PHYSOR 2004 -The Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments
Chicago, Illinois, April 25-29, 2004, on CD-ROM, American Nuclear Society, Lagrange Park, IL. (2004)

Preliminary Neutronics Design Studies for a 400 MWt STAR-LM

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Neutronics design studies for a 400 MWt high temperature fast reactor are being performed, utilizing lead coolant, transuranic (TRU) nitride fuel, and HT-9 structural material. Under the main design constraints of long fuel lifetime, natural convection heat transport, semi-autonomous control, and small unit size, parametric studies were performed to maximize the discharge burnup and minimize the burnup reactivity swing. Based on the results of these parametric studies, two point designs were developed for a single-batch once-through fuel cycle; one is a 15 full power year cycle design with core volume of 9.5 cubic meters, and the other is a 12 full power year cycle design with core volume of 7.4 cubic meters. For these two point designs, fuel cycle analyses and reactivity feedback coefficients calculations were performed. The 9.5 cubic meter design achieved an average discharge burnup of 83 MWd/kg with a maximum reactivity change over the lifetime of 0.6%. The peak fast fluence was well within the fast fluence limit of HT9, and both average and peak power densities were well below the estimated limit for natural circulation. The performances of the 7.4 cubic meter design were slightly inferior to this design. To enhance the passive safety characteristics, however, further design improvements need to be made to reduce the coolant density coefficient and to increase the radial expansion coefficient.

KEYWORDS: lead cooled fast reactor, long life core, semi-autonomous control, natural circulation, STAR-LM

1. Introduction

The feasibility of coupling of a high temperature lead-cooled fast reactor to the supercritical CO_2 Brayton cycle is being investigated under the U.S. Department of Energy Nuclear Energy Research Initiative (USDOE NERI) Program. As a part of this research, neutronics design studies for a 400 MWt high temperature lead-cooled fast reactor are being performed. This reactor system utilizes lead coolant, transuranic (TRU) nitride fuel, and HT-9 structural material. The overall design requirements are consistent with those of the Secure, Transportable, Autonomous Reactor (STAR) system. [1] Nitride fuel was considered because of its advantageous thermal properties: high melting temperature and high thermal conductivity. The nitrogen is assumed to be enriched to 100% N-15; although there are associated increases in fuel costs from the enrichment process, this eliminates parasitic reactions in N-14, and the waste disposal problems associated with C-14 production. The system relies on cooling by natural circulation, which eliminates primary system pumping requirements, but requires an open lattice design.

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In order to satisfy the overall requirements of STAR-Liquid Metal (LM), neutronics design studies focused primarily on achieving four design objectives: 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. The first and second targets require a significant core power density reduction (i.e., derated configuration) relative to conventional sodium-cooled fast systems. The semi-autonomous control requires the excess reactivity embedded in the core to be minimal within other constraints. It also requires appropriate negative temperature feedback coefficients. For long-lived systems, the size of the core is constrained mainly by reactivity loss considerations. The third goal requires the burnup reactivity swing to be minimized. In addition, to enhance the fuel cycle economics and to minimize the TRU loss to the waste stream, it is desirable to maximize the fuel discharge burnup within the constraints on the peak burnup and fast fluence limits. With these general design goals, parametric neutronics studies were performed to find the optimum reactor design.

2. Analysis Methods and Models

Reactor physics and fuel cycle analyses were performed with the DIF3D/REBUS-3 code package. [2,3] The STAR-LM cartridge core consisting of an array of fuel pins on a regular triangular pitch (rather than a typical layout of ducted assemblies) was modeled by an array of hexagonal nodes, and the nodal diffusion option in hexagonal-z geometry of the DIF3D code was used for flux calculations. [4] While the DIF3D geometry modeling would allow a pin-by-pin neutronics analysis, homogenized hexagonal nodes with a pitch of 16.14 cm (6.355 in) were used to avoid prohibitive computational expenses. Even though this nodalization requires a non-integer number of fuel pins and surrounding lead coolant be homogenized into each hexagonal node, it is sufficient for scoping neutronics analyses because of long mean free paths of neutrons in lead-cooled fast systems. In future, the hexagonal node pitch will be adjusted such that an integer number of pins are included in each node.

Fuel cycle analyses and depletion calculations were performed using the REBUS-3 code. Performance parameters were calculated for a single-batch, once-through fuel cycle, beginning with a core completely loaded with fresh pins. An enrichment search procedure was utilized to determine the beginning of cycle (BOC) Pu or TRU loading necessary to maintain criticality throughout the operating cycle. Block nuclide depletion was performed by dividing the core into depletion zones. In order to flatten the power distribution, an enrichment zoning strategy was used by employing two to three radial zones differing in the Pu or TRU mass fraction of the fuel (i.e., in "enrichment").

A multi-group cross section data library derived from ENDF/B-V was utilized in this study. Cross sections specific to individual compositions were generated using the MC^2-2 code. [5] A 33-group structure was chosen and used for the fuel cycle and reactivity coefficient calculations. For the long-lived cores investigated in this study, the multiplication factor initially increases, attains a maximum, and then decreases. Thus, reactivity coefficients and kinetics parameters were calculated for BOC, middle of cycle (MOC), and end of cycle (EOC) configurations. The MOC configuration corresponds to the time point where the multiplication factor attains its maximum value. The coolant, fuel, and structure density coefficients and the coolant void coefficient were determined using the VARI3D perturbation code. [6] 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 conditions.

3. Parametric Studies

For parametric studies, a reference TRU core design was first developed starting from a LBE cooled STAR-LM design, which utilizes weapons plutonium fuel. [7] Replacing the LBE coolant with lead and the weapons-grade Pu feed with the TRU feed separated from light water reactor (LWR) spent fuel, a series of trade studies were performed to develop a favorable configuration for the derated natural circulation design. Figure 1 shows a layout of the reference core configuration. The active core volume (including the control rod locations) is about 9.5 cubic meters. The core loaded with nitride fuel in HT-9 cladding is surrounded by a reflector/core barrel zone and the down-flowing liquid metal coolant in the down-comer. HT-9/Pb reflector and core barrel (50% HT-9/50% Pb) was used since lead is a superior reflector material but ineffective neutron shielding material.

Because of the low power density, the reference TRU core design resulted in a very low discharge burnup (~57 MWd/kg) and a peak fast fluence much lower than the limit for HT-9 structural material. This implies that the discharge burnup can be increased within the peak fast fluence limit by increasing the fuel residence time. Relatively large power peaking (~1.7) and high reactivity swing (~3.0%) were also observed. To improve these parameters, the fuel volume fraction and enrichment zoning need to be optimized. In addition, from the transportability and economics viewpoint, it is also desirable to reduce the core size within the natural circulation limit.

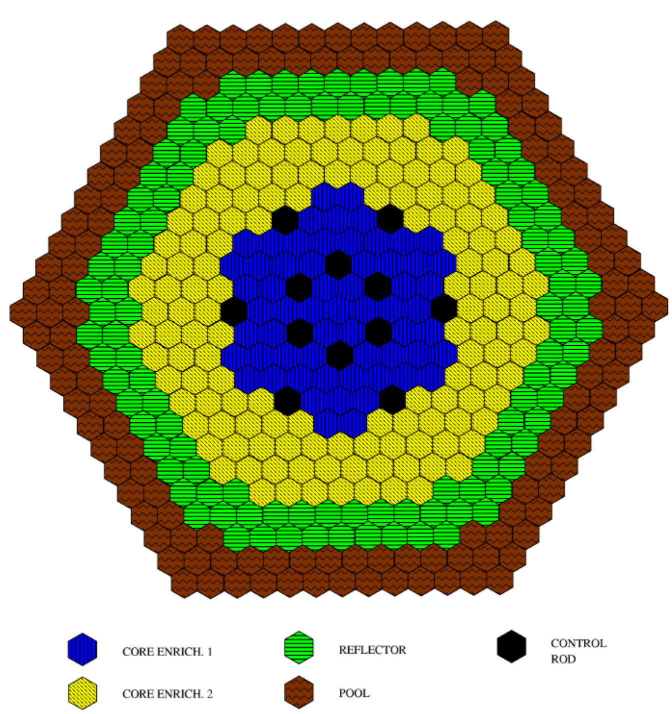
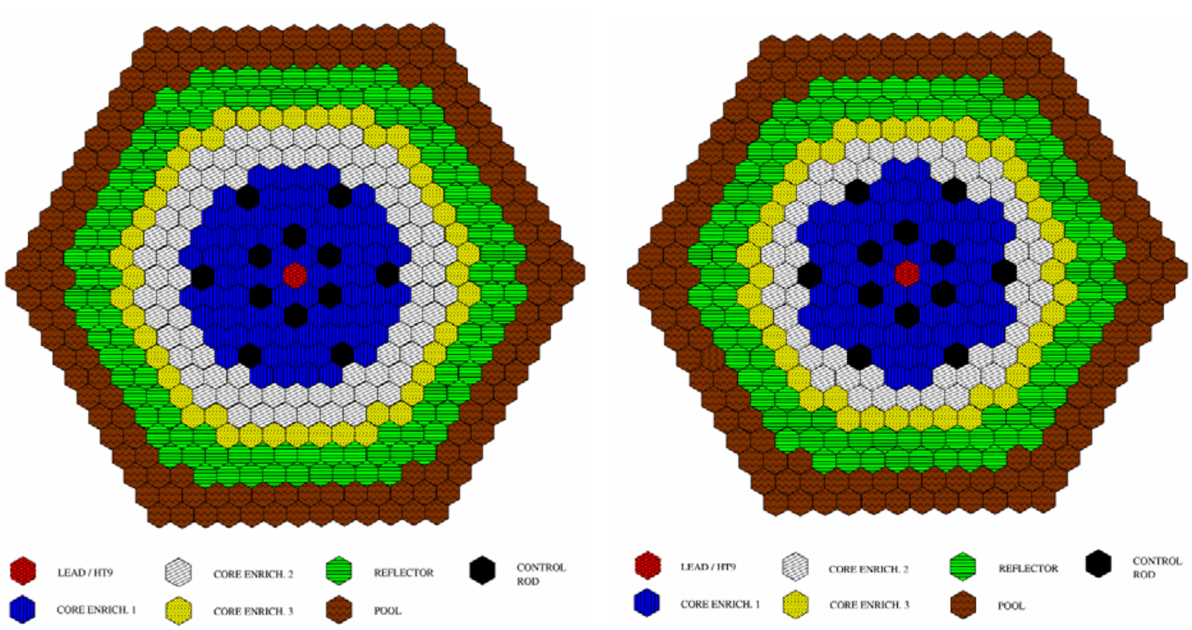


Figure 1. Core Layout of Reference Lead-Cooled STAR-LM Reactor

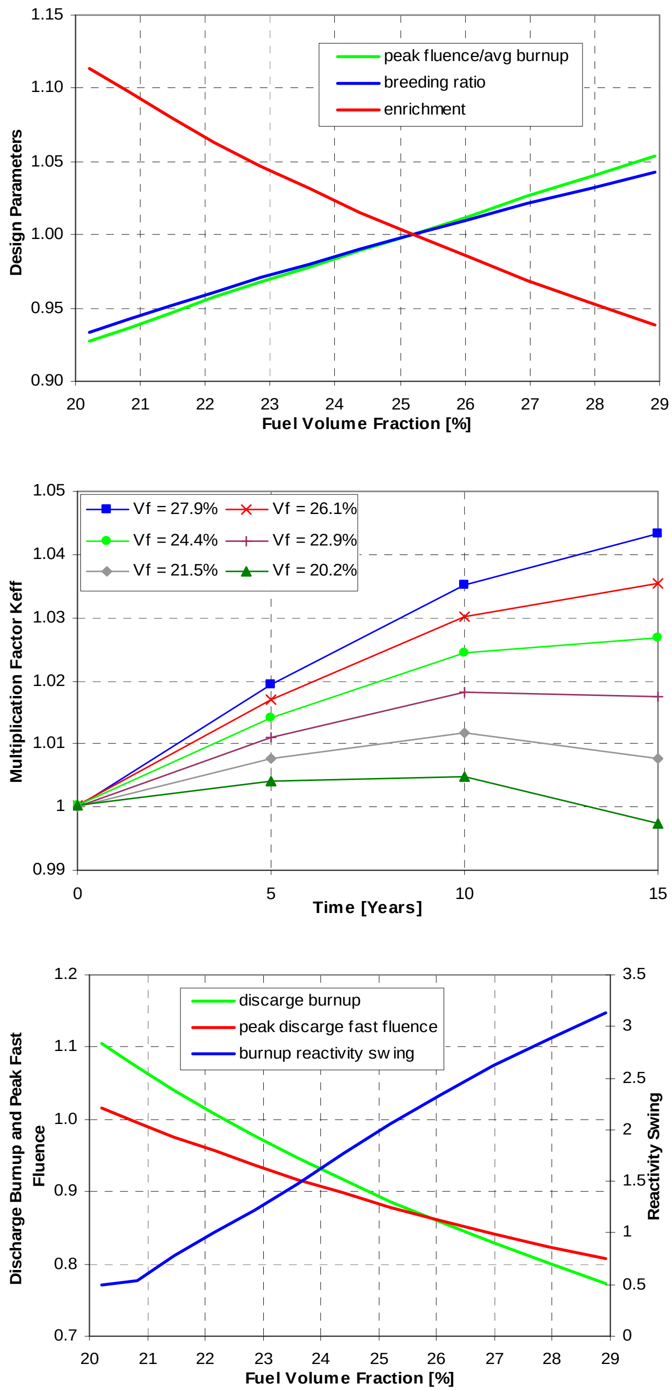
In order to enhance these core performance parameters, various design modifications were made on the reference TRU core design. The cycle length was first increased from 12 to 15 full power years to increase the discharge burnup within the peak fast fluence limit. Then, to reduce the power peaking factor, three-region enrichment zoning was employed, and HT-9/Pb reflector was introduced in the core center. The fuel volume fraction was optimized to reduce the burnup reactivity swing and increase the discharge burnup. To improve the transportability and economics, reduced core size designs were also investigated. In this attempt, a new configuration of 7.4 cubic meter size was developed. Figure 2 shows the configurations of 9.5 and 7.4 cubic meter core designs.

The performance parameters of the 9.5 cubic meters, 15 full power year TRU core design are shown in Fig. 3 as a function of fuel volume fraction. Figure 3a shows the resulting behaviors of the enrichment, breeding ratio, and peak fast fluence to discharge burnup ratio. For a fixed core size, as the fuel volume fraction increases, the neutron leakage decreases and thus the fissile enrichment required for criticality decreases. The reduced enrichment increases the fertile loading and thus enhances the conversion ratio. However, since the conversion ratio is greater than 1.0 in the range of volume fraction of interest for long-lived core, the increased breeding ratio further increases the reactivity swing. This can be seen more clearly from Fig. 3b which provides the variation of multiplication factor during the cycle for different fuel volume fractions. The multiplication factor initially increases, attains a maximum, and then decreases for a fuel volume fraction below ~23% while it increases monotonically for a higher fuel volume fraction. The reactivity swing, which is defined here as the difference between the maximum and minimum k_{eff} values, increases as the fuel volume fraction increases.



(a) 9.5 Cubic Meter Core (b) 7.4 Cubic Meter Core

Figure 2. Layout of 9.5 and 7.4 Cubic Meter TRU Core Designs



(a) Peak fast fluence to average discharge burnup ratio, breeding ratio, and enrichment (relative to 25.2% fuel volume fraction)

(b) Time evolution of multiplication factor

(c) Average discharge burnup, peak discharge burnup, and burnup reactivity swing (normalized to reference values: discharge burnup = 80 MWd/kg, peak discharge fast fluence = 4.0×10^{23} n/cm^2, burnup reactivity swing = 1.5%).

Figure 3. Performance Parameters of 9.5 Cubic Meters, 15 Full Power Year Core Design versus Fuel Volume Fraction

Figure 3a also shows that the peak fast fluence to average discharge burnup ratio increases as the fuel volume fraction increases. As can be seen from Fig. 3a, the enrichment decrease rate with fuel volume fraction is slower than linear, and hence the product of enrichment and fuel volume fraction (i.e., TRU inventory at BOC) increases with the fuel volume fraction. As a result, the discharge burnup decreases with the fuel volume fraction. On the other hand, the increased fissile loading reduces the flux level to attain the same power density, and hence the fast fluence decreases with the fuel volume fraction. However, the fast fluence decreases more slowly than the discharge burnup since the neutron spectrum becomes harder as the fuel volume fraction increases. Consequently, the peak fast fluence to average discharge burnup ratio increases with the fuel volume fraction.

Figure 3c shows the average discharge burnup, peak discharge fast fluence, and burnup reactivity swing as a function of fuel volume fraction. As discussed above, as the fuel volume fraction increases, the discharge burnup and fast fluence decreases, and the reactivity swing increases. Therefore, the maximum discharge burnup and the minimum reactivity swing can be attained at the same time by minimizing the fuel volume fraction under the peak fast fluence limit. From Fig. 3c, it can be found that about 21% is the optimum fuel volume fraction within the HT-9 fast fluence limit of 4×10^{23} n/cm^2. With this fuel volume fraction, a discharge burnup of ~85 MWd/kg and a reactivity swing of ~0.8% can be attained.

For a cycle length of 15 full power years, the discharge burnup and fast fluence of the 7.4 cubic meter design showed similar trends to those of the 9.5 cubic meter case. On the other hand, the reactivity swing showed a different behavior. It decreases initially, attains a minimum, and increases with the fuel volume fraction. This is because the conversion ratio is less than 1.0 for a low fuel volume fraction and hence the enhanced internal conversion with increased fuel volume fraction reduces the reactivity swing. With reduced core size, the main constraint became the peak fast fluence that increases significantly because of the increased power density. To reduce the peak fast fluence, the cycle length was reduced from 15 to 12 full power years. For a 12 full power year cycle length, the peak fast fluence limit is reached at a fuel volume fraction of about 22%. With this fuel volume fraction, a discharge burnup of ~85 MWd/kg and a reactivity swing of ~0.8% can be attained, which are similar to those of the 9.5 cubic meter core case.

4. Point Core Designs

Based on these parametric studies, two point core designs were developed with a single-batch once-through fuel cycle; one is a 15 full power year cycle design with a core volume of 9.5 cubic meters, and the other is a 12 full power year cycle design with a core volume of 7.4 cubic meters. For these two point designs, fuel cycle analyses and reactivity feedback coefficients calculations were performed. The important design and performance parameters of these designs are summarized in Table 1.

Compared to the reference TRU design, the 9.5 cubic meter design improves the average discharge burnup and the reactivity significantly; the average discharge burnup is increased to 83.1 MWd/kg from 57.4, and the reactivity swing is reduced to 0.6% from 3.0%. The power peaking factor is also improved from 1.72 to 1.64. The peak fast fluence value of 3.9×10^{23} n/cm2 is well within the assumed fast fluence limit of 4×10^{23} n/cm2. Based on a conservative simple thermal-hydraulic model, the maximum power density for natural circulation was estimated to be 92 W/cm^3. Both average (43.6 W/cm^3) and peak (71.2 W/cm^3) power densities are much lower than this limit. The estimated peak temperatures of clad surface and fuel centerline were 655°C and 1540°C, respectively. The peak fuel centerline temperature is much lower than the melting point of plutonium nitride fuel (2500°C). Even though it marginally exceeds the operating limit for HT-9 cladding (650°C), this peak clad surface temperature is believed to be fine since it was estimated very conservatively by applying hot-channel factors. However, more detailed thermal-hydraulic analyses need to be performed.

Table 1. Design and Performance Parameters of Point Designs

		9.5 Cubic Meter	7.4 Cubic Meter
Fuel Pin Diameter, cm		1.905	1.905
Fuel Pin Pitch-to-Diameter Ratio		1.625	1.550
Cladding Thickness, cm		0.100	0.100
Fuel Smeared Density, %		78.0	78.0
Volume Fraction	Fuel	0.2147	0.2360
	Fuel-Cladding Bond	0.0606	0.0666
	Cladding	0.0682	0.0749
	Coolant	0.6566	0.6225
Active Core Height, cm		200	200
Active Core Diameter, cm		246	216
Low Enrichment, %TRU/HM		13.30	13.69
Middle Enrichment, %TRU/HM		18.22	18.62
High Enrichment, %TRU/HM		21.28	20.40
BOC Loading, MT	HM	25.56	21.28
	U	21.47	17.89
	TRU	4.09	3.39
Driver Power Density, kW/liter	Average	43.58	57.50
	Peak	70.15	93.92
Peak Linear Heat Rate, W/cm		590.48	709.16
Discharge Burnup, MWd/kg	Average	83.11	79.90
	Peak	135.68	129.52
Peak Fast Neutron Fluence, 10^{23} n/cm^2		3.90	3.91
Peaking Factor (BOC/EOC)		1.63/1.64	1.63/1.65
Cycle Reactivity Swing, %Δk		0.61	0.88
Cycle Length, full power year		15	12

In general, the performance of the 7.4 cubic meter point design is inferior to that of the 9.5 cubic meter point design. However, compared to the reference TRU design, it is improved significantly; the average discharge burnup is increased to 79.9 MWd/kg from 57.4, and the reactivity swing is reduced to 0.9% from 3.0%. The power peaking factor is also improved from 1.72 to 1.65. The peak fast fluence value of 3.9×10^{23} n/cm2 is well within the assumed fast fluence limit. The maximum power density for natural circulation was estimated to be 78 W/cm^3, which is significantly lower than the 9.5 cubic meter point design due to the smaller pin pitch-to-diameter ratio. While the average power density (57.6 W/cm^3) is well below this limit, the peak power density (93.9 W/cm^3) is higher than this limit. The estimated peak temperatures of clad surface and fuel centerline were 710°C and 1770°C, respectively. The peak fuel centerline temperature is much lower than the melting point of plutonium nitride fuel, but the peak clad surface temperature exceeds the operating limit for HT-9 cladding (650°C). Even though it is a very conservative value, design modifications should be investigated to reduce it, including further optimization of enrichment zoning.

Table 2 presents the kinetics parameters and reactivity coefficients for both the 9.5 and 7.4 cubic meter point designs. They were calculated for BOC, MOC, and EOC configurations. The MOC configuration corresponds to the time point where the multiplication factor attains its maximum value. In both cases, the effective delayed neutron fraction β is about 350 pcm at BOC. As the fuel depletes, it decreases to ~320 pcm because of the increased fraction of minor actinides. The prompt neutron lifetime at BOC is 0.53 μs for the 9.5 cubic meter core, and it becomes slightly smaller in the 7.4 cubic meter core because of the reduced coolant volume fraction. In both systems, the prompt lifetime decreases slightly with fuel depletion due to the increased parasitic absorption.

The large amount of lead coolant due to an open lattice design results in a significantly larger coolant void worth relative to conventional sodium-cooled compact cores. For the 9.5 cubic meter core, the complete loss of coolant would lead to a reactivity increase of ~12$. The reactivity increase becomes slightly smaller in the 7.4 cubic meter core because of the reduced coolant volume fraction and increased neutron leakage. However, the coolant void worth is of no significance in this open lattice lead-cooled core since the boiling temperature of lead is well above the melting temperature of the HT9 cladding. Typically voiding is important because it can lead to a cladding failure, but in this case, it would have to occur after the cladding has already failed.

The three major material thermal expansion coefficients are typical for fast systems. With the fuel burnup, the flux peak moves inside because of the three-region enrichment zoning employed to flatten the power distribution. As a result, the coolant density coefficient increases as the fuel depletes, while the fuel and structure density coefficients become slightly more negative. The geometrical thermal expansion coefficients are sufficient to override the positive coolant density

Table 2. Kinetics Parameters and Reactivity Coefficients of Point Designs

	9.5 Cubic Meter			7.4 Cubic Meter		
	BOC	MOC	EOC	BOC	MOC	EOC
Delayed Neutron Fraction	0.0035	0.0032	0.0031	0.0035	0.0032	0.0032
Prompt Neutron Lifetime (μs)	0.534	0.504	0.498	0.503	0.473	0.468
Lead Void Worth ($)	11.64	12.20	12.20	10.76	10.68	10.66
Coolant Density Coefficient (cents/C)	0.18	0.21	0.22	0.17	0.18	0.19
Fuel Density Coefficient (cents/C)	-0.078	-0.083	-0.086	-0.078	-0.083	-0.085
Structure Density Coefficient (cents/C)	-0.009	-0.010	-0.010	-0.009	-0.010	-0.010
Radial Expansion Worth (cents/C)	-0.14	-0.15	-0.15	-0.13	-0.14	-0.15
Axial Expansion Coefficient (cents/C)	-0.06	-0.07	-0.07	-0.07	-0.07	-0.07
Doppler Coefficient (cents/C)	-0.12	-0.11	-0.10	-0.12	-0.11	-0.10
Voided Doppler Coefficient (cents/C)	-0.11	-0.09	-0.09	-0.11	-0.09	-0.09
Control Rod Worth ($/cm)	-0.04	-0.06	-0.07	-0.06	-0.07	-0.07
A, Power/Flow React. Decrement (cents)	-63	-63	-59.5	-57	-54	-51
B, Power/Flow Coefficient (cents)	-20.58	-19.85	-18.38	-20.58	-20.58	-20.58
C, Inlet Temperature Coefficient (cents/C)	-0.14	-0.12	-0.1	-0.15	-0.14	-0.13

coefficient, and they become slightly more negative with the fuel burnup. The Doppler coefficients are sufficient, and typical for fast reactors.

The integral reactivity parameters used in the quasi-static reactivity balance analysis for the passive regulation of power were also estimated and included in Table 2. [8] For this estimation, the average coolant temperature rise across the core and the increment of the average fuel temperature relative to the coolant temperature were determined using simple thermal-hydraulic and heat conduction models. The average coolant rise was estimated to be 147°C for both designs, and the average fuel temperature increment was estimated to be 350°C for the 9.5 cubic meter design and 300°C for the 7.4 cubic meter design. For both designs, the power/flow coefficient B and the inlet temperature coefficient C are similar to those of sodium-cooled, metal-fueled cores. However, the power reactivity decrement A is much more negative than that of sodium-cooled, metal-fueled cores. The magnitude of the power reactivity decrement is about three times larger than that of the power/flow coefficient. This implies that the reactivity vested in fuel temperature changes dominates that vested in coolant temperature changes. In fact, the ratio between them is out of the range for acceptable asymptotic core outlet temperatures. [8] Even though more detailed safety analyses are required to draw more decisive conclusions, design modifications may be appropriate so that the coolant density coefficient is reduced and the radial expansion coefficient is enhanced.

5. Conclusions and Future Work

Neutronics design studies for a 400 MWt high temperature lead-cooled, closed fuel cycle fast reactor have been performed. Starting from a reference 400 MWt core design utilizing weapons plutonium fuel, a reference TRU core design was developed by replacing the weapons-grade Pu feed with the TRU feed separated from light water reactor (LWR) spent fuel. Using this reference TRU core design, parametric studies were performed to maximize the discharge burnup and minimize the burnup reactivity swing. Based on the results of these parametric studies, two point TRU core designs were developed for a single-batch once-through fuel cycle; one is a 15 full power year cycle design with core volume of 9.5 cubic meters, and the other is a 12 full power year cycle design with core volume of 7.4 cubic meters. For these two point designs, fuel cycle analyses and reactivity feedback coefficients calculations were performed.

Most of the design goals have been met although there are some mentionable problems. The 9.5 cubic meter core design with a 15 full power year cycle achieves a reactivity swing of 0.6% with an average discharge burnup of 83.1 MWd/kg. The peak fast fluence value is within the fast fluence limit of HT9. Both average and peak power densities are well below the estimated limit for natural circulation. The estimated peak temperature of fuel centerline is much lower than the melting point of plutonium nitride fuel. The estimated peak clad surface temperature appears to be fine, but more detailed thermal-hydraulic analyses need to be performed. The performance of the compact 7.4 cubic meter design is slightly inferior to the 9.5 cubic meter design. The reactivity swing for a 12 full power year cycle is 0.9%, and the average discharge burnup is 79.9 MWd/kg. While the average power density is well below the estimated limit for natural circulation, the peak power density exceeds this limit. In addition, the estimated peak clad surface temperature exceeds the operating limit for HT-9 cladding. Even though it was estimated very conservatively, design modifications should be investigated to reduce it, including further optimization of enrichment zoning.

One area that needs to be investigated further is core design effects on the geometrical thermal expansion coefficients and the coolant density coefficient. At present, the reactivity vested in fuel temperature changes dominates that vested in coolant temperature changes. The ratio of the power reactivity decrement to the power/flow coefficient is out of the range for acceptable asymptotic core outlet temperatures. Even though more detailed safety analyses are required to draw more decisive conclusions, design modifications may be appropriate so that the coolant density coefficient is reduced and the radial expansion coefficient is enhanced. By utilizing a different core height to diameter ratio, the magnitude of these coefficients can be improved.

Acknowledgements

This work was supported by the U.S. Department of Energy under Contract number W-31-109-Eng-38.

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