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STUDY OF THORIA–URANIA FUEL DURING ACCIDENTS

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Abstract. *In this investigation, nuclear fuel based on mixed ceramic oxides, using (Th–U)O_2 as nuclear fuel and zirconium-based alloy as cladding, was simulated. This strategic configuration can achieve improved safety margins because of an enhanced set of thermal and mechanical properties. The Experimental Breeder Reactor built in the 1950s in Idaho introduced the concept that a reactor can generate more fissile material than it consumes. The thorium fuels have a lower cost and should decrease weapon-grade plutonium compared with conventional fuel, UO_2. The nuclear characteristic of thorium-232 or U-238 can make a converter into U-233 or Pu-239. However, using thoria fuels can avoid weapon proliferation by reducing plutonium, and it also should reduce radionuclides such as (Np, Am, Cm). This study uses an optimized composition of Th-75% wt and U-25% wt with an enrichment of 19.5%. We studied the behavior using the fuel licensing codes FRAPCON and FRAPTRAN, including many adaptations for the mixed composition choice. The results prove that thoria–urania fuel has a higher performance than pure uranium dioxide fuel during accidents.*

Keywords: *Thorium, Uranium, FRAPCON, FRAPTRAN, Loss-of-coolant accident*

1. INTRODUCTION

Thorium-based fuels are essential to supply nuclear power reactor fleets, which could run over 200 years at the current rates of consumption because of limited global uranium reserves. The advanced reactors planned for 2030 contain designs that wholly endorsed thoria-based fuels, such as the seed-and-blanket and sodium-cooled fast reactor. In recent years, research showed how thorium could be used in advanced reactor concepts as the gas-cooled reactors with designs based on thorium–plutonium carbide (Th–Pu)C fuel (Zhang et al., 2018). The primary aim of this research focus to verify the behavior of the fuel (Th–U)O_2 quantitatively and compare it with UO_2/Zr alloys, regarding the performance and aspects of safety analysis. The aim is a pattern using licensing code adapted to composition (Th-75%, U-25%)O_2 with an enrichment of 19.5% of U-235 and zircaloy as the coating.

Around 1950 years, when started the reviews of the thorium fuel cycle, predicting the future to replace UO_2. The first core reactor proposed was BORAX-IV, and an experiment, based on (Th–U)O_2 fuel, also from 1963 to 1968 used the thorium-based fuel used for the Elke River (Minnesota) Boiling Water Reactor, (Kazimi et al., 1999). The shipping port Light Water Breeder Reactor conducted tests that used (Th–U)O_2 between August 1977 and October 1982. Despite the full range of designs supporting (Th–U)O_2 as fuel applied to light-water reactors (LWRs), a fuel core based on a uniform lattice is workable and acceptable

1.1 Thoria-based fuel cycle

Presuming a normal irradiation cycle, thoria–urania (Th–U)O_2 fuel exhibits better safety margins, such as lower fuel temperature, lower gas pressure, and high-strength resistance compared with UO_2. Also, thorium-based fuels seem a lower release of fission gas because of lower diffusion rates joined with reduced fuel temperatures. The advantages offered by breeding capacity, for both thermal and fast reactors able to convert fertile to fissile material, also show a lower cost aggregating a self-sustaining system (David et al., 2007). Further, the (Th–U)O_2 fuel irradiated with thermal neutron spectral needs a reduced amount of U-233 (Gupta et al., 2008), on the order of 1 metric ton, compared with UO_2.

Thoria-based mixed oxides use Th–232 as a fertile isotope loaded into the core reactor. The nuclei of Th–232 absorb a neutron and transmute into the nucleus of Th–233 (Yang et al., 2016). The isotope Th–233 decays (negative beta decay) to Pa-233. Then, the protactinium decays (negative beta decay) to U-233, which is a perfect fissile material. Therefore, Th–233 has radioactive decay of 22 minutes and Pa-233 a half-life of 27 days. In the thermal spectrum, the breeder ratio of U-233 is 2.30 and of U-235 is 2.07, when bombarded by thermal neutrons.

1.2 Fuel performance code

The U.S. Nuclear Regulation Commission (NRC) supervises the licensing process of nuclear power units. Also, the NRC promulgates all safety rules applied to normal and abnormal scenarios. Then, the enrichment limit acceptable to fissile 235-U must be less than 20 wt% for LWRs. It carried the simulation of the composed fuel using licensing codes after changes to support the thorium dioxide. The computer code used for the calculation of steady-state thermal-mechanical behavior of oxide fuel rods for high burnup was FRAPCON (Geelhood et al., 2015). The FRAPCON developed for the NRC by Pacific Northwest National Laboratory is licensing codes that can simulate the steady-state fuel response. Also, many research groups used the Fuel Rod Analysis Program Transient (FRAPTRAN) used to simulate the loss-of-coolant accident (LOCA) scenarios (Luscher et al., 2015).

1.3 Fuel simulation under transient scenarios

The fuel rod chosen comprises a case documented used in pressurized water reactors (PWRs) used to search for LOCA experiments, IFA-650.5 (Oberlander et al., 2014). Initially, it shows an analytical overview of the test related to the Halden Reactor Program (HPR) in southeastern Sweden. HPR experiments created a sizable database based on the fuel response, also analyzing cladding failures and fragmentation of fuel. IFA-650.5 experiments involved the studied, of ballooning, burst, fuel fragmentations, fission gas release (FGR), and fuel rod failure, that occurred during LOCA scenarios performed at the Halden (Bianco et al., 2015).

IFA-650.5 showed that, during LOCA, temperatures of over 865 °C enable the phase transition of zircaloy-4. At temperatures between 850 °C and 950 °C, the mixed α-β phases of zircaloy occurred. In this region, $\alpha+\beta$ phases had the change of oxidation kinetics laws of cubic to parabolic, with a sensible breakaway effect. During regular operation, all fuel pellets can suffer cracking because of the heat stored in the ceramic fuel. When the linear heat rates reach a power of approximately 5 (kW/m), the fracture stress of the ceramic pellets begins, and following the cracking, there is fuel relocation. Several effects can create cracks and displacement, such as thermal expansion, diffusion of fission gaseous products, and fuel swelling.

2. MATERIALS AND METHODS

The goal of the fuel simulation described is to inspect the possibility of replacing the standard fuel UO_2 for thorium-uranium mixed oxide fuel applied to a PWR. During the four last decades, there are several studies for thorium mixed uranium and plutonium, but the number of research addressing physical properties is still limited. Therefore, researchers have shown that the $(Th–U)O_2$ can provide a novel fuel concept that has a breeder capacity, also showing a significant set of better physical properties than UO_2. Table 1 summarizes a few physical properties of the actinide dioxides and cladding materials (Long et al., 2004).

Table 1. Physical properties of the standard fuels UO_2, PuO_2, ThO_2, and cladding materials, Zircaloy-4 and Kanthal

Physical properties	UO_2	ThO_2	PuO_2	Zircaloy-4	Kanthal APMT
Melting point (°C)	2846.85±30	3300±100	2427.85±35	1850±10	1500±10
Density (g/cm^3) at 25 ºC	10.980±20	10.000±80	11.460±20	6.56±0.50	7.20±0.50
Elastic modulus (GPa) at 25 ºC	202±10	239±11	257±15	99±6.5	210±10
Heat capacity (J/kg-K) at 25 ºC	235±46	202±40	270±54	285±10	480±10

Nuclear fuels are materials that exhibit fission chain reactions that are sustainable and act as sources of nuclear energy. Civilian nuclear units are in operation today relying on uranium dioxide or plutonium-uranium mixed fuel based on uranium-235 as a principal fissile isotope. However, thorium-based fuels can breed fissile uranium-233, which could produce thorium-uranium mixed oxide to replace fuel to pressurized water reactors. There are many types of reactors and newer prototypes that can use thoria as fuel. Thorium-based fuels used can vary from reactors: generation IV reactors, such as the gas-cooled reactors, very-high-temperature reactors, and molten-salt reactors. Table 2 displays the fuel composition proposed, showing that the amount of fissile material is 4.31% of the total weight mass.

Table 2. Isotopic composition of mixed oxide ThO_2-UO_2 fuel with 19.5% of enrichment U-235

Materials	Isotopic vector (%)	Fission cross-section (barns)	Weight percent (%)
Th-230	0.02	9.494 m	0.01
Th-232	99.98	53.71 µ	65.48
U-234	0.005	67.02	0.0011
U-235	19.5	585.1	4.31
U-238	80.99	16.80 m	18.34
O-16	-	-	12.05

The fuel composition can use a large variety of fuel options, such as molten thorium fluoride salts or stable thorium dioxide. For ThO_2 showing a molar mass of 259 (g/mol) and UO_2 of 270 (g/mol), the composition of mixed oxide depends on isotopic vectors or true incidence. The weight mass of composition used is: (Th 75% – U 25%)O_2 of 265.71 (g/mol) and a density of 10 Kg/m^3. Fuel fragmentation followed by particle relocation are consequences of the lower thermal conductivity of uranium dioxide, which must produce a high thermal gradient during normal operating conditions. The temperature suffers a considerable reduction, considering that the fuel centerline temperature to the edge of the pellet leads to high internal stresses for linear thermal expansion. The proposed IFA-650.5 test must identify behavior under transient conditions coupled to the mechanism of fuel relocation based on the UO_2 fuel, applied to a PWR (16 × 16) configuration. The fuel rod IFA-650.5 was implemented at the Halden reactor in 2007, using fuel enrichment of 3.5%, as a double layer of Zircaloy-4 cladding. The test IFA-650.5 showed a burn cycle with six cycles of irradiation totaling 1994 days and 83 GWd/MTU. The linear power rates used during the cycles were 37.5, 28.0, 22.0, 20.0, 18.0, and 18.0 (kW/m). After irradiation, it divided the fuel rod into 480-mm segments to simulate the LOCA conditions using an appropriate apparatus. Table 3 compiled the key properties of the fuel rod used for simulations.

Table 3. Fuel design properties of IFA-650.5 case using UO_2 and (Th. 75%-U 25%)O_2

Fuel characteristics as fabricated	UO_2	(Th–U)O_2
Pellet diameter (mm)	9.132	9.132
Pellet length (mm)	11	11
Dish depth (mm)	0.28	0.28
Land width (mm)	1.2	1.2
Density (UO_2)% of TD	94.8	94.8
U-235 enrichment in UO_2	3.5	19.5
Cladding outer diameter (mm)	10.735	10.735
Cladding wall thickness (mm)	0.721	0.721
Radial pellet-clad gap (mm)	0.0805	0.0805
Fuel rod active length	480	480
Plenum volume (cm^3)	15	15
Fill gas pressure (MPa)	4	4

2.1 Thermal physical properties of (Th–U)O_2

The thermal properties of composite fuel are intermediates between the pure contents, supposing that the fabrication routes must produce a homogeneous system efficiently. The calculation of physical properties used the empirical rule of mixtures or Vegard's law, which assumes that both components may have a similar crystal structure. Vegard's law is a practical rule that holds a linear relationship between the concentrations of the constituent elements and alloy properties.

In this work, it used the Neumann–Kopp rule for the approximation of the heat capacity of mixed oxides. The Neumann–Kopp rule, the heat capacity of composite fuel results from the molar fraction of the elements, and their respective heat capacities. Figure 1 shows the fractional thermal expansion of the primary ceramic oxide fuels and the (Th–U)O_2 composition.

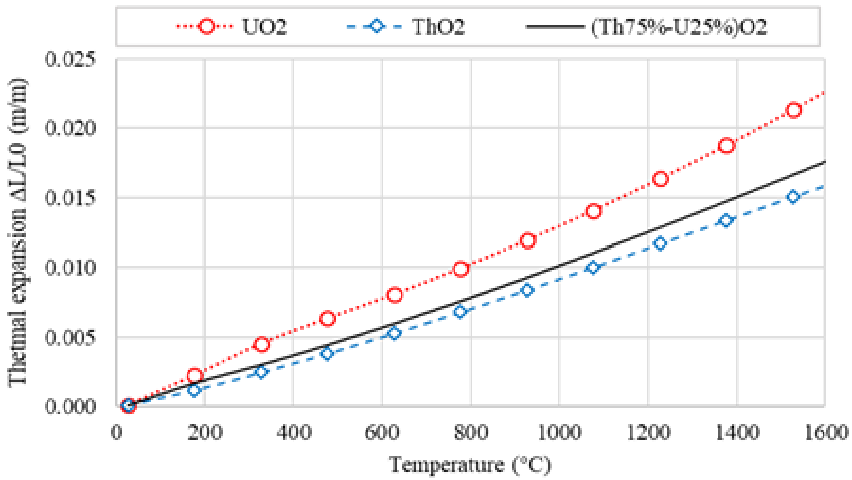

Figure 1. Linear thermal expansion of UO_2, ThO_2, and (Th–U)O_2 fuel.

Following the thermal expansion of the (Th–U)O_2 composite fuel, the open literature provides less detail than that of the pure compounds. The formulation adapted to it bases the thermal expansion on pure ThO_2 and pure UO_2 by applying

Vegard's law and updated FRAPCON code. The fuel UO_2 has a fractional length change of 10%-5% over PuO_2, and ThO_2 shows lower expansion at approximately 50% than UO_2.

Because of the lower thermal expansion of ThO_2 of approximately half that of UO_2, the $(Th–U)O_2$ should show smaller values than UO_2. The correlation adopted to expand the UO_2 is a third-degree polynomial curve fit. The numeric correlations depend on the temperature ranges, with an uncertainty lower than 10%. The linear expansion coefficient of UO_2 is higher than that of ThO_2 (Tyagi and Mathews, 2000). The function adopted used the thermal expansion coefficient in the composition's function, as shown in Eq. (1).

$$\alpha(y) = (0.973 - 1.746y) \times 10^{-6} \tag{1}$$

where α in (*µm/m-°C) y* is thorium content $(Th_yU_x)O_2$, *x* represents the uranium fraction, and *y* is thorium in weight.

Previously, Lyon and Baily measured the equilibrium between the solid and liquid phases of UO_2, reporting 2850 °C ± 20 °C, and PuO_2 is of 3650 °C ± 20 ºC. Therefore, supposing formation of a mixed homogeneous $(U\text{-}Pu)O_2$ with 20 wt%, a melting point of 2750 °C ± 20 ºC. The fusion behavior of $(Th–U)O_2$ depends on molar fractions of thorium content, with portions of 20 wt% of UO_2. It shows a melting point of 3553 °C ± 20 °C of ThO_2, superior to pure UO_2.

The experimental data came from the thermal conductivity measurement of mixtures contents. Comparatively, it could verify that the data onto thorium dioxide is minimal. The composite $ThO_2–UO_2$ may improve safety margins because of the addition of ThO_2, having a high thermal conductivity for the temperature range of operation in nuclear reactors. It validates the correlations used for thermal conductivities between room temperature and 1500 °C, with limited UO_2 content below 20 wt%. Recent studies of molecular dynamics (MD) investigated the thermal conductivity of ThO2 in the temperature range 27–2726 C. The quality of sinter process used to$(Th\text{-}U)O_2$ has essential functions as the homogenization level caused by crystal defects. The model applied to compute the thermal conductivity must apply correction factors, such as porosity and irradiation (Kutty et al., 2008). The model adopted the equations for fresh fuel given as a function of temperature and thorium content $(Th\text{-}U)O_2$, and Eq. (2) involve the thermal conductivity of thoria–urania-based fuels. Figure 2 shows the thermal conductivity of oxide fuels.

$$k(ThyUx)O_2 = \frac{1}{0.0232 + 0.1887(y) - 0.24(y)^2 + (2.37 - 0.18(y)) \times 10^{-4} T} \tag{2}$$

where *y* is the thorium content, the temperature is in K, and *k* is the thermal conductivity in (W/m-K).

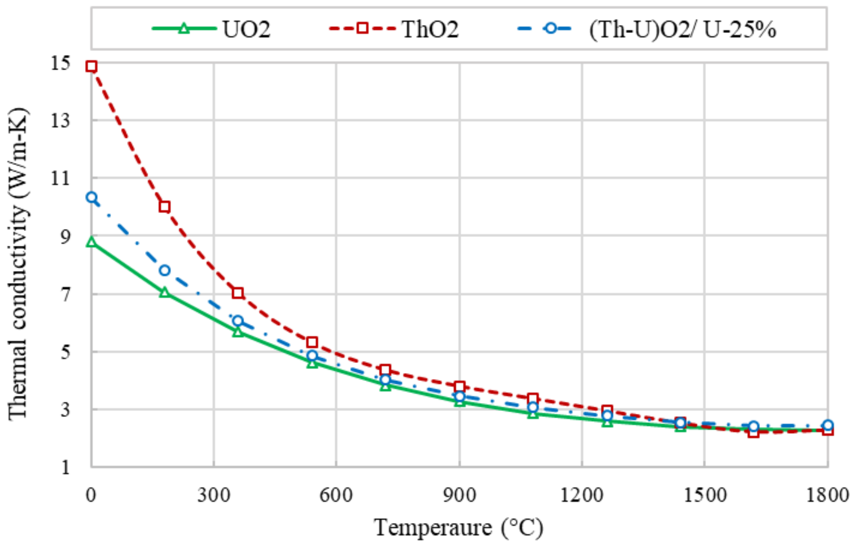

Figure 2. Thermal conductivity of pure oxides UO_2 and ThO_2 and mixed oxide (Th 75% - U 25%)O_2.

The scientific literature includes many studies of the specific heats of oxide actinides, and ThO_2, UO_2, and mixed $(Th–U)O_2$ (Xiao et al., 2016). The following empirical correlations express the specific heat as functions of the temperature in K, from room temperature and below the melting point, summarized in Eq. (3). Figure 3 displays the specific heat capacity of primary oxide fuels.

$$Cp(ThyUx)O_2 = (8.510 - 1.344y) \times 10^{-2} + (3.864 + 4.392y) \times 10^{-6} T - (3.403 - 2.281y) \times 10^{3} / T^2 \tag{3}$$

where *y* is thorium content (Th*y*U*x*)O_2, and *x* represents the uranium fraction in weight.

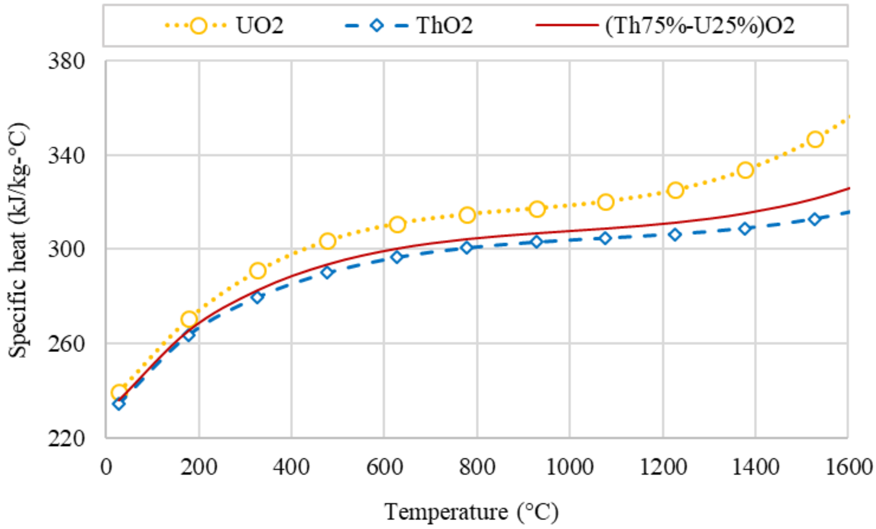

Figure 3. Specific heat of pure oxides UO_2, ThO_2, and mixed (Th 75% - U 25%)O_2.

There is a radial distribution of power generated different isotopes within pellets because of the decay chains between Th-232 and U-238. Initially, defined to UO_2 in the Transuranus Burn-up (TUBRNP), a method applied to LWRs for a burn cycle of 35 and 64 (GWd/MTU) (Lassmman et al., 1994). The shape function of Th-232 used only one-group cross-section values to calculate, spectrum-averaged of neutron absorption, and the capture used on the model. The results using empirical coefficients are $p_0 = 0.80$, $p_1 = 1.76$, $p_2 = 3.0$, and $p_3 = 0.22$. It gives the shape curve as a function radius of the pellet, as displayed in Eq. (4).

$$f(r) = p_0 + p_1 \exp(-p_2 (1-r) p^3) \tag{4}$$

2.2 Mechanical Properties

The modulus of elasticity of composite fuel hired the properties of contents UO_2 and ThO_2. Equation 5 showed the Young modulus of UO2 the Eq (6) exhibited the correlation used to ThO2, and eq (7) represents the modulus of elasticity of (ThU)O_2

$$E_{(UO_2)} = 233.(1-2.752P).(1.0-1.0915\times10^{-4}T) \tag{5}$$

$$E_{(ThO_2)} = 253.(1-2.21P).(0.96462-1.405\times10^{-4}T) \tag{6}$$

$$E_{(ThU)O_2} = 241.29-3.3018\times10^{-2}T+7.6415\times10^{-2}P.T-564.81P \tag{7}$$

where *E* is the modulus of elasticity in GPa, *T* is the temperature in K, and *P* is porosity or (1- TD%) of fuel.

2.3 Steady-state and transient simulation

A PWR fuel rod was pre-irradiated at 58 (GWd/MTU) and subjected to LOCA conditions. The rod fuel length was 0.47 mm with zircaloy-4 cladding. The outer diameter was 9.50 mm, and the wall thickness was 0.75 mm, yielding a plenum volume of 17 cm^3. The gas used to fill the system was a balanced mixture of 95% Ar and 5% He pressurized at 4 MPa. Comparatively, the internal temperature distribution of the fuel was lower for the (Th–U)O_2 than UO_2, because of higher thermal conductivity. The coating must exhibit embrittlement after irradiation because of hydrogen uptake of approximately 380 ppm. The fuel rod underwent irradiation cycles at 58 GWd/MTU. In the heating-up phase, the temperature increased abruptly from 200 °C to 1100 °C in approximately 310 seconds. The failure occurs when the peak cladding temperature of 850 °C at a time of 308 seconds after the initiation of the blowdown phase. The outer surface had a layer of the zirconia with a thickness of 75 µm (Manngård et al. 2014).

Nuclear safety analysis depends on the fuel rod response to the LOCA case studied for design basis accidents for LWRs. Forward to 10CFR50.46, the safety criteria define deterministic rules. The criteria establish that the maximum cladding temperature shall not outpace 1204.44 ºC. It describes the radial temperature gradient as a function of the pellet radius, reaching reductions of up to 200 °C in the central line, occurring at 50 days of irradiation, as shown in Fig 4.

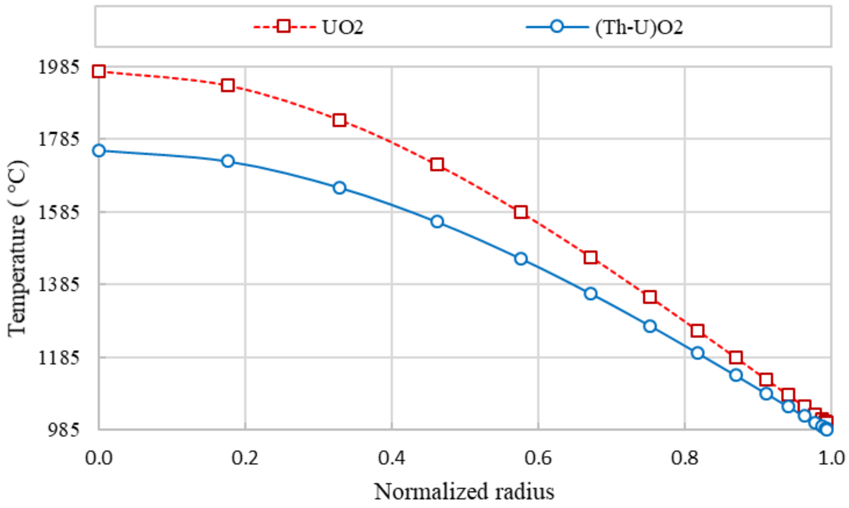

Figure 4. Fuel pellet temperature distribution of UO_2 pellet and (Th 75%–U-25%)O_2 at 6.7 MWd/kg

The computed total oxidation of cladding shall nowhere exceed 0.17 times the total cladding thickness as fabricated. The maximum hydrogen produced from the chemical reaction with water showed a limit of only 1% of the total hypothetical amount considering that it reacted whole with all the metal of the cladding. Figure 5 displays the temperature distribution of fuel centerline and average temperatures for both fuels, UO_2, and mixed oxide.

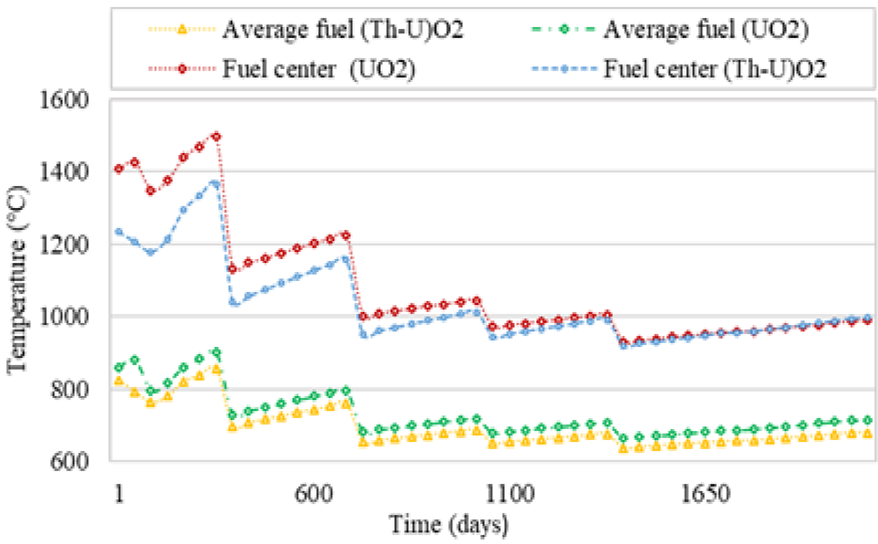

Figure 5. Burnup cycle and the fuel temperature of the rod fuel IFA-650.5 using (Th 75%-U 25%)O_2 and UO_2.

The simulation of the steady-state performed by FRAPCON was 8% in terms of the amount of built-up gas throughout 1994. Several physical models show dependencies with fuel pellet temperatures such as thermal strain, thermal creep, fission gas diffusion and release, fission gas swelling. The temperature drops into the standard fuel UO_2, because of lower thermal conductivity, but using composite fuel (Th-U) exhibits improved safety conditions. The spatial distribution of temperature of the fuel is a function of the axial power rate used, and burnup levels since the fissile are depleted more rapidly on the surface of the pellet, and thus the volumetric heat source distribution varies. In a simulation (Th-U)O_2 showed around 200 °C below of UO_2. Figure 6 shows the results of the simulation. (Th 75% - U 25%)O_2 fuel produces lower fission gas, in simulation, proving that it may reduce the fuel temperatures, and an FGR of 8.4% decreased to 3.5% using mixed oxide based on thoria fuel.

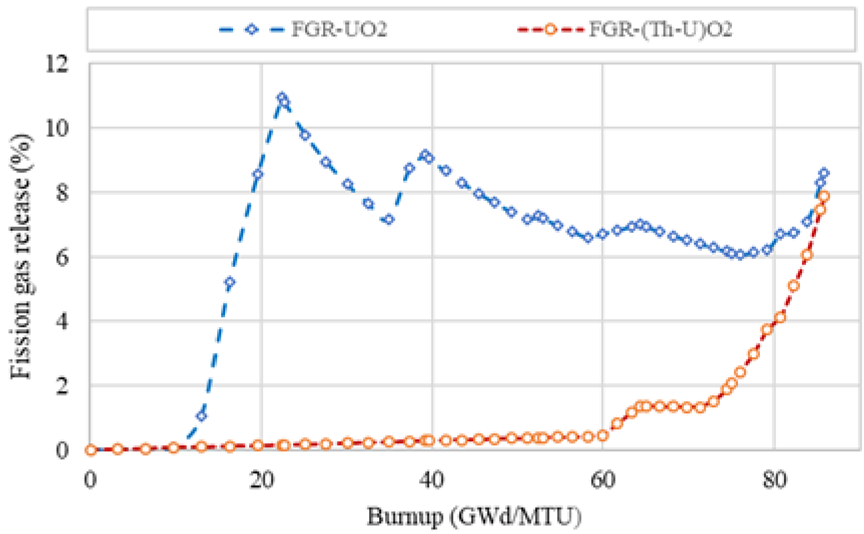

Figure 6. Fission gas release from UO_2 and (Th 75%–U 25%)O_2.

For the original configuration of the case IFA-650.5 during LOCA, temperatures of over 865 °C caused to the zircaloy-4 phase transition. The leading causes can create ceramic oxide fuel cracks and displacement because of thermal expansion, coupled with small-particle relocation, combining the effects of diffusion of fission gas release and fuel swelling. After the irradiation cycle, through FRAPCON code analysis, the transient state was simulated with FRAPTRAN code for both the proposed UO_2 and (Th–U)O_2. In these simulations, working with nine axial nodes applied to the axial rod length, also IFA-650.5 uses UO_2 as fuel showed a failure of the fourth node of dozen nodes used to axial length. The failure occurred because of the cladding rupture calculated based on the mechanical model or FRACAS-I model used by FRAPTRAN code.

During the LOCA transient, the blocking of the coolant channels could occur, and FRAPTRAN uses the ballooning model to predict any failure. The calculated swelling as a function of the differential pressure between the fuel rod gap and the coolant joined with the thermal expansion of the fuel rod, follows the plastic straining of the coating, helping produce the ballooning. This deformation must reduce hydraulic canals and increase the temperature. The fuel particles may fragment and drop into the gap because of fuel swelling. In the LOCA, the IFA-650.5 series showed a diametral strain of 13%–15%, resulting in a series of fragments relocation of the fuel. Also, the (Th–U)O_2 showed a time failure of 196 seconds, and UO_2 at approximately 189 seconds. A simple comparison shows that both responses are equivalent, and it gives the results from Table 4.

Table 4. Simulation accident scenarios of IFA-650.5

FRAPTRAN at failure point results	UO_2	(Th–U)O_2
Fuel centerline temperature (°C)	908	928
Fuel-pellet surface temperature (°C)	864	872
Cladding inside temperature (°C)	796	814
Cladding average temperature (°C)	795	813
Cladding permanent hoop strain (m/m)	0.32	0.31
Cladding hoop stress (MPa)	14.23	14.23
Equivalent cladding reacts (frac)	0.067	0.066
Failure time (s)	189	195

The axial elevation of the failure position depends on the outer-cladding surface temperature distribution. Mechanical properties are the most critical factors in estimating fuel performance. The model of pellet-cladding interaction emerges from empirical and analytical studies. In simulations, the cladding temperatures are practically identical to UO_2, and (Th-U)O_2. However, reduced gaseous produced with thorium fuel had reduced pressure on the internal wall of cladding change slight pressure and strain effects. On experiments performed on the Halden program, the fuel performance codes exhibited uncertainties. The temperature of rupture shows a deviation of 1.5%, and the diametric cladding strain is around 6.5%, cladding internal pressure show uncertainties of 5.4%. Possibly there are deviations of the same order regarding safety parameters. Figure 7 shows the hoop strain axial distribution of cladding.

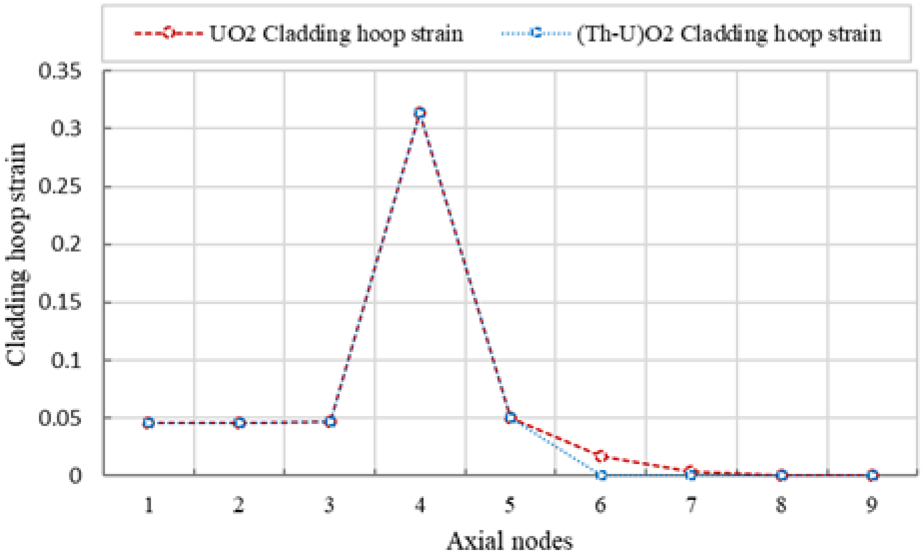

Figure 7. Axial distribution of cladding hoop strain between UO_2 and (Th 75%–U 25%)O_2

The failure of fuel results from the coupled effects of the high-over-pressure temperature of the cladding, reaching 765 °C. The FRACAS model compared the effective plastic strain of the cladding material with the stability strain limit defined material properties library MATPRO. Here, the stability strain limit was lower than produced, thus beginning the balloon model where the cladding suffered plastic deformation following a rupture because of the burst effect. The BALON-2 model uses two criteria to calculate a possible clad failure. The first condition observed is if cladding hoop stress exceeds the practical limit. The second motivation can depend on whether the permanent hoop strain found on cladding exceeded the FRAPTRAN strain limit defined for zirconium alloys. Figure 8 shows the fuel behavior under LOCA scenarios.

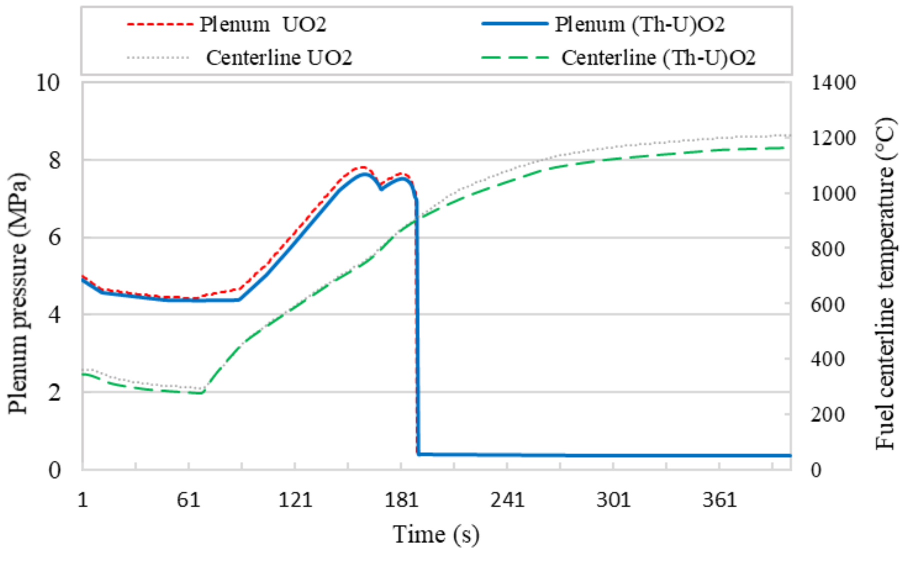

Figure 8. Plenum pressure and centerline temperature under transient for UO_2 and (Th 75%–U 25%)O_2

3. DISCUSSION

Thoria based fuels really could replacement uranium dioxide, but it needs some guidelines to take towards of use on civilian reactors fleet. The enrichment limits and a few deterministic parameters must change, such as PCT, mechanical criteriums, and corrosion, to become workable within LWRs. The optimized option tested is the fuel system based on mixed fuel containing a fraction of 25% UO_2 and 75-wt% ThO_2 can permit a simple replacement of standard UO_2 fuel, using the same dimension of fuel and pellet. The breeder characteristic of thorium dioxide applied to PWR showed the following features. (Th–U)O_2 displays a higher conversion rate of Th-232 to U-235 than conventional PWR, which should reduce the number of long-term actinides.

The simulations were performed with FRAPCON and FRAPTRAN codes adapted for composite fuel, which demonstrated important issues regarding safety parameters. Thorium can avoid a large amount of plutonium, increasing proliferation resistance. The simulations proved that the fuel designs adopted also have a safety response slightly better than that of the conventional fuel UO_2. The fuel temperatures decrease during steady-state also under accident scenarios. Internal plenum pressure also decreases in accident scenarios. In simulations failure, time after blowdown to $(Th\text{-}U)O_2$ is extensive than UO_2. Since 1993, when begun of a global effort to eliminate massive quantities of nuclear plutonium grades, Th-MOX is a solution. Thoria based fuel needs detailed full-core simulations based on core neutron kinetics of Th-MOX fuel to prove the feasibility of use for extended cycles. A lower fission gas release provides beneficial safety alternatives. A novel concept for thorium predicts as an additive to UO2 fuels for BWRs is another potentially attractive option of thorium fuel usage. Today, there is a scarce irradiation experience of ThO_2 fuel that has a slower increasing from last years. In this investigation introduced two versions of FRAPCON and FRAPTRAN to analysis the fuel behavior of $(Th\text{-}U)O_2$ containing 75 wt% of ThO_2.

4. ACKNOWLEDGMENTS

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