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Fuel Density, Uranium Enrichment, and Performance Studies for the Advanced Neutron Source Reactor

E. E. Alston, J. C. Gehin, C. D. West

June 1994

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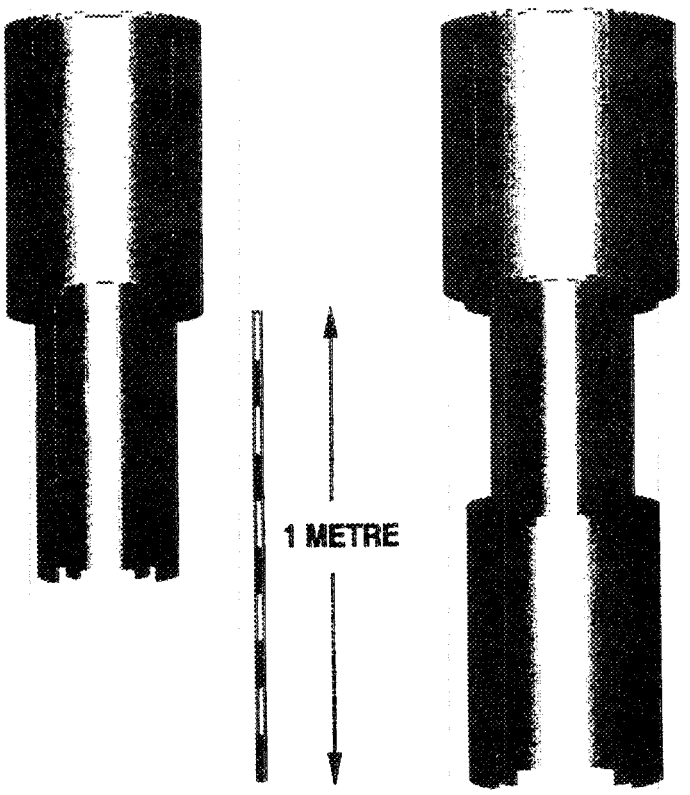

Advanced Neutron Source

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FUEL DENSITY, URANIUM ENRICHMENT, AND PERFORMANCE STUDIES FOR THE ADVANCED NEUTRON SOURCE REACTOR

E. E. Alston, J. C. Gehin, C. D. West

June 1994

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ACRONYMS AND SYMBOLS

ANS	Advanced Neutron Source
BNL	Brookhaven National Laboratory
BOC	beginning-of-cycle
DOE	Department of Energy
EOC	end-of-cycle
HEU	highly-enriched uranium
LEU	low-enriched uranium
MEU	medium-enriched uranium
ORNL	Oak Ridge National Laboratory
RERTR	Reduced Enrichment for Research and Test Reactors

A	a constant, see Eq. (1)
B	a constant, see Eq. (1)
^{235}BOC	mass of ^{235}U at beginning-of-cycle
^{Other}BOC	mass of nonfissile uranium isotopes at beginning-of-cycle
^{Total}BOC	total mass of uranium at beginning-of-cycle
C_0	a constant, see Eqs. (2) , (7)
C_1	a constant, see Eqs. (2) , (7)
C_2	a constant, see Eqs. (2) , (7)
E	enrichment
MWd	megawatt days
R	rendement

FUEL DENSITY, URANIUM ENRICHMENT, AND PERFORMANCE STUDIES FOR THE ADVANCED NEUTRON SOURCE REACTOR

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1. INTRODUCTION

Consistent with the words of the budget request for the Advanced Neutron Source (ANS), the Department of Energy (DOE) commissioned a study[1] of the impact on performance of using medium- or low-enriched uranium (MEU or LEU) in the fuel of the reactor that generates the neutrons. The department requested that Brookhaven National Laboratory (BNL) lead that study, which also involved the participation of the Argonne, Idaho, and Oak Ridge national laboratories and of independent experts on research reactor fuels.

In the course of the study, performance calculations for 19 different combinations of reactor core volume, fuel density and enrichment, power level, and other relevant parameters were carried out by the Idaho and Oak Ridge national laboratories. Some of the results were verified through independent calculations by Argonne National Laboratory.

Since then, another 14 cases have been analyzed at Oak Ridge to explore some of the more interesting and important configurations and to gain further insights into the tradeoffs between performance and enrichment. Furthermore, with the aid of the data from these additional cases, we have been able to correlate the most important performance parameters (peak thermal neutron flux in the reflector and core life) with reactor power, fuel density, and fuel enrichment. This enables us to investigate intermediate cases, or alternative cases that might be proposed by people within or outside the project, without the time and expense of doing completely new neutronics calculations for each new example. The main drivers of construction and operating costs are the reactor power level and the number of fuel plates to be fabricated each year; these quantities can be calculated from the correlations.

The correlations can be used to prepare curves or tables showing the effect of different enrichments within the entire range from LEU at 20% to highly-enriched uranium (HEU) at 93% and for any fuel density within the range from existing technology to very advanced, undeveloped, as-yet-untested fuel. The correlations relate only to the neutronic performance; **they do not give detailed information about the safety or practicality of the fuel or the core.**

The results show that the baseline two-element core design cannot be adapted to any practical fuel of greatly reduced enrichment without great performance penalties, but that a modification of the design, in which one additional fuel element is incorporated to provide extra volume for lower enrichment fuels, has the capability of using existing, or more advanced, fuel types to lower the uranium enrichment. The modification adds only about 15 litres (0.5 cubic feet) to the core volume and would have no significant impact on construction or operating costs.

2. TWO- AND THREE-ELEMENT CORES

During the preconceptual design phase of the project, many alternative core designs were studied. In particular, cores made up from two and three elements were compared.[2]

The three-element designs have a lower rendement (ratio of thermal neutron flux to reactor power) than the two-element ones, so that more power is required for a given flux. Typically, an optimized three-element core would produce 10–20% less flux than a two-element one with the same power. (The ultimate flux capability of the optimized three-element design is 10–20% higher than a two-element one but could only be reached by increasing the power by 25–50% and also increasing the power density.) Power and power density have a major influence on corrosion of the fuel plates, on fission product inventory, on margin to critical heat flux, and on other important safety-related topics, as well as on cost. These disadvantages seemed to outweigh the gain in flux possible with a three-element design, especially as the two-element design already met the flux goals established by the scientific community and by DOE.

Nevertheless, at a constant power level the 10–20% neutron flux penalty of the three-element core is offset by two potential advantages. The first is that the greater volume of the core leads to a lower power density and therefore even greater safety margins, and perhaps therefore to a quicker and more certain licensing or safety review process. The second is that the larger volume can also accommodate more uranium, leading to a longer core life and hence greater availability: the oxide growth that can limit the safe core life is reduced by the lower heat flux associated with a larger core volume, so that increased core life is a practical option for the three-element design. The ANS Project was therefore carrying out tradeoff studies for the alternative design to ascertain if these advantages might justify proposing (to DOE and the user community) a change to the conceptual design baseline. In a certain sense, that it maximizes the neutron flux available with a given margin to the incipient boiling power limit, and within certain constraints, the optimum volume for a two-element core is 67.6 L, and for a three-element core, it is 82.6 L.[2] Figure 1 shows the two designs drawn to the same scale.

Although these tradeoff studies were initiated for quite a different purpose, the results that were available were useful in the reduced enrichment study—because to reduce the enrichment of the fuel, a large quantity of nonfissile ^{238}U must be added to provide the dilution required, and also some extra fissile ^{235}U must be added to compensate for the loss of neutrons that are absorbed by the ^{238}U (forming plutonium, among other things). Therefore, some way must be found to increase greatly the amount of uranium in the core. Increasing the uranium density in the mixture of aluminum powder and fuel particles inside the core (the fuel meat) is the first approach and has been very successful in low power and low power density research reactors; the fuels developed for that purpose by the Reduced Enrichment for Research and Test Reactors (RERTR) have been successfully applied to convert a number of smaller reactors to low-enriched fuel. However, as the uranium density is increased further, fuel performance and safety margins suffer; increased swelling, blistering of the clad, high fuel temperatures because of reduced thermal conductivity, and less homogeneous distribution of the fuel particles leading to local hot spots become more and more of a problem. Once the safe limit on fuel density is reached—and that limit is lower in high performance reactors, because of their higher fission rate and heat flux—the only way to add more fuel is to make the core bigger, which unfortunately reduces the thermal neutron flux available to the beam tubes and to other experiments outside the core.

The three-element core design provides a way to increase the fuel volume and thus allow the addition of more uranium, without increasing the length or width of each individual fuel plate, which would be necessary in a modified two-element design. Wider plates are more susceptible to deflection or distortion by the high velocity cooling water flow, and longer plates have lower thermal-hydraulic safety margins because the greater pressure drop reduces the saturation temperature at the outlet end of the cooling channel.

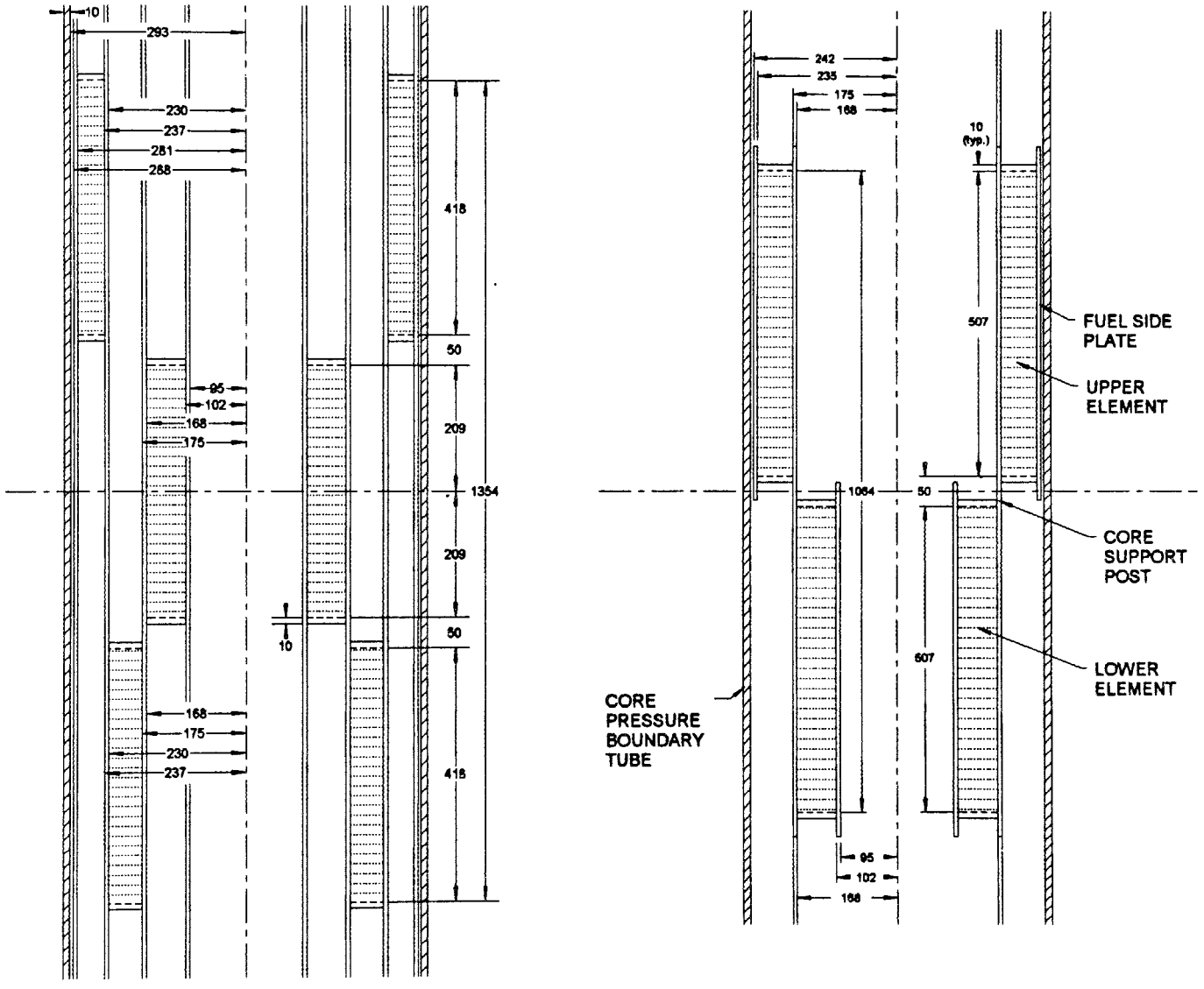


Fig. 1. Optimal three- and two-element cores.

3. CALCULATIONS

Table 1 shows the results currently available. Note that these are the results of neutronic calculations only; in some cases, thermal-hydraulic or other safety issues would limit the operating power, and therefore the attainable neutron flux, below the figures shown in Table 1.

3.1 82.6 L THREE-ELEMENT CORES

Twenty-four different cases have been computed for the 82.6-L core shown in Fig. 1. All 24 are included in Table 1, which lists the enrichment; the power level, the core life, and their product; the peak flux divided by the power; the fuel density; and the initial [beginning-of-cycle (BOC)] loading of ^{235}U and of the other, nonfissile, isotopes (mainly ^{238}U).

Inspection of the table reveals that the number of MWd achieved increases with increasing ^{235}U loading but decreases as more of the other, nonfissile isotopes are added. We tried the simplest assumption, a linear relationship:

$$\mathrm{MWd} = A \times {}^{235}\mathrm{BOC} - B \times {}^{\mathrm{Other}}\mathrm{BOC} - \mathrm{constant} \tag{1}$$

or

$${}^{235}\mathrm{BOC} = C_0 + C_1 \times \mathrm{MWd} + C_2 \times {}^{\mathrm{Other}}\mathrm{BOC}\ . \tag{2}$$

Note that we expect C_0, C_1, and C_2 all to be positive because a certain amount of ^{235}U must be present to make the core critical (i.e., to have any positive number of MWd), even in the absence of ^{238}U, and the more ^{238}U that is present, the more ^{235}U we need.

The MULREG linear regression routine in TK Solver[3] was used to fit all 24 points to Eq. (2), with the results shown in Table 2 and in Eqs. (3) and (4). The correlation was derived from data with more significant figures than shown in Table 1.

The ratio of ^{235}U mass to total uranium mass at BOC is the enrichment, E. Substituting this relationship and the coefficients from Table 2 into Eq. (2) leads to

$$\text{total uranium mass at BOC} \approx \frac{4.00 + .00225 \times \mathrm{MWd}}{1.108\mathrm{E} - 0.108}\ . \tag{3}$$

In this equation, the enrichment is expressed as a fraction (e.g., 0.93) rather than as a percentage.

Expressing the same equation with MWd as the independent variable yields

$$\mathrm{MWd} \approx (493\mathrm{E} - 48) \times {}^{\mathrm{Total}}\mathrm{BOC} - 1780\ . \tag{4}$$

Table 1. Results of neutronic calculations for the 67.6-, 82.6-, and 108-L cores

ORNL case number	BNL case number	Number of elements	Volume (L)	Enrichment (% ^{235}U)	Fuel density (g/mL)	Power (MW)	Core life (d)	Peak flux 10^{19} $m^{-2} \cdot s^{-1}$	Mass of ^{235}U BOC (kg)	Mass of $^{Other}U$ BOC (kg)	Mass of ^{235}U EOC (kg)	Plutonium production/cycle (kg)	Flux/power 10^{17}/MW
OR 1	1	2	67.6	93	1.7	330	17	7.33	23.62	1.78	16.40	0.07	2.221
OR 2	2	2	67.6	80	2.1	330	17	7.25	25.12	6.28	17.90	0.14	2.197
OR 3	3	3	82.6	50	2.0	330	17	5.87	19.50	19.50	12.50	0.23	1.779
OR 4	4	3	82.6	50	2.2	400	17	7.06	21.40	21.40	13.00	0.28	1.765
OR 5	5	3	82.6	35	3.5	405	17	6.80	23.84	44.27	15.40	0.52	1.679
OR 6	6	3	82.6	35	3.0	330	17	5.71	20.41	37.90	13.50	0.4	1.730
-	7	3	108.0	20	7.2	530	17	7.12	38.72	154.88	27.50	1.697	1.343
OR 7	8	3	108.0	20	6.0	449	17	6.04	32.22	128.88	23.10	1.293	1.345
OR 8	9	3	82.6	20	3.5	125	17	2.17	13.60	54.40	11.00	0.26	1.736
OR 9	10	3	108.0	20	4.8	330	17	4.79	25.80	103.20	19.00	0.86	1.452
OR 10	11	2	67.6	35	6.5	330	17	6.63	33.95	63.05	26.80	0.66	2.009
OR 11	-	3	82.6	93	2.0	330	42[a]	5.29	36.27	2.73	18.50	0.12	1.603
OR 12	12	3	82.6	50	2.0	340	17	5.98	19.50	19.50	12.30	0.23	1.759
OR 13	13	3	82.6	35	3.3	370	17	6.24	22.51	41.80	14.80	0.47	1.686
OR 14	14	3	108	20	5.8	430	17	6.00	31.20	124.80	22.30	1.23	1.395
OR 15	15	3	82.6	93	0.95	330	17	6.04	17.21	1.30	10.20	0.04	1.830
OR 16	16	2	67.6	45	3.5	200	17	4.35	23.49	28.72	19.19	0.29	2.175
OR 17	17	3	82.6	45	1.8	250	17	4.68	15.77	19.28	10.54	0.19	1.872
OR 18	18	3	82.6	35	2.1	200	17	3.78	14.31	26.58	10.13	0.2	1.789
OR 19	19	3	108	35	1.8	260	17	3.99	16.96	31.50	11.5	0.26	1.535
OR 20	-	3	82.6	20	4.8	200	17	3.37	18.70	74.80	14.5	0.48	1.685
OR 21	-	3	108	20	3.5	225	17	3.15	18.82	75.28	14.2	0.5	1.400
OR 22	-	3	82.6	20	2.2	85	4.25	1.51	8.58	34.32	8.12	0.04	1.776
OR 23	-	3	82.6	20	9.5	425	17	6.2	37.00	148.00	28.2	1.44	1.515
OR 24	-	3	82.6	93	1.15	330	23	5.92	20.82	1.57	11.32	0.05	1.794
OR 25	-	3	82.6	20	9.8	330	23	5.02	38.16	152.65	28.97	1.416	1.521
OR 26	-	3	82.6	50	2.4	330	23	5.67	23.35	23.35	14	0.298	1.718
OR 27	-	3	82.6	35	3.8	330	23	5.39	25.90	48.10	16.6	0.545	1.633
OR 28	-	3	82.6	50	2.55	365	23	6.22	24.85	24.85	14.3	0.337	1.704
OR 29	-	3	82.6	35	4.1	365	23	5.88	27.93	51.87	17.7	0.675	1.611
OR 30	-	3	82.6	93	1.25	365	23	6.53	22.60	1.70	11.7	0.059	1.789
OR 31	-	3	82.6	50	1.55	165	23	2.96	15.10	15.10	10.6	0.127	1.794
OR 32	-	3	82.6	50	1.1	165	11.5	3.12	10.70	10.70	8.4	0.061	1.891
OR 33	-	3	82.6	50	3.1	450	23	7.39	30.20	30.20	17.4	0.463	1.642

[a] Enough reactivity for 42d, but oxide growth would limit practical life to 40 d.

Table 2. 82.6-L core correlation for MWd and fuel mass from TK Solver MULREG

Name	Result	Comment
C_0	3.9954183	Constant
C_1	0.0022465	Coefficient for MWd variable
C_2	0.10829393	Coefficient for mass of other uranium isotopes at BOC
Syx	0.65821902	Standard error
R^2	0.99286119	Adjusted coefficient of multiple determination

Figure 2 shows the lines obtained by substituting enrichment values of 0.93, 0.50, 0.35, and 0.20 into Eq. (4) and also plots the fuel loading and MWd data points from Table 1. The correlation appears to be quite good.

Within the 82.6-L active volume of the core, half is occupied by the coolant channels. Of the remaining 41.3 L, two-fifths are taken up by the cladding. As the BNL study group agreed, by comparison with the baseline of the two-element conceptual design core, approximately 5.4 L of the remaining 24.8 L would be occupied by the aluminum filler powder used in grading the thickness of the fuel meat. Thus, approximately 19.4 L is available for the fuel/aluminum powder mixture, and the fuel density and the BOC uranium loading can be related by

$$\text{BOC mass of uranium/fuel density} = \text{fuel meat volume} \approx 19.4\ \text{L}\,. \tag{5}$$

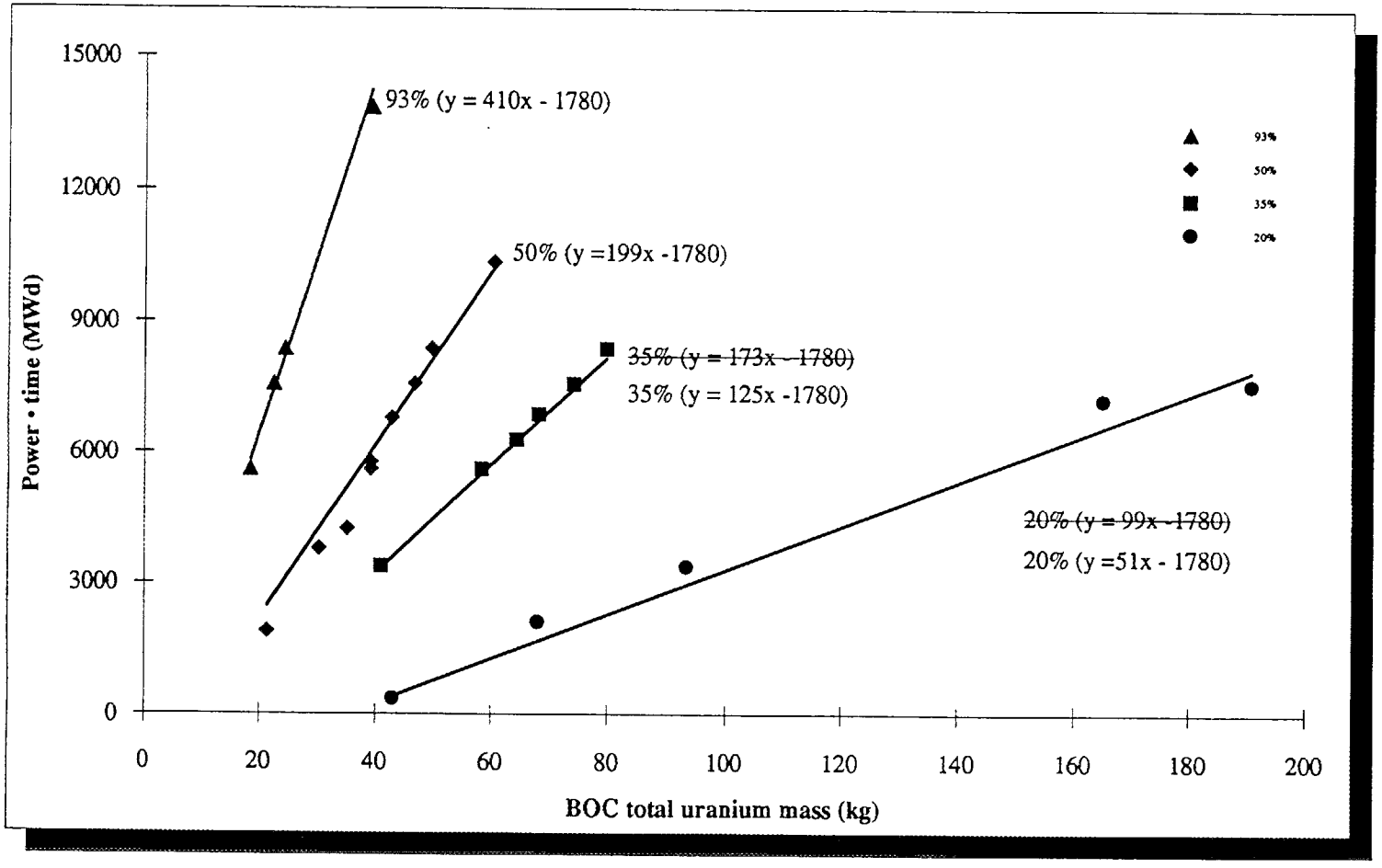
If the BOC uranium mass is expressed in kg, and the uranium density in the fuel meat is in g/mL or kg/L or Mg/m^3, then the volume must be in litres. With this relationship in mind, Eq. (4) can be recast as

$$\text{MWd} \approx (9564\text{E} - 931) \times (\text{uranium density}) - 1780\,. \tag{6}$$

Equation (4) relates the power and core life to the fuel enrichment and uranium loading (but remember, that correlation relates only to a specific 82.6-L core design). To determine the peak thermal neutron flux, the most important single measure of performance, we need to know the ratio of flux to power, called the rendement, R.

We expect that higher fuel loadings, for greater MWd, will also lead to lower rendement, because of the increased neutron absorption in the core region. Lower enrichments will also lead to lower rendement, because the additional ^{238}U will absorb neutrons that would otherwise contribute to the flux peak. We tried several such correlations, and the one described here is

$$R = C_0 - C_1 \times \text{MWd} - \frac{C_2}{E}\,. \tag{7}$$



EEA **Fig. 2. Correlations for MWd, enrichment, and beginning-of-cycle uranium mass for 82.6-L three-element cores.** 3/16/95

Using the same 24 data points, and the same TK Solver routine, we find the rendement, in units of 10^{17} $m^{-2} \bullet s^{-1} \bullet MW^{-1}$, to be

$$R = 2.128 - 3.497 \times 10^{-5} \times MWd - \frac{.070085}{E} . \quad (8)$$

The correlation coefficient is 0.894, and the standard error is .035 or only about 2% of a typical rendement. Figure 3 shows the correlation for enrichments of 0.93, 0.50, 0.35, and 0.20 compared with the data points from the 24 cases. Note this relation applies only to the core geometry and dimensions shown in Fig. 1. There are other ways to arrange the three different sized elements, and the different arrangements have different neutronic properties.

3.2 67.6-L TWO-ELEMENT CORES

Similar correlations, with far fewer data points, have been calculated for the 67.6-L two-element cores. The correlations for MWd and fuel loading are shown in Table 3 and illustrated in Fig. 4, which shows the correlation and the four data points from which it was derived. Of course, it would not have been reasonable to establish such a correlation on the basis of only four data points if the important variables, and the linear nature of the correlation, had not already been established from the much larger number (24) of cases available for the 82.6-L core.

Substituting the coefficients from Table 3 and rearranging the equation to match the format of Eq. (3) yields

$$\text{total uranium mass at BOC} \approx \frac{11.28 + .00221 \times MWd}{1.163E - 0.163} . \quad (9)$$

Or, rewriting with MWd on the left-hand side, we have

$$MWd \approx (525E - 74) \times {}^{Total}BOC - 5100 . \quad (10)$$

The volume of the fuel meat with the baseline grading in the 67.6-L core is approximately 14.9 L, so Eq. (10) can be rewritten in terms of the uranium density in the fuel meat as

$$MWd \approx (7822E - 1103) \times \text{uranium density} - 5100 . \quad (11)$$

A correlation for the rendement of the 67.6-L core is

$$R = 2.588 - 4.301 \times 10^{-5} \times MWd - \frac{0.1180}{E} , \quad (12)$$

and, in Fig. 5, the data points for rendement are compared with lines drawn from this formula. With three variable coefficients and only four data points, the correlation coefficient is naturally rather good ($R^2 = 0.999$).

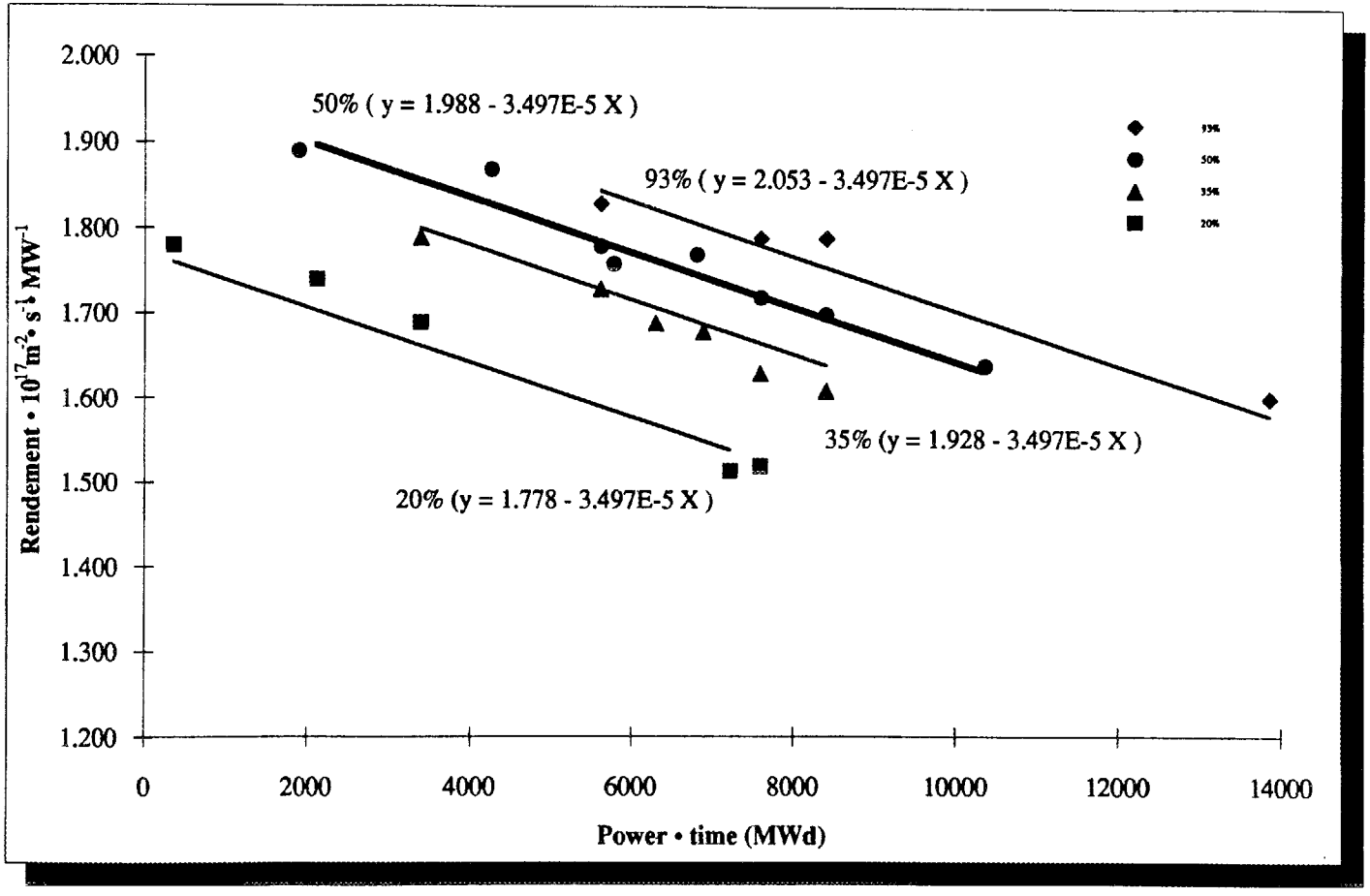


Fig. 3. Correlations for rendement, MWd, and enrichment for 82.6-L three-element cores.

Table 3. 67.6-L core correlation for MWd and fuel mass from TK Solver MULREG

Name	Result	Comment
C_0	11.281752	Constant
C_1	0.00221328	Coefficient for MWd variable
C_2	0.16306025	Coefficient for mass of other uranium isotopes at BOC
Syx	0.54298548	Standard error
R^2	0.98816775	Adjusted coefficient of multiple determination

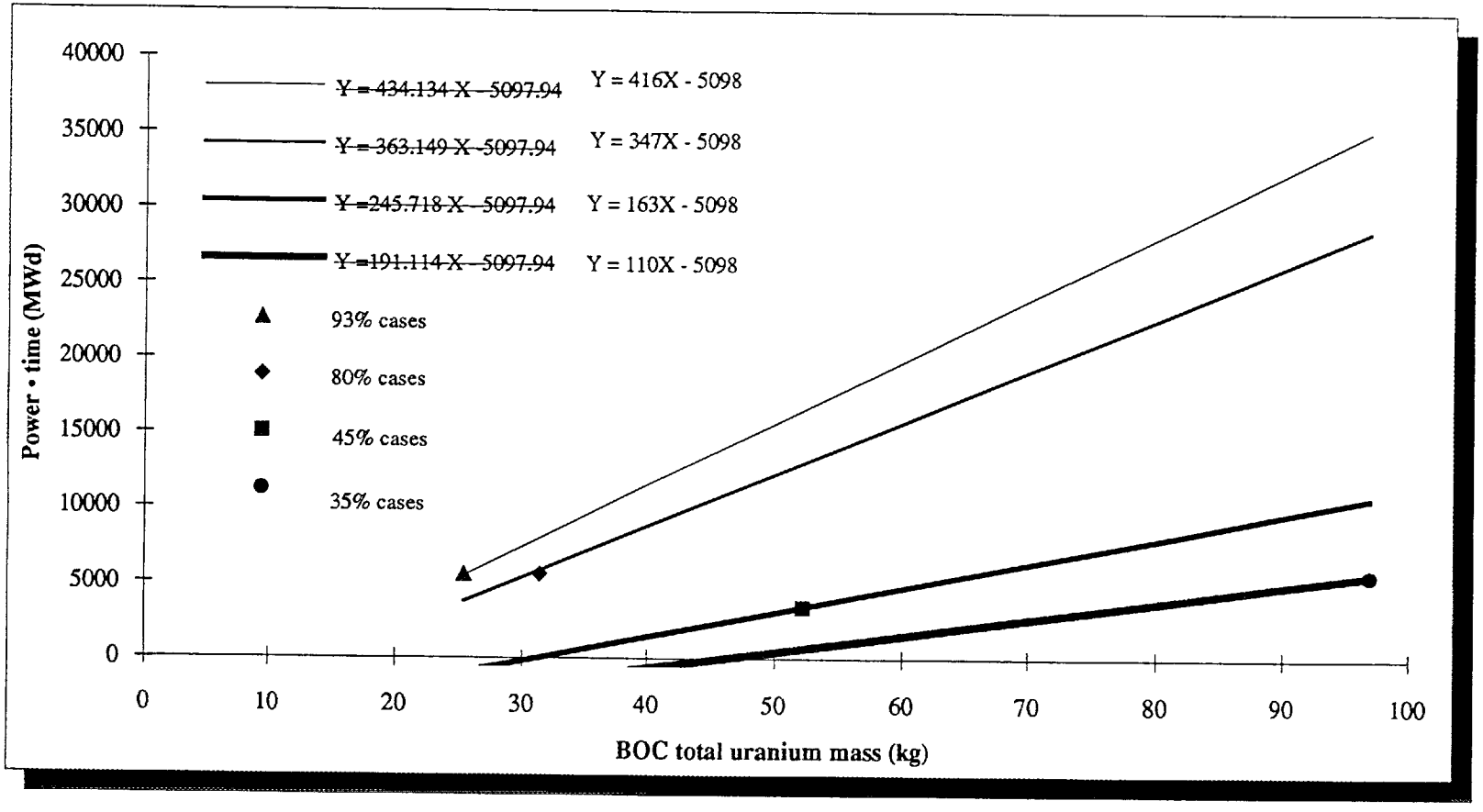


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Fig. 4. Correlations for MWd, enrichment, and beginning-of-cycle uranium mass for 67.6-L two-element cores.

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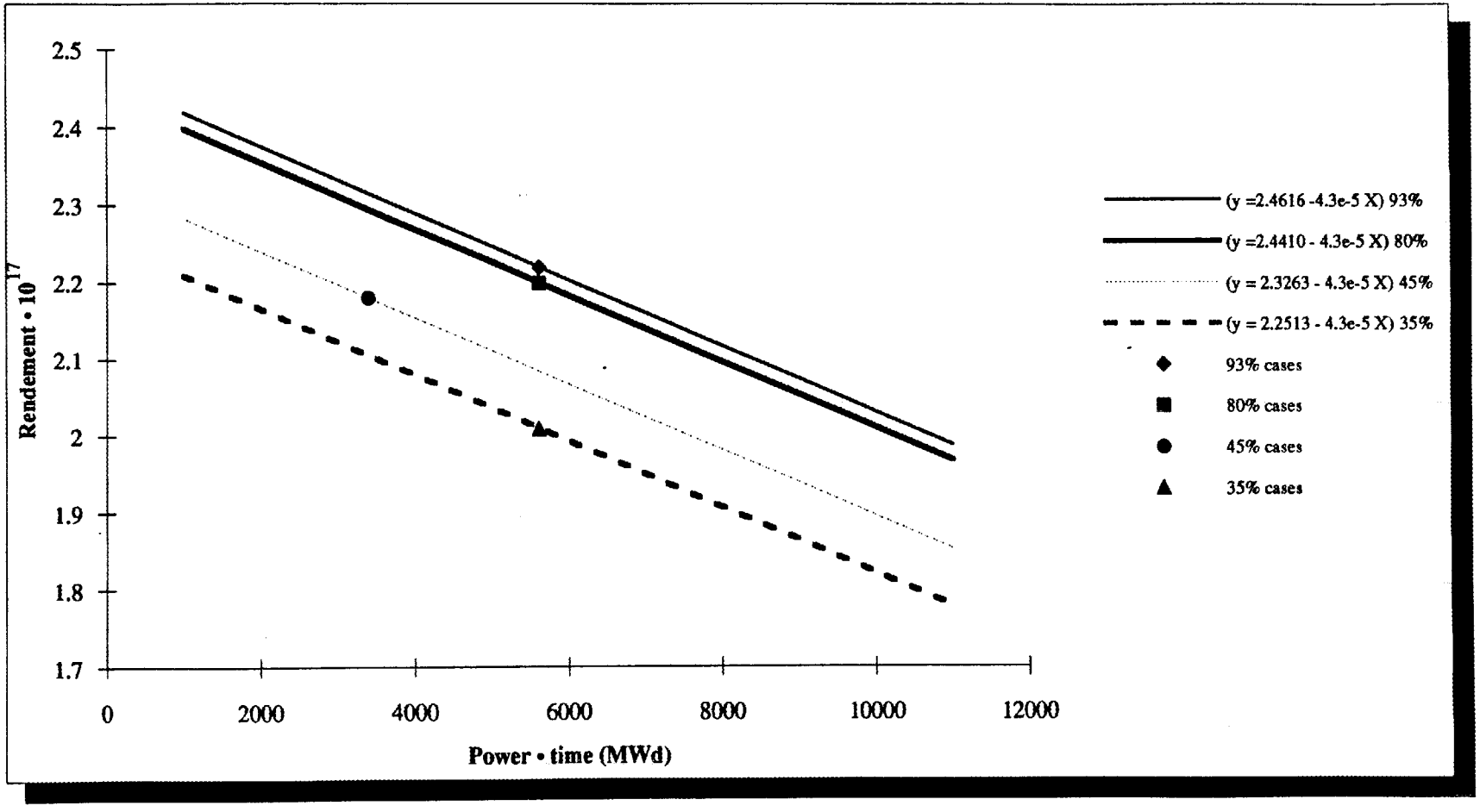


Fig. 5. Correlations for rendement, MWd, and enrichment for 67.6-L two-element cores.

4. RESULTS

Table 4 summarizes the correlations for the baseline core and for the modified design.

Table 4. Summary of correlations

Core	MWd vs density and enrichment	Rendement vs MWd and enrichment
67.6-L, two-element	$(7822E - 1103) \times \rho_U - 5100$	$2.588 - 4.301 \times 10^{-5} \times MWd - \frac{0.1180}{E}$
82.6-L, three-element	$(9564E - 931) \times \rho_U - 1780$	$2.128 - 3.497 \times 10^{-5} \times MWd - \frac{0.070085}{E}$

The minimum acceptable core life, within the constraints of the desired availability factor for the reactor, is 17 d. This results in a "cliff" in the relationship between thermal neutron flux vs enrichment (see, for example, Fig. 6). For any given uranium density, there is a maximum power, easily calculated from Eqs. (6) and (11), that can be sustained for 17 d. If a lower enrichment than that is adopted, the reactor power must be reduced accordingly, and the thermal neutron flux declines almost in proportion. For higher enrichments, the full 330 MW of the baseline design can be maintained, but the rendement is reduced somewhat for all enrichments below 93% [see Eqs. (8), (9), and (12)]. Consequently, as enrichment is reduced below the baseline HEU, the neutron flux at first declines slowly, and then, when full power cannot be maintained, it declines rapidly. Eventually, as the enrichment is lowered further, there is not enough ^{235}U in the fuel for the core to go critical at all, so that both the power level and the neutron flux are zero.

Expert, independent review groups have concluded that fuel densities below about 2.2 gU/mL require no further development, although confirmatory tests are needed and are planned as part of the ANS Project (see Appendix C for a summary of the findings of two expert fuel reviews). Higher densities, up to 3.5 gU/mL, do require some further development work, but the development would probably have minimal cost and schedule impact and is fairly low risk (up to 5% chance of failure).

Above 3.5 gU/mL, and up to 4.8 gU/mL, more development is needed, and there is a higher chance of failure. To develop fuels of 4.8 gU/mL and above would require considerable development effort and higher risks; furthermore, the thermal conductivity of such very high density fuels is so low that the reactor power might have to be reduced to keep the fuel centerline temperature within safe limits.

Figure 7 illustrates these issues. For the baseline two-element core design, and existing fuel technology (i.e., no more than 2.2 gU/mL), the "enrichment cliff" occurs at 76% ^{235}U. With 3.5 gU/mL fuel (if the riskier development were successful), the cliff occurs at 53%. Extrapolation of the correlations shows that to postpone the cliff to LEU (19.9%) fuel would require a density of more than 23 gU/mL, which is an impossibility because the density of pure uranium metal is less than 19 g/mL.

Because of the very rapid falloff in performance once the cliff is reached (a further ten percentage point reduction in enrichment beyond the edge of the cliff typically reduces the flux to only about half its HEU value), we cannot meet high performance requirements by operating far to the right of the cliff. Therefore, we must conclude that the baseline core design cannot, using any fuel that is realistically likely to be developed, accommodate LEU while retaining acceptable performance. This is precisely the reason that the larger, three-element core design was featured in the Brookhaven study.

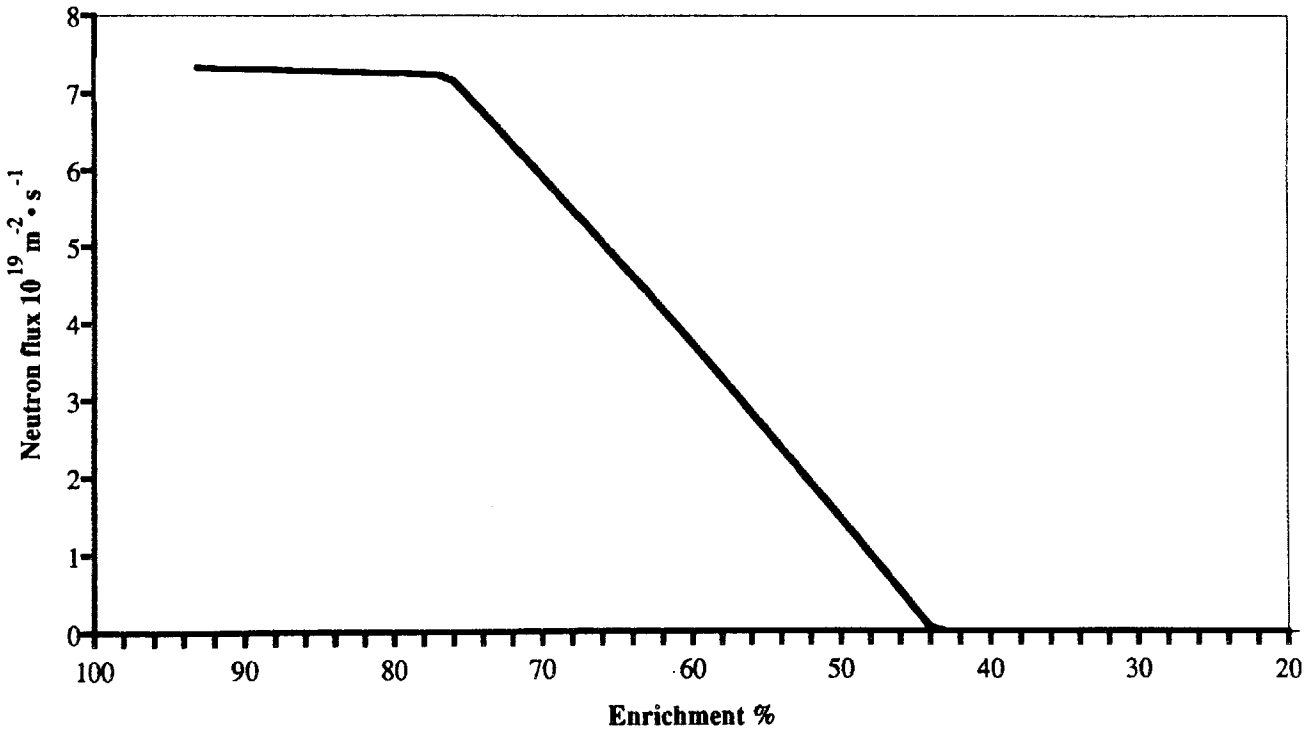


Fig. 6. Neutron flux vs enrichment for the baseline core design with 2.2 gU/mL fuel density.

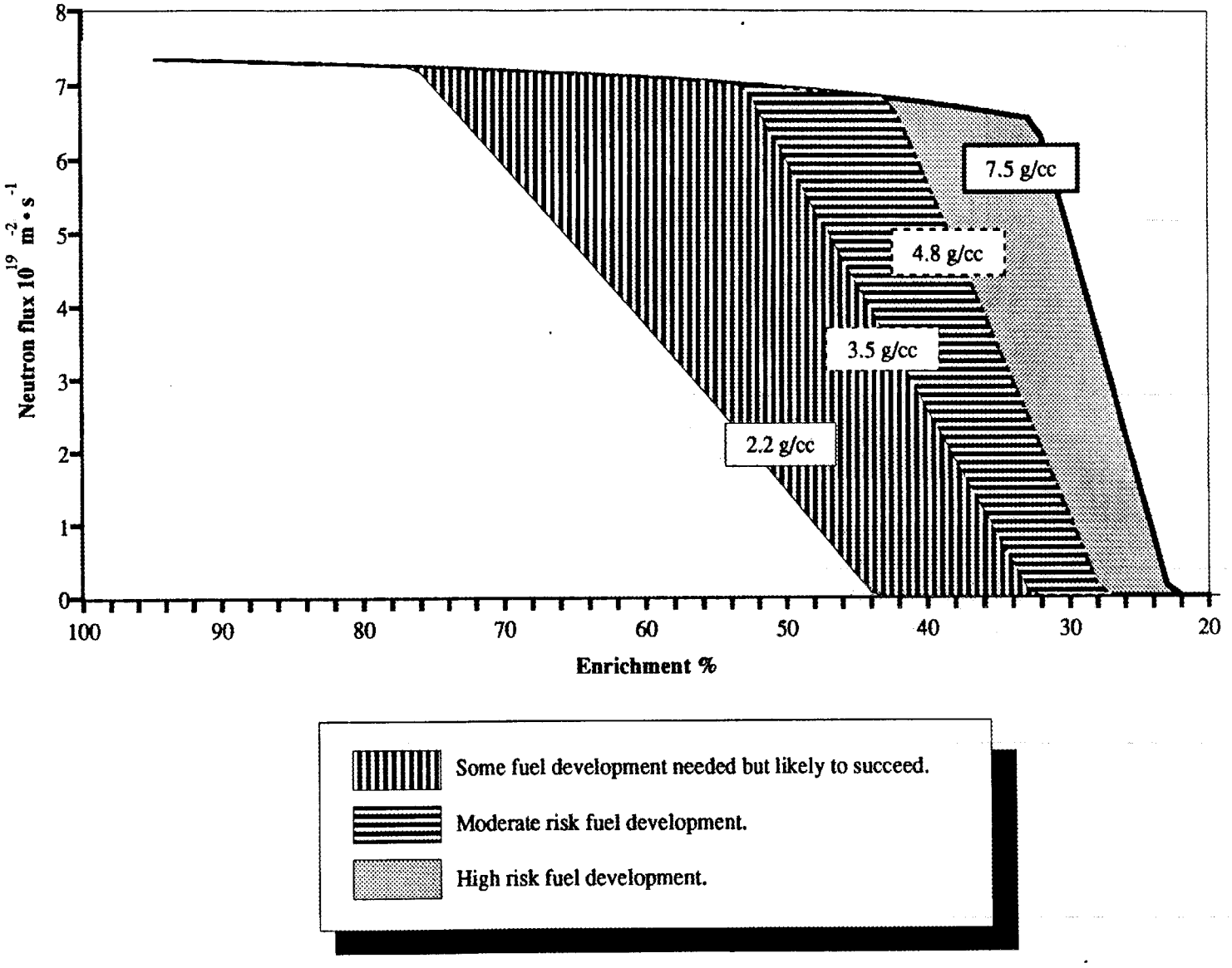


Fig. 7. Neutron flux for different fuel densities and enrichments; baseline core design; 17-d life; 330-MW maximum power level.

With the 82.6-L three-element core and 2.2 gU/mL fuel, the enrichment cliff is in the MEU region, at 45% (see Fig. 8). However, because the larger core has a lower rendement, the flux generated at a given power level is less than from the baseline design. This is also shown in Fig. 8; the loss, in the enrichment region where full power can still be maintained, is about 15–20% of the baseline flux, which is certainly a very significant loss of scientific capability.

Figure 9 compares, for the baseline and modified core designs, the fuel density needed to operate at 330 MW and thus avoid the enrichment cliff. For the modified design, a fuel density of 3.5* g/mL defers the cliff to 32% enrichment, and a 4.8* g/mL puts the cliff at 26%.* If the RERTR program successfully developed and qualified a 7.5 g/mL fuel with good thermal conductivity, the modified core design could use LEU and still run at full power for 17 d; Fig. 9 illustrates this and compares the baseline core with the modified design.

With MEU or HEU, the operating life of the 82.6-L core can be increased beyond 17 d by adding more fuel. A longer operating cycle and the increased availability factor that it permits would provide a benefit to the users that, for certain experiments only, might partially offset the lower flux. Figure 10 illustrates this possibility.

*Corrections from original copy.

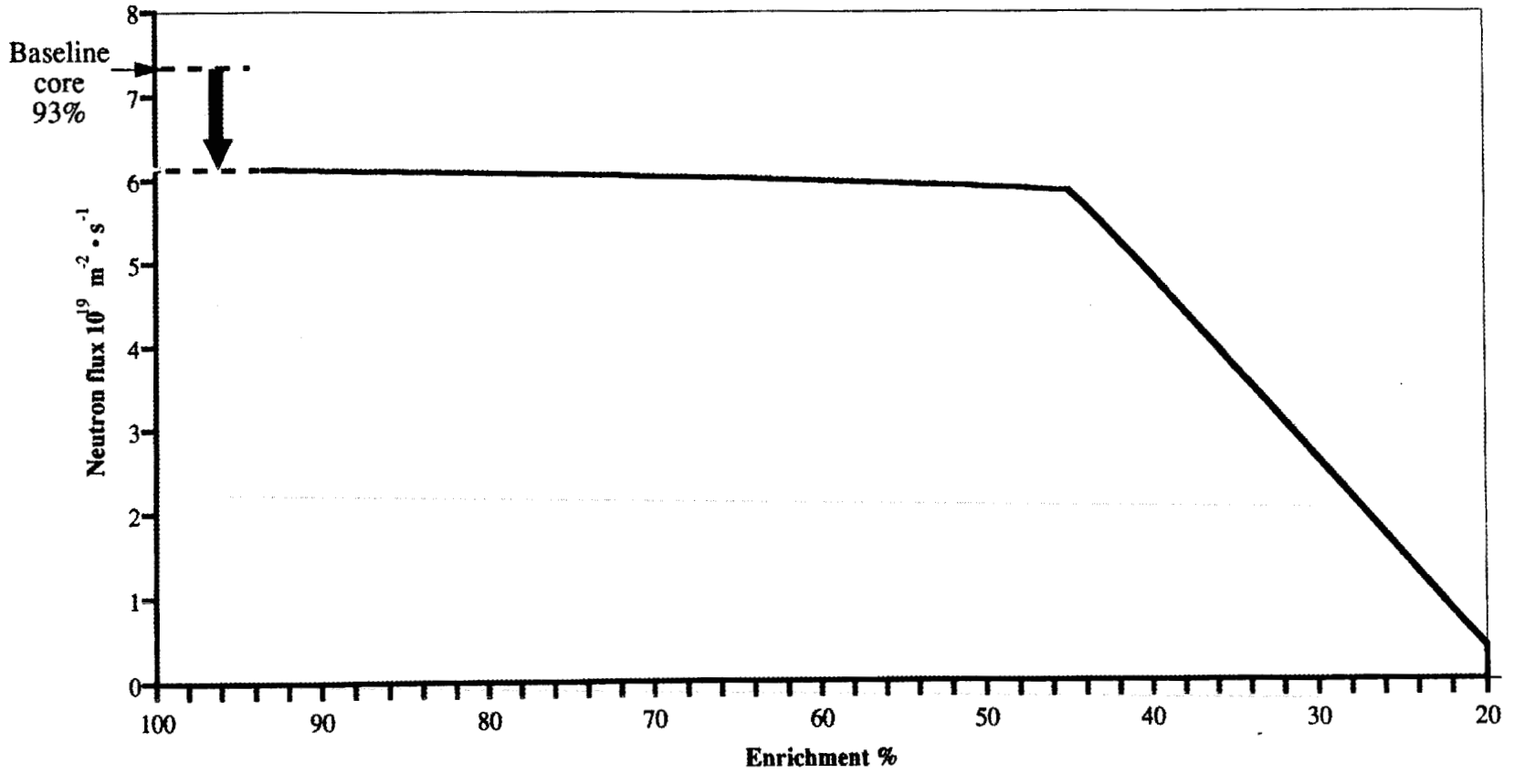


Fig. 8. Neutron flux vs enrichment for the modified core design with a 2.2 gU/mL fuel density and 330-MW maximum power.

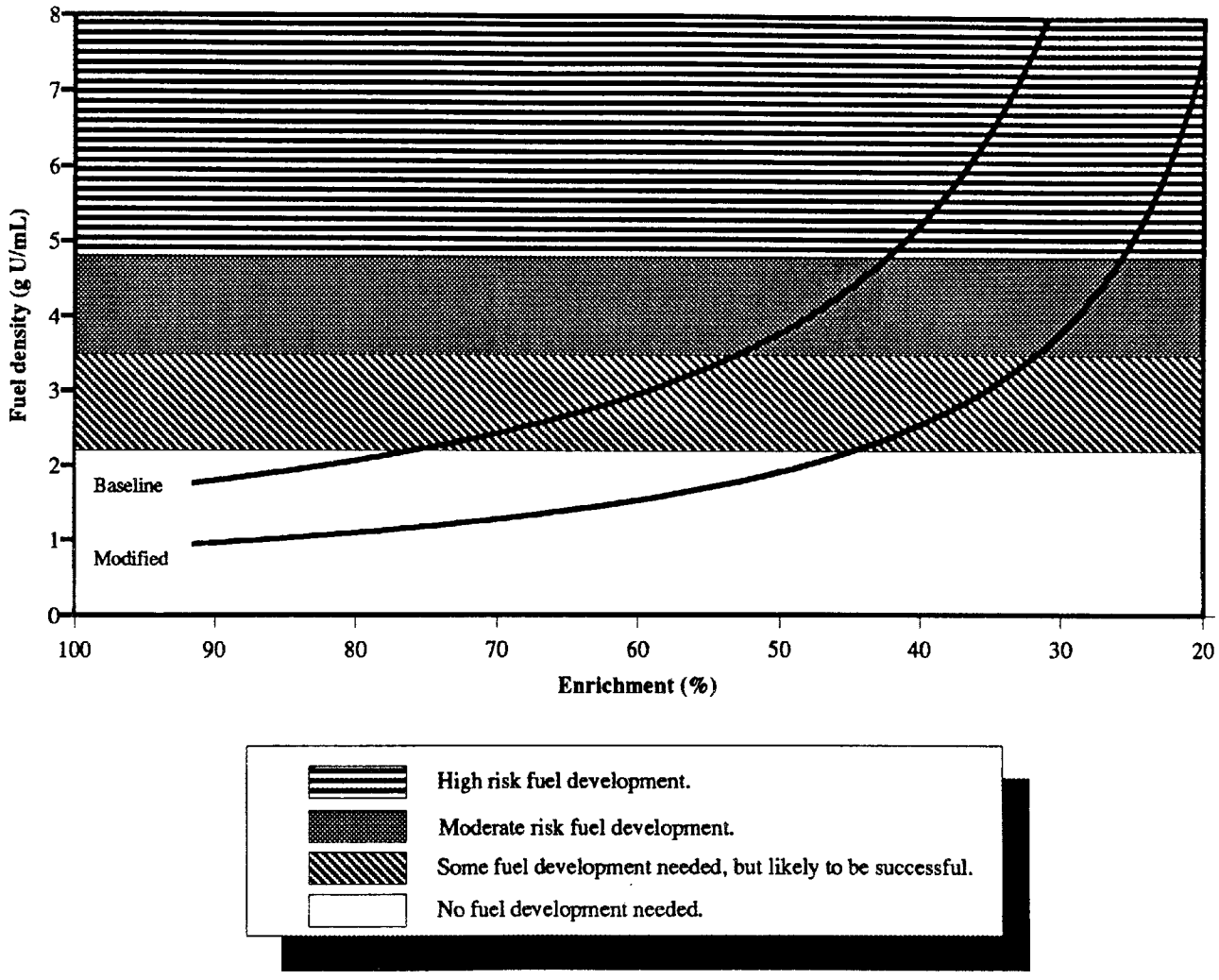


Fig. 9. Fuel density required to sustain 330 MW for 17 d vs enrichment for conceptual design report baseline and modified (three-element) core design.

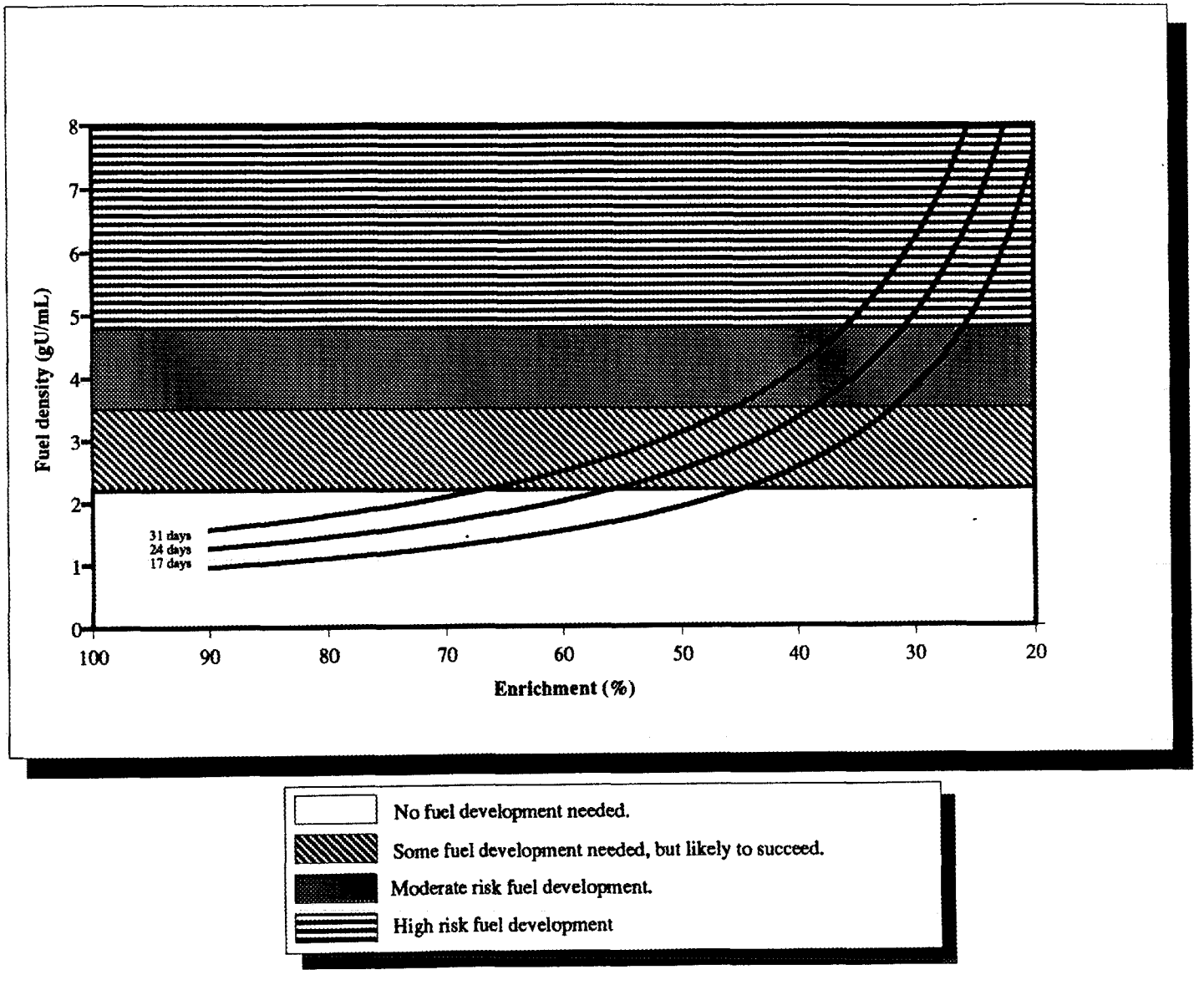


Fig. 10. Fuel density required to sustain 330 MW for 17-, 24-, and 31-d core life vs enrichment for modified core design.

5. SUMMARY

The baseline, two-element 67.6-L core design is so well optimized to provide the maximum possible neutron flux within safe operating limits that it has little tolerance for lower enrichment fuels. The modified design is fully responsive to guidance in the FY-1995 budget language (Appendix E); it provides the flexibility to lower the enrichment somewhat, even with currently available fuels, and to utilize improved fuels as they are developed and qualified to reduce the enrichment further, even to the 20% level that is characterized as LEU. Adopting the modified design would lower the flux by 15–20%, even using highly enriched fuel, and careful evaluation of the impact of such a reduction of scientific utility is needed.

6. REFERENCES

1. R. A. Bari, H. Ludewig, and J. Weeks, *Advanced Neutron Source Enrichment Study*, (draft) Brookhaven National Laboratory, Upton, New York, January 31, 1994.

2. G. L. Copeland et al., *Advanced Neutron Source Final Preconceptual Reference Core Design*, ORNL/TM-11234, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., August 1989.

3. *TK Solver Plus*, Universal Technical Systems, Inc., Rockford, Illinois, July 1989.

Appendix A: DESCRIPTION OF NEUTRONICS CALCULATIONS FOR THE THREE-ELEMENT CORE

Appendix A

DESCRIPTION OF NEUTRONICS CALCULATIONS FOR THE THREE-ELEMENT CORE

INTRODUCTION

The analysis of the three-element core configuration was first performed for the ANS fuel enrichment study.[A-1] In this study the baseline two-element core and two different three-element core configurations were considered. Several calculations were performed to find fuel loadings at several enrichments that maintain a 17-d fuel cycle. Since the completion of the fuel enrichment study, several additional three-element core calculations have been performed to provide a wider range of information from which correlations could be obtained. This appendix provides a brief description of the calculational models and methods used for the analysis of the three-element configurations.

MODEL DESCRIPTION

The conceptual ANS design[A-2] consists of two fuel elements with a total active fuel volume of 67.6 L. Each element consists of involute-shaped fuel plates containing U_3Si_2/aluminum fuel meat, aluminum-1100 filler, and aluminum-6061 clad. The fuel meat thickness is axially and radially graded to minimize power peaking during the fuel cycle, which consists of 17-d at a fission power of 330 MW. The end-caps of the fuel plates contain 12.7 g of ^{10}B in the form of boron carbide for reactivity control at BOC. This design requires a fuel density of approximately 1.7 gU/mL, resulting in a fuel loading of about 23.6 kg of ^{235}U with a fuel enrichment of 93%.

The two-element design is based upon optimization studies to achieve the highest possible flux. This design has undergone close scrutiny for engineering and safety aspects that are required for an acceptable design. The three-element design closely resembles those studied in the core optimization and design process. The element thicknesses are based upon fuel-plate stability and element thermal-hydraulic considerations so that this design is as reasonable as possible, given the limited amount of information for such a design.

The fuel cycle analysis of the conceptual core design and for the three-element core design is performed with four-group, finite-difference diffusion theory using the VENTURE code system[A-3] using two-dimensional r-z geometry. The VENTURE code system requires, as input, group cross sections for the cases to be considered. In this study, the weighted cross sections with a four-neutron group structure were created using the AMPX[A-4] and SCALE[A-5] systems and the ANSL-V 99-group master cross-section library.[A-6] Starting with the master library, the unresolved resonance region is processed with the BONAMI[A-4] module, which is followed by the resolved resonance processing in the NITAWL-II[A-4] module. The material temperatures required for resonance broadening are representative of the ANS operation conditions. The group collapse was performed using one-dimensional discrete-ordinates calculations with the XSDRNPM[A-4] module. Radial cross-section collapses are performed for each element of the reference design with and without control rods.

In the two-element baseline design, the fuel is graded in the radial and axial direction to minimize power peaking throughout the fuel cycle. This grading is modeled with 299 (13 × 23) zones in each element. For the three-element designs, the fuel grading is uniform; there was not nearly enough time during the enrichment study to define an optimized fuel grading for each design studied. Fortunately, neutron flux and core reactivity are integral properties mainly dependent on total fuel loading and rather insensitive to the fuel distribution. The same volume of fuel lost to fuel

grading in the two-element core was also removed from the three-element core to account for the fuel displaced if a grading had been used. In the fuel enrichment studies, the boron carbide was removed from the end-caps to simplify the fuel cycle comparisons.

To determine the correct amount of fuel required for the entire fuel cycle, it is important to consider the reflector components that constitute a substantial negative reactivity worth to the system. Not only does the aluminum structure absorb neutrons, it, along with large voids, displaces D_2O, which is a good neutron moderator. In the two-dimensional fuel cycle model, the reflector components are represented with the components smeared throughout the reflector. Five radial zones, and several axial zones (one corresponding to each axial fuel zone), are used. The shutdown rods and in-core irradiation facilities have not been modeled.

The current ANS control rod design consists of three hafnium rods in the central hole of the core. This geometry cannot be modeled in two-dimensional r-z geometry, so an equivalent two-ring control rod model is used. Calculations of the two-element ANS core were performed with the MCNP Monte Carlo code[A-7] with the explicit three-rod model and the equivalent two-ring model. The D_2O gaps between the materials were adjusted until the equivalent model matched the explicit model. Two rings are used to provide more degrees of freedom in matching the explicit rod and do not require the modification of the hafnium number densities and thicknesses when adjusting this model.

The fuel cycle depletion steps for the 17-d cycle are at 1, 4.25, 8.5, 12.75, and 17 d. Each of the 299 zones in each element fuel grading is used as a depletion zone. The xenon and samarium fission product chains, along with several additional fission products, are represented explicitly. The remaining fission products are represented by two fission product lumps. The xenon concentration and fluxes are adjusted at 1 d based upon an equilibrium model to avoid an excessive number of depletion steps. At each step in the cycle, the control rod is adjusted so the reactor is precisely critical (i.e., the effective multiplication factor has a value of 1) using the CTRLPOS module of the VENTURE system.

CALCULATIONS

Many calculations were performed for several combinations of core geometry, fuel density, fuel enrichment, and total core power. The optimization of the fuel cycle consists of finding the fuel density and core power that results in a 17-d fuel cycle. The system is required to be critical at EOC with the control rods fully withdrawn. For the lower enrichments, a 17-d cycle is not possible at a power of 330 MW with reasonable fuel densities, so the core power is reduced until a 17-d cycle is obtained.

The primary quantity used to characterize the different enrichment, density, and core designs is the unperturbed reflector peak thermal flux that occurs throughout the fuel cycle. The method used to calculate this flux value is

$$\begin{array}{l}\text{Peak thermal}\\ \text{reflector flux}\\ \text{throughout cycle}\end{array} = \begin{array}{l}\text{VENTURE peak reflector}\\ \text{flux throughout}\\ \text{cycle}\end{array} \times \frac{\text{MCNP unperturbed peak reflector flux (BOC)}}{\text{VENTURE peak reflector flux at BOC}} \cdot \qquad \text{(A-1)}$$

In this equation, the VENTURE calculation provides the ratio of the peak thermal flux at BOC to that at the time at which the peak occurs. Multiplying the MCNP peak thermal flux by this fuel cycle ratio gives an estimate of the peak thermal flux with corrections for transport and energy group effects. After the fuel cycle calculation is performed with the model described above, the reflector components are removed and unperturbed VENTURE calculations are performed at each depletion step with the fuel region atom densities from the perturbed fuel cycle calculation. The fuel loading, fission products, and rod positions at each step are the same as the fuel cycle calculation. The unperturbed fluxes from this calculation are used in the above equation to obtain the peak

thermal reflector flux throughout the fuel cycle. A comparison of the cases from the fuel enrichment study indicates that the MCNP/VENTURE correction factor has a value of approximately 1.06 (i.e., VENTURE underpredicts the peak thermal flux in the reflector by about 6%) and is constant for a wide range of cases.

In addition to the unperturbed peak reflector thermal flux throughout the fuel cycle, there are other parameters that can also be used to compare the different cases. The assessment of the proliferation risk of the fuel, for example, requires a knowledge of the mass of uranium at BOC and of plutonium at EOC. Note that the ^{239}Pu mass that should be considered is the sum of the ^{239}Pu and ^{239}Np that occurs at EOC since ^{239}Np decays to ^{239}Pu with a half-life of 2.35 d.

SUMMARY

In this appendix, the models and methods used to perform the calculation of the neutronics parameters of the three-element core were discussed. These methods and models are the same as applied in the ANS fuel enrichment study and were used to obtain additional data from which correlations can be determined.

REFERENCES

A-1. See Sect. 6, ref. 1.

A-2. F. J. Peretz, *ANS Conceptual Design Report Summary*, ORNL/TM-12184, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., September 1992.

A-3. D. R. Vondy, T. B. Fowler, and G. W. Cunningham, *The Bold VENTURE Computation System for Nuclear Reactor Core Analysis, Version III*, ORNL-5711, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., June 1981.

A-4. N. M. Green, W. E. Ford, III, L. M. Petrie, and J. W. Arwood, *AMPX-77: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Cross-Section Libraries from ENDF/B-IV and/or ENDF/B-V*, ORNL/CSD/TM-283, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., October 1992.

A-5. W. E. Ford, III et. al, *ANSL-V: ENDF/B-V Based Multigroup Cross-Section Libraries for Advanced Neutron Source (ANS) Reactor Studies*, ORNL-6681, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., September 1990.

A-6. *SCALE 4.1, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, CCC-545, Radiation Shielding Information Center, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., July 1992.

A-7. J. F. Briesmeister, ed., *MCNP-A General Monte Carlo Code for Neutron and Photon Transport, Version 3B*, LA-7396-M, Rev. 2, Los Alamos National Laboratory, Los Alamos, New Mexico, April 1991.

Appendix B: 108-L THREE-ELEMENT CORES

Appendix B

108-L THREE-ELEMENT CORES

As described in the Brookhaven report,[B-1] the 108-L core, shown in Fig. B.1 ". . . is a hypothetical example that was constructed to study the physics behavior of a large core. In practice, the design would suffer from large and unsafe deflections of the fuel plates due to forces acting on the wide, relatively flat, and therefore, flexible plate span." The Fuel Experts Panel Evaluation said , "The panel does not recommend adoption of the wider plates with lower curvature (i.e., the 108-L core) for this reason."

Nevertheless, the neutronic calculations are reported here for the sake of completeness. Table B.1 is the TK-Solver MULREG correlation for the uranium mass and MWd, which leads to the relationship

$$\text{MWd} \approx (425\text{E} - 31) \times {}^{\text{Total}}\text{BOC} - 1255 \,. \qquad \text{(B-1)}$$

Figure B-2 illustrates the correlation.

The volume available for fuel meat is approximately 26.9 L, so

$$\text{MWd} \approx (11{,}422\text{E} - 832) \times \text{uranium density} - 1255 \,. \qquad \text{(B-2)}$$

A correlation for the rendement of the 108-L core is

$$\text{R} = 1.741 - 1.504 \times 10^{-5} \times \text{MWd} - \frac{.050421}{\text{E}} \,. \qquad \text{(B-3)}$$

The correlation coefficient is 0.713, and the standard error is only .038, or less than 3% of a typical rendement. Figure B.3 shows the correlations for 35% and 20% enrichment and the data points.

REFERENCE

B-1. See Sect. 6, ref. 1.

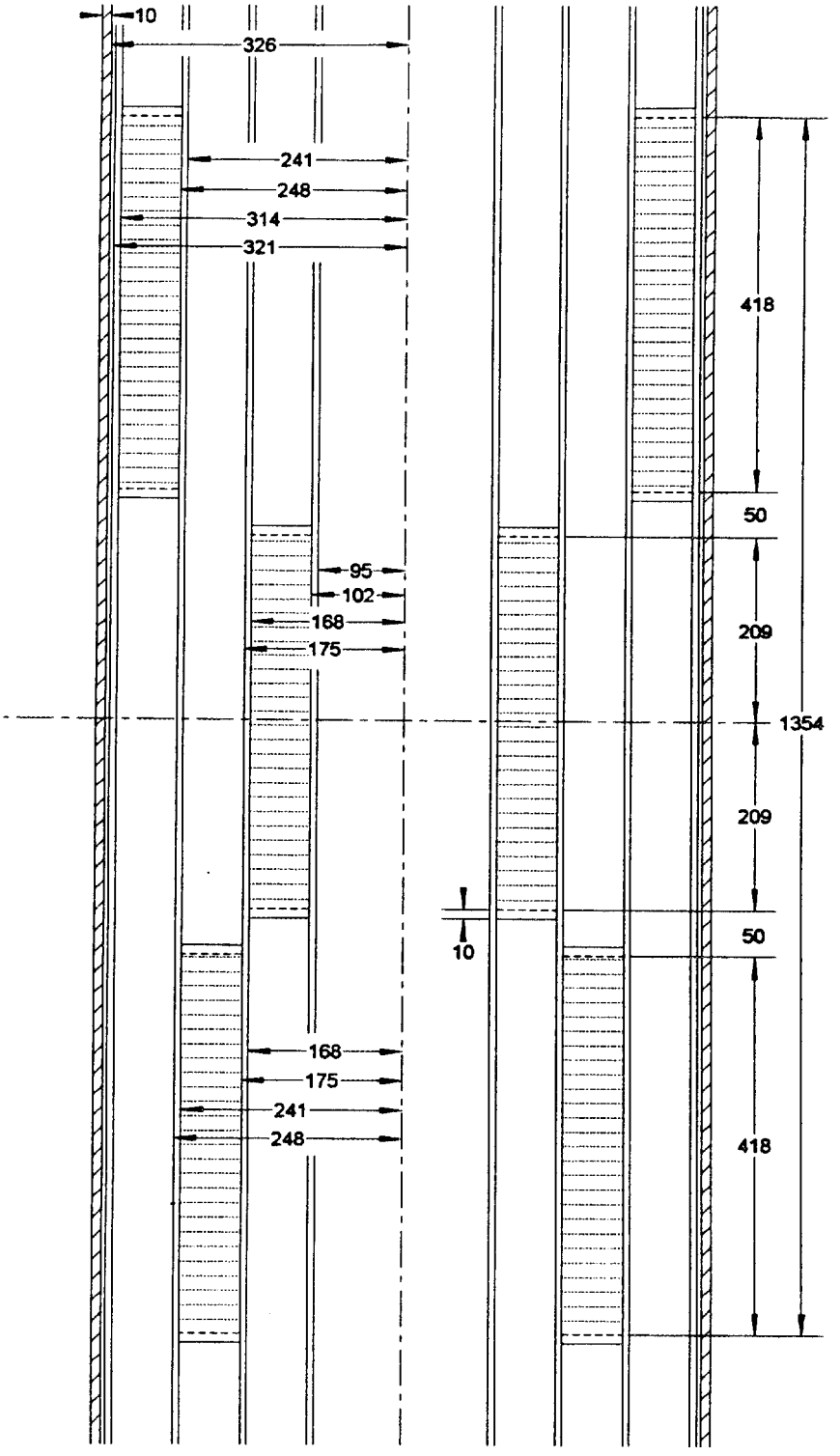


Fig. B.1. Dimensions of the 108-L three-element core design.

Table B.1. 108-L core correlation for MWd and fuel mass from TK Solver MULREG

Name	Result	Comment
C_0	3.187861	Constant
C_1	0.00253987	Coefficient for MWd variable
C_2	0.07851598	Coefficient for mass of other uranium isotopes at BOC
Syx	0.46902559	Standard error
R^2	0.99686549	Adjusted coefficient of multiple determination

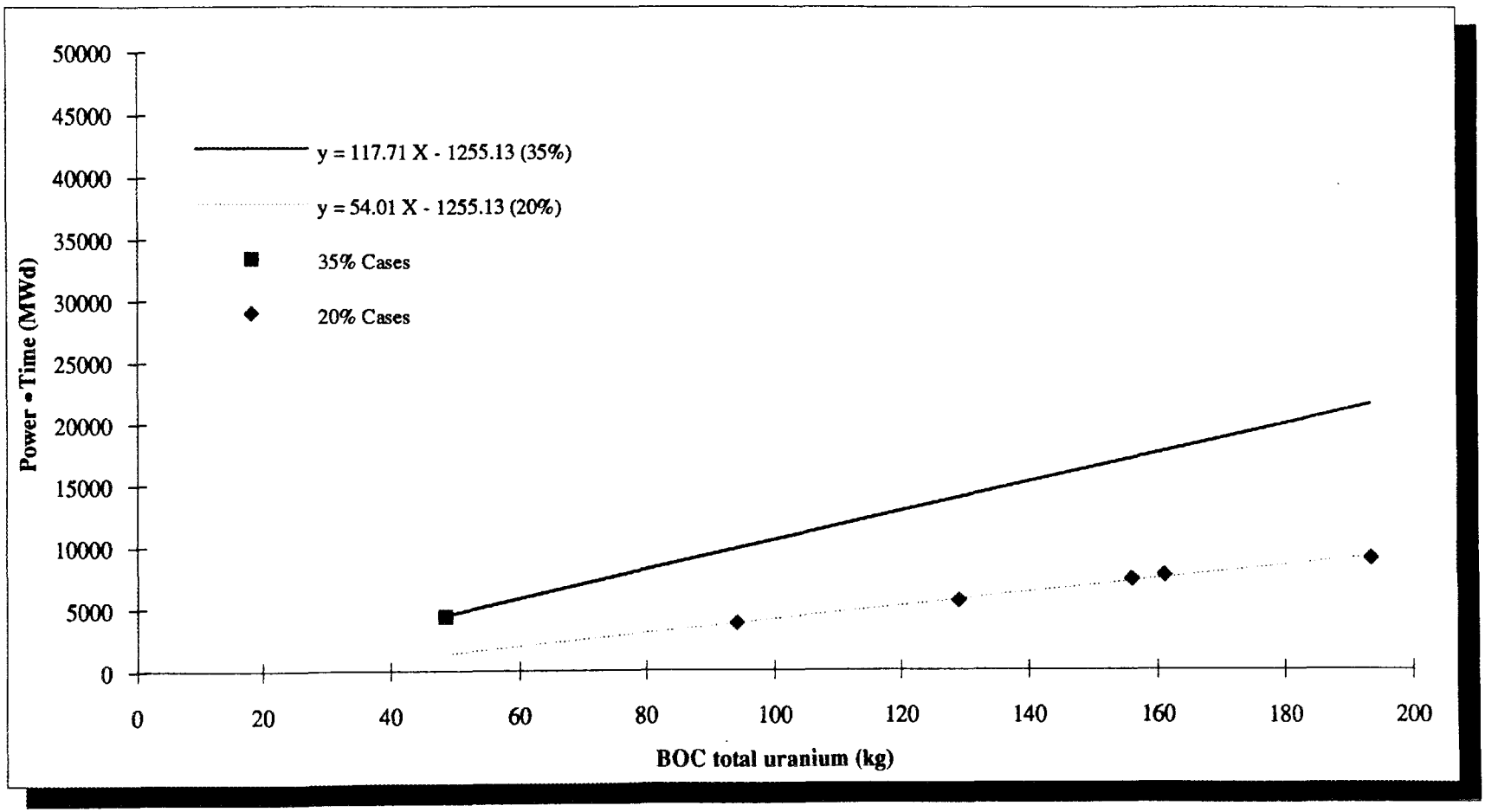


Fig. B.2. Correlations for MWd, enrichment, and beginning-of-cycle total uranium mass for the 108-L three-element core design.

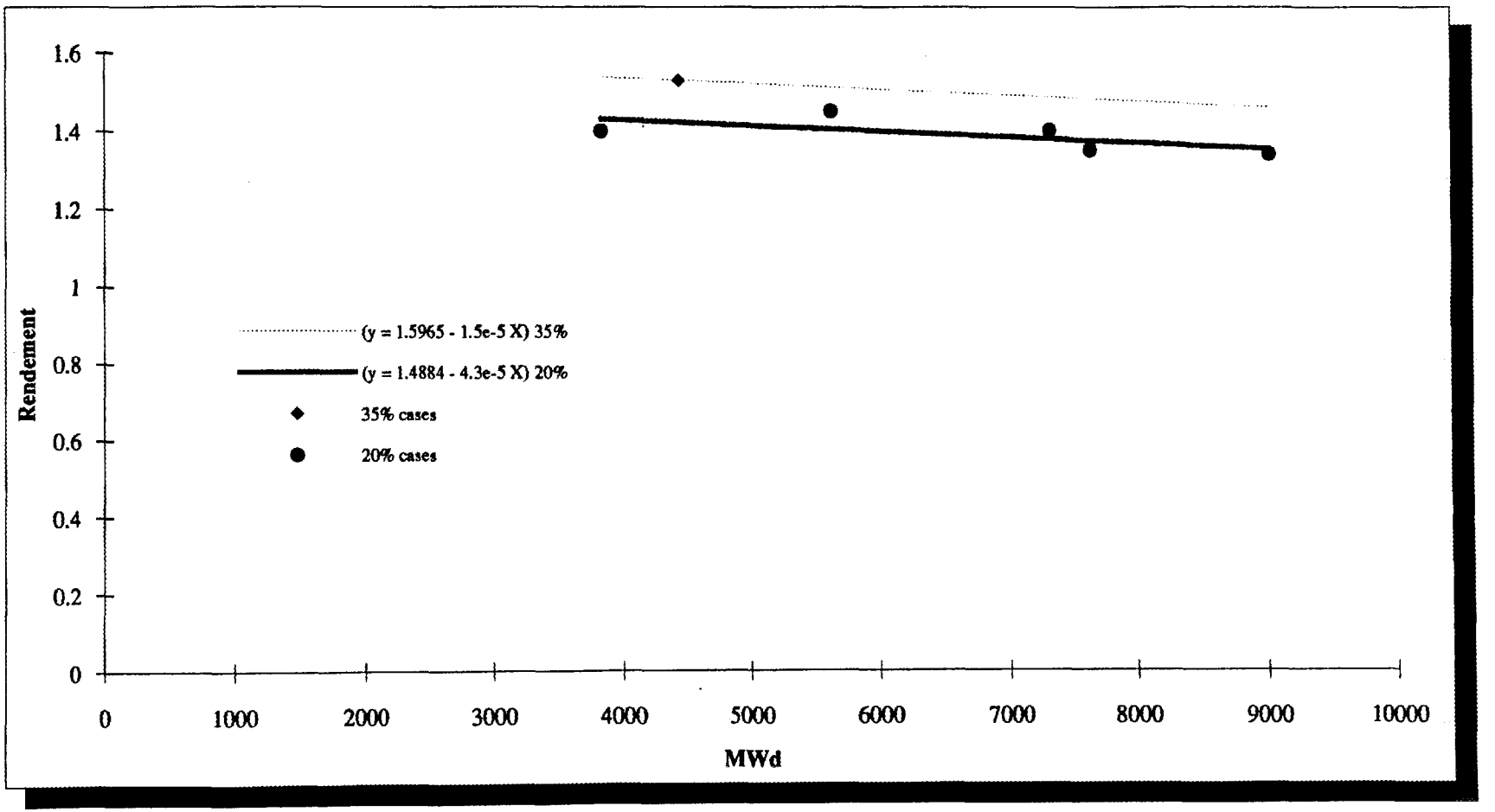


Fig. B.3. Correlations for rendement vs MWd for the 108-L three-element core design.

Appendix C: FUEL DEVELOPMENT

Appendix C

SUMMARY OF REVIEW FINDINGS

Fuel development: uranium density, technical risk, and reactor safety

U density, g/mL	BNL study expert evaluation*	DOE review of ANS fuel development program	Notes
>6.0	Less than 10% probability of success in development program		
6.0–3.5	50–95% probability of success in development programs		BNL panel also found "going to fuel loadings of 4.8 gU/mL and higher will introduce larger costs and uncertainties and require considerable development effort"
>4.8			ANS Project calculations show that the low thermal conductivity of U_3Si_2 fuels with such a density would necessitate a reduced reactor power
<3.5		Four (of five) panel members believe this fuel could be qualified with minimal cost and schedule impact	
3.5–1.3	95–100% probability of success in development programs		
<2.2		The fifth panel member believes that this fuel could be qualified with minimal cost and schedule impact	
1.7			Baseline ANS design

*See Sect. 6, ref. 1.

Appendix D: FUEL ENRICHMENT AND PLUTONIUM PRODUCTION

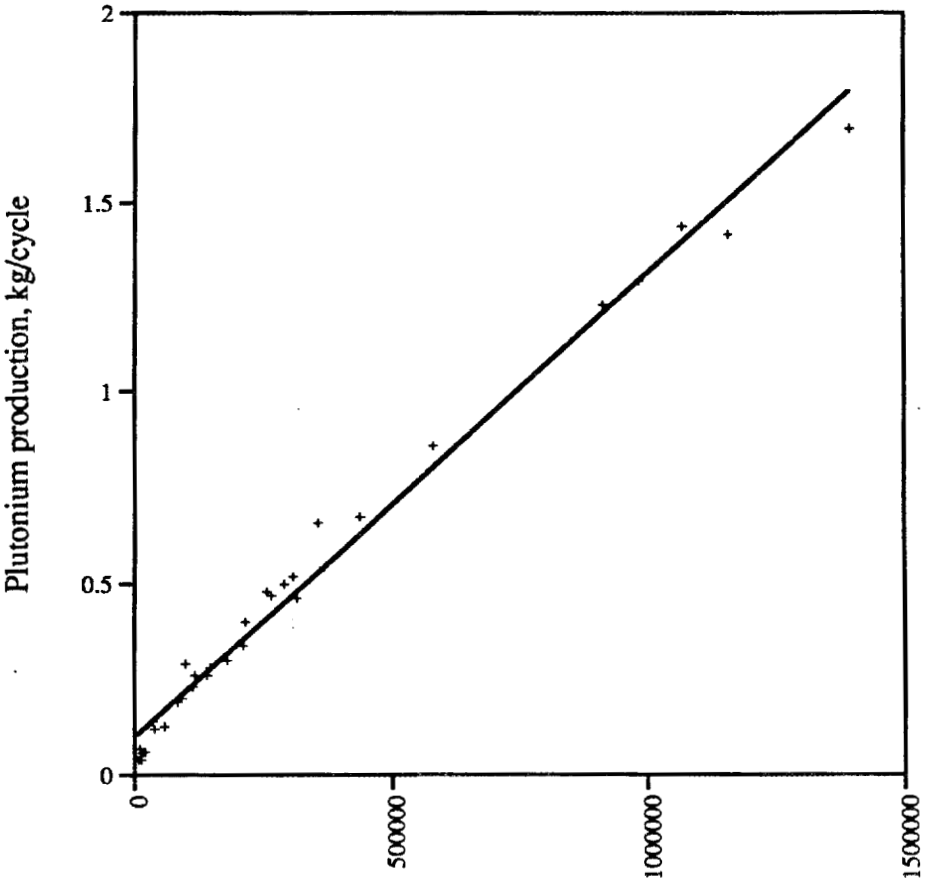
Appendix D

FUEL ENRICHMENT AND PLUTONIUM PRODUCTION

Plutonium is produced when neutrons are absorbed by the ^{238}U nuclei in the fuel. Therefore, we correlated the amount of plutonium produced per cycle with the amount of uranium isotopes other than ^{235}U present in a fresh core and the product of reactor power and core life, which is proportional to the total number of neutrons produced by each core. As shown in Fig. D-1, a fairly good straight line correlation can be made, although at the very low end of the scale, which is not of practical importance, the actual plutonium production per cycle is a few tens of grams less than the correlation implies. The TK Solver routine LINREG was used for these calculations.

The amount of plutonium produced is, of course, larger at low enrichments than at high, because there is more ^{238}U "feedstock." This is illustrated in Fig. D-2, which shows the amount produced per year (not just per cycle) for the modified core design operating at 330 MW with a 17-d core life (i.e., 5610 MWd). The initial core content of nonfissile uranium, nearly all of it ^{238}U, was calculated with the help of Eq. (4). At 50% enrichment, the production is about 4 kg per year, and with an LEU core (i.e., 20% enrichment), some 15 kg per year is produced.

$y = 1.214E\text{-}06x + 9.825E\text{-}02$ $r^2 = .9867$ Standard Error = .0528



Reactor power (MW) • cycle length (d) • amount of nonfissile uranium isotopes (kg)

+ ORNL case runs with various core sizes and power levels.

Fig. D.1. Plutonium production vs (MWd × ^{Other}BOC).

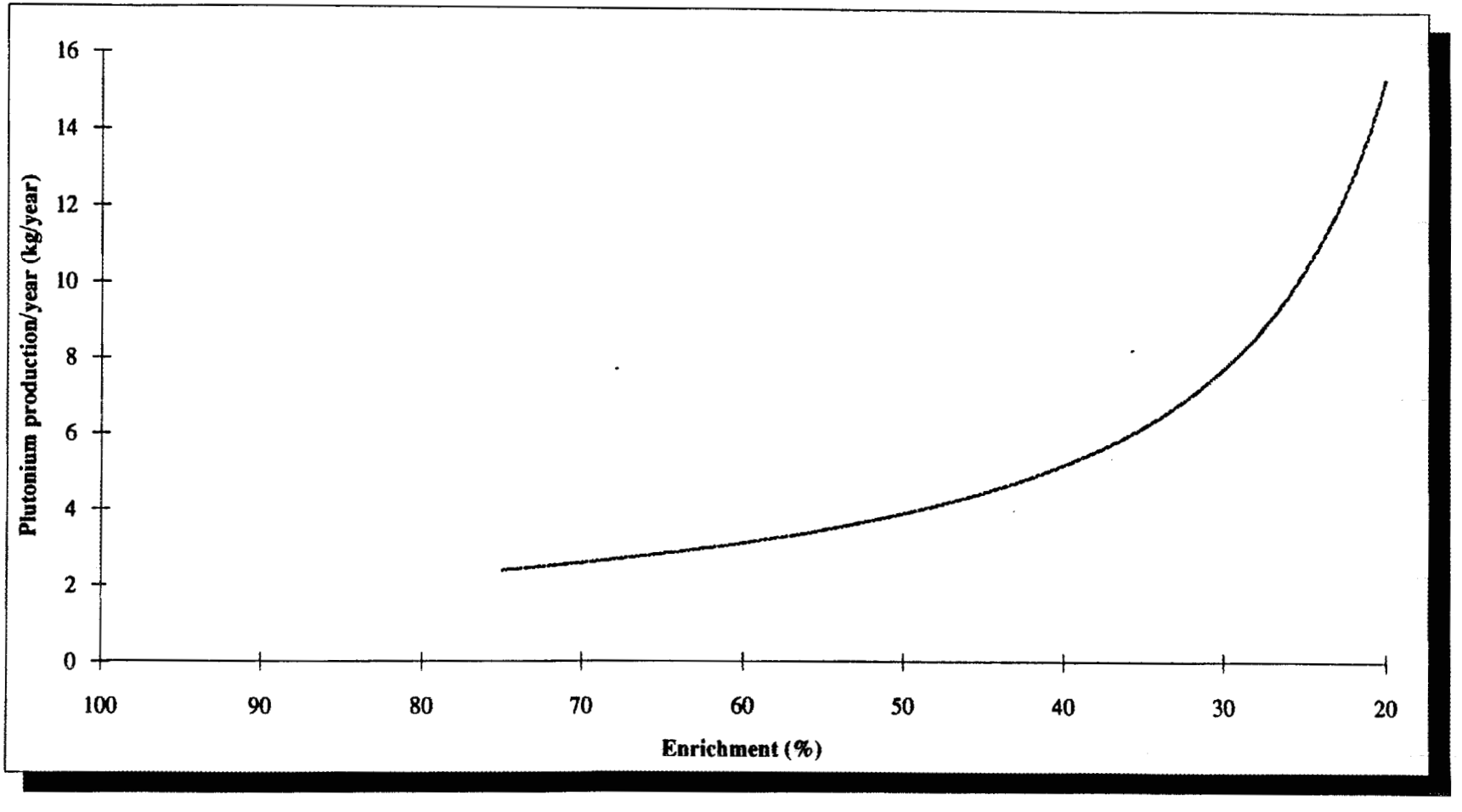


Fig. D.2. Annual plutonium production by the modified core design (330 MW, 17-d core life).

Appendix E. DOE REQUEST TO CONGRESS FOR FY-1995 BUDGET

FY 1994	FY 1995
The FY 1994 research and development (R&D) program will be a continuation of designs, tests, and modeling of ANS components to provide further results prior to ANS construction. Activities in FY 1994 will include fuel element R&D such as the irradiation of the second miniplate in the reflector region of the High Flux Isotope Reactor facility. Upper and lower fuel elements without uranium in fuel plates will be fabricated for use in core flow tests. Dynamic tests of some reactor core elements, the control rods, and the reflector shutdown rods will be carried out. A study will be undertaken to examine the impact on the ANS performance goals if low- or medium-enriched fuel is used rather than the highly-enriched fuel used for the design. The study will build on existing information using currently developed fuels and focus on low (20%) and medium (35%) enriched fuel. Continue work on Environmental Impact Statement.	Continue research and development leading to the construction of the Advanced Neutron Source. Complete corrosion tests under irradiation conditions at the High Flux Isotope Reactor. Fabricate dummy prototype core for flow tests. Begin operation of a natural circulation core facility. Continue effort on preliminary safety analysis report, Phase 2 of the Level 1 Probabilistic Risk Assessment and Phase 1 of the Level 2 Probabilistic Risk Assessment. Continue work on safety analysis report. Continue work on Environmental Impact Statement. Continue work on the fabrication of fuel elements for use in ANS critical experiments. Cold source tests will be performed. The current ANS design is based on a core fueled with highly enriched (93%) uranium fuel. Modification of the core design to incorporate low-enriched uranium fuel will be considered based on: the results of the FY 1994 study on the feasibility of fueling with medium- or low-enriched uranium; and the cost effectiveness of such a modification. If fueling the reactor with existing low-enriched uranium fuel would result in an unacceptable degradation of performance, the DOE will consider initiating a research program to attempt to develop a low-enriched uranium fuel option that would meet necessary performance standards. It is recognized that changing the current design could result in an increase to the total project cost. The cost to develop a new fuel has not been incorporated into the current estimate.

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