A ONCE FOR LIFE CORE DESIGN FOR THE ENCAPSULATED NUCLEAR HEAT SOURCE (ENHS) REACTOR

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ABSTRACT

The reference core for the Encapsulated Nuclear Heat Source (ENHS) reactor has been redesigned so as to maintain the fuel rods clad integrity up to the peak discharge burnup of over 100 GWd/tHM. It was found possible to design a once-for-life uniform composition and blanket free core that will provide the design goals of nearly zero burnup reactivity swing throughout ~20 years of full power operation, natural circulation at $125MW_{th}$ without exceeding permissible temperatures, and maintaining of clad integrity. The newly designed reference core has 1.3cm in diameter smeared fuel diameter, 0.13cm clad thickness, fission gas plenum volume identical to the fuel volume, lattice p/d ratio of 1.35 and $11.344^{w}/_{o}$ Pu. The core reactivity coefficients, reactivity worth, and power distributions are nearly constant throughout the core life.

1. INTRODUCTION

The Encapsulated Nuclear Heat Source (ENHS) is a Pb or Pb-Bi cooled innovative 125 MW_{th} reactor suitable for use in countries with small and medium electricity grid [1-6]. Unique features of the ENHS that affect the core design include 20 years of operation without refueling, no fuel shuffling, 100% natural circulation and autonomous operation. In order to achieve these features it was decided to design the core to maintain a nearly zero burnup reactivity swing and to have a significantly negative temperature reactivity feedback. In addition, the fuel rods integrity should be maintained throughout the fuel life.

Applying a recently developed model for evaluating the mechanical integrity of metallic fuel rods as a function of burnup and clad temperature [8] it was found [9] that the fuel rod design previously adopted for the reference ENHS core can not maintain the fuel rod clad integrity up to the discharge burnup goal. The original reference ENHS cores featured fuel rods having a diameter of 1.0cm, clad thickness of 0.1cm, and 75% or 50% fission gas plenum length to fuel length ratio [1-4]. Two fuel rod design modifications can increase the discharge burnup for which the clad will maintain its integrity: increasing the ratio of fission gas plenum volume to fuel volume, and increasing the clad thickness to fuel diameter ratio [9]. The former is more effective than the latter. Unfortunately, increasing the clad thickness or increasing the fission gas plenum length both increase the friction of coolant flow through the core. Increased friction losses interferes with the natural circulation resulting in a reduction of the power level the coolant can remove from the core without exceeding acceptable outlet temperatures. However, by increasing the fuel rod diameter and maintaining the coolant-to-fuel volume ratio it is possible to reduce the coolant friction losses through the core to compensate for the increase in the friction losses due to increase in the length of the fission gas plenum and in the clad thickness. Thermal hydraulics and fuel clad integrity analysis led to a new fuel rod designs of the following dimensions: The fuel rod is 1.3cm in diameter (smeared), its clad thickness is 0.13cm and the fission gas plenum length is 100% of the fuel length. This fuel can maintain the clad integrity up to the peak target burnup[9] and can provide acceptable friction losses.

The primary purpose of this work was to neutronically redesign the core using the new fuel rod dimensions and to determine the reactor physics characteristics of the new reference core. Another purpose of this work was to optimize the design of the shutdown assembly and peripheral absorber so as to provide a sufficiently low subcritical multiplication ($k_{eff} < 0.95$) at shutdown.

2. NEW REFERENCE CORE DESIGN

Figure 1 and Table I describe the geometry, dimensions and composition used for modeling the ENHS reactor. The core is a bundle of fuel rods positioned at the bottom on grid plate pedestals. There are no blanket rods and no reflector rods. The unit cell is triangular. For neutronics calculation the core is homogenized into a single annular cylindrical region. The reference core has 125cm long fuel and is designed to generate 125MW_{th} thermal power with an average linear heat generation rate of 101.4W/cm. The number of fuel rods needed is 9862 [=125MW/(125cm*101.4W/cm)]. The corresponding effective core outer radius is 111.01cm. The fuel in all the rods is of a uniform composition (U-Pu-10Zr) and the fuel smear density is assumed to be 75% of the nominal density. The uranium is depleted to 0.2$^{w}/_{o}$ U-235 and the plutonium initial composition is assumed to be: 67.2 $^{w}/_{o}$ ^{239}Pu, 21.7 $^{w}/_{o}$ ^{240}Pu, 6.4 $^{w}/_{o}$ ^{241}Pu, and 4.7 $^{w}/_{o}$ ^{242}Pu.

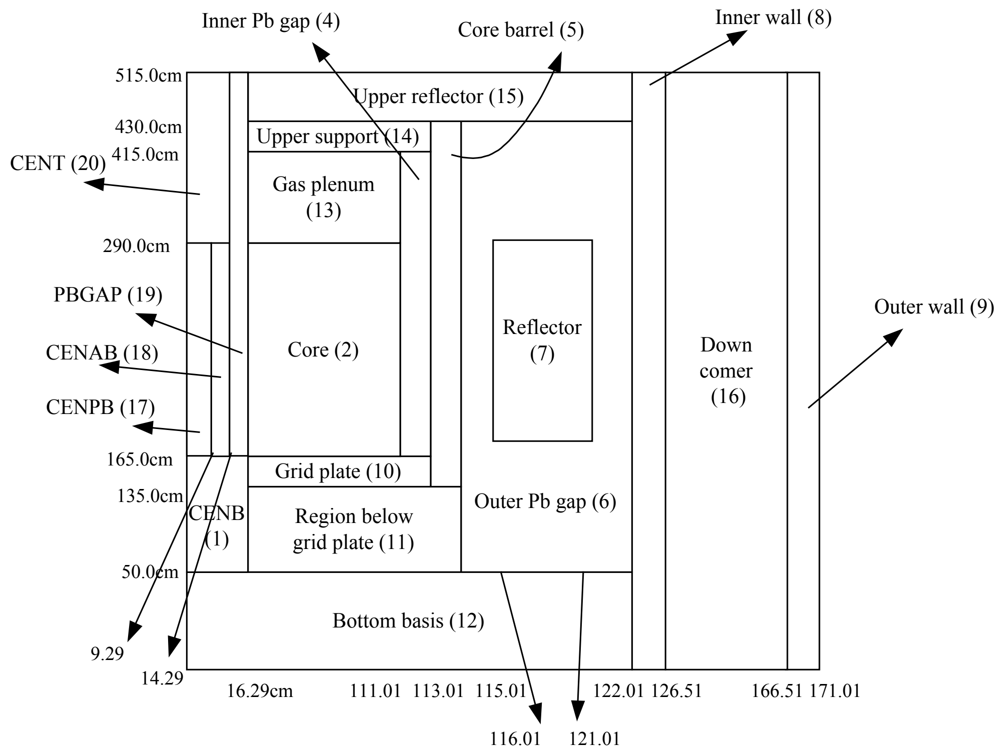


Figure 1. Schematic configuration of the reference ENHS reactor used for neutronic calculations

Table I. Material Compositions for the Reference ENHS Reactor

Region No.	Region name	Material volume fractions	Temp (K)
1	Bottom of absorber	99%Pb-Bi + 1%SS	698
2	Core region	50.24%Pb-Bi + 34.56% fuel + 15.20%SS	698
4	Inner Pb-Bi[a] gap	100%Pb-Bi	698
5	Core barrel	100%SS	698
6	Outer Pb-Bi gap	100%Pb-Bi	623
7	Space for peripheral absorber	100%Pb-Bi	623
8	Inner structural wall	100%SS	623
9	Outer structural wall	100%SS	623
10	Lower grid plate	50%Pb-Bi + 50%SS	623
11	Below grid plate	80%Pb-Bi + 20%SS	623
12	Bottom base	100%SS	623
13	Fission gas plenum	50.24%Pb-Bi + 15.20%SS	773
14	Upper grid plate	50%Pb-Bi + 50%SS	773
15	Upper reflector	100%Pb-Bi	773
16	Down comer	100%Pb-Bi	624
17	Pb-Bi region of absorber	100%Pb-Bi	698
18	Space for central absorber	100%Pb-Bi	698
19	Gap of absorber	100%Pb-Bi	698
20	Top of absorber	20%SS	698

[a] The results presented in this paper were calculated using Pb rather than Pb-Bi coolant.

The coolant is either Pb or Pb-Bi. In this study Pb is used for the calculations; the neutronic characteristics of Pb and Pb-Bi are nearly the same. The structural material is HT-9. However stainless steel having 64.8w/o Fe, 17w/o Cr, 14w/o Ni, 2.8w/o Mo, and 1.5w/o Mn is used in this study. This leads to a conservative design as replacing the SS with HT-9 increases BOL k_{eff} by 0.75%.

A central cylindrical region having a cross section area that is equivalent to that of 217 unit cells is located at the center of core. It is the location of the “central absorber” assembly – the primary safety element used for scramming the ENHS and for keeping it subcritical. During reactor operation, this central site is filled with coolant. The effective radius of the central absorber region is 16.29cm. There are no control rods in the core. The operational reactivity control is done by six annular segments of a peripheral absorber the shutdown position of which is denoted in Fig. 1 as the “reflector” (Region 7).

3. COMPUTATIONAL METHOD

All the neutronic calculations are done with the DIF3D code [10] and the REBUS-3 code [11] using R-Z geometry. The DIF3D code is used to solve the multigroup diffusion equation using the finite difference method. The REBUS-3 code gets the neutron flux from DIF3D and solves the transmutation equations using the matrix-exponential technique. For the depletion analysis the core is divided into nine equal volume zones: three radial zones and three axial zones. In each zone, the isotopic atomic number densities are assumed to be constant. The multigroup cross sections for the DIF3D calculations are generated with an ENDF/B-VI based 150 group cross section library (MATXS format) for master nuclides [12] and with an ENDF/B-VI 80 group cross section library (MATXS format) for fission products [13]. The MATXS format library is transformed into the multigroup cross section set (ISOTXS format) needed for DIF3D using the TRANSX code [14], accounting for the proper operating temperature. The 150 group cross section set is used for BOL calculations while the EOL calculations are done with an 80 group cross section set prepared by an approximate 150 group EOL neutron spectrum. This is because 150 group cross sections are unavailable for fission product nuclides. A lumped fission product cross section, based on data for 172 isotopes, is used to account for the effect of fission products. There is one lumped fission product for 17 different fissionable isotopes. The 80 group microscopic cross section of the lumped fission products is generated by weighting the fission products cross section with the fission yield (ENDF/B-VI based). The DIF3D calculations in REBUS-3 are performed using an 80 group cross section set. The decay chain spans the range from U-234 to Cm-246. This computational method was benchmarked against MOCUP; good agreement was found[15,16].

4. NEW REFERENCE CORE DESIGN AND PERFORMANCE

To begin with we search for a combination of lattice pitch-to-diameter (p/d) ratio and initial plutonium weight percent that offers a BOL k_{eff} of 1.0042 and as flat as possible k_{eff} over 20 years of full power operation. The above target k_{eff} value is slightly larger than unity so as to accommodate uncertainties in data and computations. Figure 2 shows the k_{eff} evolution with burnup for the reference core and its sensitivity to variation of the p/d ratio. It is found that a flat k_{eff} core is obtained with p/d of 1.35 and initial plutonium enrichment of 11.344wt%. The maximum value of burunup reactivity swing over 20 years is less than 0.2% ; this is less than 0.5$.

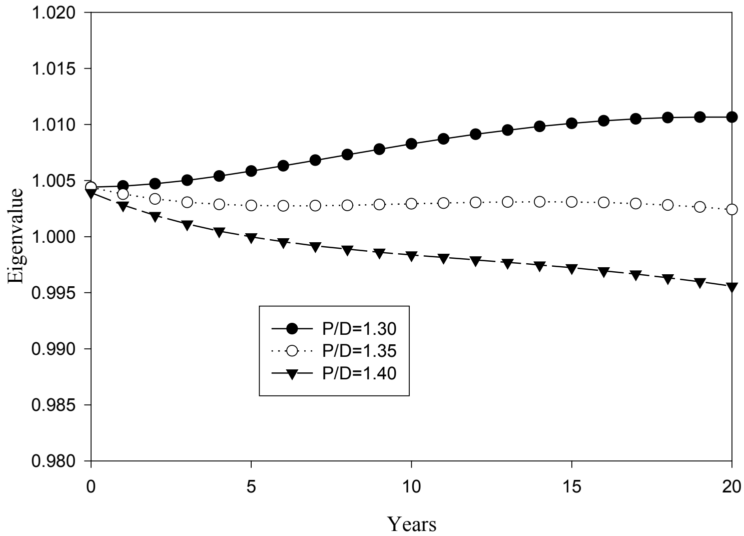


Figure 2. k_{eff} evolution over core life

Table II summarizes selected characteristics of the new reference core at BOL and EOL. The temperature coefficients of reactivity are all negative except for that of the coolant expansion. However, the negative reactivity coefficient associated with the radial expansion of the core structure can compensate for the coolant thermal expansion effect. The void coefficients for 100% voided inner and middle cores are positive. However, creation of voiding in the ENHS core is practically impossible since the boiling temperature of Pb-Bi is 1943K; that is 1000K higher than the operating temperature and even higher than the melting temperature of the metallic fuel and clad. The fast neutron fluence (E > 0.1MeV) at 20 effective full power years (EFPY) is below 4.0×10^{23} n/cm^2 that is assumed to be the radiation damage limit of HT-9. In fact, it is possible to operate this core for ~ 22EFPY. Alternatively, it appears possible to operate the ENHS reactor for 20EFPY at close to 140MW$_{th}$ up to a peak burnup of ~104GWD/tHM.

Table II. ENHS Reference Core Physics Data

Performance Parameter	BOL	EOL
Peak-to-average power density	1.767	1.778
Peak linear heat rate (W/cm)	177.8	178.7
Peak-to-average channel power	1.425	1.435
Peak-to-average power density in hot channel	1.241	1.235
Peak burnup after 20 (21.9) EFPY (GWD/tHM)		95.2 (104.4*)
Average burnup after 20 (21.9) EFPY (GWD/tHM)		50.7 (55.6*)
Peak fast (E>0.1MeV) neutron flux (n/cm^2-s)	5.681E+14	5.895E+14
Peak fast (E>0.1 MeV) fluence at 20 (21.9) EFPY (n/cm^2)		3.646E+23 (4E+23)
Conversion ratio	1.153	1.065
Effective delayed neutron fraction	0.0046	
Maximum Δk_{eff} with burnup (%)	0.197	
Doppler effect (dk/kk'-$_o$C)	-5.78547E-06	-4.55582E-06
Axial fuel expansion (dk/kk'-$_o$C)	-4.42189E-06	-4.39633E-06
Coolant expansion (dk/kk'-$_o$C)	+5.17092E-07	+9.68071E-07
Grid-plate radial expansion (dk/kk'-$_o$C)	-8.97388E-06	-6.80369E-06
Cold (350$_o$C) to hot (480$_o$C; fuel: 700$_o$C) ρ swing (dk)		
Doppler effect	-2.03373E-03	-1.65610E-03
Axial fuel expansion	-0.56615E-03	-0.57690E-03
Coolant expansion	+0.04453E-03	+0.14029E-03
Grid-plate expansion	-1.15292E-03	-0.89738E-03
Total	-3.70827E-03	-2.99013E-03
Void reactivity effect (dk)		
Voiding inner 1/3 core	+2.54917E-02	+2.65734E-02
Voiding middle 1/3 core	+0.88586E-02	+0.94506E-02
Voiding outer 1/3 core	-0.31642E-02	-0.29761E-02
Voiding whole core	+3.11058E-02	+3.28106E-02
Tungsten annulus reactivity worth (dk)	0.01130	0.01047
Central absorber reactivity worth (dk)	0.042	0.042
Total plutonium mass (kg)	1985.9	2057.5
Fissile-to-total plutonium ratio	0.7360	0.7288

* Limited by radiation damage to clad of 4x10^{23} fast (E>0.1 MeV) neutrons per cm^2.

The peripheral absorber is designed to have sufficient reactivity worth that can compensate for the reactivity variations due to reactivity deficiency from the startup temperature of 350C at BOL to full power at EOL. The total reactivity deficiency that needs to be compensated is ~1.07%. The peripheral absorber referred to in Table I is made of 80$^v/_o$ tungsten at 75% nominal density and of 20$^v/_o$ SS. Tungsten is chosen for the absorber as its density is higher than that of the coolant so that the peripheral absorber can be scrammed by gravity. Figure 3 shows the reactivity worth of the tungsten-based reflector as a function of its thickness for two core heights – 125cm and 150cm. It is concluded that 5cm is sufficient to provide for the needed reactivity worth.

The central absorber is designed to have sufficient reactivity worth to be able to shutdown the core. In addition, the combined insertion of the central and peripheral absorbers is to bring k_{eff} to below 0.95. Table III shows the reactivity worth of the central absorber for various combinations of W, B_4C, and $ZrH_{1.6}$. The boron is enriched to 92% ^{10}B. The solid hydride $ZrH_{1.6}$ can maintain its hydrogen at the ENHS operating temperatures. The results show that a central absorber of 24$^v/_o B_4C$+56$^v/_o ZrH_{1.6}$ +20$^v/_o$SS offers a maximum reactivity worth of 4.18%; this is adequate for core shutdown. The combined reactivity worth of the central absorber and of the peripheral absorber can be as high as 7.52% -- quite adequate for safe shutdown and suppression of the spontaneous fission neutron multiplication. Both absorbers are 5cm thick. The small sensitivity to the amount of hydrogen is surprising. The preferred design would probably use B_4C and tungsten rather than $ZrH_{1.6}$ so as to make the absorber element density larger than that of the coolant.

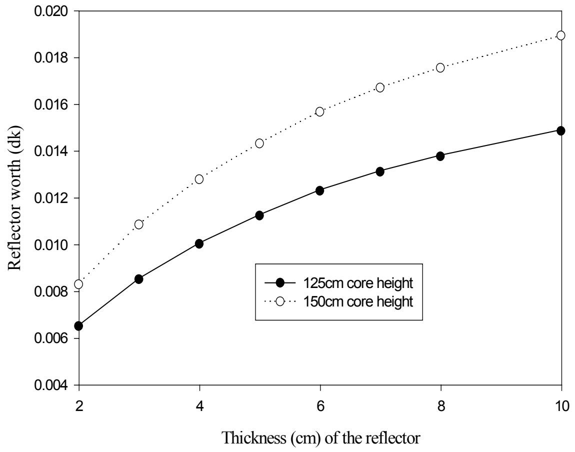


Figure. 3 Reactivity worth of $80^v/_oW+20^v/_oSS$ peripheral absorber ("reflector") versus thickness

Table III. BOL Reactivity Worth of the Central Absorber with and without the Peripheral Absorber

Absorber (5.0cm thick)	Reflector (5.0cm thick)	Worth (dk)
$80^v/_oW+20^v/_oSS$	-	0.0172
$50^v/_oW+30^v/_oB_4C+20^v/_oSS$	-	0.0349
$10^v/_oW+70^v/_oB_4C+20^v/_oSS$	-	0.0402
$8^v/_oB_4C+72^v/_oZrH_{1.6}+20^v/_oSS$	-	0.0408
$24^v/_oB_4C+56^v/_oZrH_{1.6}+20^v/_oSS$	-	0.0418
$40^v/_oB_4C+40^v/_oZrH_{1.6}20^v/_oSS$	-	0.0418
$56^v/_oB_4C+24^v/_oZrH_{1.6}+20^v/_oSS$	-	0.0415
$72^v/_oB_4C+8^v/_oZrH_{1.6}+20^v/_oSS$	-	0.0412
$80^v/_oW+20^v/_oSS$	$80^v/_oW+20^v/_oSS$	0.0287
$50^v/_oW+30^v/_oB_4C+20^v/_oSS$	$50^v/_oW+30^v/_oB_4C+20^v/_oSS$	0.0609
$24^v/_oB_4C+56^v/_oZrH_{1.6}+20^v/_oSS$	$24^v/_oB_4C+56^v/_oZrH_{1.6}+20^v/_oSS$	0.0752

One of the unique features of the ENHS core is that its reactivity coefficients, reactivity worth and power distribution are almost constant throughout the core life. Figure 4a shows the axial power distributions in the hottest channel at BOL and EOL and Figure 4b shows the channel averaged radial power distribution. The peak-to-average power density varies from BOL to EOL by <1%.

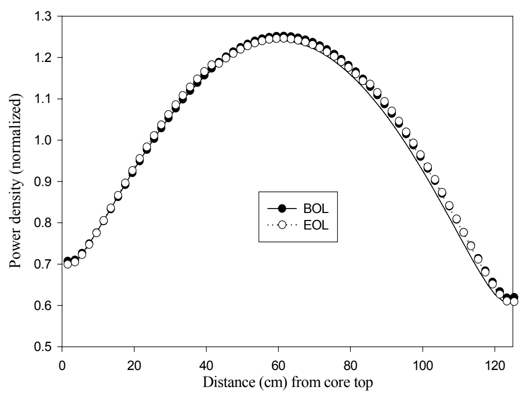


Figure 4a. Axial power distribution at BOL and EOL (normalized)

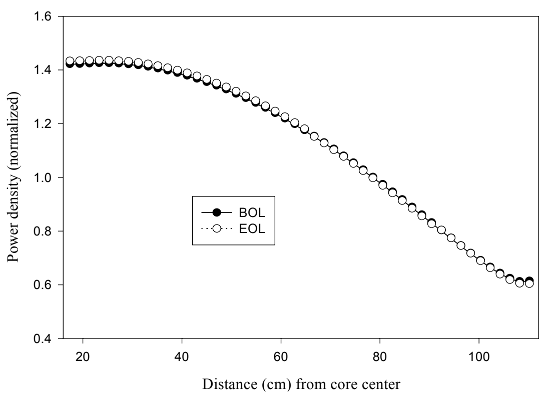
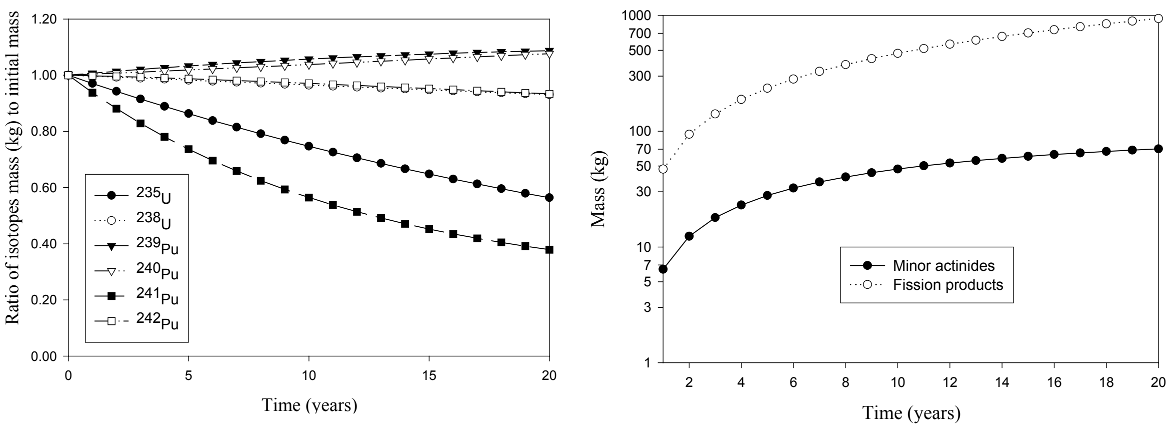


Figure 4b. Radial power distribution at BOL and EOL (normalized)

Table IV and Figure 5 show the evolution with burnup of the fuel isotopes of the reference ENHS core. Table V shows the evolution of the total plutonium and uranium inventories, of the concentration of their fissile isotopes, of k_{eff}, and of the conversion ratio (CR). The CR is defined as the ratio of the thermally fissile isotopes generated per thermally fissile isotopes consumed. In a fast spectrum reactor such as the ENHS this definition is not a good measure of the effect of transmutation on the fuel quality evolution. The fissile plutonium fraction stays nearly constant over life. The CR decreases by about 7% as does the U-238 inventory. The plutonium inventory increases by 3.6%. However, after adding depleted uranium to make up for the fuel fissioned: about 5.5% of the initial fuel inventory, and after removing the fission products, the left-over fuel will have just about the right reactivity as needed for the fuel loading for a new ENHS module.

Table IV. Evolution of the Fuel Isotope Inventory (kg) Over the Core Life

Isotopes	0 year	5 years	10 years	15 years	20 years
^{234}U	.000E+00	8.157E-03	6.248E-02	1.890E-01	3.952E-01
^{235}U	3.104E+01	2.680E+01	2.319E+01	2.012E+01	1.750E+01
^{236}U	.000E+00	8.407E-01	1.529E+00	2.091E+00	2.548E+00
^{238}U	1.549E+04	1.521E+04	1.494E+04	1.468E+04	1.442E+04
^{238}PU	.000E+00	6.533E-01	2.369E+00	4.587E+00	6.966E+00
^{237}NP	.000E+00	9.384E-01	1.773E+00	2.512E+00	3.165E+00
^{239}PU	1.335E+03	1.378E+03	1.411E+03	1.435E+03	1.451E+03
^{240}PU	4.309E+02	4.389E+02	4.473E+02	4.557E+02	4.640E+02
^{241}PU	1.271E+02	9.361E+01	7.174E+01	5.747E+01	4.818E+01
^{242}PU	9.334E+01	9.213E+01	9.060E+01	8.888E+01	8.708E+01
^{241}AM	.000E+00	2.431E+01	4.006E+01	5.030E+01	5.695E+01
^{242m}AM	.000E+00	2.177E-01	6.917E-01	1.249E+00	1.803E+00
^{243}AM	.000E+00	2.110E+00	3.968E+00	5.601E+00	7.038E+00
^{242}CM	.000E+00	1.677E-01	2.932E-01	3.719E-01	4.215E-01
^{243}CM	.000E+00	1.100E-03	3.877E-03	7.133E-03	1.026E-02
^{244}CM	.000E+00	8.582E-02	2.990E-01	5.879E-01	9.165E-01
^{245}CM	.000E+00	1.531E-03	1.048E-02	3.028E-02	6.165E-02
^{246}CM	.000E+00	8.414E-06	1.173E-04	5.173E-04	1.427E-03



(a) Ratio of isotopes mass to initial mass (U, Pu)　　(b) Isotopes mass (kg, MA, FP)

Figure 5. Evolution of fuel isotopes inventory

Table V. Evolution of U and Pu inventories and Fissile Concentration, k_{eff}, and Conversion Ratio

Years	Total Pu (kg)	$(Pu^{239}+Pu^{241})/Pu$	Total U (kg)	U^{235}/U	K_{eff}	CR
0	1.98591E+03	.7360	1.55203E+04	.0200	1.00438	1.15296
1	1.98905E+03	.7357	1.54640E+04	.0195	1.00379	1.14907
2	1.99240E+03	.7355	1.54079E+04	.0190	1.00336	1.14496
3	1.99593E+03	.7352	1.53520E+04	.0185	1.00306	1.14068
4	1.99960E+03	.7349	1.52965E+04	.0180	1.00287	1.13626
5	2.00337E+03	.7346	1.52411E+04	.0176	1.00277	1.13174
6	2.00722E+03	.7343	1.51860E+04	.0171	1.00273	1.12715
7	2.01111E+03	.7340	1.51312E+04	.0167	1.00274	1.12251
8	2.01501E+03	.7336	1.50767E+04	.0163	1.00279	1.11786
9	2.01892E+03	.7333	1.50224E+04	.0159	1.00285	1.11321
10	2.02280E+03	.7329	1.49684E+04	.0155	1.00292	1.10857
11	2.02664E+03	.7326	1.49146E+04	.0151	1.00299	1.10397
12	2.03043E+03	.7322	1.48611E+04	.0147	1.00305	1.09941
13	2.03414E+03	.7318	1.48078E+04	.0144	1.00308	1.09491
14	2.03779E+03	.7314	1.47548E+04	.0140	1.00310	1.09047
15	2.04134E+03	.7310	1.47021E+04	.0137	1.00308	1.08610
16	2.04478E+03	.7306	1.46496E+04	.0134	1.00303	1.08181
17	2.04814E+03	.7301	1.45973E+04	.0130	1.00294	1.07760
18	2.05137E+03	.7297	1.45452E+04	.0127	1.00281	1.07348
19	2.05449E+03	.7292	1.44934E+04	.0124	1.00263	1.06944
20	2.05749E+03	.7288	1.44418E+04	.0121	1.00241	1.06550

5. CONCLUSIONS

It is possible to design a once-for-life uniform composition and blanket free core for the Encapsulated Nuclear Heat Source that will provide the design goals of nearly zero burnup reactivity swing throughout ~20 years of full power operation, natural circulation at 125MW_{th} without exceeding permissible temperatures, and maintaining clad integrity up to the peak discharge burnup of 104 GWd/tHM. The newly designed reference core has 1.3cm in diameter smeared fuel diameter, 0.13cm clad thickness, fission gas plenum volume identical to the fuel volume and p/d ratio of 1.35. With 11.344$^{w}/_{o}$ Pu it maintains k_{eff} constant within less than 0.2%. The temperature coefficients of reactivity are all negative except for that of the coolant expansion. However, the negative reactivity coefficient associated with the radial expansion of the core structure can compensate for the coolant thermal expansion. The core reactivity coefficients, reactivity worth, and power distributions are almost constant throughout the core life.

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