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Thorium Based Fuel Performance Modelling and the Development of the Multi-Pellet Material Fuel and Sheath Modelling Tool (MPM-FAST)

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ABSTRACT –A pathfor achieving greater sustainability in the nuclear power production industry would be to develop thorium based fuel cycles. The Canadian Super-Critical Water cooled Reactor (SCWR) concept is using this route to achieve the sustainability requirement of the Gen-IV Forum (GIF). However the study of Th-based fuel irradiation behaviour is less advanced than that of UO_2 fuel. With the majority of studies focused on $(Th,U)O_2$, and very limited work examining the $(Th,Pu)O_2$fuel type proposed in the SCWR project.

The Fuel and Sheath Modelling Tool (FAST), a fuel performance model for CANDU UO_2 fuel was developed at the Royal Military College of Canada (RMCC). Its capabilities could be extended to includeThO_2,$(Th,U)O_2$ and $(Th,Pu)O_2$as fuel pellet materials and potentially be used as a tool to aid in the design and performance assessment of Th-based fuels, including the SCWR $(Th,Pu)O_2$ fuel.

The development of the $(Th,U)O_2$ and $(Th,Pu)O_2$ models that were added to FAST to produce the Multi-Pellet Material Fuel and Sheath Modelling Tool (MPM-FAST) was performed in co-operation with Canadian Nuclear Laboratories (CNL). Presented below is an outline of the $(Th,U)O_2$ and $(Th,Pu)O_2$ models as well as modelling results from MPM-FAST compared to the Post Irradiation Examination (PIE) data from two CANDU fuel like experimental irradiation tests (DME-221 and BDL-422) conducted at Chalk River Laboratories (CRL).

Introduction

The Generation IV International Forum was established to undertake the research required to determine the feasibility and performance capabilities of the next generation of nuclear energy systems. GIF identified six reactor designs to focus research upon. One of these reactor designs is the Super Critical Water cooled Reactor (SCWR). Canada's participation in GIF is led by CNL, with the design of a pressure-tube-based SCWR. The proposed fuel is comprised of ceramic thorium-plutonium mixed-oxide ($(Th,Pu)O_2$, 13 wt% Pu content) pellets. The fuel will operate in supercritical water coolant, pressurized to 25 MPa and temperatures ranging from 315-625^{0}C. Linear element rating limits and target exit burnup goals have been established for the fuel [1]. With coolant and fuel conditions provided, a fuel performance model is being developed to support a feasibility study on fuel designs [2].

Work on the development of a fuel performance model for Canadian SCWR fuel is ongoing at RMCC. Models for ThO_2, $(Th,U)O_2$,and $(Th,Pu)O_2$fuel pellets havebeen in development [3,

4].These Th-based pellet models have been added to the Fuel And Sheath modelling Tool (FAST). FAST is aCOMSOL Multiphysics based fuel performance model for UO_2 CANDU fuel elements [5].The new combined model is referred to as the Multi Pellet Material Fuel and Sheath modelling Tool (MPM-FAST). Presented within this work is the proof of concept validation exercises undertaken for these Th-based fuel modelsin MPM-FAST. Theexercise is to have the ThO_2, $(Th,U)O_2$, and $(Th,Pu)O_2$models replicate the irradiation behaviour of two fuel irradiation experiments conducted by Canadian Nuclear Laboratories (CNL).

The validation exercise of the ThO_2 and $(Th,U)O_2$ models in MPM-FAST was to apply the model to reproduce the Post Irradiation Examination (PIE) data from DME-221. DME-221 is an experimental fuel irradiation test conducted by CNL [8]. DME is short hand for "demountable element"; this refers to how the experimental fuel is inserted into the U1 and U2 test loops of the National Research Universal (NRU) reactor. The experimental fuel is fabricated as elements that can be inserted into or removed (demounted) from the outer ring of a fuel bundle assembly. Six variations of experimental fuel were fabricated for DME-221: three different fuel compositions (ThO_2, $(Th,U)O_2$ with 1.0 wt.% ^{235}U and $(Th,U)O_2$ with 1.5 wt.% ^{235}U) having two different pellet geometries, one set of pellets has a "standard" CANDU Length-to-Diameter (L/D) pellet ratio (~1.3) and the other set has a reduced L/D ratio (~0.7). The initial average grain size of the fuel was reported as ranging between 6 - 8 µm, as well as initial densities of ≥95% theoretical density [6]. The experiment is ongoing, with twelve elements having undergone PIE and the three remaining elements to undergo further irradiation. The twelve elements that have undergone PIE were removed from the outer element ring on two separate occasions, leading to elements being either a "low" or a "high" exit burnup case.

The MPM-FAST $(Th,Pu)O_2$ model validation exercise was to attempt to replicate the irradiation behaviour of BDL- 422. BDL- 422 was a fuel irradiation experiment that was undertaken to demonstrate the ability of $(Th,Pu)O_2$ bundles to operate to high burnups (>1000 MWh/kgHE) (Heavy Element [HE]) [7, 8]. Six bundles were irradiated in test loops of the NRU reactor. Each bundle was fueled exclusively with $(Th,Pu)O_2$ with 1.53 wt% Pu. In order to avoid potential over pressurization caused by fission gas release, the outer ring elements were fabricated with plena at both ends of each element. The BDL-422 fuel pellets had an average initial grain size of 3-4 µm and an average initial pellet density of 9.469 g cm^{-3}. The six BDL-422 bundles were labelled ADA – ADF. A selection of outer ring elements (OE) and an additional element from the intermediate bundle ring (IE) were chosen for PIE from ADA, ADC, ADD, ADE and ADF [7 - 9].

The modelling assumptions directly relevant to the discussed results will be presented in context of physical phenomena MPM-FAST is calculating. This will be followed by model predictions of: pellet centerline temperature, and fission gas release, with the gas release results compared to PIE data.

1. Modelling Assumptions

Centerline Temperature

The model calculates the fuel teperature by approximating the solution to the heat conduction equation (equation (1)) using finite element methods.

$$\rho C_p \frac{\partial T}{\partial t} = \nabla \cdot (k \nabla T) + Q_{prod} \quad (1)$$

Here, T is the temperature (K), t is time (s), ρ is the density of the material (kg m^{-3}), C_p is the specific heat capacity of the material (J K^{-1}), k is the thermal conductivity of the material (W m^{-1} K^{-1}), and Q_{prod} accounts for the heat produced within the fuel pellet (J $m^{-3}s^{-1}$).

Since DME-221 and BDL-422 experiments experienced Normal Operating Conditions (NOC) during their irradiation histories. $C_p$is more significant during fuel transient conditions, but during NOC time steps tend to be large (i.e. the fuel is nearly at thermal equilibrium) and has little effect on the temperature calculation, and will be left out of the discussion. Also the volumetric heat production Q_{prod} is kept the same from one material model to the next and is excluded from the discussion for the sake of brevity but described in [5]. The discussion around fuel temperature will focus on the thermal conductivity$k$.

Lucuta presented a method to describe the effects of fuel burnup on the thermal conductivity of UO_2 [10]. The underlying assumption in this derivation of UO_2 thermal conductivity is that separate physical changes in the fuel during irradiation act individually on the thermal conductivity. The effect of physical changes on thermal conductivity can then be accounted for as a multiplication "correction" factor on the thermal conductivity of 100% theoretical density, unirradiated UO_2. The primary modelling assumption for the thermal conductivity of (Th,U)O_2, and (Th,Pu)O_2 are that the correction factors developed for UO_2 are applicable to the Th-based fuel. The application of Lucuta's thermal conductivity factors to (Th,U)O_2 simulated the temperature conditions required to model the fission gas release of that fuel composition [11].

The three correlations for (Th,U)$O_2$and (Th,Pu)O_2unirradiated thermal conductivity (k_0 in W m^{-1} K^{-1}) have been tested in MPM-FAST; are presented in the form of a semi-empirical correlation given in equation (2).

$$k_0 = \frac{1}{A+BT} \quad (2)$$

$A$and$B$ are dependent on the proportion of uranium or plutonium mixed with thorium.

(Th,U)O_2

The first of the (Th,U)O_2 models used a correlation presented by [12]for 100% theoretical density (Th,U)O_2 up to 0-30% UO_2 content below 2200 K, with the $A_1$and B_1 components in equation (3).

$$A_1 = \frac{1}{46.948 - 112.072 M_U}$$
$$B_1 = 1.597 \times 10^{-4} + 6.736 \times 10^{-4} M_U - 2.155 \times 10^{-3} M_U^2 \quad (3)$$

M_U is the mole percent of UO_2 present in the ceramic mixture. The use of this thermal conductivity is similar to the work on a Th version of FRAPCON [11].

The second thermal conductivity correlation tested in the model was presented by Bakker et al. for thermal conductivity of $(Th_{1-y},U_y)O_2$ at 95% theoretical density, where *y* is the wt% of U up to 10% ($0 \leq y \leq 0.1$) and temperatures between 300-1800 K (outlined in equation (4)) [13].

$$A_2 = 0.0004195 + 1.112y - 4.499y^2$$
$$B_2 = 0.0002248 - 0.000917y + 0.004164y^2 \quad (4)$$

A correction factor is applied to the Bakker correlation to make it applicable to 100% dense $(Th,U)O_2$.

(Th,Pu)O2

The $(Th,Pu)O_2$ fuel performance model uses the Cozzo correlation for unirradiated $(Th,Pu)O_2$ for a ceramic at 95% theoretical density as given in equation (5) [14].

$$A_3 = 0.006071 + 0.572 wtPu - 0.5937 wtPu^2$$
$$B_3 = 0.00024 \quad (5)$$

Here, *wtPu* is the wt% of PuO_2 in the fuel matrix (0-100% PuO_2), with the correlation being derived from a series of measurements in the temperature range of 500 K to 1600 K. Acorrection factor is applied to the Cozzo correlation in order for it to reflect the thermal conductivity of 100% dense $(Th,Pu)O_2$.

Fission Gas Release

Fission Gas Release (FGR) is modeled as a two-step process. The first step is to solve for the concentration of fission gas within fuel grains using the Booth diffusion model. This solution leads to the release rate of fission gas at the grain boundaries. Fission gas then accumulates at the grain boundaries. The second step within the model is the determination of the boundary saturation conditions; once the boundaries become saturated with fission gas, any fission gas that reaches the grain boundary past this point is released to the free volume of the element. The Booth diffusion equation is given in equation (6).

$$\frac{\partial C}{\partial t} = D\nabla^2 C + P_{fg} \quad (6)$$

In equation (6), C is the concentration of fission gas atoms (atoms m^{-3}), ∇^2 is the Laplacian in spherical coordinates, and P_{fg} is the volumetric production rate of fission gas (atoms $m^{-3}s^{-1}$), and D is the net fission gas diffusion rate ($m^2\ s^{-1}$). P_{fg} is determined by the yield of stable fission gas per fission multiplied by the rate of fission (F_{rate}). In the development of the UO_2 fuel performance model at RMCC, the determination of D is based on the work by Turnbull et al. and given by equation (7) (a full description of this model is presented in [5]).

$$D = \frac{D_0 b'}{b' + g_a} \tag{7}$$

Here, b' is the intragranular resolution rate in s^{-1}, g_a is the trapping rate in s^{-1}, and D_0 is the single fission gas atom diffusion coefficient for a fully dense UO_2 crystal as given. The value of D_0 for UO_2 is found by weighted sums of diffusion coefficients that describe the contribution of three separate mechanisms given in equation (8). The details of the b' and g_a are omitted for the sake of brevity.

$$D_0 = D_{thrm} + 4D_{irr} + 4D_{athrm} \tag{8}$$

D_{thrm} is the diffusion coefficient due to thermally activated processes (dominant term T > 1700 K), D_{irr} is the diffusion coefficient due to irradiation induced vacancies (dominant term 1100 K $\leq$ T $\leq$ 1700 K), and D_{athrm} is the diffusion coefficient due to athermal effects (dominant T < 1100 K).

(Th,U)O_2

For (Th,U)O_2, the change in the FG diffusion behaviour is scaled by a factor of 0.1 applied to D_0, as seen in equation (9).

$$D_{0(Th,U)O_2} = 0.1D_0 \tag{9}$$

It was found in CNL irradiation experiments that the observed FGR from Th-based fuels is significantly less than that observed in UO_2 fuel [6,15]. During the irradiation test of a defected (Th,U)O_2 fuel element, online monitoring of the ^{133}Xe released was found to be 1 to 2 orders of magnitude less than that of UO_2 fuel. The ingress of coolant into a defected element leads to fuel oxidation in UO_2, which affects the thermal conductivity and leads to an increase in FGR. It is uncertain what chemical reactions occur with defected Th-based fuel. Some reactions may alter the release rate of ^{133}Xe; however, in post-irradiation annealing tests on thoria-urania fuel (35% ThO_2, 65% UO_2) by [16], it was found that the diffusion of ^{133}Xe in poly-crystalline (Th,U)O_2 was approximately an order of magnitude less than that of UO_2. These findings provided the rationale for selecting the 0.1 scaling factor for the (Th,U)O_2 fission gas diffusion model used.

(Th,Pu)O_2

MPM-FAST uses the single fission gas diffusion coefficient presented in equation (10) when (Th,Pu)O_2 is modelled.

$$D_{0(Th,Pu)O_2} = 1.05D_{thrm} + 0.25D_{irr} + 0.25D_{athrm} \quad (10)$$

Equation (10) takes the same form as equation (8), the modelling assumption is that the physical phenomena that enable the fission gas to diffuse through the fuel matrix of UO_2 will be present in (Th,Pu)O_2. However reduced FGR from (Th,Pu)O_2 fuel irradiation tests compared to UO_2 fuel under NOC havebeen reported [10,11], and it is believed that the this difference in behaviour can be replicated by reducing weighting factors applied to the components of D_0.

Theweighting factors presented in equation (10) were found by fitting the FGR model to the PIE results from the intermediate elements from BDL-422, while testing various weighting factors within the single fission gas diffusion coefficient. This was done by using non-linear,least squares optimization software.As the software used for this series of optimization is in its testing phase, this fitting will be repeatedwith COMSOL's built-in optimization tools as a means of verifying the fitting results.

Unfortunately there are very few irradiation tests of (Th,Pu)O_2discussed in open literature, and the literature reviewhas as of yet to find diffusion data on any gas withingrains of (Th,Pu)O_2. The only justification for the selection of these weighting factors is that it fits the FGR data from BDL-422.Th-based ceramics have a higher melting temperature than UO_2 which would indicate that more energy is required to break apart the solid structure of this fuel type. It is reasonable to assume that producing interstitial defects that enable fission gas diffusion within a Th-based fuel grain would similarly require more energy. For the low temperature dependent terms in equation (10), that are less than 1, there is only a mild reassurance that this setof weighting factors reaffirms this assumption.

2. (Th,U)O_2 Results

This section will focus on the modelling results from the (Th,U)O_2 model. First a comparison of the model's temperature predictions will be presented, followed by the calculated FGR compared to the PIE data from DME-221.

Centerline Temperature

Two variations of the (Th,U)O_2 MPM-FAST model were examined: Case 1) uses the Belle and Berman thermal conductivity correlation and Case 2) uses the thermal conductivity presented by Bakker et al.[14, 15]. The modelling results from Case 1) and Case 2) will be compared to measured FGR, and end-of-life strain PIE data from DME-221.Table 1 displays a comparison of the highest achieved centerline temperatures (calculated) between the UO_2, Case 1), and Case 2) for each DME-221 element.The left most column in Table 1 indicates the amount of U-235 within the fuel, followed the exit burnup of each DME-221 fuel element in (MWh $kgHE^{-1}$), the middle three columns display the modelling results for UO_2 and bothTh-based modelling case, and the right most column contains the temperature difference between the twoTh-based cases.

Table 1Comparison of Modelled Centerline Temperatures for DME-221 Elements

% U-235	Burnup (MWh kgHE^{-1})	Modeled Max. Temp.of Fuel (K)			Temp. Diff.Between Th-based Models (K)
		UO_2	Case 1)	Case 2)	
0	**361**	1314	1144	1244	100
0	**619**	1680	1317	1460	143
0	**375***	1537	1174	1281	107
0	**618***	1792	1344	1520	176
1	**499**	1455	1261	1421	160
1	**525***	1554	1315	1411	96
1	**839***	3127	1611	1820	209
1.5	**929**	2487	1698	1860	162
1.5	**549**	2921	1592	1860	268
1.5	**593***	1950	1592	1731	139
1.5	**914***	2396	1669	1831	162
1.5	**903***	2386	1653	1804	151

* Fueled with “short” pellets (L/D = 0.7)

The Th-based fuel model using either the Belle and Berman correlation or the Bakker et al. thermal conductivity correlation, consistently predict lower centerline temperatures than the UO_2 model.Centerline temperature results from Case 1) are lower than those modeled in Case 2), with growing temperature differences occurring with increased burnup; there is an additional increase in temperature difference within the highest uranium enrichment fuel. This temperature difference arises because the Bakker et al. model predicts lower thermal conductivity values for Case 2) models compared to the Belle and Berman model (used in Case 1)). Figure 1 displays the conductivity curves from both models for the 1.5% U-235 fuel (assuming fully dense fuel, porosity is accounted for via Lucuta factor). Consensus within the literature is in favour of using the Bakker correlation over the Belle and Berman correlation, due to later measurements displaying larger degradation in thermal conductivity with U enrichment [13].

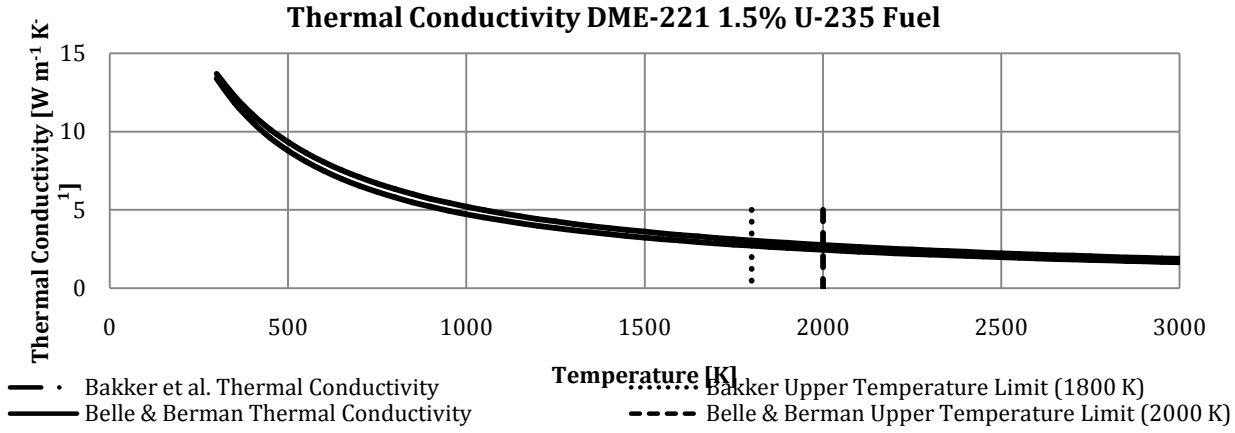

Figure 1: Presents a comparison of Thermal Conductivity correlations used in each case for 1.5% U-235 DME-221 Fuel.

In Figure 1, the vertical lines indicate the uppermost validation temperature for each of the models. It should be noted that all the elements from the 1.5% U-235 fuel and the high burnup element from the 1.0% U-235(839 MWh kgHE^{-1}) fuel in the Case 2) models exceed the range of the Bakker et al. conductivity curve, which it is based upon. This leads to uncertainty in the temperature behaviour for these cases because it is not known if extrapolating the Bakker correlation is physically representative above this temperature.

Fission Gas Release

Table 2 compares the modelled FGR for UO_2, Case 1), and Case 2) to measured data from [6].

Table 2: Modelled Percent FGR for UO_2 Case 1 and Case 2) compared to the PIE results of DME-221.

% U-235	**Burnup (MWh kgHE^{-1})**	**Modelled % FGR**			**Measured % FGR Range**
		UO_2	**Case 1)**	**Case 2)**	
0	**361**	0.5	0	0	0.05-0.1
0	**619**	15.1	0.2	1.0	
0	**375***	5.0	0	0	
0	**618***	18.1	0.4	0	
1	**499**	5.0	0	1.6	0.06-1.2
1	**525***	6.3	<0.1	0	
1	**839***	36.6	2.4	<0.1	
1.5	**929**	22.3	3.3	4.3	0.08-2.8
1.5	**549**	21.4	0.1	5.0	
1.5	**593***	10.3	0.2	1.1	
1.5	**914***	21.1	2.9	0.5	
1.5	**903***	20.1	2.5	4.3	

* Fueled with “short” pellets (L/D = 0.7)

Data presented in Table 2 is presented in a similar format to Table 1 except the far right column contains the FGR from DME-221. Results of the FGR from the UO_2 are substantially greater than the measured FGR from the experiment. The Th-based fuel model that uses the Belle and Berman thermal conductivity correlation best replicates the experimental results for FGR from DME-221.

3. $(Th,Pu)O_2$ Results

The $(Th,Pu)O_2$ modelling results will be presented in the same fashion as the $(Th,U)O_2$. In that the temperature calculation of the new $(Th,Pu)O_2$ model will be compared to the UO_2 irradiated under the same conditions, followed by comparing the FGR calculations to the experimental results of BDL-422.

Centerline Temperature

Table 3 below contains the maximumpredicted centerline temperature of the UO_2 and $(Th,Pu)O_2$experiencing the irradiation conditions of the BDL-422 fuel. The table indicates results from outer ring elements and intermediate ring elements for various bundles. The left most column indicates the specific bundle, followed by the max sustained linear power rating and exit burnup. In the other half of the table the maximum predicted temperaturesof UO_2 and $(Th,Pu)O_2$ are presented followed by the temperature difference between the two results.

Table 3:A comparison of the modelled max.temperature of the fuel during the BDL-422 irradiation test.

Bundle	**Max Linear Power [KW m^{-1}]**	**Exit Burnup [MWh $kgHE^{-1}$]**	**Modelled Max. Fuel Temp. [K]**		**Temperature Difference[K]**
			UO_2	**$(Th,Pu)O_2$**	
Outer Elements					
ADA	**54**	**1181**	1953	2127	-174
ADC	**67**	**451**	2394	2572	-178
ADD	**73**	**1082**	2468	2687	-219
ADE	**64**	**597**	2156	2246	-90
ADF	**52**	**856**	1887	1869	18
Intermediate Elements					
ADA	**27**	**718**	1364	1354	10
ADC	**22**	**288**	1047	1046	1
ADD	**45**	**665**	1578	1549	29
ADE	**32**	**308**	1223	1218	5
ADF	**39**	**531**	1358	1348	10

It is expected that Th-based fuel will experience lower fuel temperatures than UO_2. Predicted temperatures for the outer fuel elements from BDL-422 fuel with power histories excluding ADF are all predict higher temperature than UO_2 irradiated in the same conditions ($\geq$ 220 K). The intermediate elements and the outer elements fromADF have modelled temperature results that meet the expectation of reduced fuel temperatures.

In order to determine if the temperature behavior displayed by the model is physical, the thermal conductivity correlation needs to be examined. Figure 2 presents the unirradiated thermal conductivity for BDL-422 fuel as predicted by the Cozzocorrelationaltered for 100% dense

ceramic, in comparison to the unirradiated UO_2 thermal conductivity predicted by the ELESTRES fuel code for the same temperature range [5, 14].

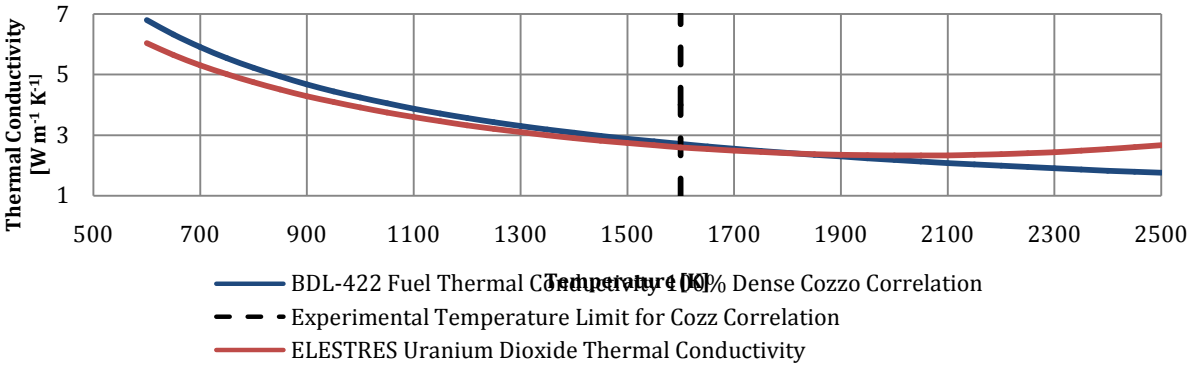

Figure 2: A comparison of 100% dense$(Th,Pu)O_2$ thermal conductivity correlation, to the conductivity of UO_2.

With the Cozzocorrelation adjusted to account for 100% dense ceramic fuel, the temperature predicted by the model will likely fall in line with the expectation of Th-based fuel will have lower centerline temperatures until the fuel reaches ~1850 K. The fuel temperature results in Table 3 reflect this behavior.

Fission Gas Release

Results from the UO_2 and $(Th,Pu)O_2$model for %FGR are compared to the FGR data from BDL-422, as indicated in Table 4. Table 4 is of a similar format to Table 3,with the major differences being, the PIE data is presented in the right most column, and the FGR modelling results are in the middle right columns.

Table 4: Presents the %FGR model results for the $(Th,Pu)O_2$, and UO_2 compared to the PIE results of BDL-422

Bundle	**Max Linear Power [KW m^{-1}]**	**Exit Burnup [MWh $kgHE^{-1}$]**	**Modelled %FGR**		**Measured % FGR**
			UO_2	**$(Th,Pu)O_2$**	
Outer Elements					
ADA	**54**	**1181**	27.7	21.8	26.2 -32.8
ADC	**67**	**451**	3.6	3.7	5.3
ADD	**73**	**1082**	26.3	18.3	19.6 - 26.9
ADE	**64**	**597**	8.5	4	1.2
ADF	**52**	**856**	19.4	8.7	2.8
Intermediate Elements					
ADA	**27**	**718**	11.4	1	1.0
ADC	**22**	**288**	0	0	0.3
ADD	**45**	**665**	10	0.5	0.5
ADE	**32**	**308**	0.3	0	0.1
ADF	**39**	**531**	4.9	0	0.2

As discussed above, the weighting factors for equation (11) were determined by attempting to fit the fission gas diffusion behavior to the intermediate element FGR results.The FGR model results are only able to replicate the PIE measurements from IE ADA and ADD. Even with the lack of agreement with PIE measurements for FGR from ADC, ADE, and ADFthe new $(Th,Pu)O_2$model is an improvement over the UO_2 model.

The $(Th,Pu)O_2$ model results over predict the FGR for the OE from ADE and ADF (again they are an improvement over the UO_2 model), but under predict the PIE data from ADA, ADC, and ADD. Recall that the three components of the fission gas diffusion in equations (8) and (11) are each the dominant component within different temperature regimes (D_{thrm} is dominant when T > 1700 K, D_{irr} when 1100 K ≤ T ≤ 1700 K, and D_{athrm} when T < 1100 K). It should be noted that from the temperature results in Table 3, the IE used to fit the FG diffusion behavior never achieve temperatures that have allow D_{thrm} to act as the dominate term. This causes the high temperature diffusion behavior to be under represented in determining the current FG diffusion weighting factors. With further examination of high temperature behavior diffusion it is believed that the modelwill be able to replicate the FGR of OE ADA, ADC, and ADD.

4. Summary and Conclusions

Th-based fuel performance models for ThO_2, $(Th,U)O_2$, have been compared to the DME-221 PIE data and demonstrated improved agreement to the FGR results in comparison to a UO_2 model. A $(Th,Pu)O_2$ is nearing the end of its development phase. Further edits are required to fission gas diffusion model. In its current iteration the $(Th,Pu)O_2$model is better at replicating the FGR exhibited by the BDL-422 experiment, than the UO_2 model.

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