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Preliminary Neutronics Design Studies of a Lead Cooled, Small Modular Reactor

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This paper explores the feasibility of a small, lead-cooled modular reactor. The main design criteria that must be satisfied are long fuel lifetime, natural convection heat transport, semi-autonomous control, and small unit size. These design goals inherently require a derated power level, low burnup swing, and open pin bundle design. Parametric neutronics studies were performed to determine the minimum core size required to satisfy these design criteria for several different fuel pin configurations.

Targeting a 20 year lifetime and 25 MWt power level, the natural circulation design objective was easily met. The semi-autonomous control design objective is also possible, but modifications to the core geometry are likely necessary to facilitate better reactivity feedback from geometrical expansion. The transportability design objective is the only point of contention due to the large size and weight of the core. More precise limits on the thermal hydraulic assessment of the reactor design can help reduce the core size and fuel loading.

KEYWORDS: lead cooled fast reactor, 20 year fuel lifetime, 25 MWt power, secure, transportable, autonomous reactor system, neutronics study

1. Introduction

A neutronics study was initiated for a small, lead-cooled modular reactor (SMR). This reactor is designed to meet the requirements of the Secure, Transportable, Autonomous Reactor (STAR) system [1]. The four major design objectives of the STAR system are: 1) long fuel lifetime to eliminate on-site refueling and fuel access, 2) natural convection heat transport, 3) semi-autonomous control, and 4) small unit size enabling transportability. A preference for a transuranic (TRU) based fuel derived from recycled light water reactor fuel is also specified. These design goals inherently require a derated power level, low burnup swing, and open pin bundle design. The semi-autonomous control goal excludes the use of power regulating rods, and inherently requires that the excess reactivity be small for the duration of the core lifetime.

Parametric neutronics studies were performed to find the optimum reactor design that satisfied the project design goals. The initial design work focused on a power level of 25 MWt (10 MWe) and a core lifetime of 20 years (20 years of continuous operation). Parametric studies showed that the goal of using a transuranic (TRU) feed could not be met at this power level. A weapons grade Pu feed and an enriched uranium feed were considered as alternatives. The enriched uranium feed was more favorable and was chosen for more detailed study. Detailed fuel cycle characteristics and reactivity coefficients were generated for a 25 MWt reactor with an enriched uranium feed.

2. Computational Methods and Models

A nitride fuel form was chosen because of its high melting temperature and high thermal conductivity. The nitrogen is assumed to be enriched to 100% N-15. Although there are obvious increases in fuel costs caused by this enrichment requirement, it is necessary to eliminate parasitic reactions in N-14 and the waste disposal problems associated with C-14 production. The pin geometry chosen for this study (1.905 cm outer pin diameter) is based upon previous work. A more detailed thermal-hydraulics analysis is required to ascertain the optimum pin geometry and the one chosen for this work is only meant to be representative.

Multi-group cross section library derived from ENDF/B-V data were utilized in this study. For the simplified RZ parametric studies, a 21-group region-dependent cross section library was utilized. The actinide and structural isotope cross sections of this library had previously been generated for a compact, U/30TRU/10Zr-fueled, sodium-cooled core design [2] using the MC^2-2 code [3]. For the final designs done in hexagonal-z geometry, cross sections specific to the compositions in each reactor design were generated and used for the fuel cycle and reactivity coefficient calculations. A 33-group cross section structure was chosen and the MC^2-2 code was used to obtain the cross sections for each composition.

The reactor physics and fuel cycle analyses were performed using the DIF3D/REBUS-3 code package [4-6]. The initial parametric studies on core sizing were carried out using an idealized RZ model with later design refinements performed in hexagonal geometry (used to model discrete control rod structures). The nodal diffusion option [4,5] of the DIF3D code was used for flux calculations in hexagonal geometries, while the finite difference diffusion option was used in RZ geometries. Performance parameters were calculated for a single-batch, once-through fuel cycle, beginning with a clean core (unirradiated). An enrichment search procedure was utilized to determine the beginning of cycle (BOC) enrichment needed to maintain criticality throughout the operating cycle. Block nuclide depletion was performed by splitting the core into depletion zones. In order to flatten the power distribution, an enrichment zoning strategy was used by employing three cylindrical zones of fuel enrichment.

Reactivity coefficients and kinetics parameters were calculated for the beginning of cycle (BOC) and end of cycle (EOC) configurations. The coolant, fuel, and structure density coefficients and the coolant void coefficient were determined using the VARI3D perturbation calculation code [7]; the linear perturbation theory option was used for density coefficients, while the exact perturbation theory option was employed for the coolant void coefficient. The effective delayed neutron fraction and prompt neutron lifetime were also calculated using the VARI3D code. The radial and axial expansion coefficients and the control rod worth were determined by direct eigenvalue differences of the base and perturbed configurations.

The limitation placed upon the reactor design by the natural convection cooling was estimated using a simple thermal-hydraulic model. This model is based upon the balance between frictional pressure drop and buoyancy force.

3. Parametric Studies

Parametric studies were performed using a simplified RZ reactor model. The reactor is assumed to be exactly critical at BOC and must be cooled using natural circulation. These limitations constrain the pin pitch to diameter (P/D) ratio and core diameter, which are the two main variables considered in the parametric study. Using the REBUS-3 code, the end of cycle criticality state k_{eff}^{EOC} was determined and the reactivity swing was calculated using

$$\rho = \frac{k_{eff}^{EOC} - k_{eff}^{BOC}}{k_{eff}^{EOC} \cdot k_{eff}^{BOC}} = \frac{k_{eff}^{EOC} - 1}{k_{eff}^{EOC}}. \tag{1}$$

Those designs that have a negative reactivity swing are subcritical at the end of life (20 years) while those with positive reactivity swings are supercritical at the end of life. To meet the design goal of semi-autonomous control, the reactivity swing needs to be close to zero. Therefore, for any P/D ratio or fuel volume fraction (FVF), the core diameter is determined to be the near zero burnup swing point.

Figure 1 plots the burnup swing results for several reactors with different fuel volume fractions and core diameters (the core height to diameter ratio was fixed at 0.8 for minimal leakage). The limitation placed on the power density by natural circulation (NC Limit) is also included in Fig. 1 for each FVF line as a vertical dashed line with the symbol of the corresponding FVF line. The acceptable designs are therefore the intersection points of 0.0% burnup to the right of the vertical natural circulation lines. From Fig. 1, it is easy to see that for a FVF significantly less than 0.56, or a P/D ratio greater than 1.0, the natural circulation limit is not generally important. As a consequence, the FVF and core size are constrained more by engineering limitations on fuel pin spacer design than by natural circulation.

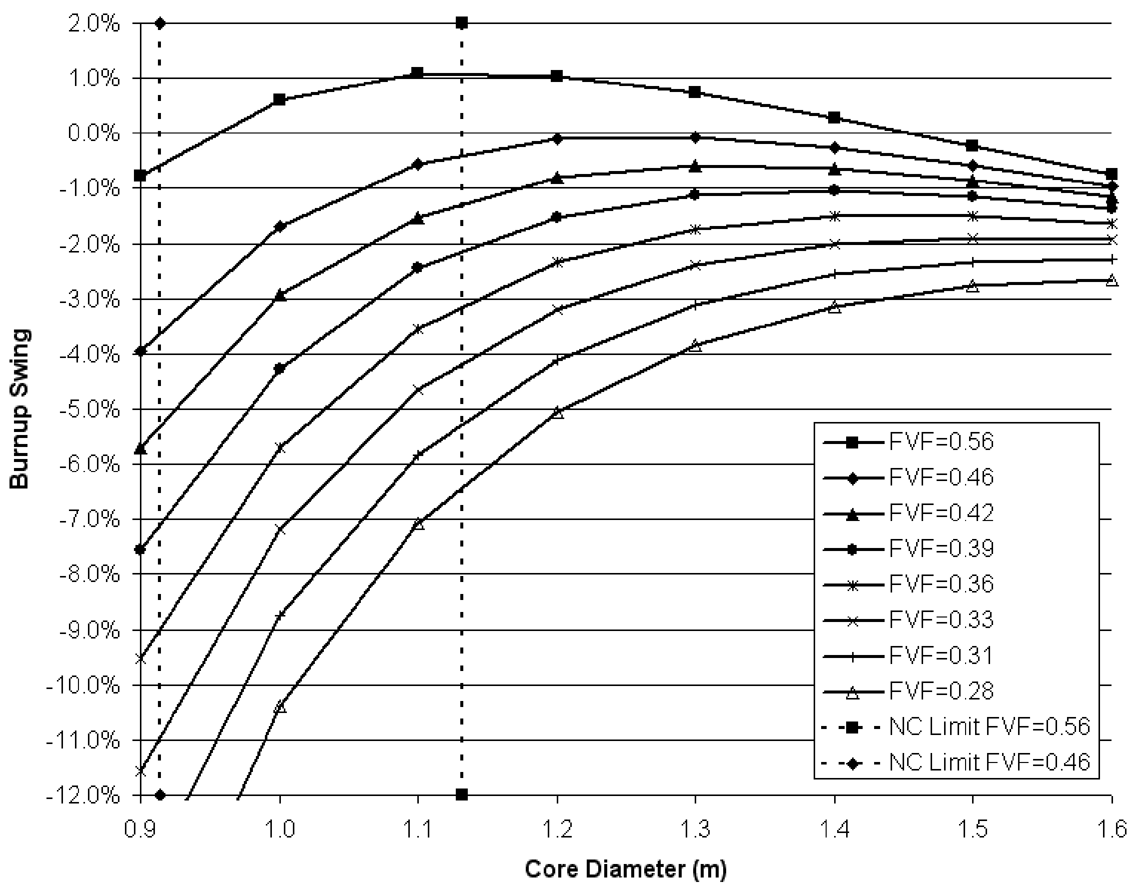

Fig. 1. Burnup Swing Dependence of a 25 MWt SMR Using TRU Enrichment Feed

The variation of the enrichment split was found to affect the burnup swing by at most ±0.5%. Modifications to the core geometry (core height to diameter ratio) did not improve the performance. The feasible designs in Fig. 1 appear to be between a FVF of 0.56 (P/D ratio of 1.0) with a core diameter of ~1.45 m and a FVF of 0.46 (P/D ratio of 1.1) with a core diameter of ~1.25 m. The engineering limitation on the P/D ratio was estimated to be larger than 1.15, consequently, a 25 MWt design utilizing the TRU enrichment feed does not appear to be feasible.

The reason for the failure of the reactor design utilizing a TRU feed has been traced to insufficient breeding. A significant amount of the fissile material in the TRU fuel is ^{241}Pu, which decays to fertile ^{241}Am with a 14.4 year half-life. At 25 MWt, the breeding of ^{239}Pu fuel cannot keep up with the loss of ^{241}Pu over the 20 year lifetime. Therefore, a 25 MWt reactor with a P/D ratio of ~1.2 can only operate for about 5 years before becoming subcritical. It was found that the power level needs to be increased to ~30-35 MWt to meet the 20 year lifetime.

Given the problems with the TRU feed, additional calculations were performed using an enriched uranium feed and a weapons grade Pu feed. Both fissile feeds have significant proliferation problems with the weapons grade Pu being worse (problem at both BOC and EOC). In terms of their impact on the reactor design and reactivity coefficients, these fissile feeds differ significantly. For small P/D ratios (1.0 and 1.1), the weapons Pu feed allows for a smaller core diameter than that of the enriched uranium feed and the TRU feed. The enriched uranium feed results in a core diameter slightly smaller than that of the TRU feed. Because of the impact on core size and proliferation problems, it was decided that the use of an enriched U fuel in place of TRU would be preferable.

4. 25 MWt Designs

Based on the preceding parametric study results, more detailed design studies were performed using a hexagonal geometry. In the hexagonal model of Fig. 2, the center assembly was replaced with a control rod channel to remove the central flux peak. Also, additional control rods (labeled "S") were added in the third ring where the shading indicates two separate control drive systems. These "S" control rods are defined to make up 50% of the volume of the hexagons. When combined with the center control rod, the control rod layout meets the design goal of three independent control systems each of which has enough reactivity to shutdown the reactor. If necessary, for power regulating reactivity control, any one of the control systems can be used as a conventional control rod assembly.

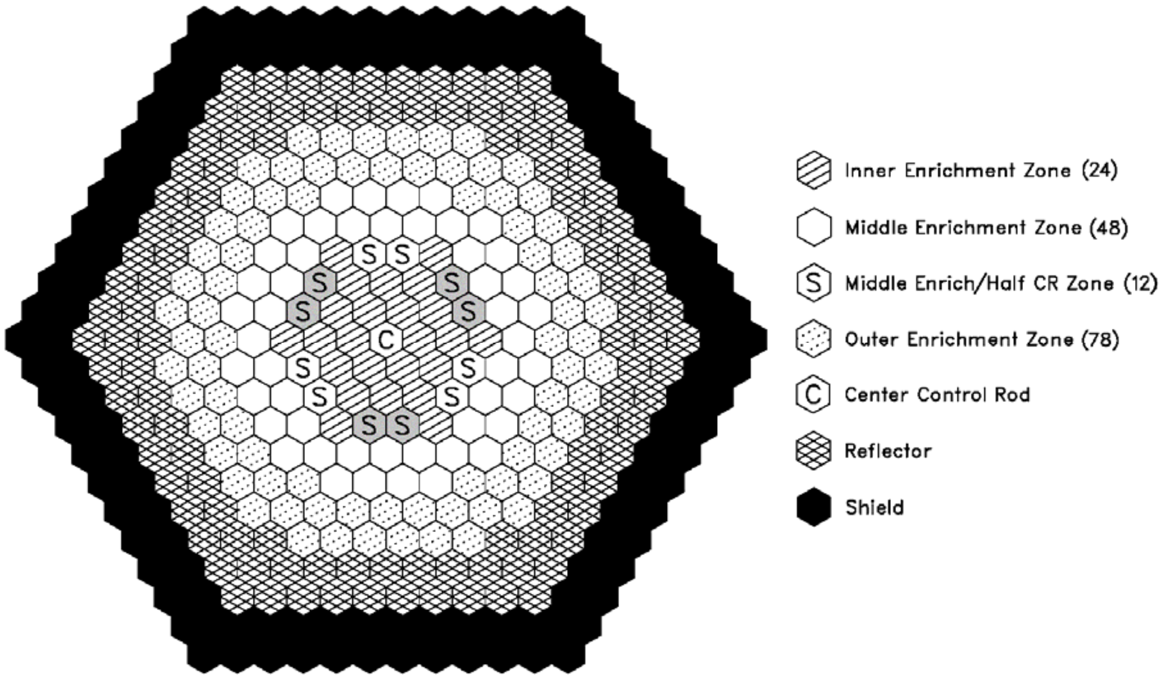

Fig. 2. Hexagonal Geometry Core Model

Similar to the RZ model, a parametric study was carried out to determine the necessary size of the core with the control rod positions included. As was found with the RZ model, the natural convection constraint is not important for the range of P/D ratios considered acceptable for this work. The P/D ratios of 1.15, 1.20, and 1.25 (FVF of 0.42, 0.39, 0.36) were selected, the optimum enrichment splitting was determined, and new cross sections were generated. The reactor operating characteristics and reactivity coefficients were calculated for all three designs (P/D ratios). Table 1 gives the reactor design parameters for the three P/D ratio point designs, each of which have the assembly loading shown in Fig. 2 (the hexagonal pitch is varied to increase the diameter of the core).

Table 1. 25 MWt Core Design Data

Pin Pitch-to-Diameter Ratio	1.15	1.20	1.25
Fuel Volume Fraction	0.42	0.39	0.36
Cladding Volume Fraction	0.14	0.13	0.12
Bond Volume Fraction	0.12	0.11	0.10
Coolant Volume Fraction	0.31	0.37	0.42
Core Height (m)	1.08	1.20	1.28
Core Diameter (m)	1.38	1.53	1.63
Core Volume (m^3)	1.60	2.20	2.67
Inner Zone Enrichment (%)	10.4	10.2	10.3
Middle Zone Enrichment (%)	12.5	12.3	12.4
Outer Zone Enrichment (%)	15.6	15.3	15.4

As the P/D ratio increases, the fuel volume fraction decreases. The reduced fuel volume fraction increases the neutron migration length and thus the neutron leakage. To maintain criticality in this situation, the core diameter must be increased to compensate the increase in leakage. For a given fuel composition, the core size required to maintain criticality increases more rapidly than the decrease of fuel volume fraction. Consequently, the initial heavy metal loading increases as the P/D ratio increases. It may appear as though the increases in the core diameter are small, but they translate to significant changes in the core volume since the core height to diameter ratio was fixed at 0.8. The combined increase in core volume, decrease in power density, and increase in fuel loading allow the fuel enrichments to be nearly identical for all three cores.

Table 2 gives the performance characteristics of the three reactor designs over the 20 year lifetime. Neither the peak linear powers nor the peak power densities seen in Table 2 are of any concern from a safety aspect. In addition, the burnup swing is near to zero as desired. The discharge burnup and peak fast fluence are typical of a low power reactor such as this and are clearly within engineering limits. In fact, they are less than 25% of the limiting values. This indicates that the discharge burnup can be increased by a factor of ~4 within current design limits by increasing the power density or the fuel residence time. The discharged fuel can be of concern since there is a significant amount of Pu discharged at the end of the 20 year lifetime. This discharged Pu is 98% ^{239}Pu which raises a significant proliferation issue. However, since the project goal preferred the use of a TRU enrichment feed, this problem should be easy to avoid given an increase in the reactor power.

Table 2. 25 MWt Core Performance Data

Pin Pitch-to-Diameter Ratio	1.15	1.20	1.25
Initial Uranium Loading (kg)	8479	10681	11947
Discharged Plutonium (kg)	133	140	141
Discharged Minor Actinides (kg)	1	1	1
Discharged Uranium (kg)	8157	10354	11618
Total Power Density (kW/L)	15.0	11.4	9.4
Peak Power Density (kW/L)	23.2	17.6	14.6
Peak Linear Power (W/cm)	50.0	39.9	35.8
Average Discharge Burnup (MWd/kg)	21.1	16.7	15.0
Peak Discharge Burnup (MWd/kg)	31.9	25.2	22.6
Peak Fast Fluence (10^{23} n/cm^2)	0.94	0.73	0.64
Burnup Swing (%)	-0.12	0.08	0.05

The reactivity coefficients of the three reactor designs at beginning of cycle (BOC) are given in Table 3 (the values at end of cycle are similar). The delayed neutron fraction is typical for a fast reactor fueled with uranium. The prompt neutron lifetime is also typical for a fast system. The three major material thermal expansion coefficients are all generally small for fast systems, especially the coolant density coefficient. The sign change in the coolant density coefficient can be explained by the decreasing importance of leakage as the core size increases.

Table 3. 25 MWt BOC Reactivity Coefficients

Pin Pitch-to-Diameter Ratio	1.15	1.20	1.25
Delayed Neutron Fraction	0.0072	0.0071	0.0071
Prompt Neutron Lifetime	3.7E-07	4.1E-07	4.5E-07
Coolant Density (cents/C)	-0.004	0.001	0.004
Fuel Density (cents/C)	-0.029	-0.028	-0.028
Structure Density (cents/C)	-0.003	-0.003	-0.003
Radial Expansion (cents/C)	-0.06	-0.06	-0.06
Axial Expansion (cents/C)	-0.02	-0.02	-0.02
Center Control Rod ($/cm)	-0.010	-0.009	-0.009
Doppler (cents/C)	-0.12	-0.13	-0.13
Voided Doppler (cents/C)	-0.13	-0.13	-0.13
Coolant Void Worth ($)	-1.21	-0.85	-0.75

The geometrical thermal expansion coefficients are also rather small for fast systems. In particular, the radial expansion coefficient is quite small and may place in jeopardy the ability of the reactor system to operate under autonomous control. Modifications to the reactor design may be appropriate so that more radial or axial leakage is introduced. One of the initial design goals of this reactor is to avoid the use of a control rod for reactivity control, and hence a control rod worth is not necessary (the control rods should be removed from the core). However, assuming that a control rod is necessary, the center control rod was assumed to be a power regulating rod and the rod worth was computed. As can be seen, this control rod worth is quite small and well within safe operating limits. If the other control rods were chosen as regulating power rods, then they would most likely have a slightly higher worth (approximately -0.06 $/cm).

The Doppler coefficients are sufficient, and typical for fast reactors with this fuel type. At this point, neither the flooded Doppler coefficient nor the voided Doppler coefficient is of any concern since they are acceptable values from a reactor safety point of view. The void worth is particularly not important since the lead boiling temperature is well above the melting temperature of the cladding. Typically, voiding is important because it can lead to cladding failure, but in this case, it would occur after the cladding has already failed.

5. Conclusions and Future Work

A TRU based enrichment feed could not be used for the targeted 25 MWt power and 20 year lifetime, but an enriched uranium and weapons Pu feed could. The enriched uranium feed had slightly better qualities and thus was selected for more detailed studies. To meet the natural circulation and semi-autonomous control design objectives, the reactor had to have a derated power level and low burnup swing. Using an estimate of the natural circulation thermal hydraulic limitation the feasible core diameter was found to be between 1.4m and 1.6m with a core height between 1.0m and 1.3m. Detailed reactivity coefficients were generated for use in a future safety analysis. Preliminary analysis indicates that some modifications to the core geometry would be advisable to allow for larger reactivity feedback from geometrical thermal expansion.

The ability of any of the point designs to meet the transportability design goal is the only problem area. Future work is focused on a more thorough study of the transportation limitations.

Relaxation of the semi-autonomous control goal may be necessary to meet the transportation goal. This would allow for compaction of the core design, but necessitate the use of a regulating power control rod(s). Additional focus is also being placed on higher power applications that allow the use of a TRU based enrichment feed.

Acknowledgements

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