ASSESSMENT OF HIGH CONDUCTIVITY CERAMIC FUEL CONCEPT UNDER NORMAL AND ACCIDENT CONDITIONS

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Abstract

After the Fukushima Daiichi accident, the high conductivity ceramic concept fuel has been revisited. The thermal conductivity of uranium dioxide used as nuclear fuel is relatively low, as consequence fuel pellet centerline reaches high temperatures, high fission gas release rate, increase of fuel rod internal pressure reducing the safety thermal margin. Several investigations had been conducted in framework of ATF (Accident Tolerant Fuel) using different additives in ceramic fuel (UO_2) in order to enhance thermal conductivity in uranium dioxide pellets. The increase of the thermal conductivity of fuel can reduce the pellet centerline temperature, consequently less fission gas releasing rate and the low risk of fuel melting, hence improving significantly fuel performance under accident conditions. The beryllium oxide (BeO) has high conductivity among other ceramics and is quite compatible with UO_2 up to 2200°C, at which temperature it forms a eutectic. Moreover, it is compatible with zircaloy cladding, does not react with water, has a good neutronic characteristics (low neutron absorption cross-section, neutron moderation). This work presents a preliminary assessment of high conductivity ceramic concept fuel considering UO_2-BeO mixed oxide fuel containing 10 wt% of BeO. The FRAPCON and FRAPTRAN fuel performance codes were conveniently adapted to support the evaluation of UO_2-BeO mixed oxide fuel. The thermal and mechanical properties were modified in the codes for a proper and representative simulation of the fuel performance. Theobtainedpreliminary results show lower fuel centerline temperatureswhen compared to standard UO_2 fuel, consequently promoting enhancement of safety margins during the operational condition and under LOCA accident scenario.

1. INTRODUCTION

The uranium dioxide has been used as nuclear fuel for pressurized and boiling water reactors (PWR and BWR) for a long time with an outstanding performance. Nonetheless, the research for accident tolerant fuels (ATF) as a consequence of the Fukushima Daiichi accident brought the necessity to improve the performance of nuclear fuels aiming to develop fuels which present enhanced accident tolerance in comparison with the standard and existing UO_2/zircaloy system widely used by the nuclear industry [1]. Nowadays, a significant effort has been conducted in the cladding material research and investigation in order to accomplish the ATF criteria such as corrosion resistance, good mechanical properties, lower hydrogen generation etc. Additionally, the performance can be improved considering the fuel pellet itself, specially enhancing the thermal conductivity and reducing fission gas release.

The low thermal conductivity of uranium dioxide used as nuclear fuel in PWR induces a high fuel pellet centerline temperature, stores energy consequently reducing the safety thermal margins during steady state irradiation, transients, and under accident scenarios, as for instance loss-of-coolant accident (LOCA). Due to the poor thermal conductivity of UO_2, the high fuel centerline temperature in the pellet promotes different phenomena such as high fission gas release, steep temperature gradient resulting in high thermal stress, increase of the fuel pellet swelling, plastic deformation, cracking, anticipation of pellet cladding mechanical interaction occurrence, and consequently reduction of operational safety limit. In this sense, the increase of the uranium dioxide thermal conductivity by means of the use of specific additives to the fuel matrix can reduce the fuel centerline temperature, consequently the amount of fission gas release would be lower, less thermal stress, less swelling and deformation allowing high burnup and enabling improved safety due to less stored energy.

In the last decades several investigations have been conducted in order to enhance the thermal conductivity of the fuel pellet and one of the promising techniques is associated to doping the UO_2 pellet with high conductivity additive. The beryllium oxide (BeO) have shown as good candidate due to its high thermal conductivity among oxides, high melting point, low neutron thermal absorption cross section, easily fabricated without significant impacts in the conventional manufacturing process with an acceptable fuel cost. Additionally, to enhancing thermal conductivity, beryllium oxide presents a large stability and chemical compatibility with uranium at high temperatures as well as some neutron moderation capacity.

In order to perform a preliminary assessment of the effect of the BeO addition to enhance the nuclear fuel behaviour, it was necessary to include the relevant properties, mainly the thermal conductivity into the FRAPCON and FRAPTRAN fuel performance codes.

2. METHODOLOGY

2.1. Modelling Thermal and Mechanical Properties of UO_2 with BeO addition

In the present paper, initially the necessary materials properties to be changed were identified considering the MATPRO (Material Properties Data Library) existing materials properties for fuel pellet correlated directly to temperature and a preliminary literature survey was conducted in order to assess the existing information about Be (beryllium) and BeO (beryllium oxide), especially those related to thermal properties. Moreover, information regarding fabrication process in order to verify the feasible amount of beryllium oxide to be mixed to the UO_2 matrix [2–4]. As begin of assessment it was defined the enhanced fuel pellet with following composition: 10% of BeO and 90% of UO_2 in volume fractions, giving almost 3.10% of BeO in weight percent (wt%).

In order to define the new material properties correlations to be applied in the modified versions of the codes, the first step consists in stablish the weight fraction of each fuel pellet component contribution, according to following expressions [5]:

$$\omega_1 = \frac{v_1 \cdot \rho_1}{v_1 \cdot \rho_1 + v_2 \cdot \rho_2} \tag{1}$$

$$\omega_2 = \frac{v_2 \cdot \rho_2}{v_1 \cdot \rho_1 + v_2 \cdot \rho_2} \tag{2}$$

where $\boldsymbol{v_i}$ are the volume fractions, ρ_i are the densities and ω_i are the mass fractions of the two components.

2.1.1. *Thermal Conductivity UO_2-BeO*

The thermal conductivity of UO_2 pellet usually is a function of burnup, temperature, porosity, theoretical density, and radiation damages due to change in the crystalline network or irradiation defects. The thermal conductivity of UO_2, 8.4 $W \cdot m^{-1} \cdot K^{-1}$, is significantly lower than that of BeO, 260 $W \cdot m^{-1} \cdot K^{-1}$, at room temperature. Two type of correlations were considered before implementation in the FRAPCON code, the analytical model proposed by D. Chandramouliand S. T. Revankar [7–9] and experimental correlations [10, 11] presented by HALDEN (Norway) and Nippon Nuclear Fuel Development Co., Ltd (Japan). The thermal conductivity of the UO_2-BeO proposed in this work is a fitting curve of existing data considering a given BeO composition (10%) in the fuel, this fitting curve became function only of temperature according to following expression:

$$K_{UO2-B} \quad = 3348\, T^{-0.928} \tag{3}$$

where K_{UO2-B} is the UO_2-BeO fuel pellet thermal conductivity in $W \cdot m^{-1} \cdot K^{-1}$, and T is the temperature in Kelvin unit, the function was implemented in subroutine FTHCON of FRAPCON code.

2.1.2. *Specific Heat Capacity*

The heat capacity of UO_2-BeO fuel pellet is given by the straightforward result of weight fraction applied to both oxides specific heat capacities UO_2 [13] and BeO [9, 12]:

$$Cp_{UO_2-BeO} = \omega_{UO_2} \times Cp_{UO_2} + \omega_{BeO} \times Cp_{BeO} \tag{4}$$

where:

$$Cp_{BeO} = 0.036\left(\frac{T-650}{360}\right)^3 - 0.12\left(\frac{T-650}{360}\right)^2 + 0.2\left(\frac{T-650}{360}\right) + 1.9 \tag{5}$$

Cp_{UO_2} was considered in equation 2.2-1 from [13].

The function above was implemented in subroutine FCP of FRAPCON code.

2.1.3. *Enthalpy*

The enthalpy of UO_2-BeO fuel pellet is given by the straightforward result of weight fraction applied to both oxides enthalpies as presented below [7] and the function was implemented in subroutine FENTHL:

$$H_{UO_2-BeO} = \omega_{UO_2} \times H_{UO_2} + \omega_{BeO} \times H_{Be} \tag{6}$$

$$H_{BeO} = 11.1084 + 7.1245.10^{-4}T^2 + \frac{840705}{T} - \frac{53124500}{T^2} - 5453.21 \tag{7}$$

H_{UO_2} was considered in equation 2.2-2 from [13].

2.1.4. *Melting Point and Heat of Fusion*

The melting temperature of UO_2 pellet is 3113 K and its heat of fusion is 2.74×10^5 J/kg [6, 15]; for UO_2-BeO fuel pellet, these values are 3104 K and 3.19×10^5 J/kg [7], respectively.

2.1.5. *Thermal Expansion*

Based on the weight fraction presented above, the thermal expansion of the UO_2-BeO fuel pellet is given by the following straightforward expression [7]:

$$\frac{\Delta L}{L}_{UO_2-BeO} = \omega_{UO_2} \times \frac{\Delta L}{L}_{UO_2} + \omega_{BeO} \times \frac{\Delta L}{L}_{BeO} \tag{8}$$

The function was implemented in subroutine FTHEXP of FRAPCON code.

2.1.6. *Mechanical Properties*

The main mechanical properties for the UO_2-BeO fuel pellet are obtained using volume fraction of each component to calculate following properties: Young modulus, Shear modulus, Poisson coefficient, and strength, as presented below [14]:

$$\rho_{UO_2-BeO} = \omega_{UO_2} \times \rho_{UO_2} + \omega_{BeO} \times \rho_{BeO} \tag{9}$$

$$v_{UO_2-BeO} = \omega_{UO_2-BeO} \times \rho_{UO_2-BeO} \tag{10}$$

Then, the mechanical properties for the UO_2-BeO fuel pellet are given by the following straightforward expressions:

$$Y_{UO_2-BeO} = v_{UO_2} \times Y_{UO_2} + v_{BeO} \times Y_{BeO} \tag{11}$$

$$Shear_{UO_2-BeO} = v_{UO_2} \times Shear_{UO_2} + v_{BeO} \times Shear_{BeO} \tag{12}$$

$$Poisson_{UO_2-BeO} = v_{UO_2} \times Poisson_{UO_2} + v_{BeO} \times Poisson_{BeO} \tag{13}$$

2.2. Fuel Performance Codes

The well-known FRAPCON and FRAPTRAN [15] codes, sponsored by the United States Nuclear Regulatory Commission (U.S. NRC) for the licensing of PWR and BWR nuclear power plants, were conveniently modified to support the evaluation of BeO/UO_2 mixed oxide fuel containing 10 wt% of beryllium oxide. The FRAPCON and FRAPTRAN codes have a material data packagecompilation, namely MATPRO [6], which contains material properties, theoretical and experimental models and correlations considered in the simulation of the fuel performance. The implemented modifications mainly addressed the role of the BeO to enhance the thermal conductivity of the fuel pellet. The modifications of the previously identified subroutines related to UO_2-BeO fuel pellet properties in the fuel performance codes were carried out considering step by step manner in order to assess the contribution of each property to the final fuel global performance. The first modified subroutine was related to the thermal conductivity property, after following by the specific heat, the enthalpy, and finally, the mechanical properties. For each property modification, new versions of the codes were generated and tests were carefully performed. The verification of implemented modification was conducted by means of LOCA experiment (IFA 650-5) performed in HALDEN. It is worthwhile to mention that similar work [7] exist but consider only

FRAPTRAN code modification for UO_2-BeO as fuel in the LOCA condition, this work considers modification on both codes: FRAPCON and FRAPTRAN to consider the LOCA analysis.

2.3. Fuel Rod Analysis (Test Case: IFA 650-5)

The modified version of the codes (FRAPCON and FRAPTRAN) was evaluated using as test case, the data available in the open literature related to the experiment IFA 650-5 performed in the framework of Halden Reactor Project [16] to study the behaviour of UO_2/zircaloy fuel rod under LOCA scenario.

The IFA 650-5 test fuel rod was re-fabricated from an irradiated PWR UO_2/Zircaloy-4 fuel rod. The fuel had a high average burnup of 83 MWd/kgU. The base irradiation of the full-length rod comprised 6 reactor cycles corresponding to about 2000 effective full power days. The properties of the IFA-650.5 fuel rod are summarized in Table 1 below. Initially, steady state condition was simulated using FRAPCON code for UO_2 and UO_2-BeO.

TABLE 1. FUEL ROD PROPERTIES OF IFA 650-5 TEST FUEL ROD

Fuel type	PWR
Fuel material	UO_2
Fuel pellet diameter (mm)	9.132
Fuel pellet length (mm)	11
Fuel dish depth (mm)	0.28
Fuel dish width (mm)	1.2
Fuel density (% TD)	94.8
Fuel enrichment (w/o %)	3.5
Cladding material	DX ELS0.8b
Cladding outer diameter (mm)	10.735
Cladding wall thickness (mm)	0.721
Fuel rod burnup (MWd/kgU)	83
Fuel rod total length (mm)	480
Fuel rod gap (mm)	0.0805
Fuel Rod plenum volume (cm^3)	15
Fuel rod fill gas	90% Ar +10%He
Fill pressure (MPa)	4.0

2.4. Results and Discussion

The Figs 1, 2 and 3 present the evolution of the fuel centerline temperature, the internal pressure, and the fission gas release as function of burnup for IFA 650-5 considering UO_2 fuel pellet as reference and enhanced fuel UO_2-BeO pellet. For UO_2-BeO fuel pellet, the results are shown according to the modified properties into correspondent subroutines.

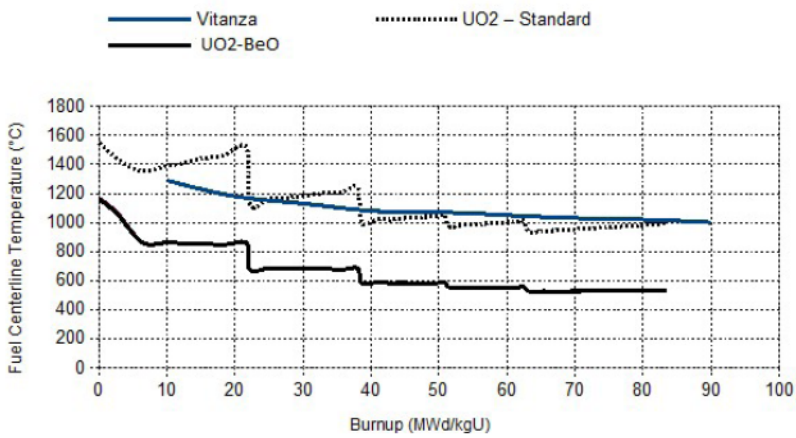

FIG. 1. Fuel centerline temperature for IFA 650-5 as function of burnup for reference UO_2 fuel pellet and UO_2-BeO fuel pellet using FRAPCON codes (original and modified version) for steady state condition (burnup accumulation).

Figure 1 shows that the fuel centerline temperature is about 500°C lower for UO_2-BeO fuel pellet compared to the reference UO_2 fuel pellet. The figure also shows that the governing property in the fuel pellet behaviour is mainly due to the thermal conductivity; the modification of the other subroutines related to other properties (specific enthalpy, specific heat and mechanical) does not promote significantly changes in the fuel pellet centerline temperature. The result confirms that thermal conductivity plays a very important role in the fuel temperature profile.

The Figure 1 additionally presents the Vitanza threshold [17] curve which is associated to fission gas release rate due to the fuel centerline temperature. As it can be seen from the Vitanza curve, the fission gas release for UO_2-BeO fuel pellet do not exceed the threshold, consequently the dominant phenomena governing the fission gas release process will be athermal up to approximatelly 40 MWd/kgU.

The evolution of the internal fuel rod pressure as function of burnup for IFA 650-5 presented in Figure 2 show that the internal pressure for the fuel rod with UO_2-BeO fuel pellet is lower than that observed for the reference UO_2 fuel pellet during all the irradiation period, even considering the high burnup reached during the steady state irradiation of the fuel rod.

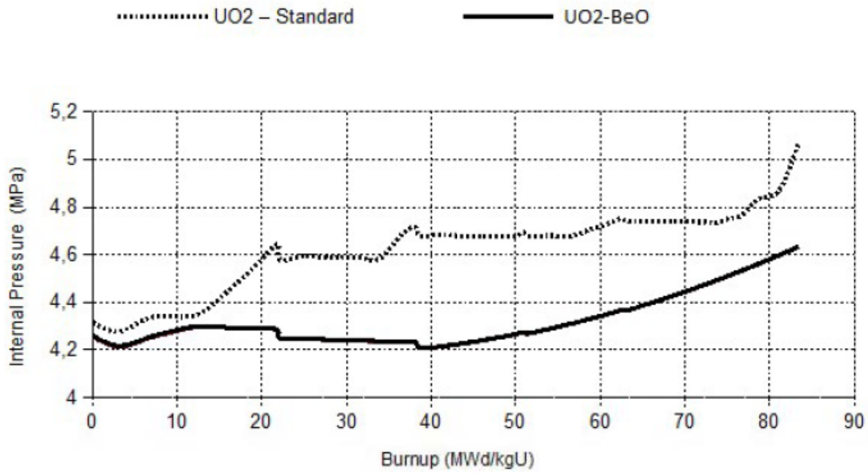

FIG. 2. Internal fuel rod pressure for IFA 650-5 as function of burnup for reference UO_2 fuel pellet and UO_2-BeO fuel pellet using FRAPCON codes (original and modified version) for steady state condition (burnup accumulation).

The evolution of the fission gas release as function of burnup presented in Figure 3 show that the amount of fission gas released by UO_2-BeO fuel pellet is significantly lower than that of the reference UO_2 fuel pellet. Then, the lower temperatures experienced by the UO_2-BeO fuel pellet during the entire irradiation period enable the fuel centerline temperature not to exceed the Vitanza threshold.

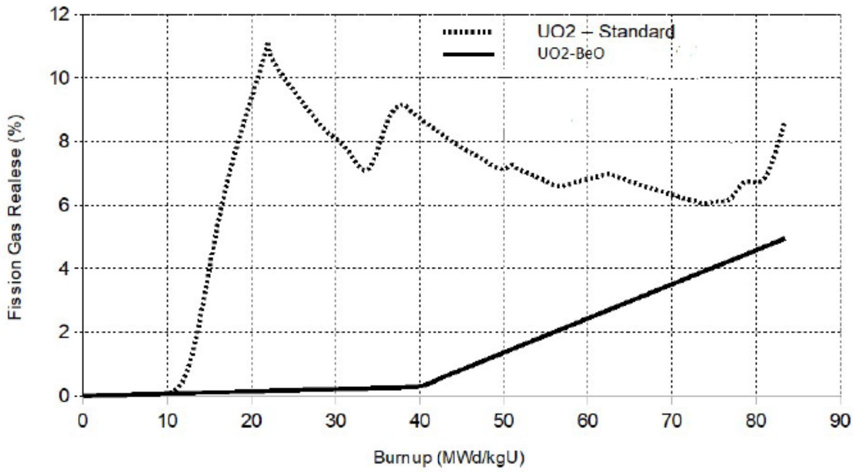

FIG. 3. Fission gas release for IFA 650-5 as function of burnup for reference UO_2 fuel pellet and UO_2-BeO fuel pellet using FRAPCON codes (original and modified versions) for steady state condition (burnup accumulation).

Figure 4 shows the evolution of the internal fuel rod pressure as function of time during the LOCA transient. Initially, this assessment was performed using coupled simulation (FRAPCON and FRAPTRAN), the original version of FRAPCON and FRAPTRAN codes for the reference UO_2 fuel pellet and, for UO_2-BeO, the modified versions of FRAPCON and FRAPTRAN codes taking to account new properties implementation. Moreover, the steady state condition simulated using modified FRAPCON and original version of FRAPTRAN in order to verify the consistency of FRAPTRAN modification. The curves in the figure show a slight increase of the cladding rupture time for the UO_2-BeO fuel pellet. The possible reason for a slight improvement shall be associated to temperature boundary condition considered in the FRAPTRAN input, which is conservative assumption considering the previous results (Figure 1) of fuel temperature. Again, it is worthwhile note the importance of results presented by modified FRAPCON compared to the results obtained from modified FRAPTRAN in order to evaluate the effects of the BeO addition in the global fuel performance during steady state irradiation prior to LOCA scenario.

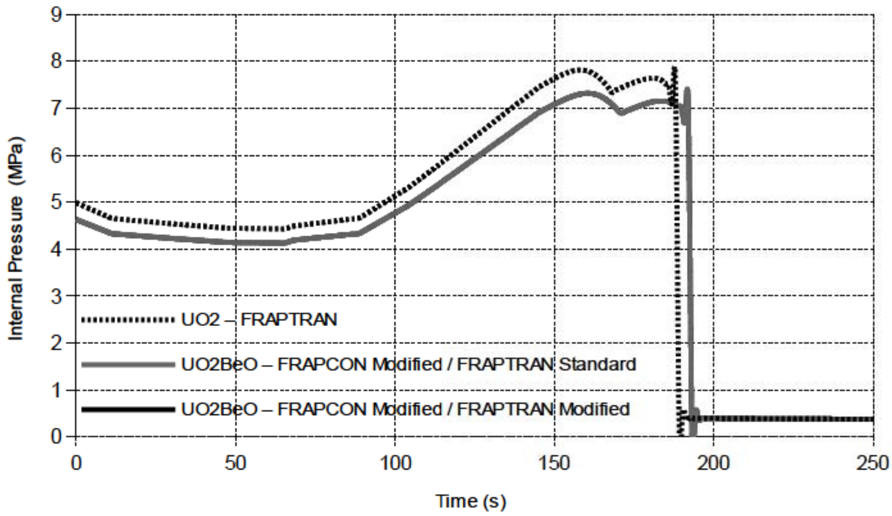

FIG. 4. Internal fuel rod pressure for IFA 650-5 as function of time for reference UO_2 fuel pellet and UO_2-BeO fuel pellet using FRAPCON-FRAPTRAN codes and the modified versions for LOCA condition.

3. CONCLUSION

All results obtained using modified version of FRAPCON code show an improvement of the parameters directly associated to safety margins, specially the fuel centerline temperature reduction, consequently reduced thermal gradient inside fuel pellet, small amount of fission gas release, lower internal pressure, all effects resulting in the improvement of safety margins during the steady state operational condition. The results of modified FRAPTRAN shown a slight improvement for burst time during the LOCA accident scenario. Some improvement expected for UO_2-BeO fuel during the LOCA was not clearly achieved mainly due to the assumption of same fuel cladding temperature as boundary condition adopted for both simulations at beginning of transient. Moreover, the future work can address the appropriate cladding temperature profile and the importance of burnup associated to the thermal conductivity degradation and some evaluation of neutronic penalty due to reduction of uranium loading in the UO_2-BeO fuel pellet.

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REFERENCES

[1] GOLDNER, F., Development Strategy for Advanced LWR Fuels with Enhanced Accident Tolerance, http://energy.gov/sites/prod/files/061212%20Goldner%20%20NEAC%20Presentation%20(FINAL).pdf (2012).

[2] NISHIGAKI, S., MAEKAWA, K., Fabrication of BeO-UO_2 Fuel Pellet, Journal of Nuclear Materials, **14** (1964) 453-458.

[3] BEAVER, W.W., THEODORE, J.G., BIELAWSKI, C.A., Effects of Powder Characteristics, Additives and Atmosphere on the Sintering of Sulfate-Derived BeO, Journal of Nuclear Materials **14** (1964) 326.

[4] LI, D.S., GARMESTANI, H., SCHWARTZ, J., Modeling Thermal Conductivity in UO_2 with BeO Additions as a Function of Microstructure, Journal of Nuclear Materials **392** (2009) 22.

[5] SMITH, R. J., Enhanced Thermal Conductivity UO_2-BeO Nuclear Fuel: Neutronic Performance Studies and Economic Analyses, Master Thesis. Texas (2012).

[6] ALLISON, C.M., et alli., SCDAP/RELAP5/MOD3.1 Code Manual Volume IV: MATPRO - A Library of Materials Properties for Light-Water-Reactor Accident Analysis, NUREG/CR-6150, EGG-2720 Washington (1993).

[7] CHANDRAMOULI, D., REVANKAR, S.T., Development of Thermal Models and Analysis of UO_2-BeO Fuel during a Loss of Coolant Accident, International Journal of Nuclear Energy (2014).

[8] ZHOU, W., LIU, R., REVANKAR, S.T., UO_2-BeO Composite Fuel Thermal Property and Performance Modeling, Journal of Energy and Power Engineering **8** (2014) 7.

[9] CHANDRAMOULI, D., REVANKAR, S.T., Study of Performances of UO_2-BeO Composted Fuel Using FRAPTRAN, Purdue University School of Nuclear Engineering Report PU/NE-12-16, Purdue University School of Nuclear Engineering (2012).

[10] ISHIMOTO, S., HIRAI, M., ITO, K., KOREI, Y., Thermal Conductivity of UO_2-BeO Pellet, Journal of Nuclear Science and Technology **33** 2 (1996) 134–140.

[11] MCGRATH, M.A., et. alli., In-reactor Investigation of the composite UO_2-BeO fuel: background, results and perspectives, The Nuclear Materials Conference, Montpellier, France, (2016).

[12] DOUGLAS, A.C., Thermodynamic Properties of Magnesium Oxide and Beryllium Oxide from 298 to 1200 K, Journal of Research of the National Bureau of Standards A. Physics and Chemistry **67A** 4 (1963) 325–329.

[13] LUSCHER, W.G., GEELHOOD, K.J., Material Properties Correlation: Comparison Between FRAPCON 3.4, FRAPTRAN 1.4 and MATPRO, NUREG/CR-7024, (2010).

[14] ZHOU, W., and ZHOU, W., Thermophysical and Mechanical Analyses of UO_2-36.4vol % BeO Fuel Pellets with Zircaloy, SiC, and FeCrAl Claddings. Metals **8** (2018) 65.

[15] GEELHOOD, K.J., LUSCHER, W.G., BEYER, C.E., FLANAGAN, M.E., FRAPCON-3.4: A Computer Code for the Calculation of Steady state Thermal mechanical Behavior of Oxide Fuel Rods for High Burnup, U.S.NRC, NUREG/CR-7022, Washington (2011).

[16] MANNGARD, T., STENGARD, J., Evaluation of the Halden IFA-650 Loss-of- Coolant Accident Experiments 5, 6 and 7, Swedish Radiation Safety Authority, Sweden (2014).

[17] VITANZA, C., KOLSTAD, E., GRAZIANI, V., Analysis of Fission Gas Release Tests to High Burnup, Proc. ANS Topical Meeting on Light Water Reactor Fuel Performance, Portland, Oregon (1979).