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PARAMETRIC NEUTRONIC ANALYSIS OF THERMIONIC REACTORS WITH U-ZrH DRIVER FUEL

by

M. H. Merrill and M. R. Hackney

Prepared under
Contract AT(04-3)-840
for the
San Francisco Operations Office
U. S. Atomic Energy Commission

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ABSTRACT

A series of parametric nuclear calculations have been done for a U-ZrH moderated thermionic reactor design. The critical size, thermal efficiency, control drum worth, and radial power distribution have been determined as a function of the number of thermionic fuel elements, the emitter thickness, the control drum thickness, and the weight percent of uranium in the U-ZrH driver fuel. In addition to the direct numerical results, some simple correlations have been derived for the critical size, thermal efficiency and control drum worth. They are useful for interpolating between the results of other calculations and should be applicable to a wider class of moderated thermionic reactors than those included in this report.

1. INTRODUCTION

In the past year, neutronic calculations have been done in some detail for two U-ZrH moderated thermionic reactor designs, a reactor with 36 thermionic fuel elements⁽¹⁾ and a space station power system with 60 thermionic fuel elements.⁽²⁾ The nuclear analysis of such systems is quite complex because of the large number of resonance absorbing materials and the very heterogeneous lattice. It has therefore been difficult to rapidly estimate the effects of proposed design changes or extrapolate performance data to other reactor sizes on the basis of the two reference designs studied. To improve this situation, a large number of parametric calculations have been done in which the neutronic properties of greatest interest have been investigated for a wide range of reactor sizes. In particular, the critical size, thermal efficiency, control drum worth and radial power distribution have been determined as a function of the number of thermionic fuel elements (TFEs), the emitter thickness, the control drum thickness, and the weight percent of uranium in the U-ZrH driver fuel. In addition to the numerical results, which are most directly applicable to estimating variations around the two reference designs, some simple correlations have been derived which should be applicable to a larger class of moderated thermionic reactors than those included in the present study.

The basic reactor concept and the parametric variations considered are described in Section 2 and the results are given in Section 3.

2. DESCRIPTION OF REACTOR AND PARAMETRIC VARIATIONS

A plan view of the space station reactor⁽²⁾ is shown in Fig. 2.1. The central region consists of 60 TFEs of about 3.3 cm outer diameter arranged in a lattice with U-ZrH fuel elements of about 1.1 cm outer diameter. A hexagonal unit cell of this lattice contains six U-ZrH fuel elements per TFE. The TFE design, apart from the parametric variations considered, is that of the 6F series of prototype thermionic elements now beginning testing at GGA. The U-ZrH fuel element design follows that developed in the SNAP-8 reactor program. In the axial direction, the TFE consists of six fueled thermionic emitters about 5 cm in length, separated by intercell connector regions about 2.3 cm long. Ten centimeter long axial reflector pieces of BeO are included on each end of both the TFEs and the U-ZrH elements.

The central lattice of TFEs and U-ZrH elements is designated as the "core region" in the subsequent discussion even though it is not the only region containing fuel. Surrounding the core region is additional U-ZrH fuel arranged on a hexagonal lattice and separated from the core region by a thin shroud. This region is designated as the "driver" region, since it provides much of the excess reactivity necessary to make a critical system with a smaller number of TFEs. Two additional benefits provided by the external driver region are the flattening of the radial power distribution in the core region, and the enhancement of the prompt negative temperature coefficient. A disadvantage, of course, is the additional waste heat produced in this region.

Outside of the driver is a radial reflector consisting of 18 reflector-control drums about 8.6 cm in diameter. The drums are made of solid BeO with inserts of B_4C on one side for control poison. The

Fig. 2.1 - Space Station Reactor Midplane Cross Section (from Ref. 2) (Dimensions in centimeters)

thickness of the B_4C segment is one-sixth of the drum diameter. Filler pieces of BeO and steel occupy the spaces between the drums.

The core, driver, and reflector drums are enclosed within a steel pressure vessel. The NaK coolant flows in the spaces between the TFEs and U-ZrH elements and in the channels surrounding the control drum drywells. The minimum spacing between fuel elements of either type is .38 mm.

Neutronic calculations for this reactor, described in Ref. 2, have established the basic feasibility of the design with respect to excess reactivity, power distribution, control worth and temperature coefficients. The present study was concerned with variations of this design, such as those which would be expected if a more detailed design of the system were carried out, or those involved in utilizing the reactor type for other space missions. To keep the scope of the study within reasonable limits, the basic TFE design and the core lattice arrangement of six U-ZrH elements per TFE was kept unchanged. Many other lattice arrangements could, of course, be devised, employing different sizes and numbers of TFEs and U-ZrH elements. Because of the severe heterogeneity of these lattices, the nuclear calculations required for a comprehensive survey would be very expensive, and the limited number of calculations done in the past for other lattice arrangements have not indicated that significant advantages could be obtained.

The following parameters were selected as variables in the present study:

1. Number of TFEs. The number of TFEs would normally be fixed by the electrical power and reliability requirements of the mission. For the hexagonal core lattice of the present design, the multiple of six nearest to the required number would be chosen, and the most desirable numbers from the standpoint of power distributions would probably be those which completed a hexagonal ring (6, 18, 36, 60 ... etc.). Since the TFEs are usually operated in pairs to obtain higher voltage, an even number is required, and the central lattice position is available for

sources or instrumentation. (In the present survey calculations this position contains a TFE.) Above a certain number of TFEs, an external driver region is no longer required for criticality. This criticality limited configuration (CL) is the upper bound on the number of TFEs considered in the present calculations. It is often a very large number (several hundred) which would not be of practical interest, since one would probably select an all-TFE fast reactor for an application requiring such a large number of TFEs. As will be shown in the results, however, the CL case is a useful point for normalizing some simple correlations. The other numbers used in the calculations were 19 and 61 TFEs.

2. Tungsten Emitter Thickness. Tungsten is the most important neutron absorber after uranium, and the amount in the core has a strong influence on the critical size. Emitters are subject to distortion under prolonged irradiation so that the thickness for a particular design is determined by the mission lifetime and available TFE test data. In the parametric calculations, thicknesses of 1 and 2 mm have been assumed.

3. Control Drum Thickness. In the present study, the boron carbide in the control drum occupies a segment one-sixth of the diameter in thickness, and this volume fraction remains constant as the drum size changes. Increasing drum thickness, therefore, increases control worth. It also improves thermal efficiency, since the increased BeO reflector thickness reduces the volume of driver fuel required for the same excess reactivity. Since the driver fuel adds much more reactivity per unit thickness than the BeO however, the overall reactor diameter is greater for thicker drums. In most cases one would therefore select the smallest drum thickness that gives adequate control margin. In the present study, drum thicknesses of 7.6, 11.4, and 15.2 cm have been used. Only natural boron was considered and the reactivity effects of shield materials outside the drums were not investigated.

4. Uranium Weight Percent in U-ZrH. Variations of uranium concentration in the U-ZrH elements are of interest in tradeoffs between thermal efficiency and size, power distribution adjustments, and temperature coefficient enhancement. Uranium concentrations up to 20 weight percent are considered feasible for this reactor system. Values of 5, 10.5 and 15 weight percent were selected for the present study.

5. Fuel Type. Several cases were included in which U-233 replaced U-235 in both TFEs and U-ZrH elements. Only criticality limited systems were considered.

6. Excess Reactivity. The excess reactivity of the cold clean system must be sufficient to cover the temperature defect to power, the burnup of fuel and buildup of fission products, the redistribution and loss of the hydrogen in the U-ZrH fuel, and uncertainties in the calculations remaining at the time of construction. Based on experience with the two reference designs, a multiplication factor of $k_{\text{eff}} = 1.05$ with all control drums turned full out has been assumed to be adequate. Since the uncertainty in the calculations at the present time is substantial, several cases were recalculated to determine the size required for $k_{\text{eff}} = 1.10$.

All calculations were done with six-group P_1S_4 transport theory in one-dimensional radial geometry. Further details on the calculational methods are given in Ref. 2.

3. RESULTS

The results of greatest interest are the critical reactor diameter, the thermal efficiency (fraction of total fissions occurring within the TFE diodes), the control drum worth, the reactivity worth per unit thickness of the driver region, and the peaking factors for the radial power distributions. These are summarized for a set of 24 consistent cases with U-235 fuel in Table 3.1. Cases with U-233 fuel are given in Table 3.2. (The parametric variables in the first three columns of these tables are used without dimensions for case identification in several of the subsequent figures.)

There are numerous ways to plot all of these data. One method that may be useful is to consider the most likely sequence of design choices. The number of TFEs is chosen first according to the electric power output required, with some extra elements provided to allow for failures. The thinnest emitter thickness is then selected consistent with the mission lifetime and available test data. The uranium weight percent selected for the U-ZrH fuel would probably be the highest allowed by the metallurgists, since the results of Table 3.1 show no advantage in thermal efficiency for lower weight percents. One then goes to Fig. 3.1* on which is plotted control worth as a function of number of TFEs, emitter thickness, and drum thickness for 15 w/o U-ZrH. This figure is used to select a drum thickness that will give adequate control margin. Consistent with the specification of $k_{\text{eff}} = 1.05$ for all cases in the set, the control margin should be at least $.05 \Delta\rho$ plus the worth of whatever number of

*The calculations in this study were done before the recent project decision to report all results in the metric system. Units have been converted to the extent feasible, but for ease of interpolation in the figures, the drum size in inches is retained.

Table 3.1 - Summary of Results for U-235 Fueled Cases ($k_{eff} = 1.05$)

Design Variables				Principal Results					
Emitter Thickness (mm)	w/o U in U-ZrH	Control Drum Diameter (cm)	Number of TFEs	Reactor Diam. (cm)	Thermal Efficiency	Control Worth ($\Delta\rho$)	Driver Worth ($\Delta\rho/cm$)	Max/Min Radial Power Peak	
								Core	Driver
1	15	7.6	94 (CL)	62.76	.728	.0439	-	1.70	-
			61	59.87	.545	.0563	.0227	1.26	1.23
			19	51.03	.270	.0890	.0371	1.27	1.22
		15.2	72 (CL)	72.47	.728	.1148	-	1.32	-
			61	72.09	.623	.1293	.0222	1.23	1.35
			19	63.27	.314	.2369	.0356	1.29	1.40
	5	7.6	135 (CL)	71.68	.890	.0304	-	1.93	-
			61	66.14	.550	.0477	.0186	1.32	1.44
			19	57.12	.280	.0721	.0262	1.55	1.43
		15.2	107 (CL)	81.05	.890	.0763	-	1.48	-
			61	78.33	.606	.1106	.0192	1.34	1.50
			19	69.37	.312	.1880	.0252	1.56	1.47
		7.6	160 (CL)	76.53	.689	.0246	-	2.09	-
			61	63.20	.409	.0534	.0245	1.15	1.21
			19	52.32	.215	.0855	.0392	1.28	1.20
2	15	15.2	128 (CL)	85.55	.689	.0604	-	1.56	-
			61	75.51	.460	.1227	.0255	1.19	1.42
			19	64.57	.249	.2258	.0382	1.29	1.37
		7.6	345 (CL)	103.68	.867	.0097	-	2.78	-
			61	70.76	.396	.0451	.0198	1.43	1.44
			19	59.08	.218	.0681	.0194	1.60	1.49
		15.2	292 (CL)	112.09	.867	.0227	-	2.11	-
			61	83.03	.433	.1054	.0229	1.43	1.50
			19	71.35	.242	.1763	.0270	1.59	1.46
	5	7.6	94 (CL)	62.76	.728	.0439	-	1.70	-
			61	59.87	.545	.0563	.0227	1.26	1.23
			19	51.03	.270	.0890	.0371	1.27	1.22
		15.2	72 (CL)	72.47	.728	.1148	-	1.32	-
			61	72.09	.623	.1293	.0222	1.23	1.35
			19	63.27	.314	.2369	.0356	1.29	1.40

Table 3.2 - Summary of Results for U-233 Fueled Cases

Design Variables					Principal Results				
Emitter Thickness (mm)	w/o U in U-ZrH	Control Drum Diameter (cm)	Number of TFEs	k_{eff}	Reactor Diameter (cm)	Thermal Efficiency	Control Worth ($\Delta\rho$)	Core Region Worth ($\Delta\rho/cm$)	Max/Min Radial Power Peak
1	15	7.6	17	1.04	37.59	.723	.2037	.067	1.21
		7.6	18	1.05	37.85	↓	.1956	.051	1.21
		9.47	16	1.05	40.39		.2700	.049	1.14
		15.2	13	1.05	50.04		.6816	.048	1.23
		15.2	16	1.10	51.56		.5279	.043	1.24
2	5	7.6	22	1.05	39.88	.777	.1680	.064	1.29
	15	7.6	23	1.05	40.39	.679	.1647	.048	1.29
		7.6	26	1.08	41.66	↓	.1441	.067	1.31
		9.47	21	1.05	42.93		.2254	.044	1.19
		15.2	17	1.05	52.32		.5460	.047	1.22
	5	7.6	32	1.05	44.20	.730	.1281	.045	1.41

Fig. 3.1. Control Drum Worth as Function of Emitter Thickness, No. of TFEs and Drum Thickness.
15 w/o U in U-ZrH

stuck drums is assumed conceivable plus a modest allowance for uncertainty. A control margin of .10 $\Delta\rho$ would be a good starting guess for most of these systems. With the control drum thickness selected from Fig. 3.1, the size of the reactor is found from Fig. 3.2 and the thermal efficiency from Fig. 3.3. The total thermal power of the system can then be determined and the size and weight of shield and radiator estimated. If the system still looks attractive for the proposed mission, some detailed nuclear calculations can be started.

All of the results in the tables are very approximate because of the large calculational uncertainties mentioned previously. In addition, other lattice geometries and material combinations of possible interest are not included in the survey. We are therefore interested in seeking more general results and relationships from the study which would continue to be valid and useful as cross section data and calculational models are modified, and as other design concepts are studied. Results of this type obtained to date are discussed in the subsections below for each of the items in the results columns of Fig. 3.1.

3.1 CRITICAL REACTOR DIAMETER

Plots of critical reactor diameter as a function of the number of TFEs for a particular set of the other variables give nearly straight lines on log-log paper. This is illustrated in Fig. 3.4 in which the numbers have been normalized to the criticality limited cases. The relationship is

$$\frac{D}{D_{CL}} = \left(\frac{N}{N_{CL}} \right)^x \quad (3.1)$$

The data are listed in Table 3.3* including values of the exponent x.

*It will be noted in the table that fractional numbers of TFEs occur. In the CL cases this is because the computer code searches for a critical core radius which may not correspond to an integral number of lattice unit cells. In the cases of 61 TFEs, the space station reactor dimensions were used for the core region, and the outer TFEs in this design do not have the full complement of associated U-ZrH elements. The correlation is really based on the number of lattice unit cells.

Fig. 3.2. Reactor Diameter as a Function of Emitter Thickness, No. of TFEs and Drum Thickness.
15 w/o U in U-ZrH

Fig. 3.3. Thermal Efficiency as a Function of Emitter Thickness, No. of TFEs and Drum Thickness.
15 w/o U in U-ZrH

Fig. 3.4. Reactor Diameter vs. Number of TFEs Normalized to Criticality Limited Case

Table 3.3 - Correlation of D/D_{CL} vs. N/N_{CL}

Case	N	D _{cm}	N/N_{CL}	D/D_{CL}	$x = \frac{\ln(D/D_{CL})}{\ln(N/N_{CL})}$
1 mm-15 w/o - 7.6 cm - 15.2 cm	93.50	62.76	1.0	1.0	-
	57.86	59.87	.6188	.9538	.0985
	19.00	51.03	.2032	.8128	.1300
	71.63	72.47	1.0	1.0	-
	57.86	72.09	.8077	.9950	.0235
	19.00	63.27	.2653	.8731	.1023
	134.70	71.68	1.0	1.0	-
	57.86	66.14	.4295	.9228	.0950
	19.00	57.12	.1411	.7968	.1160
- 15.2 cm	106.75	81.05	1.0	1.0	-
	57.86	78.33	.5419	.9665	.0556
	19.00	69.37	.1780	.8559	.0902
2 mm-15 w/o - 7.6 cm	160.35	76.53	1.0	1.0	-
	57.86	63.20	.3608	.8256	.1880
	19.00	52.32	.1185	.6836	.1783
	127.85	85.55	1.0	1.0	-
	57.86	75.51	.4525	.8828	.1573
	19.00	64.57	.1486	.7550	.1475
2 mm-5 w/o - 7.6 cm	344.60	103.68	1.0	1.0	-
	57.86	70.76	.1680	.6825	.2141
	19.00	59.08	.0551	.5697	.1941
	291.59	112.09	1.0	1.0	-
	57.86	83.03	.1984	.7408	.1855
	19.00	71.35	.0652	.6366	.1654

Deviations from the simple approximation of Eq. (3.1) are notable for some cases with very large control drums and large N_{CL} . For purposes of rough estimates however, the relationship provides an easy way to interpolate between a limited number of calculations. If the criticality limited size is not known, it appears possible to estimate it by extrapolating the driver thickness to zero on a linear plot. This procedure is illustrated in Fig. 3.5 and is seen to work quite well for cases with less than 100 TFEs. The actual calculated N_{CL} for each case is written on the figure in parenthesis beside the estimate. Another method for estimating N_{CL} is given in the next section.

3.2 THERMAL EFFICIENCY

The thermal efficiency given in Table 3.1 is the product of three factors

$$\eta_{th} = f_{core} \times f_{axial} \times f_{cell} \quad (3.2)$$

f_{core} is the fraction of the total reactor fissions occurring in the core region. f_{axial} is the fraction of the core region fissions occurring in the TFE fuel and in those parts of the six associated U-ZrH elements adjacent to the emitter. (The remaining core region fissions are from the U-ZrH fuel adjacent to the intercell regions.) The value of f_{axial} is about 0.9 for 15 w/o U in U-ZrH and of course becomes 1.0 for unfueled ZrH. Linear interpolation was used for intermediate uranium weight percents. f_{cell} is the fraction of fissions in the TFE fuel in a one-dimensional radial cell calculation including the TFE fuel and its six associated U-ZrH elements. Values of f_{cell} and f_{axial} are plotted in Fig. 3.6 as a function of uranium weight percent.

Of the three factors, f_{core} has the greatest effect on the efficiency. It has been found, as with the critical diameter, that f_{core} plots as a straight line on log-log paper as a function of the number of TFEs. The normalized form of the relationship is

Fig. 3.5. Prediction of N_{CL} by Extrapolation of Driver Thickness to Zero (Calculated N_{CL})

Fig. 3.6. Efficiency Factors f_{cell} and f_{axial} (Eq. 3.2 in Text)

$$f_{\text{core}} = \left(\frac{N}{N_{\text{CL}}} \right)^y \quad (3.3)$$

The data are plotted in Fig. 3.7 and the values of the exponent y , which do not vary widely, are given in Table 3.4. It is worth noting that since f_{core} equals 1.0 for the criticality limited case, the relationship may be used to approximately predict N_{CL} if f_{core} is known for at least two other values of N .

3.3 CONTROL DRUM WORTH

At an early point in the study, a simple method was derived for predicting control drum worth using one-group diffusion theory constants for the core composition and an empirical constant.

The multiplication factor k is given in the one-group model by

$$k = \frac{k_{\infty}}{1 + L^2 B^2} \quad (3.4)$$

We consider only radial leakage and assume the axial leakage is represented by an absorption cross section included in the calculation of k_{∞} and L^2 . The radial buckling B^2 is given by

$$B^2 = \left(\frac{2.405}{\tilde{R} + \delta} \right)^2 \quad (3.5)$$

where $\tilde{R}$ is the core radius R plus the extrapolation distance $.71 \lambda_t$, and δ is the reflector savings. The reactivity change $\Delta\rho$ between the drums-in and drums-out positions is

$$\Delta\rho = \frac{k_{\text{out}} - k_{\text{in}}}{k_{\text{out}} k_{\text{in}}} \quad (3.6)$$

Fig. 3.7. Core Region Power Fraction as a Function of No. of TFES

Table 3.4 - Correlation of f_{core} and N/N_{CL}

Case	N/N_{CL}	f_{core}	$y = \frac{\ln f_{core}}{\ln(N/N_{CL})}$
1 mm - 15 w/o - 7.6 cm - 15.2 cm	1.0	1.0	-
	.6188	.7487	.6030
	.2032	.3711	.6220
	1.0	1.0	-
	.8077	.8543	.7371
	.2653	.4313	.6336
	1.0	1.0	-
	.4295	.6187	.5683
	.1411	.3143	.5910
1 mm - 5 w/o - 7.6 cm - 15.2 cm	1.0	1.0	-
	.5419	.6807	.6280
	.1780	.3507	.6071
	1.0	1.0	-
	.3608	.5935	.5117
	.1185	.3125	.5453
	1.0	1.0	-
	.4525	.6677	.5094
	.1486	.3610	.5345
2 mm - 15 w/o - 7.6 cm - 15.2 cm	1.0	1.0	-
	.1680	.4564	.4395
	.0551	.2513	.4765
	1.0	1.0	-
	.1984	.4987	.4301
	.0652	.2789	.4676

which, with our definition of k , gives

$$\Delta\rho = \frac{L^2}{k_\infty} \left(B_{in}^2 - B_{out}^2 \right) \quad (3.7)$$

The only way to represent a change in B^2 for drums in or out is with a change in the reflector savings, since all other parameters in the equation are constant properties of the core. Textbook equations for the reflector savings generally consider only slab geometry for a single region core and a reflector of pure moderator. This equation is complex enough without introducing multiple regions and black absorbers. Under certain ideal conditions (never encountered in thermionic reactors) the slab geometry equation reduces to the approximation that the reflector savings equals the reflector thickness. While this approximation is not true in the present case, there appears to be a reasonable proportionality between changes in effective reflector thickness and changes in actual reflector savings. The effective reflector thickness for the drums-out condition is defined as the total thickness of reflector materials including shrouds, poison and pressure vessel. When all the drums are full-in, the effective thickness is defined as the distance between the fuel region outer radius and the poison region outer radius. We are in effect assuming that with the drums in, any fast neutron leaking out through the poison section will never be able to return. The change in effective reflector thickness is the difference of these numbers and is only a function of geometry.

The actual reflector savings, δ , for drums in and out has been computed for a wide variety of cases. Using values of k_∞ and L^2 for the core composition (homogenization is required for two-region cores), we find the value of δ in Eq. (3.5) which will force Eq. (3.4) to give the same k as the multigroup transport calculation. The difference in δ between the drums-in and the drums-out condition is plotted as a function of the change in effective reflector thickness in Fig. 3.8.

Figure 3.8. Change of One-Group Reflector Savings (δ) as a Function of Change in Effective Reflector Thickness (R_{eff})

It is seen that cases from the current parametric studies may be fitted to a nearly straight line. (The approximate relation is $\Delta\delta = .5 \Delta R_{\text{eff}}$.) All of the cases in the parametric survey have the same reflector cross section data, but significantly different drum sizes and core compositions. Cases from other studies with different reflector cross section data do not fall as close to this line. In the case of the TREX fast reactor⁽³⁾ we are probably underestimating the effective reflector thickness for the drums-in condition, since there are a greater proportion of high energy neutrons which can traverse the poison segment twice. This suspicion is confirmed by the fact that the TREX case with fully enriched boron is closer to agreement with the hydride cases. Rather than complicate the model to make all reactor types fit one curve, it would be easier to derive a second curve for fast reactors.

A relationship as given by the curve of Fig. 3.8 seems to be generally valid for any reactor of a particular type (fast, intermediate or thermal), as long as the reflector cross sections are not different between cases. Large variations of core composition and size may be treated as well as variations of drum thickness and poison segment thickness. A different curve would be required if the poison concentration (e.g., boron enrichment) changes.

3.4 DRIVER REACTIVITY WORTH

The large reactivity worth of the driver fuel per unit thickness is an important characteristic of the reactor, since it gives some confidence that the presently large uncertainties in the nuclear calculations can be compensated for without greatly increasing the size of the system. Values of the driver worth coefficient α_d given in Table 3.1 vary from .02 to .04 $\Delta\rho$ per cm. depending primarily on reactor radius. They are obtained in the present study from the edit of the computer code which searches for a driver thickness giving the desired value of $k_{\text{eff}} = 1.05$. Since the coefficient is most often used for estimating purposes, an extremely accurate value is not usually needed, and one would not go too far wrong in simply averaging the column of results

in Table 3.1 (obtaining .027 $\Delta\rho/\text{cm.}$). A better method is to correlate the coefficient with radius. It has been found that the relationship

$$\alpha_d = \frac{1}{R^z} \quad (3.8)$$

works very well, where R is the outer driver region radius in cm. Values of the exponent z are given in Table 3.5. In this case the exponents are all very close and using a single average value of 1.2 would make a good rule of thumb for all cases in the study. For estimating core size changes for large reactivity adjustments, the equation may be easily integrated. This procedure has been checked against the calculations mentioned earlier in which the driver thickness was adjusted to give $k_{\text{eff}} = 1.10$.

3.5 RADIAL POWER DISTRIBUTION

The maximum/minimum power density ratios for the core and driver regions are given in Table 3.1 for all cases. The complete power distributions are plotted in Fig. 3.9 for the set of cases having 2 mm emitters and 15 w/o U in U-ZrH. The great variety of possible shapes illustrated suggests that it may not be easy to characterize a satisfactory or unsatisfactory power distribution by single parameters like the two max/min ratios.

In the case of the core region, the TFE fuel would be zoned axially and radially in any design of interest in order to optimize thermionic performance. This usually results in a loss of reactivity since fuel is removed from the center to locations of lower reactivity worth. The driver region is then made somewhat thicker to compensate this loss. (In criticality limited cores, the control drums may be enlarged or additional TFEs added.) It would be helpful if the core region max/min power density ratio in these unzoned survey calculations could measure the amount of zoning required or the reactivity loss to be anticipated. To date no useful correlation has been found; in part because the fairly expensive zoning calculations have only been done for the two reference designs. Another difficulty is that for small numbers of TFEs, the core

Table 3.5 - Reactivity Worth of Driver Region

Case	No. of TFEs	Mean Driver Radius in Criticality Search - R-cm	Driver Worth $\Delta\rho/\Delta R$	$z = \frac{\ln(\Delta\rho/\Delta R)}{\ln R}$
1 mm-15 w/o - 7.6 cm	61	21.95	.0227	-1.226
	19	15.95	.0371	-1.189
	- 15.2 cm	61	.0222	-1.247
		19	.0356	-1.227
1 mm-5 w/o - 7.6 cm	61	24.47	.0186	-1.246
	19	18.91	.0262	-1.239
	- 15.2 cm	61	.0192	-1.265
		19	.0252	-1.269
2 mm-15 w/o - 7.6 cm	61	22.78	.0245	-1.187
	19	16.28	.0392	-1.161
	- 15.2 cm	61	.0255	-1.186
		19	.0382	-1.192
2 mm-5 w/o - 7.6 cm	61	26.68	.0198	-1.194
	19	21.90	.0194	-1.277
	- 15.2 cm	61	.0229	-1.189
		19	.0270	-1.234

max/min power ratio mainly measures the power peak at the core-driver interface which significantly overestimates the power density of the outer ring of TFEs (note Fig. 3.9).

In the case of the driver region, a large max/min power density ratio suggests thermal hydraulic design problems, but no quantitative criteria have been developed yet. One of the complications is the fact that the control drum positions have a strong influence on the radial power distribution in the driver, and the reactor would, of course, not be operated with all drums out as in the survey calculations. In particular, the driver power distributions for the cases with large control drums shown in Fig. 3.8 would probably look much more attractive if the drums were in an operating configuration.

3.6 USE OF U-233 AS FUEL

Only a limited number of cases were considered for U-233 fueled reactors since the range of interest for moderated designs is fairly narrow. Previous studies have indicated that a fast spectrum all-TFE reactor using U-233 would be criticality limited at about 50 to 60 TFEs. Moderated U-233 systems would presumably only be considered for missions with relatively low power requirements (15-50 TFEs) for which the advantages in size and efficiency over a U-235 system outweigh the handling problems associated with U-233.

The main objective in the present calculations was to provide a few comparisons with the U-235 cases. The cases summarized in Table 3.2 show substantial reductions in diameter and large gains in efficiency compared to the corresponding 19 TFE cases in Table 3.1. The control margins (calculated from Eq. 3.7) are more than adequate.

The nuclear calculations must be regarded as more uncertain for the U-233 cases since the cross sections of U-233 have not been completely measured or well tested in the intermediate energy range. For example, there have been no measurements at all to date of the U-233 capture cross section above 2 Kev, which is close to the median fission energy of the systems considered here. Fortunately, the reactivity change

produced by small increases in core radius (listed in Table 3.2) is large enough to compensate major uncertainties without greatly increasing the reactor size.

3.7 FAST FLUENCE FOR RADIATION DAMAGE

The fast neutron fluence in the core zone above an energy of 0.1 Mev is of interest as a measure of radiation damage to the insulators. Peak and average values of this quantity are given in Table 3.6 in units of nvt per 10^4 hours. To put all cases on a comparable basis, we assume an electrical power production of 1 kWe per TFE and a thermionic efficiency of 10%. The thermal power level required for this electrical output is computed from the thermal efficiencies in Table 3.1.

The average nvt values are mainly of interest to compare the effects of compositional changes on the core region spectrum. Of these, the effect of varying uranium weight percent in U-ZrH is the most important. Lowering the uranium weight percent from 15 to 5 drops the average nvt about 25% for cases with external drivers and about 16% for criticality limited cases. Unfortunately, this reduction in damage flux would come at the expense of increased reactor diameter.

The maximum nvt is of interest for estimating the useful reactor lifetime. Values in the range of $3-4 \times 10^{21}$ nvt are presently assumed to be limiting with respect to insulator swelling. The variations of maximum nvt with parametric variables follow the same trends observed for the average values.

Table 3.6 - Maximum and Average Fast Fluence for All Cases

Case	No. of TFEs	Power kwt	Avg. nvt in 10^4 hr	Max. nvt in 10^4 hr
1 mm-15 w/o - 7.6 cm	94	1291	$.979 \times 10^{21}$	1.39×10^{21}
	61	1119	1.01	1.24
	19	704	.902	.967
	- 15.2 cm	72	.938	1.22
	61	979	.987	1.22
	19	605	.893	.958
1 mm-5 w/o - 7.6 cm	135	1517	.832	1.23
	61	1109	.817	.958
	19	697	.700	.741
	- 15.2 cm	107	.806	1.09
	61	1007	.811	.952
	19	609	.699	.740
2 mm-15 w/o - 7.6 cm	160	2322	1.10	1.65
	61	1491	1.13	1.30
	19	884	1.02	1.06
	- 15.2 cm	128	1.07	1.48
	61	1326	1.11	1.28
	19	763	1.01	1.05
2 mm-5 w/o - 7.6 cm	345	3979	.927	1.52
	61	1540	.888	.961
	19	872	.772	.795
	- 15.2 cm	292	.912	1.38
	61	1409	.886	.959
	19	785	.770	.793

4. REFERENCES

1. A. J. Gietzen, et al, "25 kWe Thermionic Power System for Space Base Application," Proceedings of the 1970 IEEE Thermionic Conversion Specialist Conference, October 26-29, 1970, p. 145.
2. C. Fisher, et al, "Thermionic Reactor System for Auxiliary Power and Electric Propulsion," Proceedings of the 1971 IEEE Thermionic Conversion Specialist Conference, October 4-7, 1971.
3. C. R. Fisher, et al, "Thermionic Reactor Experiment Design," Proceedings of the 1970 IEEE Thermionic Conversion Specialist Conference, October 26-29, 1970, p. 130.