NUCLEAR REACTOR DESCRIPTIONS FOR SPACE POWER SYSTEMS ANALYSIS*

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For the small, high performance reactors required for space electric applications, adequate neutronic analysis is of crucial importance, but in terms of computational time consumed, nuclear calculations probably yield the least amount of detail for mission analysis study.

It has been found possible, after generation of only a few designs of a reactor family in elaborate thermomechanical and nuclear detail to use simple curve fitting techniques to assure desired neutronic performance while still performing the thermomechanical analysis in explicit detail. The resulting speed-up in computation time by three orders of magnitude, permits a broad detailed examination of constraints by the mission analyst.

INTRODUCTION

Space research in general and Space nuclear electric systems in particular are in a period of priority reassessment and technological retrenchment. Even so, new concepts and different combinations of older concepts are being put forth for possible application. In order to provide any hope of progress, the mission analyst, faced with a restricted list of missions and of vehicles, must develop a set of analyses more consistent and credible than those to date. One of the chief obstacles in his path is the state of nuclear reactor heat source data which is usually presented as a pointwise design or as a set of parametric statements so overgeneralized as to be effectively useless.

Mission analysis can hardly optimize a point, or give a reasonable estimate of specific research potential on systems where it has neither breadth nor depth of knowledge. On the other hand, the designer of the nuclear reactor heat source has insufficient information to optimize his system to the particular mission since it in turn lacks definition.

To be effective, the reactor descriptions must be presented to the mission analyst in considerable detail. However, the parametric constraints must be held to an absolute minimum or the surface to be fitted will be as full of confusing depressions as a lunar landscape.

Some years ago the Lawrence Radiation Laboratory developed a method of parallel-optimization-path reactor design for cylindrical reactors, which coupled thermo-mechanical design to multi-group neutron transport analysis to allow rapid assessment of the effects of acquired research data and potential lines of research on reactor performance. It was found practicable to rapidly convert from one type of reactor to another. Beyond this, the consistency inherent in a single pass computerized design produced configurations significantly superior to those that the same group achieved by traditional reactor design methods.

This method requires approximately 5 to 10 minutes of CDC-6600 time to produce a design point which is too long to be suitable for inclusion in mission analysis work. The neutronic routines are called 30 to 40 times in a typical iterative design sequence and require three to four orders of magnitude more computation time than the thermomechanical portion.

The obvious approach is then to attempt to eliminate the detailed neutronic analysis if possible. As a demonstration case, we have selected a heatpipe-cooled reactor type previously studied at LRL.[1] We selected the criticality-limited design region and designed a family of 12 reactors in detail at power levels of .6, 2., and 4.6 Mwth. Fueled core aspect ratios (L/D) were chosen as .8, 1.2, 1.6, and 2.0.

This paper purposely avoids extensive tables and reactor descriptions so that it may not be construed as a description of a particular reactor system, but as an approach to a particular design problem, applicable to a variety of systems, whether liquid metal convectively cooled or heatpipe cooled cores for Rankine systems, or in- core or out-of-core thermionic systems. However, it is necessary to consider some parameters and features of the system selected to demonstrate that the study was not made on an over-simplified students model.

DESCRIPTION OF THE HEAT SOURCE

The model selected was as detailed and sophisticated as any published in connection with the LRL Space Reactor Technology Program. The corresponding model here actually has its outermost radius within the fueled core radius of the SPR-4 design, yet it has been designed with a greater safety margin.

A cross section is shown in Fig. 1. Both the reactor and the heatpipe boiler sections on the ends were designed in the code. The design includes a dual control system, either of which could carry the mission to completion. The core is not represented by a simple fuel matrix but is designed

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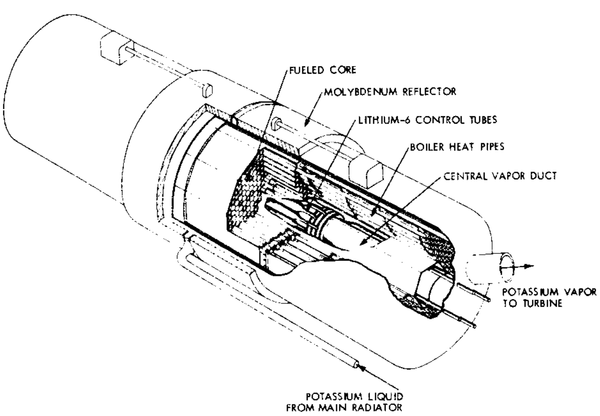

Fig. 1. Reactor Model - Double Ended Heatpipe, Integral Boiler Type

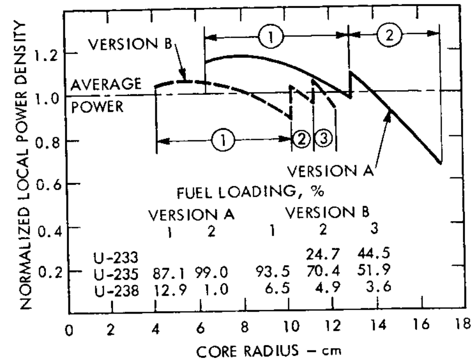

Fig. 2 Power Density Profiles

as a discrete number of fuel elements arranged in a hexagonal pattern in a discrete number of dodecagonal rows. Zonal boundaries coincide with fuel element boundaries and fuel loadings of U-238, U-235, and U-233 are varied to give radial power flattening. Material properties are temperature dependent and in cases where creep is involved, both time and temperature dependent.

Fuel elements were designed with both thermal stress and peak temperature limits. Provision was made for differential thermal expansion and burnup dependent fuel swelling. The heatpipes for each system were tailored to the reactor and optimized for the specific ase. Their design limits included peak or boiloff flux limits, and a safety margin to carry the load on failure of any single adjacent heatpipe as well as a 10% factor for power surges, 20% for transport property data scatter, and 10% for radial power variations.

The units were designed to have a maximum multiplication factor at room temperature of .96 when fully immersed in water and flooded in the absence of blowoff side reflectors. Appropriate neutron filters to accomplish this were placed between core and pressure vessel in this case. The design code could and in the past has, designed these filters into the metallic structure within the core as alloying elements or as plating.

The design code also checks zero power temperature coefficients as a function of temperature, and power coefficients for control as a function of lifetime including fuel burnup effects, production of daughter fissile species, etc.

The pressure vessel and structure were designed for each case using time and temperature dependent creep data.

The remaining figures show additional details of the system. Figure 2, previously published[2] shows the compactness and the degree of power flattening achieved by the U-233, U-235 system (version B) compared with the SPR-4 design (version A). The remaining carpet plots, Figures 3 - 10 show some of the more significant parameters of the system. In particular the readers attention is directed to the nonlinear behavior of many of the variables.

It should be obvious from the detail described above that thermionic elements could have been designed within the core instead of heatpipes, or outside the core in the boiler regions. Similar families of liquid metal cooled reactors have been designed to similar criteria and others including specific pressure drops, coolant temperature rises, etc. These have been even better behaved than the heatpipe reactors shown. They could just as easily have been gas cooled reactors for Brayton cycle applications.

This listing of the detail considered in the design is not presented with a view to convincing the reader of the credibility of this particular set of design calculations or the feasibility of this particular design concept which has been published in the past; it is presented as a demonstration that a rather complex design can be described in considerable detail employing a reasonable number of limits or weighting factors.

SHORT FORM OF THE COMPLETE SYSTEM

In place of the neutronic analysis we predicted the fuel mass using average core fuel burnup data from the twelve designs. This was purposely done in the simplest fashion by fitting second order curves through the twelve data points to produce expressions for burnup and the spacing parameter β as a function of power and aspect ratio. The details of this fitting are shown in Appendix A. Using these functionals to replace the neutronic analysis, the thermo-mechanical code portion produced fifteen additional designs at points intermediate to those of the previous detailed analysis. These fifteen cases were repeated using the full neutronic analysis and the data were compared. They agreed within a few percent in all cases. We therefore conclude that this analysis could be used for this family of reactors within the specified parameter ranges for mission studies. Running times averaged less than 1/10 second per "fitted" design.

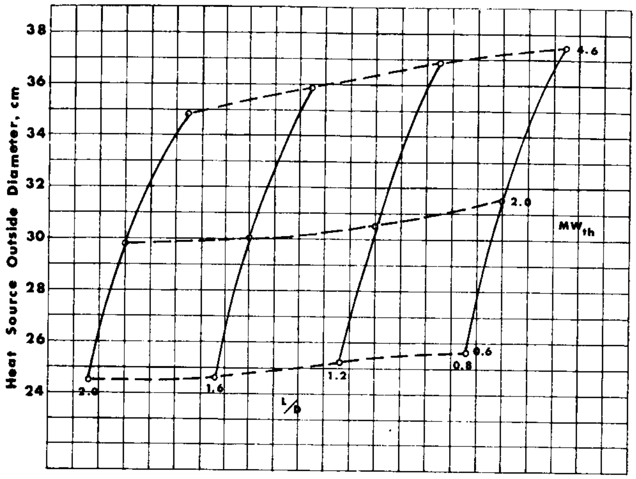

Fig. 3. Heatsource Outside Diameter

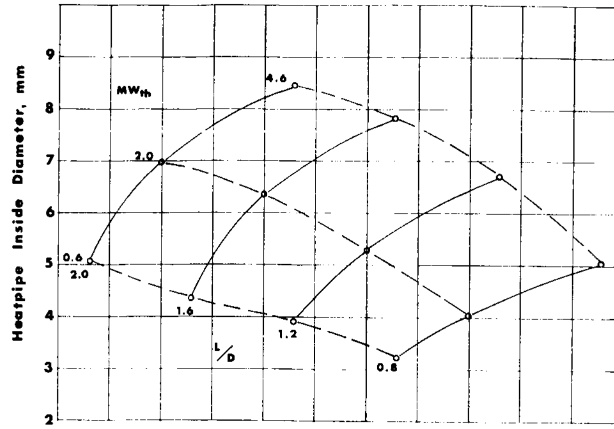

Fig. 6. Heatpipe Inside Diameter

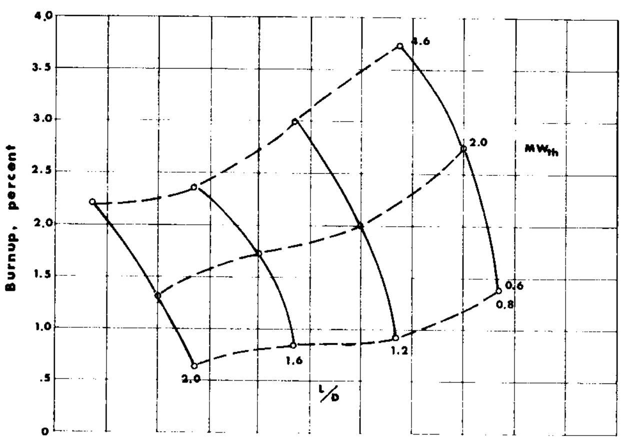

Fig. 4. Average Fuel Burnup

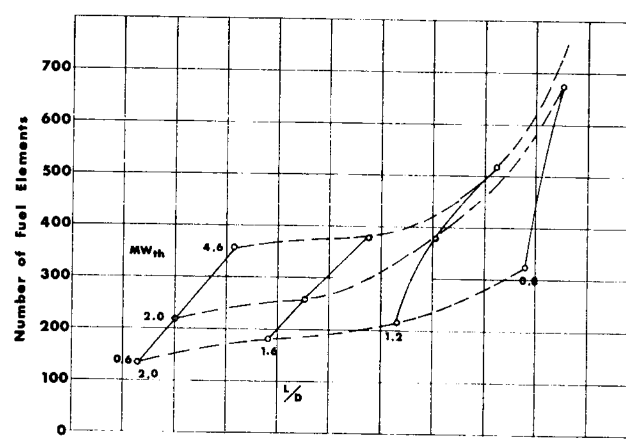

Fig. 7. Number of Fuel Elements

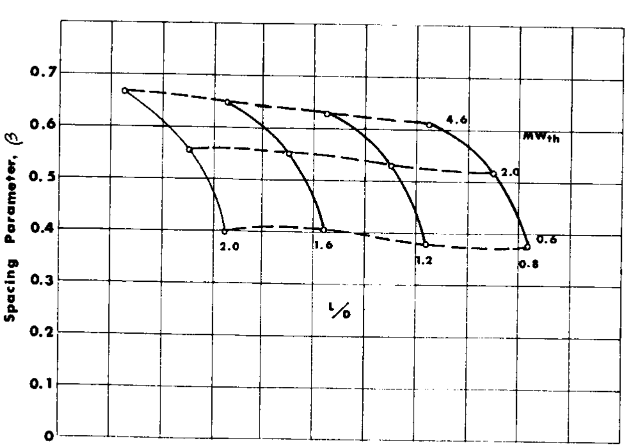

Fig. 5. Spacing Parameter, β.

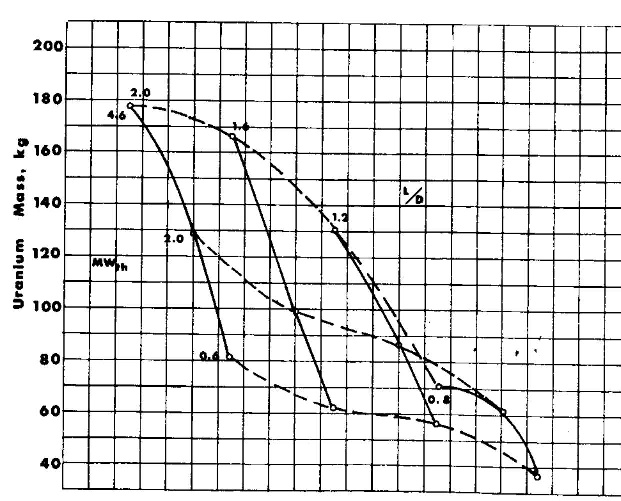

Fig. 8. Total Uranium Mass

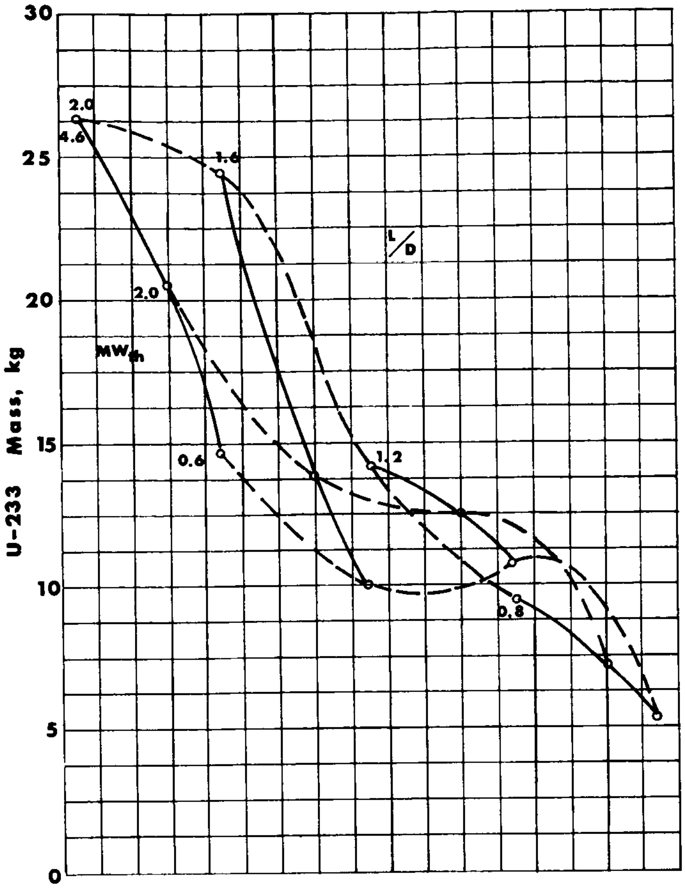

Fig. 9. U-233 Mass

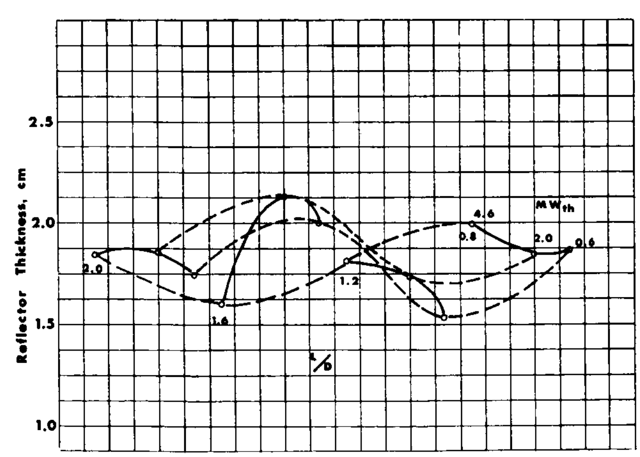

Fig. 10 Reflector Thickness

Table I shows typical results. These data were selected from the larger set as quantities of particular interest to the mission analyst, reactor physicist, and engineer. For the mission analyst fueled core radius, maximum package radius, and reactor-boiler mass were selected. For the physicist the hot-core full-power start of life and cold-core water-immersed and flooded multiplication factors are presented. For the engineer the heatpipe inside radius and a fuel element porosity parameter were selected as variables representative of areas subject to possible future constraints in view of, e.g., fabrication difficulties. The factors shown in percentages are the maximum and rms deviation of the predicted value from that established by the complete analysis. Also shown where applicable is the ratio of the maximum value of the parameter to the minimum encountered in the field, demonstrating that we were not fitting near constants. Fuel mass varied by a factor of 5.4.

Table I

Quantity(F)	Core Radius	React Radius	Mass	Keff Hot	Keff Submerged	Heatpipe Radius	$\beta^{(a)}$
Max Dev %	2.74	2.57	5.86	2.52	2.21	.255	2.39
Rms Dev %	1.27	1.13	3.20	1.62	1.23	.12	1.13
F max/F min	1.79	1.60	5.18			2.89	1.80

(a) β is a fuel element porosity parameter representative of a design detail and is the ratio of the fuel inside radius to the outside hexagonal-flat radius.

CONCLUSIONS

As a demonstration the results indicate the promise of the method. This becomes more evident when the reactors are examined in detail. The simplicity of the fitting method selected is not meant to apply to the designs. The simplest possible representation was used to predict a complex design, to get across the point that it is neither impossible nor impractical to implicitly formulate reactor designs so that they can be meaningfully employed by the mission analyst. It is obvious that a more sophisticated set of formulae could be developed covering such parameters as reflector thickness, zone loading fraction, etc., to give even closer predictions. As an example of this, the parameter β was included above as a starting point to reduce the number of iterations in the thermomechanical analyses.

By using these methods, reactor analysis and reactor system surveys can be broadened in scope so as to pass from company and academic circles to interact meaningfully and credibly with the space systems analysts who must provide input data for the decision making process.

BIBLIOGRAPHY

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2. Walter, C.E.; Brown, N.J; Hampel, V.E.; McCauley, E.W.; and Wilcox, T.P., Jr.; An Advanced 2000 kWth Nuclear Heat Source, Lawrence Radiation Laboratory, Livermore, Rept. UCRL-70978 (1968).

Appendix A

THE FITTED FUNCTIONALS β AND BU

In order to provide a direct path to the explicit, mechanical characterization of an implicitly critical and optimized nuclear reactor heat source of power Q and ratio L/D, it is necessary to describe the spacing parameter, β and the burnup fraction, BU as functionals. The data used are abstracted from the detailed analyses.

As is clear from Figure 5, β is a function of L/D and Q, or

$$\beta = \beta(L/D,Q) = D(Q) + E(Q)(L/D) + F(Q)(L/D)^2 \qquad 1.1$$

with

$$D(Q) = \sum_{i=0}^{2} d_i Q^i$$

$$E(Q) = \sum_{i=0}^{2} e_i Q^i \qquad 1.2$$

$$F(Q) = \sum_{i=0}^{2} f_i Q^i$$

where, for constant Q,

$$\beta(L/D)_Q = \sum_{j=0}^{2} g_i (L/D)^j \qquad 1.3$$

Performing the indicated least squares fit in equation 1.3, the g_i are used to fit the d_i, e_i, and f_i, respectively, of equations 1.2 thus producing the coefficients of equation 1.1. They are listed, for reference, in Table A.1.

The burnup (Figure 4) is treated in a similar manner with

$$BU = BU(L/D,Q) = A(Q) + B(Q)(L/D) + C(Q)(L/D)^2 \qquad 1.4$$

where

$$A(Q) = \sum_{k=0}^{2} a_k Q^k$$

$$B(Q) = \sum_{k=0}^{2} b_k Q^k$$

$$C(Q) = \sum_{k=0}^{2} c_k Q^k$$

and

$$BU(L/D)_Q = \sum_{\ell=0}^{2} \alpha_\ell (L/D)^\ell$$

The resulting coefficients of equation 1.4 are listed in Table A.2.

Table A.1

The Functional Coefficients of β

	i=0	i=1	i=2
D (d)	2.59217E-1	1.23627E-1	-1.19133E-2
E (e)	4.27773E-2	3.74926E-2	-7.96442E-3
F (f)	-8.41595E-2	-1.06819E-2	2.64808E-3

Table A.2

The Functional Coefficients of BU*

	k=0	k=1	k=2
A (a)	1.20680E-2	2.39719E-2	-3.53083E-3
B (b)	-1.07593E-2	-1.36532E-2	2.53387E-3
C (c)	3.09971E-3	2.41979E-3	-5.07874E-4

*Burn Up