameter gap obtained with FEMAXI-6 are higher then those obtained with TRANSURANUS, but no contact between fuel and clad was observed, the gap remained open for all the irradiation period.

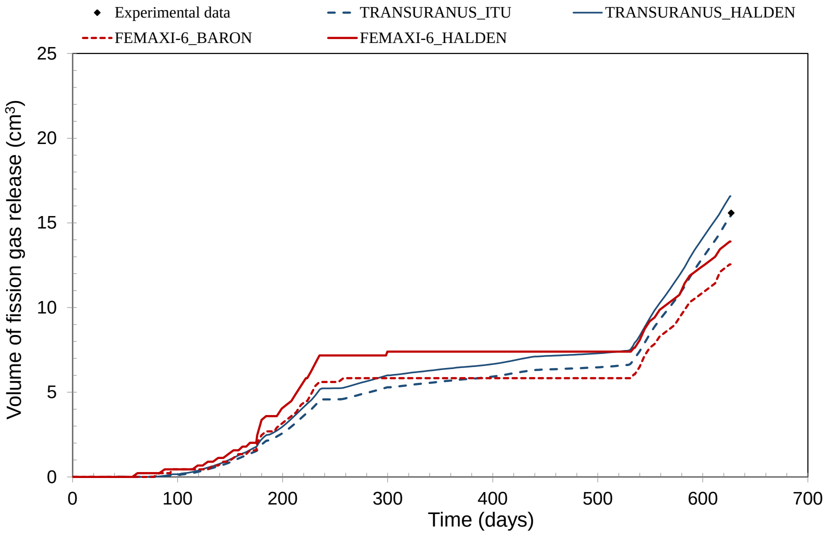


Figure 7. Measured and calculated volume of fission gas release for ROD2

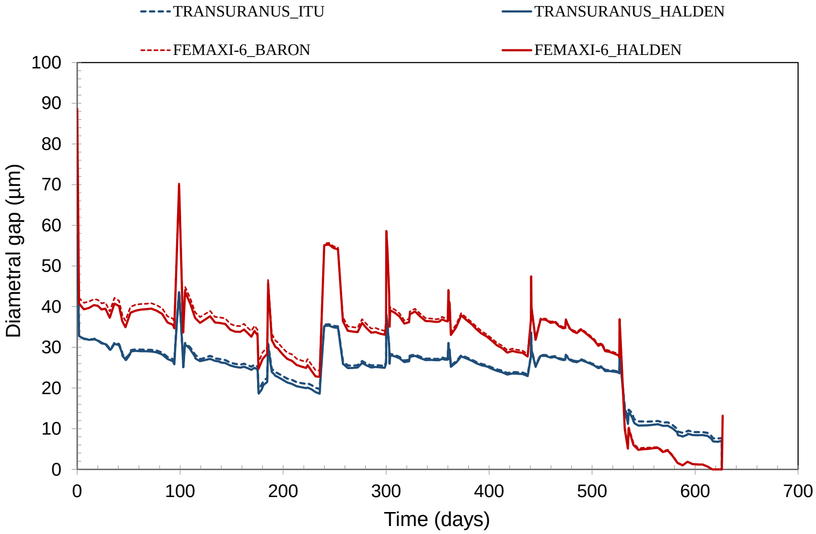


Figure 8. Calculated pellet – clad diameter gap for ROD1

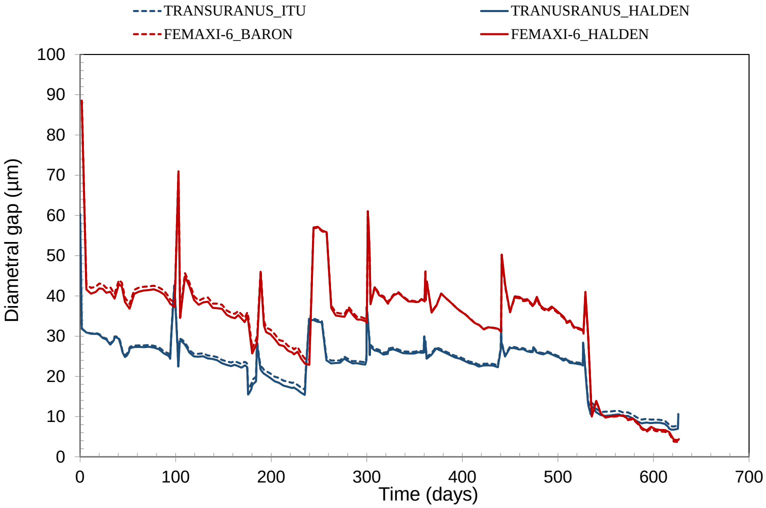


Figure 9. Calculated pellet – clad diameter gap for ROD2

5. Conclusions

In this study, we evaluated FEMAXI-6 and TRANSURANUS capability to predict MOX fuel behaviour under normal operating conditions. In such analyses, a peculiar attention should be given to some models and parameters, such as fuel thermal conductivity, fuel relocation, fuel swelling, fission gas release and gap thermal conductance. In order to reasonably predict the MOX fuel behaviour, a proper selection of specific models is required. The comparison between TRANSURANUS and FEMAXI-6 codes results with experimental data proved that the applicability of the selected models in predicting MOX fuel behaviour is appropriate. Future perspectives include predictions of fuel rods behaviour proposed for irradiation in liquid metal fast reactors.

REFERENCES

[1]. M. Suzuki, H. Saitou, Light Water Reactor Fuel Analysis Code FEMAXI-6 (Ver. 1), JAEA-Data/Code 2005-003, Japan Atomic Energy Agency (JAEA) (2005).

[2]. K. Lassmann, J. van de Laar, TRANSURANUS Handbook, V1M2J15, European Commission, Joint Research Centre, Institute for Transuranium Elements, Karlsruhe, Germany, 2015

[3]. Terje Tverberg, Mixed-oxide (MOX) Fuel Performance Benchmark, Summary of the Results for the Halden Reactor Project MOX Rods, OECD Halden Reactor Project.

[4]. Akifumi YAMJI , Motoe SUZUKI & Tsutomu OKUBO, FEMAXI-6 Code Verification with MOX Fuels Irradiated in Halden Reactor, Journal of Nuclear Science and Technology, Vol 26 No.12, p 1152 -1161 (2009).

[5]. R. Roman, D.V. Ionescu, Gh. Olteanu, D.F. Oproiu, Study of Thermal Conductivity Influence on MOX Fuel Behaviour, The 10th Annual International Conference on Sustainable Development through Nuclear Research and Education May 24-26, Pitesti, Romania.