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PHYSICS EVALUATIONS AND APPLICATIONS

QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JANUARY 31, 1978

APPLIED TECHNOLOGY

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PHYSICS EVALUATIONS AND APPLICATIONS
QUARTERLY PROGRESS REPORT FOR PERIOD ENDING

JANUARY 31, 1978

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1.0 OBJECTIVES

- 1.1 Advanced System analyses of the heterogeneous core design concept for small and large LMFBRs.
- 1.2 Full implementation and utilization of the ERDA supported three dimensional burnup diffusion theory code, VENTURE, at ARD for the design verifications and development of future tools required for LMFBR designs.
- 1.3 Utilization of the ERDA developed two-dimensional kinetics code, FX-2, for the transient analyses and HCDA energetics studies of advanced LMFBR design concepts.
- 1.4 Continued evaluations and applications of the ENDF/B-nuclear data files for general LMFBR analyses.
- 1.5 Continued participation in the Cross Section Evaluation Working Group (CSEWG), the Computer Code Coordination and Method testing committees.

2.0 CURRENT PROGRESS

2.1 EFFECT OF AN ~20% REDUCTION IN Th^{232} CAPTURE CROSS SECTION ON CSDT OF A 1000 MWe HETEROGENEOUS LMFBR

An analysis of the effect of an ~20% reduction in the ENDF/B-IV Th^{232} capture cross section on the nuclear design parameters of a Heterogeneous LMFBR core fueled with recycled uranium and thorium was performed.

The reactor core analyzed was a radial parfait core consisting of 252 fuel assemblies and 241 (97 inner and 144 radial) blanket assemblies. The height of the core was four feet and the axial blanket thickness was 14 inches. The fuel material was recycled uranium and thorium and the blanket material was thorium oxide.

The analysis was performed in two-dimensional diffusion theory using a 23 neutron group cross section set based on ENDF/B-IV data. Compound system doubling times were calculated based on a cycle length of 584 full power days. Table 1 gives a detailed fissile mass breakdown of the ENDF/B-IV Th^{232} and reduced Th^{232} cases. Figure 1 shows the ENDF/B-IV and reduced Th^{232} capture cross section as a function of energy. This Figure was distributed at the February 2, 1978 ZPPR Planning Meeting.

Due to the reduction in Th^{232} capture cross section, the fissile mass required at the SOC and the fissile mass gained were reduced by ~5%. Consequently, the compound system doubling time (CSDT) remained unchanged.

A similar analysis performed with denatured (U^{233} , U^{238}) oxide fuel and denatured ThO_2 blankets with 20% U^{238} showed negligible differences in the fissile mass requirement at SOC, and the fissile gain per year. The SOC and EOC fissile masses and related parameters are given in Table 2. Again, there was essentially no change in the compound system doubling time (CSDT).

2.2 IMPACT ON NUCLEAR DESIGN PARAMETERS OF USING RECYCLED URANIUM IN A LARGE HETEROGENEOUS LMFBR

Using a large heterogeneous LMFBR configuration with $(U^{233}, Th)O_2$ fuel and ThO_2 blankets, the effect of recycled uranium on fissile enrichment, breeding ratio and compound system doubling time was determined by substituting recycled uranium for the fuel material. The LMFBR design analyzed consisted of a four foot core with 14 inch thick axial blankets containing 252 fuel assemblies, 241 blanket assemblies (97 inner and 144 radial) and 18 control assemblies. Comparative results were obtained using two dimensional diffusion theory and 23 group ENDF/B-IV neutron cross section data. The results of the analysis are summarized in Table 3.

Comparing the $(U^{233}, Th)O_2$ and (Recycled U, $Th)O_2$ cases, one observes that the use of recycled uranium in the fuel lowered the compound system doubling time approximately 30%, due primarily to improved breeding performance. The reactivity change due to depletion is approximately 13% less for the recycled case, which required adjustments in the SOC fissile inventory. An additional case was analyzed wherein the recycled uranium was utilized with a denatured blanket. This reduced the compound system doubling time to 19.5 years and the SOC fissile inventory to 4338 KG. This assumes that the total fissile mass includes Pu^{239} .

2.3 RELATIVE MERITS OF HOMOGENEOUS VERSUS HETEROGENEOUS U-Th OXIDE FUELED CORE DESIGNS

Studies at W-ARD^(1,2) showed that a reduction in the sodium void reactivity by a factor of two or three is feasible in comparison with the conventional $(Pu-U)O_2$ homogeneous core designs. This significantly reduced sodium void reactivity in a heterogeneous core design (\$3 void worth in fuel assemblies) makes the HCDA energetic benign even under pessimistic assumptions about fuel-coolant interactions, during hypothetical under-cooled transients without scram, see References 2, 3, and 4.

LMFBRs using the (U-Th) fuel cycle are known to have a small or even negative sodium void reactivity⁽⁵⁾. The significance of the heterogeneous core designs in reducing the sodium void reactivity therefore does not necessarily apply to (U-Th) oxide designs or (U-Th) carbide fueled LMFBRs. A comprehensive study of the relative merits of homogeneous and heterogeneous (U-Th) oxide fueled LMFBR designs was performed to identify the inherent differences between the two design concepts. As a part of this study, the beginning of life fissile requirement, the equilibrium cycle breeding ratio and the compound system doubling time were computed as a function of fuel volume fraction for both the core designs.

Due to the effect of the slow Pa^{233} decay on the built-in reactivity and the increased burnup requirements, a core height of four feet was chosen. However, three foot cores were also analyzed to compare their merits relative to the four foot design. The selection of a three foot core was based on the assumption that the linear heat rate limit could be raised from ~ 20 kw/ft to ~ 25 kw/ft due to the higher melting point of ThO_2 compared to that of UO_2 . A complete evaluation of the performance of ThO_2 fuel pins at high temperatures is necessary, however, to confirm the applicability of the higher heating rate limit assumption.

The heterogeneous reactor analyzed consisted of 252 fuel assemblies and 241 blanket assemblies (97 inner and 144 radial). The axial blanket thickness was 14 inches. The homogeneous core had 276 fuel assemblies and 168 blanket assemblies. The homogeneous core had three fissile enrichment zones, while the heterogeneous core had the same fissile enrichment in all fuel zones. Both pure U^{233} and recycled uranium with ~ 22 a/o U^{234} were used as fuel. The analyses were performed using two dimensional diffusion theory with a 23 neutron energy group cross section set based on ENDF/B-IV data. Both the cores were designed to produce 1000 MWe. The beginning of life fissile mass, the breeding ratio and the compound system doubling time for both the homogeneous and heterogeneous core designs are shown in Figures 2 through 4 as a function of fuel volume fraction. As shown in Figure 4, the homogeneous design's CSDT was extremely sensitive

to variations in the fuel volume fraction, whereas the heterogeneous cases remained fairly stable. Figure 2 shows that a homogeneous core requires about 20% less fissile material at the start of life compared to that of a heterogeneous core, but the compound system doubling time of a homogeneous core is longer than that of heterogeneous core due to its lower breeding ratio (Figure 3). However with large fuel volume fractions it is possible to achieve a compound system doubling time for a homogeneous core in the range of 30 years.

Utilization of recycled uranium instead of pure U^{233} has the following effect:

- (1) Decreases the start of life fissile mass by 1.5% in a heterogeneous core and by 2.5% in a homogeneous core,
- (2) Increases the breeding ratio ~ 0.04 in a heterogeneous core and ~ 0.025 in a homogeneous core,
- (3) Decreases the compound system doubling time by 10-12 years in both core designs.

Decreasing the height of the core from four feet to three feet has the following advantages in a (U,Th) oxide fueled core:

- (1) Decreases the start of life fissile mass by 20% in a heterogeneous core and by 17% in a homogeneous core;
- (2) Increases the breeding ratio by a maximum of 0.03 in both the homogeneous and heterogeneous cores;
- (3) Decreases the compound system doubling time by as much as 12 years in a heterogeneous core and ~ 20 years in a homogeneous core.

It can be concluded that it is possible to achieve a compound system doubling time in the range of 30-40 years in both the homogeneous and the heterogeneous core by utilizing recycled uranium (with ~ 22 a/o U^{234} content) and/or reducing the core height to three feet. The maximum breeding ratio in the homogeneous core (~ 1.17) can be achieved only by using recycled uranium in a three foot core with a very high fuel volume fraction ($\sim 54\%$). However, a breeding ratio of 1.20 can be achieved in a four foot heterogeneous core using recycled uranium with a fuel volume fraction of only 44%.

2.4 COMPARISON OF RUNNING TIME AND COST BETWEEN 2DB and VENTURE FOR SMALL RZ PROBLEM

A small RZ problem (35 x 14 mesh, 9 group) was run on both VENTURE and the local 2DB neutronics code to compare running time and cost. A small problem was chosen for comparison so that VENTURE could utilize any of its data handling modes to analyze the problem. Table 4 shows the resulting times and cost of analysis. As shown in the table, 2DB was considerably faster and cheaper than VENTURE, in the mode of operation chosen. This mode consisted of the VENTURE automated procedures for extrapolation, initialization acceleration and data handling with the number of inner iterations per group fixed at four, consistent with ARD's present strategy for three dimensional VENTURE analyses.

The cost differential between the two codes appears to be directly related to differences in the central processing time required to obtain the eigenvalue and flux solution. It was observed that a large difference existed between the overrelaxation coefficient used by 2DB and the values calculated in VENTURE. Rerunning the VENTURE analysis with the 2DB overrelaxation value reduced the number of outer iterations to 19 and the CP time for the outers to 14.46 seconds. Removing the restriction on the number of inner iterations did not alter the results. Similar comparisons will soon be made for a larger problem.

2.5 VENTURE OUTER ITERATION COST ANALYSIS

Initial cost analysis of an outer iteration in the VENTURE code was completed. The problem chosen for analysis was a large (84 x 42 x 33 mesh), 9 group eigenvalue problem utilizing the "multi-level" mode of data storage. Table 5 gives a breakdown by cost category of the flux calculation overlay.

It should be noted from Table 5 that the fraction of the total cost attributable to the use of direct access disc files includes the rotating mass storage and disc access categories (26.6%) plus a fraction of the cost attributable to the other four categories. Therefore, the flux calculation overlay, and the use of direct access disc files within the overlay, are the primary areas for optimization within the code.

2.6 LCCEWG SECOND BENCHMARK CALCULATIONS

The second benchmark calculations for the Large Core Code Evaluation Working Group (LCCEWG) were completed during this quarter. The primary objective of this benchmark was to quantify the accuracy of current neutronics codes and methods, as employed by the various LCCEWG participants, for calculating the sodium void reactivity worth in large, homogeneous LMFBRs. See Appendix A for additional details.

TABLE 1

Effect of an ~20% Reduction in Th^{232} Capture
 Cross Section on CSDT of a 1000 MWe Heterogeneous
 LMFBR with Recycled Uranium Fuel and ThO_2 Blankets

	With ENDF/B-IV Cross Sections	With Reduced Th^{232} Capture Cross Section (Refer Figure 1)
Fuel Material	(Recycled Uranium, Thorium) Oxide	(Recycled Uranium, Thorium) Oxide
Blanket Material	Thorium Oxide	Thorium Oxide
SOC Mass Inventory (kg)		
U^{233}	4868.0	4636.0
U^{235}	195.8	186.5
U^{234}	1416.6	1349.4
Fissile Enrichment	17.8%	17.0%
Average Breeding Ratio	1.20	1.19
EOC Mass Inventory (Kg)		
Core U^{233}	4178.0	3934.0
Core U^{235}	218.8	210.2
Core U^{234}	1390.8	1325.4
Axial Blanket U^{233}	133.0	134.0
Radial Blanket U^{233}	297.8	295.6
Inner Blanket U^{233}	537.6	533.2
Burnup Swing/Cycle (% Δk)	2.3	2.6
CSDT (years)	42	42

TABLE 2

Effect of an ~20% Reduction Th²³² Capture
Cross Section on the CSDT of a Large Heterogeneous
LMFBR with Denatured Fuel and Blankets.

	With ENDR/B IV Cross Sections	With Reduced Th ²³² Capture Cross Section (Refer Figure 1)
Fuel Material	(U ²³³ , U ²³⁸)O ₂	(U ²³³ , U ²³⁸)O ₂
Blanket Material	(Th.8U.2) O ₂	(Th.8U.2) O ₂
SOC Fissile Mass (kg)	4340.0	4286.0
Fissile Enrichment	14.9%	14.7%
Average Breeding Ratio	1.29	1.28
EOC Mass Inventory (kg)		
Core U ²³³	3052.0	3000.0
Core U ²³⁵	3.3	3.4
Core Pu ²³⁹	709.0	721.0
Axial Blanket U ²³³	113.6	111.3
Axial Blanket Pu ²³⁹	25.7	26.7
Radial Blanket U ²³³	258.2	248.8
Radial Blanket Pu ²³⁹	58.2	59.5
Inner Blanket U ²³³	461.2	448.8
Inner Blanket Pu ²³⁹	107.4	111.1
Burnup Swing/Cycle (% Δk)	3.7	3.7
CSDT (years)	24	24

TABLE 3

Comparative Results of Using Recycled Uranium
in a Large Heterogeneous LMFBR

Fuel Material	(U ²³³ ,Th ²³²) Oxide	(Recycle U,Th ²³²) Oxide	(Recycle U,U ²³⁸) Oxide
Blanket Material	ThO ₂	ThO ₂	Th _{.8} U _{.2} O ₂
SOC Fissile Mass (kg)	5136.0	5064.0	4338.0
Fissile Enrichment (%)	17.9	17.7	15.18
Average Breeding Ratio	1.16	1.20	1.31
Burnup Swing Per Cycle (%Δk) (584 FPD)	2.63	2.32	3.39
Net Fissile Gain Per Year (kg)	91.6	116.6	204.9
CSDT (years)	51.7	40.0	19.5

TABLE 4

SMALL RZ COMPARISON OF 2DB AND VENTURE

	<u>2DB</u>	VENTURE <u>"Core Contained" Mode</u>	
		<u>Calculated Relaxation Coefficients</u>	<u>Input Relaxation Coefficients</u>
CP Time (sec)*	8.0	22.0	17.90
Total Cost (\$)	10.37	24.54	21.29
Fast Core Data Storage Locations	7379	25841	25841
Slow Core Data Storage Locations	15594	25841	25841

*CP = Central Processing Time

TABLE 5

Venture Outer Iteration Cost Breakdown During Flux Calculation
Large Eigenvalue Problem in "Multi-Level" Mode

<u>Cost Category</u>	<u>Percentage Cost During Flux Calculation</u>
CP	10.46
SCM	20.79
LCM	23.45
IO	18.73
RMS	19.13
DA	7.44

CP = Central Processing Time
SCM = Small Core Memory
LCM = Large Core Memory
IO = Input/Output
RMS = Rotating Mass Storage
DA = Disc Accesses

Figure 1. Th^{232} Capture Cross Section

Figure 2. Beginning of Life Fissile Mass Requirement as a Function of Fuel Volume Fraction in a Large LMFBR.

0517-2

Figure 3. Breeding Ratio as a Function of Fuel Volume Fraction in a Large LMFBR

0517-3

Figure 4. Compound System Doubling Time as a Function of Fuel Volume Fraction in a Large LMFBR.

0517-4

3.0 REFERENCES

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APPENDIX A

LARGE CORE CODE EVALUATION WORKING GROUP CALCULATIONS

The second benchmark problem for the Large Core Code Evaluation Working Group (LCCEWG) was completed during January, 1978. The objective of this particular benchmark problem was to quantify the accuracy of current neutronics codes and methods for calculating the sodium void reactivity worth in large, homogeneous LMFBs.

The reactor model is a simple, two enrichment zone homogeneous core with a central control rod channel based on the first benchmark, which is representative of a 1200 MWe LMFB. The RZ model is illustrated in Figure A1. GE-BRD made available to all LCCEWG participants a 21-group ISOTXS file of macroscopic cross sections generated from the ENDF/B-IV library using MINX and TDOWN. The library includes cross sections for the sodium-in and sodium-out configurations based on homogeneous compositions.

The basic calculations were performed in two-dimensional (R,Z) geometry using diffusion theory approximation. The eight basic calculations which were performed are briefly described in Table A.1. Some of the principal results of these eight calculations, which will be used for comparison purposes, are listed in Table A2. Based on these calculated values of k_{eff} , both sodium void worths and control rod worths can be determined. Table A3 gives the sodium void reactivity worths for the three different voiding configurations with the central control rod in and out. Table A4 gives the worth of the central control rod with the whole core flooded and with only the inner core voided.

Additional calculations of the sodium voiding reactivity effect were made using the first-order perturbation theory technique. Tables A5 and A6 compare these results to those obtained from the direct k calculations for the control rod in and control rod out configurations, respectively. For each voided region the sodium void worth is broken down into its leakage, moderation, absorption and fission source terms.

Significant differences exist between the perturbation and direct k calculations for sodium voiding. For total core voiding (inner core and outer core and axial blankets) the perturbation theory results underpredict the direct k calculations by about 7%. This effect is dominated by the discrepancy in the sodium void worth calculated for the inner core zone.

TABLE A1

LCCEWG - Second Benchmark
Description of Diffusion Theory Calculations

(2DB, 21 groups)

Case 1	Control Rod In	-	All Zones Flooded
Case 2	Control Rod In	-	Inner Core Voided
Case 3	Control Rod Out	-	All Zones Flooded
Case 4	Control Rod Out	-	Inner Core Voided
Case 5	Control Rod In	-	Inner and Outer Core Voided
Case 6	Control Rod In	-	Inner and Outer Core and Axial Blankets Voided
Case 7	Control Rod Out	-	Inner and Outer Core Voided
Case 8	Control Rod Out	-	Inner and Outer Core and Axial Blankets Voided

TABLE A2
LCCEWG - Second Benchmark
Principal Results from 21-Group RZ Calculations

<u>Case</u>	<u>Sodium Void</u>	<u>Control Rod</u>	<u>k_{eff}</u> ⁽³⁾	<u>Core Leakage</u> ⁽¹⁾ <u>Excluding CR</u>	<u>Total Blanket Absorption</u> ⁽²⁾
1	None	In	1.00653	0.18264	0.18466
2	IC	In	1.02300	0.20233	0.18673
3	None	Out	1.00951	0.17968	0.18252
4	IC	Out	1.02702	0.19976	0.18477
5	IC+OC	In	1.02710	0.21867	0.20072
6	IC+OC+AB	In	1.02593	0.21914	0.19838
7	IC+OC	Out	1.03089	0.21560	0.19854
8	IC+OC+AB	Out	1.02975	0.21603	0.19633

(1) Leakge = Fission Source - Absorption

(2) Absorption = Fission + Capture - (n, 2n)

(3) Convergence criteria = 10^{-6} on integral fission source rate

Table A3

LCCEWG - Second Benchmark

Sodium Void Worth (21 group, diffusion theory, 2DB)

Control Rod In

Case 2 - Case 1 = 0.01647 Δk (Inner Core Voided)

Case 5 - Case 1 = 0.02057 Δk (IC+OC Voided)

Case 6 - Case 1 = 0.01940 Δk (IC+OC+AB Voided)

Control Rod Out

Case 4 - Case 3 = 0.01750 Δk (Inner Core Voided)

Case 7 - Case 3 = 0.02138 Δk (IC+OC Voided)

Case 8 - Case 3 = 0.02024 Δk (IC+OC+AB Voided)

Table A4

LCCEWG - Second Benchmark

Control Rod Worth (21 group, diffusion theory, 2DB)

Sodium Flooded

$$\text{Case 3} - \text{Case 1} = 0.00299 \Delta k$$

Sodium Voided from Inner Core

$$\text{Case 4} - \text{Case 2} = 0.00402 \Delta k$$

Table A5
LCCEWG - Second Benchmark

Perturbation Results⁽¹⁾ - Control Rod Out

Inner Core Voiding (Δk)

Fission Source ⁽²⁾	0.21164 E-03	
Absorption ⁽²⁾	0.39387 E-02	
Moderation ⁽²⁾	0.15514 E-01	
Leakage ⁽²⁾	-0.39821 E-02	
Net $\Delta k/k$ ⁽²⁾ (PERT-V)	0.15682 E-01	(-10.4%) ⁽³⁾
Direct k calculation (2DB)	0.17503 E-01	

Outer Core Voiding (Δk)

Fission Source	0.29705 E-03	
Absorption	0.20289 E-02	
Moderation	0.83290 E-02	
Leakage	-0.66715 E-02	
Net $\Delta k/k$ (PERT-V)	0.39834 E-02	(+2.7%)
Direct k calculation (2DB)	0.38779 E-02	

Axial Blanket Voiding (Δk)

Fission Source	0.25050 E-05	
Absorption	0.21752 E-03	
Moderation	0.15214 E-02	
Leakage	-0.26005 E-02	
Net $\Delta k/k$ (PERT-V)	-0.85899 E-03	(-24.8%)
Direct k calculation (2DB)	-0.11421 E-02	

- (1) First order perturbation theory; forward and adjoint fluxes taken from case 3; 21 groups; PERT-V.
 (2) Terms are defined in Section II of BNWL-1162.
 (3) Percent difference as compared to direct k calculation.

Table A6
LCCEWG - Second Benchmark
Perturbation Results⁽¹⁾ - Control Rod In

Inner Core Voiding (Δk)

Fission Source ⁽²⁾	0.20682 E-03	
Absorption ⁽²⁾	0.37231 E-02	
Moderation ⁽²⁾	0.14853 E-01	
Leakage ⁽²⁾	-0.40058 E-02	
Net $\Delta k/k$ ⁽²⁾ (PERT-V)	0.14777 E-01	(-10.3%) ⁽³⁾
Direct k calculation (2DB)	0.16470 E-01	

Outer Core Voiding (Δk)

Fission Source	0.31468 E-03	
Absorption	0.21464 E-02	
Moderation	0.87660 E-02	
Leakage	-0.70283 E-02	
Net $\Delta k/k$ (PERT-V)	0.41987 E-02	(+2.4%)
Direct k calculation (2DB)	0.41006 E-02	

Axial Blanket Voiding (Δk)

Fission Source	0.24539 E-05	
Absorption	0.21226 E-03	
Moderation	0.14943 E-02	
Leakage	-0.25915 E-02	
Net $\Delta k/k$ (PERT-V)	-0.88250 E-03	(-24.6%)
Direct k calculation (2DB)	-0.11713 E-02	

- (1) First order perturbation theory; forward and adjoint fluxes taken from case 1; 21 groups; PERT-V.
 (2) Terms are defined in Section II of BNWL-1162, "PERT-V, A Two-Dimensional Perturbation Code for Fast Reactor Analysis," September, 1969.
 (3) Percent difference as compared to direct k calculation.

FIGURE A1
LCCENG - SECOND BENCHMARK - RZ MODEL

() - NUMBER OF MESH INTERVALS