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A Numerical Study of Excursions in UHTREX
Loaded with Uranium-Thorium Fuel Elements

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A NUMERICAL STUDY OF EXCURSIONS IN UHTREX LOADED WITH URANIUM-THORIUM FUEL ELEMENTS

by

George C. Hopkins

ABSTRACT

A uranium-thorium loading has been proposed for the Ultra High Temperature Reactor Experiment (UHTREX). Calculations have been made for this loading for excursions resulting from step reactivity inputs of 20¢ to \$2.00. The presence of thorium retards the transient significantly.

INTRODUCTION

Fuel elements now being used in the Ultra High Temperature Reactor Experiment (UHTREX) consist of UC_2 spheres coated with pyrolytic carbon and embedded in a graphite matrix.¹ A design incorporating ThC_2 into the coated particles has been proposed for experiments on the effect of large Doppler coefficients on transients. Calculations of transients were done for this design for step reactivity inputs of 20¢ to \$2.00.

MODEL DESCRIPTION

The fuel consists of a graphite matrix with two kinds of coated particles. The fissile particle has a 200- μ -diam kernel of UC_2 and ThC_2 ($Th:U = 4$); the fertile particle has a 350- μ -diam kernel of ThC_2 . Both types of particles have buffer and isotropic layers of carbon, each 30 μ thick.

The critical mass is 12.0 kg of uranium (93.5% ^{235}U). Each fuel element then contains 9.615 g uranium and 94.0 g thorium.

An average cell configuration, shown in Fig. 1, was used in core parameter calculations. The resulting atom densities are given in Table I. The DTF-finite² and ONEDAC calculations were performed with a 6-group cross-section set. From the DTF calculation, it was found that $k_{eff} = 1.0501$. The temperature coefficients, from ONEDAC, are:

bulk moderator: $-7.9544 \times 10^{-5}/^{\circ}K$
Doppler: $-1.8454 \times 10^{-5}/^{\circ}K$
fuel moderator: $-5.9762 \times 10^{-6}/^{\circ}K$

TABLE I
ATOM DENSITIES
(10^{24} atoms/cm³)

	Fuel	Moderator
Al	1.7410×10^{-4}	1.3020×10^{-3}
B	1.9628×10^{-7}	9.5476×10^{-8}
C	7.3019×10^{-2}	8.6744×10^{-2}
Fe	1.7861×10^{-5}	4.3806×10^{-5}
$^{234}U^*$	3.9997×10^{-6}	----
^{235}U	3.2380×10^{-4}	----
^{238}U	1.999×10^{-5}	----
^{232}Th	3.4471×10^{-3}	----

* ^{236}U was included as an equivalent amount of ^{234}U .

Fig. 1. One-dimensional average cell geometry.

The neutron lifetime was calculated to be 7.2822×10^{-4} sec, and $\beta_{\text{eff}} = 6.7656 \times 10^{-3}$.

The transient analysis was done using ANCON³ with step reactivity insertions from 20¢ to \$2.00.

Two thermal feedback models were studied. The first is a four-lump model, described in Table II. The second is an eight-lump model, described in

Table III. Both models assume that the coolant channels are evacuated.

DETERMINATION OF DOPPLER COEFFICIENT

The Doppler temperature coefficient was obtained from the differences between two thorium cross-section sets output from RMITU⁴ at 300 and 400°K.

TABLE II
FOUR-LUMP THERMAL FEEDBACK MODEL

Description	Lump	Fraction of Fission Energy	Heat Capacity (MW-sec/°K)	Temperature Coefficient $\left(\frac{\Delta k/k}{^\circ\text{K}}\right)$
Fissile kernels	1	0.873	0.01984	-8.259×10^{-6}
Isotropic and buffer layers, plus binder	2	0.033	0.08361	-7.828×10^{-6}
Fertile kernels	3	0.001	0.02339	-1.203×10^{-5}
Moderator	4	0.093	2.280	-7.586×10^{-5}

		Heat Transfer Coefficient	
		Convective	Radiative
		h_c^{i+j}	h_r^{i+j}
<u>i</u>	<u>j</u>	(MW/°K)	(MW/°K)
1	2	1.615	---
2	3	1.032	---
2	4	---	7.895×10^{-13}

TABLE III
EIGHT-LUMP THERMAL FEEDBACK MODEL

Description	Lump	Fraction of Fission Energy	Heat Capacity (MW-sec/°K)	Temperature Coefficient $\left(\frac{\Delta k/k}{^\circ\text{K}}\right)$
Fissile kernels	1	0.874	1.984×10^{-2}	-8.259×10^{-6}
Fissile buffer layers	2	0.031	4.605×10^{-3}	-9.739×10^{-7}
All isotropic layers plus binder	3	0.003	7.613×10^{-2}	-6.239×10^{-6}
Fertile buffer layers	4	0.001	2.868×10^{-3}	-6.154×10^{-7}
Fertile kernels	5	0.001	2.339×10^{-2}	-1.203×10^{-5}
Cylindrical moderator				
1.898 to 3.287 cm	6	0.021	0.4947	-1.646×10^{-5}
3.287 to 4.676 cm	7	0.032	0.7598	-1.646×10^{-5}
4.676 to 6.066 cm	8	0.043	1.026	-3.412×10^{-5}

		Heat Transfer Coefficient	
		Convective	Radiative
		h_c^{i+j}	h_r^{i+j}
<u>i</u>	<u>j</u>	(MW/°K)	(MW/°K)
1	2	2.577	---
2	3	4.329	---
3	4	2.451	---
3	6	---	7.895×10^{-13}
4	5	1.782	---
6	7	0.4355	---
7	8	0.6091	---

The value for σ_p , the effective potential scattering cross section for the reactor fuel region, which is required in the RMITU input, was obtained from the relation

$$\sigma_p = \frac{\Sigma_{p,eff}}{N_{a,eff}} + \frac{\tau_{eff}}{N_{a,eff}} + \frac{\tau_{f.e.}}{N_{a,f.e.}} \quad (1)$$

The first term, which accounts for the homogeneous "self-shielding" effect due to the flux depression at resonance energies, is made up of

$$\Sigma_{p,eff} = \frac{\Sigma_{p,fissile} V_{fissile} + \Sigma_{p,fertile} V_{fertile}}{V_{fissile} + V_{fertile}}$$

and

$$N_{a,eff} = \frac{N_{Th,fissile} V_{fissile} + N_{Th,fertile} V_{fertile}}{V_{fissile} + V_{fertile}}$$

The other two terms of Eq. 1 treat spatial self-shielding effects. The second term, for particles, is made up of $N_{a,eff}$, defined above, and

τ_{eff} = interaction probability per unit flux for fissile and fertile particles

$$= \frac{\Sigma_{fuel mod} S_{eff}}{\Sigma_{fuel mod} + S_{eff} \left(\frac{V_{fissile} + V_{fertile}}{V_{fuel element}} \right)}$$

where

S_{eff} = averaged effective inverse mean chord length, for the two types of particles

$$= N_{a,eff} \left(\frac{N_{Th,fissile}}{S_{fissile}} + \frac{N_{Th,fertile}}{S_{fertile}} \right)^{-1}$$

S_f = inverse mean chord length, sphere, for fissile or fertile kernel

$$= \frac{3.9}{4R_f}, \quad R_f = \text{radius.}$$

The third term of Eq. 1, for the fuel element, is made up of

$N_{a,f.e.}$ = atom density of thorium in fuel element

and

$\tau_{f.e.}$ = interaction probability/unit flux for fuel and moderator

$$= \frac{\Sigma_{s,bulk mod} S_{f.e.}}{\Sigma_{s,bulk mod} + S_{f.e.} \left(\frac{V_{f.e.}}{V_{cell}} \right)}$$

where

$S_{f.e.}$ = inverse mean chord length, cylinder

$$= \frac{1.3}{2R_{f.e.}}$$

RESULTS

The results of the numerical analyses are presented in Tables IV through VI, and transient characteristics of the standard loading and the uranium-thorium loading are contained in Table VII.

TABLE IV
RESULTS FOR FOUR- AND EIGHT-LUMP THERMAL FEEDBACK MODELS
FOR STEP INPUTS OF REACTIVITY

Reactivity (c)	Peak Power (MW)		Time at Peak Power (sec)		Energy Release to Peak (MW-sec)	
	4 Lumps	8 Lumps	4 Lumps	8 Lumps	4 Lumps	8 Lumps
20	0.0577		469		5.22	
40	0.295	0.297	148	146	8.29	7.72
60	0.915	0.926	59.9	59.7	11.4	11.2
80	2.28	2.35	22.8	23.1	6.90	7.50
100	5.88	6.02	10.3	10.4	6.67	7.03
120	15.8	16.7	5.89	5.99	9.41	11.1
140	35.7	36.9	4.01	4.05	14.1	15.7
160	62.4	62.9	2.97	2.98	16.2	16.3
180	104.4	103.2	2.40	2.39	21.9	21.0
200	155.4	152.5	2.00	1.99	26.6	25.0

Reactivity (c)	Av Fuel Temp Rise at Peak Power (°C)		Max Av Fuel Temp Rise (°C)		Av Moderator Temp Rise at Peak Power (°C)	
	4 Lumps	8 Lumps	4 Lumps	8 Lumps	4 Lumps	8 Lumps
20	36.2		78.0		0.30	
40	59.1	54.6	158		0.39	0.36
60	81.9	80.0	236	236	0.50	0.49
80	49.5	53.5	312	312	0.29	0.31
100	48.5	50.5	387	387	0.28	0.29
120	69.4	80.3	461	461	0.39	0.46
140	105	116	533	534	0.59	0.66
160	123	123	606	607	0.67	0.68
180	169	160	680	680	0.92	0.87
200	208	194	751	752	1.11	1.04

TABLE V

FRACTION OF FEEDBACK DUE TO
FUEL AT PEAK POWER

Reactivity (c)	4 Lumps	8 Lumps
20	0.974	
40	0.979	0.979
60	0.980	0.980
80	0.981	0.981
100	0.981	0.981
120	0.981	0.981
140	0.981	0.980
160	0.980	0.980
180	0.980	0.980
200	0.979	0.979

TABLE VI
FRACTION OF FUEL FEEDBACK
DUE TO DOPPLER EFFECT

Reactivity (ρ)	4 Lumps	8 Lumps
20	0.755	
40	0.755	0.755
60	0.755	0.755
80	0.755	0.754
100	0.750	0.750
120	0.742	0.744
140	0.736	0.736
160	0.725	0.723
180	0.717	0.715
200	0.707	0.703

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TABLE VII
TRANSIENT CHARACTERISTICS FOR STEP INPUTS OF REACTIVITY

	Unit	\$1.00 Insertion		\$2.00 Insertion	
		Standard	Uranium-Thorium	Standard	Uranium-Thorium
Peak power	(MW)	12.5	6.02	389	152
Time to peak power	(sec)	14.4	10.4	2.82	1.99
Energy release to peak power	(MW-sec)	29.7	7.03	89.7	25.0
Av fuel temp rise at peak power	(°C)	206	50.5	495	194
Max av fuel temp rise	(°C)	526	387	1171	752
Av moderator temp rise at peak power	(°C)	1.18	0.29	2.77	1.04
Feedback due to fuel at peak power ^a		0.930	0.981	0.932	0.979

^aValues for a 20¢ insertion are 0.500 for the standard loading and 0.97 for the uranium-thorium loading.

The principal conclusion to be drawn from these results is that the presence of the fertile thorium considerably retards the transient compared to the standard loading transient results. Melting of the UC_2 kernel (2530°K) is approached in the \$3.00 step input for the standard loading, but a \$5.00 step input is required to achieve the same temperature for the uranium-thorium loading.

The fraction of feedback due to the fuel, at peak power, varies from 30% (10¢ reactivity insertion) to 93% (\$2.00 insertion) for the standard loading, whereas it remains fairly constant at about 98% for all reactivity insertions for the uranium-thorium loading. For the uranium-thorium loading, most of the feedback (70-75%) is due to the Doppler coefficient.