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APPLICATION OF ENDF DATA TO THE AVR REACTOR WITH HIGHLY ENRICHED URANIUM FUEL AND THORIUM FEED

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D. R. Vondy

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Prepared by the
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APPLICATION OF ENDF DATA TO THE AVR REACTOR WITH HIGHLY ENRICHED URANIUM FUEL AND THORIUM FEED

D. R. Vondy

TECHNOLOGY EXCHANGE

This report provides results of analyses performed within a cooperative project involving Oak Ridge National Laboratory (ORNL), Arbeitsgemeinschaft Versuchs Reaktor GmbH (AVR) and Kernforschungsanlage Jülich GmbH (KFA) in reactor physics, performance and safety. This project has been established within the frame of the HTR Implementing Agreement between the U.S. Department of Energy and the Ministry of Research and Technology of the Federal Republic of Germany.

ABSTRACT

Calculations were done applying ENDF data to the German AVR pebble bed reactor at KFA. Several core models were used, and the results obtained with ORNL methods for the multiplication, reaction rates, temperature coefficient of reactivity, and fuel temperature distributions are reported and compared. Only a small difference in multiplication is found for this core in going from ENDF/B-IV to ENDF/B-V cross-section data. The temperature coefficients calculated with the ENDF/B-V are somewhat smaller in magnitude. The worth of control rods was obtained and only a small difference was found with the data, but the calculated results are high compared with experiment. Neutron reaction rates with the key actinides are reported for three-dimensional core calculations.

INTRODUCTION

Calculations were done for the AVR pebble bed reactor located at KFA in Germany. Figure 1 shows a vertical view of the core.

The two primary objectives of this effort were:

1. Comparing the nuclear behavior predicted with ENDF/B-V¹ and with ENDF/V-IV² data, and
2. Testing ORNL methods in use for analysis of high-temperature graphite-moderated reactor performance.

It should be noted that these ORNL methods have been implemented for rather general application, and no tailoring was done for this specific application which would be normally done if the reactor operation was being supported directly. In particular, the thermal hydraulics modeling could be improved. No bias was applied to effect a critical system as would be usual in analysis. Special representation of the voided region above the pebble bed was not done. Care should be taken to avoid absolute interpretation of these results. Thus that model producing a value of the multiplication factor nearest to unity may or may not be best. It will be interesting to compare these results with data from the operation and its supporting analysis effort.

A two-dimensional (R-Z) geometric core model was selected as a reference geometry to satisfy a number of requirements. It is similar to one in use at AVR. The ability to easily extend this description to a three-dimensional model was desirable for calculating control rod worths. Not only was a realistic representation of the detailed core contents desired, but the capability to extend the model to represent the flow of the pebbles for testing methods was of interest. The modeling was also influenced by the requirements for the thermal hydraulics in (RZ) geometry. Three cases were treated with different core contents but with essentially the same geometric arrangement:

1. AVR MODEL — the nuclide concentrations supplied by AVR³ for 100 core regions,
2. CORE-AVERAGE — a simple average of the core contents from AVR,⁴ and
3. CALCULATED — nuclide concentrations were obtained by calculation applying the steady-state continuous-fueling model⁵ for the geometry modeling pebble flow, the results depending on the nuclear data in use. Information about the details of the control over the fresh pebble feed and the recycling were not available leaving the accuracy of this modeling somewhat uncertain.

Additional information about the application of methods, modeling and the nuclear data processing is included in the Appendix.

In some cases results are reported that are subject to interpretation or have large uncertainties and these are shown in parentheses. $1.5 + 13$ equals 1.5×10^{13} .

Values for the temperature coefficients were calculated from results for two states often involving rather large temperature differences using the equation

$$\frac{T \partial k}{k \partial T} \approx \frac{\ln(k_2/k_1)}{\ln(T_2/T_1)},$$

a) Gemessene Bahnkurven und
Absinkprofile

Fig. 1. AVR elevation indicating pebble residence.

and values are simply reported at the average temperature between T_1 and T_2 . Reactivity changes, as for control rod worth, are calculated as $\Delta k/k = \ln(k_2/k_1)$.

Extension of these results to address specific aspects would be possible. Such information as temperature distributions is not practical to include in this short report form, but such data can be made available.

CORE CHARACTERISTICS

Primary calculations were done in (RZ) geometry for 46 MW(t). An initial test was done to indicate the effect of homogenization of the graphite noses that project inward from the radial reflector to hold the control rods. A (θ R) model set up to explore control rod modeling yielded a k of 0.94205 compared with 0.97091 for a one-dimensional (R) geometry model that is similar to the (RZ) cases. This indicates a lower effectiveness of the graphite noses than will be realized with homogenization in the (RZ) models.

The two-dimensional core results obtained by applying the three models and two sets of nuclear data are shown in Table 1 for the reference fuel and an average core graphite temperature of 800°K. Inventories of key materials are shown at the bottom of the table for comparison. A small decrease in the multiplication factor of this core is found in going from ENDF/B-IV data to ENDF/B-V. Data obtained in the FRG⁴ with a recent model are shown in the table for comparison. The current results seem to lack about 0.02 absorption fraction in the fission products for the AVR model in comparison with FRG results, the multiplication is high by a contribution of 0.02 or more from the two-dimensional model, and the effective temperature used is a bit low increasing multiplication.

The geometric representation for the continuous fueling model tends to overestimate the multiplication so the fissile inventory is low for the critical state. The core content of ²³⁸U is considerably lower for this model and the chosen feed enrichment than shown in the German data. The calculated external fissile feed rate was 47.3 grams/day (²³⁵U) or 0.788 grams per fresh pebble with ENDF/B-V data, and the discharge rate 10.9 grams/day fissile.

Thermal hydraulics results are shown in Table 2 for the cases discussed above. Note that trivial differences are found in the fuel temperatures for the different nuclear data, but effects of different modeling are quite significant. The model that distributes average core contents over the core results in extreme elevated temperatures.

TEMPERATURE COEFFICIENTS

The nuclear data were processed at core temperatures of 400, 800, 1200, and 1600 or 2000°K to generate information about the temperature coefficient of reactivity. These calculations were made for gross effects in the core including the effects of temperature on the resonance shielding, thermal spectrum, and leakage. A more negative temperature coefficient of reactivity is associated with raising the temperature in more and more of the graphite in the core. The most positive value (lowest in magnitude) of the coefficient is associated with an increase in only the kernel temperature, the graphite remaining unheated. Here the temperature change is assumed to occur without concern about how this might come about. The core temperature was held at 800°K except when it was varied, and the reflector was held at 600°K.

Table 1. AVR core characteristics

Cross sections		ENDF/B-IV		ENDF/B-V			(FRG) AVR model
Nuclide data	AVR model	Core-average	Calculated	AVR model	Core-average	Calculated	
Multiplication	1.04666	1.03364	1.0	1.04646	1.03276	1.0	
Fissile conversion ratio							
Calculated	0.3186	0.3345	0.3705	0.3258	0.3424	0.3782	
Critical	(0.4103)	(0.4013)	0.3705	(0.4176)	(0.4078)	0.3782	
Peak power density (w/cm ³)							
Core	4.006	3.663	3.598	3.994	3.653	3.625	
Pebble	-	-	10.76	-	-	10.52	
Peak fast flux (>8 x 10 ⁵ eV, n/cm ² -s)							
Core	4.53+13	6.36+13	6.37+13	4.53+13	6.34+13	6.40+13	
Rad. reflector	1.86+13	1.53+13	1.62+13	1.85+13	1.53+13	1.62+13	
Fraction Absorption							
²³² Th	0.1378	0.1429	0.1734	0.1400	0.1453	0.1759	0.1575
²³³ Pa	0.0015	0.0016	0.0025	0.0015	0.0016	0.0026	0.0016
²³³ U	0.0986	0.0945	0.1077	0.0990	0.0949	0.1097	0.1063
²³⁴ U	0.0048	0.0048	0.0027	0.0051	0.0052	0.0030	0.0056
²³⁵ U	0.3988	0.3977	0.3720	0.3952	0.3940	0.3665	0.4157
²³⁶ U	0.0074	0.0076	0.0063	0.0075	0.0077	0.0063	0.0087
²³⁸ U	0.0171	0.0182	0.0033	0.0171	0.0183	0.0032	0.0073
²³⁹ Pu	0.0058	0.0061	0.0029	0.0058	0.0062	0.0029	0.0071
²⁴⁰ Pu	0.0026	0.0026	0.0009	0.0026	0.0027	0.0009	0.0027
²⁴¹ Pu	0.0021	0.0022	0.0007	0.0022	0.0023	0.0008	0.0025
²⁴² Pu	0.0002	0.0002	-	0.0002	0.0002	-	0.0002
²⁴³ Am	0	0	-	0	0	-	-
Fission Products	0.0847	0.1046	0.0847	0.0835	0.1040	0.0834	0.1037
Inventory (kg)							
²³² Th	515.1	524.9	578.0		577.7	551.5	
²³³ U	7.32	7.30	7.13			7.22	7.81
²³⁵ U	31.40	31.49	25.46			25.26	33.18
²³⁸ U	27.75	30.24	5.17		5.13	31.54	
Fissile (including ²³³ Pa)	39.40	39.54	33.40		33.31	41.76	

Table 2. AVR thermal hydraulics results

Cross sections		ENDF/B-IV			ENDF/B-V		
Nuclide data		AVR model	Core-average	Calculated	AVR model	Core-average	Calculated
Outlet coolant temperature 850°C, inlet 265°C							
Bed pressure drop (ATM)		0.0499	0.0542	0.0520	0.0499	0.0542	0.0520
Temperatures (°C)							
Peak coolant outlet		944.	1184.	1142	943.	1183.	1137.
Mean fuel		632.	662.	633.	632.	662.	633.
Effective fuel ^a		645.	654.	651.	645.	654.	650.
Peak fuel		1011.	1232.	1234.	1010.	1230.	1227.
Fraction fuel							
>1200°C			0.0011	0.0003		0.0011	0.0003
>1150			0.0032	0.0025		0.0031	0.0022
>1100			0.0068	0.0061		0.0067	0.0058
>1050			0.0114	0.0123		0.0113	0.0118
>1000		0.0025	0.0168	0.0258	0.0024	0.0167	0.0254
>950		0.0598	0.0249	0.0513	0.0597	0.0247	0.0509
>900		0.127	0.0374	0.0898	0.126	0.0373	0.0895
>850		0.192	0.0507	0.136	0.192	0.0506	0.136
>800		0.253	0.0748	0.198	0.253	0.0747	0.198
Outlet coolant temperature 950°C, inlet 265°C							
Bed pressure drop (ATM)		0.0396	0.0417	0.0408	0.0396	0.0417	0.0420
Temperatures (°C)							
Peak coolant outlet		1058.	1323.	1299.	1058.	1352.	1295.
Mean fuel		687.	723.	690.	687.	723.	690.
Effective fuel ^a		701.	713.	710.	701.	713.	709.
Peak fuel		1123.	1397.	1393.	1122.	1398.	1387.
Fraction fuel							
>1300°C				0.0024			0.0022
>1250			0.0063	0.0055		0.0064	0.0053
>1200			0.0098	0.0100		0.0098	0.0097
>1150			0.0143	0.0187		0.0144	0.0185
>1100		0.0115	0.0189	0.0337	0.0114	0.0190	0.0336
>1050		0.0668	0.0272	0.0592	0.0667	0.0273	0.0592
>1000		0.124	0.0381	0.0928	0.124	0.0381	0.0929
>950		0.181	0.0493	0.131	0.181	0.0493	0.131
>900		0.232	0.0667	0.180	0.232	0.0666	0.180
>850		0.288	0.0981	0.244	0.287	0.0975	0.245
>800		0.345	0.221	0.315	0.345	0.221	0.316

^aPower density weighted.

Auxiliary one-dimensional core traverses were done applying 28 groups with discrete ordinates or 4-group diffusion theory using collapsed data from the discrete ordinates cases. An axial leakage (buckling) component was included. Multiplication values obtained for a temperature change of the whole fuel element using ENDF/B-V data are shown here using two diffusion models (A) and (B),

Temperature (°K)	Multiplication		
	Discrete ordinates	Diffusion	
		(A)	(B)
400	1.05847	1.08923	1.01719
800	1.01361	1.04392	0.97091
1200	0.99342	1.01880	0.94443
1600	0.97482	0.99544	0.91978

A primary cause for k to be higher in the diffusion (A) cases than the discrete ordinates cases has not been identified. One difference of the (B) from the (A) cases was the use of a larger axial buckling term. These results lead to the following values of $T\partial k/k\partial T$,

Temperature (°K)	Coefficient		
	Discrete ordinates	Diffusion	
		(A)	(B)
600	-0.0625	-0.0613	-0.0684
1000	-0.0496	-0.0601	-0.0682
1400	-0.0657	-0.0806	-0.0919

The average of all of these values is -0.0676.

Comparable values from discrete ordinates cases using ENDF/B-IV data are -0.0558 at 600°K and -0.0512 at 1000°K. (The 1600°K processed ENDF/B-IV data seems to be bad.)

Partial heating of the fuel elements was also considered, namely heating of only the kernel (fueled particles) or of the inner fueled part of the pebbles. ENDF/B-V results are shown below [discrete ordinate and diffusion (A) results were averaged],

Portion heated	$T\partial k/k\partial T$			$\partial k/k\partial T \times 10^5 (^\circ\text{C}^{-1})$		
	600°K	1000°K	1400°K	600°K	1000°K	1400°K
Kernel only	-0.0135	-0.0140	-0.0138	-2.25	-1.40	-1.00
Fueled part	-0.0318	-0.0383	-0.0491	-5.30	-3.83	-3.51
Whole pebble	-0.0619	-0.0548	-0.0732	-10.3	-5.48	-5.22

Note that heating of the graphite contributes considerably to the temperature coefficient.

Information about the temperature coefficient that was obtained from two-dimensional (RZ) diffusion theory calculations for a temperature rise of the whole fuel element is shown in Table 3. Note that the ENDF/B-V data produces coefficients that are lower in magnitude.

Table 3. Temperature coefficient information^a

Cross sections		ENDF/B-IV		ENDF/B-V		
Nuclide data	AVR model	Core-average	Calculated	AVR model	Core-average	Calculated
Multiplication at temperature (°K)						
400	1.09788	1.08640	1.05117	1.09285	1.07486	1.04414
800	1.04666	1.03365	1.0	1.04645	1.03276	1.0
1200	1.01145	0.99783	0.96502	1.01800	1.01098	0.97346
1600	—	—	—	0.99228	0.99126	0.94875
2000	0.95217	0.93835	0.90324	0.96816	0.97203	0.92544
Coefficient $T\partial k/k\partial T$ at temperature (°K), total						
600	-0.0689	-0.0714	-0.0720	-0.0626	-0.0576	-0.0623
1000	-0.0844	-0.0870	-0.0878	-0.0680	-0.0526	-0.0663
1400	—	—	—	-0.0890	-0.0685	-0.0894
1600	-0.1182	-0.1203	-0.1295	-0.0983	-0.0769	-0.0990
1800	—	—	—	-0.1102	-0.0878	-0.1115
Coefficient $\partial k/k\partial T \times 10^5$ °C ⁻¹ at temperature (°K), total						
600	-11.48	-11.90	-12.00	-10.43	-9.61	-10.38
1000	-8.44	-8.70	-8.78	-6.80	-5.26	-6.63
1400	—	—	—	-6.35	-4.89	-6.38
1600	-7.39	-7.52	-8.09	-6.14	-4.81	-6.19
1800	—	—	—	-6.12	-4.88	-6.19

^aLacking good ENDF/B-IV data at 1600°K, cases were run at 2000°K and the coefficients determined from 1200 and 2000°K data appropriate to 1600°K for comparison.

For comparison with these results, values calculated at the AVR for $\partial k/k\partial T$ are -5.94×10^{-5} °C⁻¹ at a coolant outlet temperature of 850°C and -5.60×10^{-5} at 950°C outlet temperature.⁴

SUPPLEMENTAL INFORMATION

The core calculation with the steady-state continuous-fueling model was also done using the calculated fuel temperature distributions and a simple linear interpolation of nuclear data between that for 400°K and for 1200°K. Results for a 20% power level increase are shown in Table 4 based on a reference outlet coolant temperature of 850°C. Conditions are shown without and with an associated increase in the coolant flow rate. These results are for a stabilized flow condition, not instantaneous. The results for a power level increase were also obtained with an available arctangent fit to the same two ENDF/B-V cross section sets. The reference k was 1.00559, and a bit smaller coefficients $P\partial k/k\partial P$ were obtained: -0.0295 with fixed coolant flow and -0.00172 with the coolant flow increased.

Table 4. Results obtained for a 20% power level increase

Data	ENDF/B-IV		ENDF/B-V	
State	k	$P\partial k/k\partial P$	k	$P\partial k/k\partial P$
Reference	1.00829	—	1.00722	—
Elevated power, fixed coolant flow	1.00116	−0.0389	1.00143	−0.0316
Elevated power, coolant flow proportionally increased	1.00791	−0.00207	1.00688	−0.00185

CONTROL ROD WORTH

Calculations were done for highly enriched uranium fuel with thorium to estimate the reactivity worth of the insertion of the four control rods. The AVR MODEL nuclide concentrations were used for the three-dimensional calculations. Certain preliminary calculations were first done in (θ R) geometry to study the modeling. In these calculations 0.1 was added to the diffusion coefficients in the core to simulate streaming in the voids.

Exploratory Two-dimensional Results

The model is a one-eight core (θ R) traverse simulating the conditions of full removal or insertion of all four 8.4-cm diameter rods. The CORE-AVERAGE nuclide concentrations were used and the four-group cross sections came from ENDF/B-V data at a reference temperature of 800° K except where shown otherwise.

Consider that the surface area per unit length of an 8.4 cm diameter control rod is $\pi D = 8.4 \pi = 26.39 \text{ cm}^2$. Modeling this with a 10 x 10 meshpoint grid in (θ R) geometry yields the following effective surface areas for a black neutron absorber at thermal energy,

Radial Dimension (cm)	6.0	7.0	8.0
Radians			
0.064	24.8	26.8	28.8
0.072	26.4	28.4	30.4
0.080	28.0	30.0	32.0

Test calculations were done in (θR) geometry to show the effect on the rod worth of varying these dimensions. Results obtained for $\partial k/k$ for rod insertion are

Radial Dimension (cm)	6.0	7.0	8.0
Radians			
0.064	-0.0708	-0.0720	-0.0732
0.072	-0.0717	-0.0726	-0.0734
0.080	-0.0726	-0.0730	-0.0736

These results indicate that due to a high thermal flux suppression, there is but a modest dependence of the results on the discrete representation. Note that increasing the rod size also increases the volume increasing epithermal absorptions. A 7.0-cm radial dimension and 0.072 radians were used for the discrete representation in three dimensions.

Of interest was the consequence of a coarse representation in the fueled core along the θ coordinate. The high cost of obtaining three-dimensional problem solutions prompts the use of as few meshpoints as necessary. Test calculations produced the following results in (θR) geometry,

Fueled Meshpoints	Radians Per Point	Rods Out k	Rod Worth ($\partial k/k$)
5	0.115	0.94508	-0.0723
7	0.082143	0.94517	-0.0725
10	0.0575	0.94525	-0.0726
15	0.038333	0.94522	-0.0725

A significant effect was not found, and eight points were used in the three-dimensional model.

A calculation was done to check the magnitude of the error from a discrete arrangement of meshpoints. A reference case was rerun doubling the numbers of meshpoints along each coordinate. Note that this was done to reveal the discretization error, and was not an attempt to test the representation of the control rod. The resulting multiplication for the rods out state was slightly higher and the control rod insertion worth was just under 1% larger in magnitude.

Calculations at three temperatures produced the following results,

Core temperature ($^{\circ}K$)	400	800	1200
k , rods out	0.99105	0.94525	0.91960
k , rods inserted	0.92727	0.87910	0.85210
$\Delta k/k$ rod insertion	-0.0665	-0.0726	-0.0762

It is noted that these results show an increase in the control rod worth with increase in temperature. This infers an increase in the amount of reactivity associated with temperature change when the rods are inserted compared to the situation with them out.

The U^{235} concentration was increased 25% yielding the following results,

Core Temperature ($^{\circ}$ K)	400	800	1200
Multiplication			
Rods Out	1.05136	1.00607	0.98031
Rods In	0.98800	0.94004	0.91277
$\Delta k/k$	-0.0622	-0.0679	0.0714

These are somewhat lower. It appears very desirable to represent the actual contents of the reactor and a desired range of applicability in multiplication.

There was some concern about modeling epithermal absorptions by the control rods. Cross sections were not made available for all of the materials. Test calculations were done varying the B^{10} concentration. Note that effects in the thermal range are not affected directly by this since a black absorber rod surface condition was used. The following results were obtained relative to a rods out condition of $k = 0.94520$, where the concentration is for smearing over the discrete area,

B^{10} (A/Bn-cm)	Rod In k	Rod Worth ($\Delta k/k$)
0	0.88742	-0.0631
0.0002	0.88430	-0.0666
0.0006	0.88069	-0.0707
0.002	0.87573	-0.0763
0.006	0.87144	-0.0812
2.0	0.87910	-0.0725

The result at a high B^{10} concentration is a little surprising. Adding pure absorption at any energy can only reduce multiplication. Evidently there are subtle transport effects. Epithermal effects are significant.

Information is reported here on effects from adjusting the diffusion coefficients, useful to simulate neutron streaming:

Adjustment in fueled core	NO	YES	YES
Adjustment in non-fueled	NO	NO	YES
k , rods out	0.96012	0.94525	0.94428
$\Delta k/k$, rod insertion	-0.0722	-0.0726	-0.0980

Thus there is but a modest effect from such a small change in the fueled core, but rather a large effect associated with changing it in the reflector. This latter effect seems to come primarily from increased transmission to the control rods. Modeling of the geometry in the vicinity of the control rods seems to be of considerable importance.

Results for Control Rod Worth

Results obtained from three-dimensional AVR core calculations using AVR MODEL nuclide concentrations are given for reactivity worth of the four 312.4 cm long control rods,

Cross Section Data:	ENDF/B-IV		ENDF/B-V	
Temperature (°K)	400	800	400	800
Reference k	1.07611	1.02650	1.07080	1.02545
Rods Inserted, k	0.98462	0.93300	0.98026	0.93149
$\Delta k/k$	-0.0889	-0.0955	-0.0883	-0.0961

The three-dimensional control rod worth is larger than was obtained with the two-dimensional model, likely due largely to the longer length of the rods than the effective fueled bed height. The calculated result reported by AVR for a temperature of 403°K is -0.0906, slightly higher than the value of -0.08873 they obtained with an earlier model.⁴ Values inferred from measurements are expected to be made available.

An inverse kinetics measurement of the total control rod bank worth gave 0.0819 $\partial k/k$ at temperatures corresponding to full power operation, outlet coolant temperature 850°C. A systematic inaccuracy may be the consequence of location of the flux measurement point outside of the reflector. A cold measurement of total rod bank worth by N₂ replacement gave $\partial k/k$ of 0.0782 at 130°C. The AVR staff believe inaccuracies are also associated with this method.¹⁰ The calculated values are sufficiently higher than measured worths to invite further investigation. At the least, the 4-group representation should be improved by solving the three-dimensional problem with 28-group nuclear data to improve the representation over the thermal energy range.

Values of $T\partial k/k\partial T$ are estimated from the above data for 600°K,

Cross Section Source	ENDF/B-IV	ENDF/B-V
Condition		
Rods Out	-0.0681	-0.0624
Rods In	-0.0777	-0.0736

ADDITIONAL REACTION RATE DATA

Reaction rate data from three-dimensional calculations are presented here. These include data for the state with the control rods removed and also for a continuous fueling problem described below. The results shown in Table 5 are for a reference temperature of 800°K.

**Table 5. AVR core characteristics from
three-dimensional calculations**

Cross sections	ENDF/B-IV	ENDF/B-V	
Nuclide data	AVR model	AVR model	Calculated
Multiplication	1.02650	1.02545	1.0
Fissile conversion ratio			
Calculated	0.3268	0.3348	0.3859
Critical	(0.3798)	(0.3861)	-
Peak power density (w/cm ³)			
Core	5.616	5.598	3.882
Pebble	-	-	10.30
Peak fast flux (>8 x 10 ⁵ eV,n/cm ² -s,x10 ⁻¹³)			
Core	5.41	5.40	6.89
Rad. reflector	2.30	2.30	1.96
Fraction absorption			
²³² Th	0.13871	0.14126	0.16160
²³³ Pa	0.00151	0.00153	0.00217
²³³ U	0.09788	0.09835	0.09084
²³⁴ U	0.00478	0.00513	0.00228
²³⁵ U	0.39182	0.38824	0.37389
²³⁶ U	0.00748	0.00756	0.00618
²³⁸ U	0.01736	0.01747	0.01971
²³⁹ Pu	0.00573	0.00580	0.01574
²⁴⁰ Pu	0.00255	0.00257	0.00454
²⁴¹ Pu	0.00205	0.00218	0.00374
²⁴² Pu	0.00020	0.00023	0.00021
²⁴³ Am	-	-	0.00005
Fission Products	0.08378	0.08312	0.08012
Inventory (kg)			
²³² Th	510.5		550.5
²³³ U	7.23		6.43
²³⁵ U	31.35		27.82
²³⁸ U	27.55		31.47
Fissile (including ²³³ Pa)	39.26		35.31

A three-dimensional model of continuous fueling was solved. The intent was to improve the two-dimensional results. Only ENDF/B-V cross sections were used at a reference temperature of 800° K. The three-dimensional model explicitly represents the reflector noses into which the rods insert. Modeling of the pebble flow was somewhat poorer than was realized with the two-dimensional model, however, because the volumes assigned to nuclide concentrations were not changed from what were used for the AVR MODEL. What was done in two dimensions should more accurately represent the local residence of the pebble fuel elements, although rather large zones of material were involved in the outer part of the core. In each case, the continuous fueling model involved 990 subzones of fuel material (no azimuthal variation), and the fissile feed rate was established as necessary for the core to be critical with the passage of pebble fuel elements represented explicitly. Th^{232} and U^{238} feed rates were selected for the three-dimensional case to cause the core loading of these fertile nuclides to agree with the contents reported from the FRG calculations.

The fission product poisoning is naturally of interest and just how much different it would be with the three-dimensional model. Reaction rates in the actinides are of interest. The calculated U^{235} feed rate of 46.4 grams/day compares with the two-dimensional result of 47.3 grams/day discussed earlier.

Regarding the Multiplication

The multiplication factor for the AVR core calculated with ENDF/B-V cross sections is found to be 0.025 high. Part of this may be a low estimate of the core neutron leakage through the gas filled space above the pebble bed. Part of it may be the approximations made of the location of materials in the core. Compared with results obtained in the FRG, the current estimate of fission product absorptions seems to be low. ENDF data was not used for this. Some local study should be done to clarify the situation.

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APPENDIX

METHODS

The AVR core calculations were done applying diffusion theory using 4-group cross sections. The VENTURE⁶ neutronics and associated codes^{7,8} were used. Results produced with these methods are reported directly.

The steady-state condition associated with continuous fueling with pebble feed, recycle, and discharge is modeled directly to obtain a solution iteratively to establish the necessary core conditions and the feed composition. Since a global iteration process is used, the results are converged only to a reasonable degree of accuracy. Representative paths of the pebbles are followed and the increase in the residence time with increase in the radial location was modeled. Early models that were tested proved somewhat inadequate. The 100 core regions were subdivided into 990 locations treated explicitly for exposure but homogenized to the 100 locations for the neutronics calculation. The fuel pebble management was assumed to be discharge of 600 pebbles/day and makeup of 60 pebbles/day fresh fuel. A core residence time of 1,585 days was used. The modeling was somewhat complicated with variation in residence time and no fresh fuel elements fed to the central port. The thermal hydraulics calculation treats 990 pebble fuel elements for reference conditions in this model (instead of one for each zone), and then treats these again for zone extreme conditions. Additional effort would likely improve the modeling of the core, its contents, and the representation of the fuel element traverses.

NUCLEAR DATA

Nuclear data in multigroup form from ENDF/B-V¹ was used for the nuclides shown in Table A. Revised ²³²Th and graphite data has replaced the original. The ENDF/B-V data were used from a 28-group library collapsed from the original 228-group set of data. These data were supplemented with data from a 123-group library in local use as indicated in the table, making up a set useful for core analysis, generally not ENDF data. Also, data available were used from ENDF/B-IV,² substituted as shown. A 27-group set of ENDF/B-IV cross sections were available that were collapsed from an original 218-group structure. Revised ENDF/B-IV graphite data is in use. Emphasis was placed on the actinides. Local processing of fission product point data is incomplete and under further study. Much resonance data is involved impacting application since shielding changes with exposure.

The cases only reconstruct the German fission product inventory in part, so some differences will result here. The properties of a pseudo lumped fission product in use at ORNL were used for the German data that actually should be different. A small amount of 1/v absorber was included to represent impurities in the graphite. (Nuclides ¹³⁵I and ¹⁴⁷Nd identify pseudo data in common local use to allow slow transients to be followed).

Table A. Nuclear data source for nuclides

Number	Nuclide	ENDF/B-V (basis)	Supplemented (123 group library)	ENDF/B-IV (replaced)
1	²³² Th	X		X
2	²³³ Pa	X		X
3	²³³ U	X		X
4	²³⁴ U	X		X
5	²³⁵ U	X		X
6	²³⁶ U	X		X
7	²³⁸ U	X		X
8	²³⁷ Np	X		X
9	²³⁹ Np		X	
10	²³⁹ Pu	X		X
11	²⁴⁰ Pu	X		X
12	²⁴¹ Pu	X		X
13	²⁴² Pu	X		X
14	²⁴³ Am	X		
15	⁸² Kr	X		
16	⁸³ Kr	X		
17	¹⁰³ Rh	X		
18	¹⁰⁵ Rh		X	
19	¹⁰⁹ Ag	X		X
20	¹³⁵ I		X	
21	¹³¹ Xe	X		
22	¹³³ Xe		X	
23	¹³⁵ Xe	X		
24	¹³³ Cs	X		
25	¹³⁴ Cs		X	
26	¹³⁵ Cs		X	
27	¹⁴³ Nd		X	
28	¹⁴⁵ Nd		X	
29	¹⁴⁷ Nd		X	
30	¹⁴⁷ Pm		X	
31	¹⁴⁸ Pm		X	
32	^{148m} Pm		X	
33	¹⁴⁹ Pm		X	
34	¹⁴⁹ Sm	X		
35	¹⁵⁰ Sm		X	
36	¹⁵¹ Sm		X	
37	¹⁵² Sm		X	
38	¹⁵³ Eu	X		
39	¹⁵⁴ Eu	X		
40	¹⁵⁵ Eu		X	
41	FP1		X	
42	FP2		X	
43	C	X		X
44	C(R)	X		X
45	O	X		X
46	He	X		X
47	¹⁰ B	X		X
48	1/v	X		X

CROSS SECTION PROCESSING

Resonance shielding was done with a single resonance model⁹ at the fuel kernel level applying a Dancoff factor of 0.40 to account for the environment. A one-dimensional discrete ordinates⁹ cell calculation was done for the fuel element (2.5-cm radius fueled sphere inside) in the group structure of the data. The results were weighted with the cell neutron flux to produce effective cross sections (27 or 28 group). Then a one-dimensional discrete ordinates⁹ core traverse was done and the data were then collapsed with the flux to generate 4-group data subsequently used with diffusion theory.

Note that calculations treating the fuel cell and one or more core dimensions could be done in a finer group structure, as in 227 groups for the processed ENDF/B-V data. This was not done, so we refer to earlier work where more elaborate representation was done for comparison. Nor were multidimensional calculations done treating more than four groups.

MODELING

The diffusion coefficients in the core were increased by 0.01 to approximate the effect of neutron streaming. A maximum value of the diffusion coefficient allowed was 60 cm^2 . It should be noted that the modeling used for calculating the temperature coefficient of reactivity involved a discrete temperature change over the whole core. (What happens with a power level change or upon reactivity addition is somewhat more complicated.) Temperature variation over the reactor core was not considered in these early calculations. (Holding the reflector at 600°K causes the results to be incomplete in regard to the total defect, for example.)

A noteworthy limitation in the modeling is the thermal hydraulics application. The pebbles piled at the top of the bed and the conical core bottom were only crudely modeled. Excessive fuel temperature peaking may result from the approximation that is a distortion. Also, the presence of the control rod holder nose pieces was ignored, increasing the cross sectional area as handled. Whereas with the continuous fueling model the fluence was calculated for correlating the thermal conductivity in the fuel element pebbles, a rather high number of $2.5 \times 10^{21} \text{ n/cm}^2 > 1.85 \times 10^5 \text{ eV}$ was used for the other models. Further effort would be required to assess the situation if results of a high quality are needed; indeed some methods improvement is desirable and might be found to be essential.

In some cases equilibrium ^{135}Xe was calculated when the diffusion theory neutronics problem was solved. This was done on a region-average basis applying an average total fission yield fraction.

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