RADIAL POWER FLATTENING IN SODIUM FAST REACTORS

by

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SUBMITTED TO THE DEPARTMENT OF NUCLEAR SCIENCE AND ENGINEERING
IN PARTIAL FULFILLMENT OF THE REQUIREMENTS FOR THE DEGREE OF

BACHELOR OF SCIENCE IN NUCLEAR SCIENCE AND ENGINEERING
AT THE
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

JUNE 2012

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Submitted to the Department of Nuclear Science and Engineering on May 18, 2012

In Partial Fulfillment of the Requirements for the Degree of

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Abstract

In order to improve a new design for a uranium startup sodium cooled fast reactor which was proposed at MIT, this thesis evaluated radial power flattening by varying the fuel volume fraction at a fixed U-235 enrichment of 18.5%. Of particular interest was how best to reduce the radial power peaking at the center of the reactor. Two cores were modeled: one with a uniform fuel volume fraction of 37% and one with fuel volume fractions which increased with the core radius. The ERANOS code was used to determine the power density, reactivity, and material values at the beginning and end of life. The varied fuel was shown to have a flatter radial power profile, but slightly lower reactivity and more mass. The power in each cell was normalized with respect to the average power; the peak power ratio in the uniform fuel volume core was 1.59, while the peak power ratio in the varied fuel volume core was 1.16, a significant improvement. The reactivity at beginning of life dropped from 12573.9 pcm to 11734.0 pcm, and stayed about 500pcm lower over the cycle, which is not a very large amount. The total mass of the heavy metals increased from the uniform core to the varied core by less than 0.9% and the mass of U-235 by 1.2%, so the varied fuel does not significantly impact the overall fuel cycle cost.

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1 Background

1.1 Traditional Fast Reactors

Physicists recognized the ability of fast reactors to breed fissile materials as early as 1940. However, the technology was not developed at the time because natural uranium ore was found in sufficient quantities and breeding was not a priority. Thermal reactors were preferred because they were economically advantageous.

However, as only thermal reactors have been built, natural uranium has been used, leaving large amounts of depleted and reprocessed uranium. Fast breeder reactors make it possible to use all of the available uranium: not only natural, but also depleted and reprocessed. Liquid sodium is the preferred liquid-metal coolant because it has a very favorable heat extraction capacity. It also has high thermal inertia because of its large heat capacity and high boiling point. Sodium technology has been selected by many countries for the most promising new reactor designs, including the USA, France, Japan, Germany, Russia, and India [Knief, 2008].

Fast reactors are a type of nuclear fission reactor which use fast neutrons to sustain the fission process. Traditional reactors use a moderator material to slow down fission neutrons from around 2 MeV, the energy at which they are born, to 0.025 eV. Fast reactors do not have a moderator, and fission can take place at much higher energies. Because of this, fast reactor fuel must have a high fissile content. However, U-235 enriched fuel can be made directly from enrichment plants just as that for Light Water Reactors (LWRs). Such designs are the focus of work by a research group in CANES at MIT, of which the present thesis was part.

The initial attraction of fast reactors was the possibility of breeding new fuel, according to the neutron capture sequence:

$$^{238}\mathrm{U} + \mathrm{n} \rightarrow {}^{239}\mathrm{U} \xrightarrow{\beta} {}^{239}\mathrm{Np} \xrightarrow{\beta} {}^{239}\mathrm{Pu}$$

where U-238 is used to produce the fissile material Pu-239 via neutron absorption and beta decays. This process is achieved by surrounding the reactor core with U-238 blankets which absorb neutrons leaving the driver fuel. The resulting Pu-239 can then be recycled as fuel into the fast reactor. These reactors are "breeder" reactors because they create much more Pu-239 than they consume.

1.2 Current Design

The design in this project differs from conventional fast reactors because it has no U-238 breeding blanket. Instead, the core is surrounded by a highly efficient neutron reflector made from magnesium oxide (MgO) which both helps achieve a low neutron leakage rate and avoid the security concern from the production of

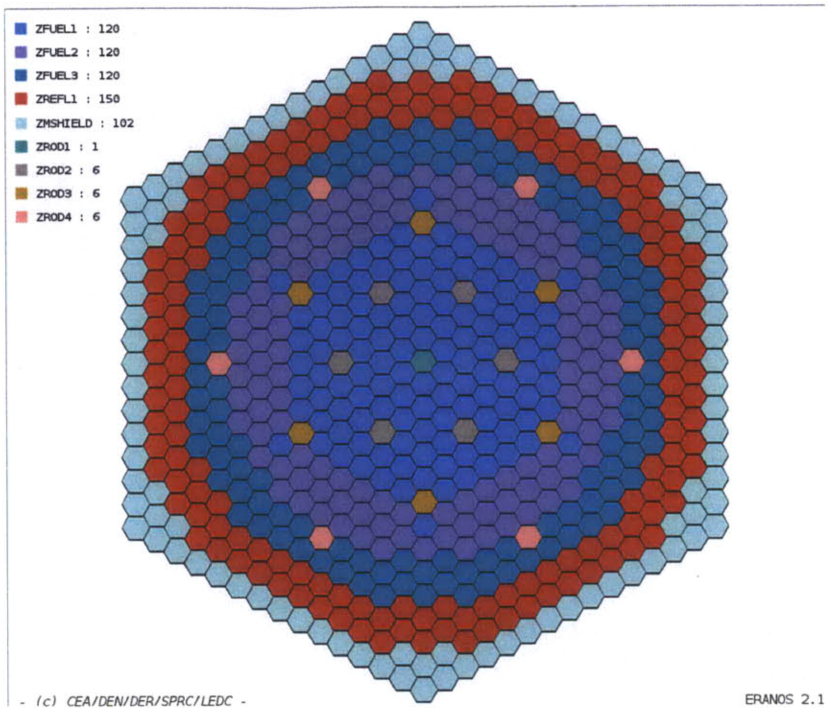

Figure 1: Uranium Oxide core using three radial zones

weapons grade plutonium. Previous work determined that MgO is the optimal material for reflecting neutrons of the energies of interest, with superior performance when compared to common reflector materials such as steel. It also has good thermal properties [R. Macdonald, 2010].

The basic reference design is sodium-cooled and fueled with uranium oxide (UO_2). The design is based on ABR-1000 [Kim, 2009] and is one of the cores studied in the Ph.D. thesis by Fei [Fei, 2012]. The fuel is surrounded by MgO reflector zones both radially and axially, which is then surrounded by B_4C shielding. Its thermal power is 2400 MW. The core layout is shown in Fig.1 and key dimensions are given in Tables 1 and 2. The pin diameter is constant even when the fuel volume fraction changes if an annular pellet is used to reduce the volume fraction.

1.3 Objective of Present Work

It is desirable to reduce the power density peaking in nuclear reactors. A flat radial power density allows higher core average power density and significantly improved economics. In prior work done investigating

Parameter	Value
Core Thermal Power (MW)	2400
Assembly Gap-Gap Distance (cm)	16.1417
Assembly Outer Flat to Flat Distance (cm)	15.7099
Assembly Innter Flat to Flat Distance (cm)	14.9225
Assembly Duct Wall Thickness (cm)	0.3937
Assembly Total Height (cm)	342
Fuel Height (cm)	102
Number of Fuel Assemblies	360
Number of Primary Control Assemblies	13
Number of Secondary Control Assemblies	6
Number of Reflecting Assemblies	150

Table 1: Key Parameters of Core Assemblies

Parameter	Value
Pitch (cm)	0.748
P/D	1.1300
Outer Cladding Diameter (cm)	0.6619
Outer Cladding Radius (cm)	0.3310
Inner Cladding Diameter (cm)	0.5619
Inner Cladding Radius (cm)	0.2810
Outer Fuel Diameter (cm)	0.5181
Outer Fuel Radius (cm)	0.2590
Unit Cell Area (cm^2)	0.4845
Bond Thickness (cm)	0.022
Number of Pins	397
Fuel VF	0.3709
Bond VF	0.1182
Cladding and Duct VF	0.2617
Coolant VF	0.2492

Table 2: Key Parameters of Fuel Pins

uranium startup SFRs, the power density peaked in the center due to the uniform enrichment and fuel volume fraction [Richard, 2012, Fei, 2012]. In the present work to achieve radial power flattening, the fuel volume fraction was varied in different sections of the core. The fuel volume fraction refers to the fraction of the volume of fuel material confined inside a fuel element. Knowing the fuel volume allows for calculation of the absolute fission density in a fuel, which is proportional to the power density. It is possible but not as desirable to vary the enrichment throughout the core since this leads to variation in local conversion ratio, hence a larger change in local power as burnup and fissile breeding takes place.

2 Method

The reactor core was modeled using ERANOS (the European Reactor Analysis Optimized code System), a neutronic code developed for fast neutron reactors by a European collaboration. ERANOS contains data libraries, codes, and calculation procedures which provide reliable neutronic calculations for thermal and fast reactor cores. It uses the ECCO code for cell calculations, which performs 2D and 3D calculations on homogenized assemblies and is based on subgroup methods combined with a fine group transport calculation. ERANOS is a deterministic code which solves the Boltzmann equation with the transport method or the diffusion approximation. The main advantage of this code is that it is very flexible and can run fast calculations.

All runs were completed using ERANOS v2.2 homogeneous RZ diffusion mode [CEA, 2006].

Six different cores were modeled: one with a uniform fuel volume fraction of 37%, and five where the fuel volume increased from the center outwards in the different fuel regions, as shown in Table 3 . The uniform 37% fuel volume fraction was based on prior work done at MIT [Fei, 2012]; an upper limit of 43% was chosen based on the reference SFR V2B core, which was designed to break even with reduced sodium voiding effect [Rimpault, 2009]. The effect of varying the fuel volume fraction was compared at both the beginning and end of life. Of particular interest are the radial power peak and the reactivity at the beginning of life and the reactivity lifetime, when the fuel has experienced full burnup. One other important number is fluence. Radiation damage on the cladding is a limiting factor in this design. As the fuel volume fraction is varied, it is important to make sure the flattened power density is not linked with an increase in flux that exceeds the fluence limits. HT9 steels have been tested in the Fast Flux Test Facility and experienced no breach up to a neutron fast fluence of 4E23 neutrons/cm^2. Oxygen Dispersion Strengthened (ODS) steels are claimed to withstand an even higher fast fluence level up to 5E23 neutrons/cm^2 [S. Ukai, 2000]. This design uses ODS steel as cladding material.

	Fuel Region 1	Fuel Region 2	Fuel Region 3
Core 1 (Uniform)	37%	37%	37%
Core 2	35%	37%	39%
Core 3	33%	37%	41%
Core 4	31%	37%	43%
Core 5	32%	37%	43%
Core 6	32%	35%	43%

Table 3: Varied fuel volume fraction cores

3 Results

3.1 Power Peaking

The objective of the work was to reduce the power peaking in the center of the reactor. As shown in Fig2 and Fig3, varying the fuel volume fraction greatly reduced the power peaking at the center and made for a much flatter radial power profile. The graphs are normalized to the average power. For the initial startup cycle at beginning of cycle, power in the uniform fuel volume fraction core peaked in the center and decreased radially outward, while the power in the varied fuel volume fraction cores was relatively flat and peaked near the middle of the core radius and very subtly peaked again near the outside. This peak is near the transition from the first fuel region to the second fuel region and again from the second fuel region to the third, as marked by the dashed lines on the figures. At the end of the cycle, the radial power profiles are lower and flatter than those at the beginning of the cycle. The uniform fuel volume fraction core still peaks in the center, but is comparatively flatter overall. The varied fuel volume fraction cores show a similar change and has a smaller power range at EOC as compared to BOC. At both BOC and EOC, the more greatly varied fuel volume fraction cores have flatter radial power profiles.

3.2 Reactivity

Reactivity was calculated for the initial startup cycle for both fuel setups. As shown in Fig.4, the reactivity of the cores with varied fuel volume fractions are lower than the reactivity of the core with uniform fuel volume fractions. This is to be expected since higher peripheral (radial) power leads to an increase in leakage losses. However, the greatest difference is about 1000 pcm, which is equivalent to about 1% in heavy metal loading and is not a significant difference.

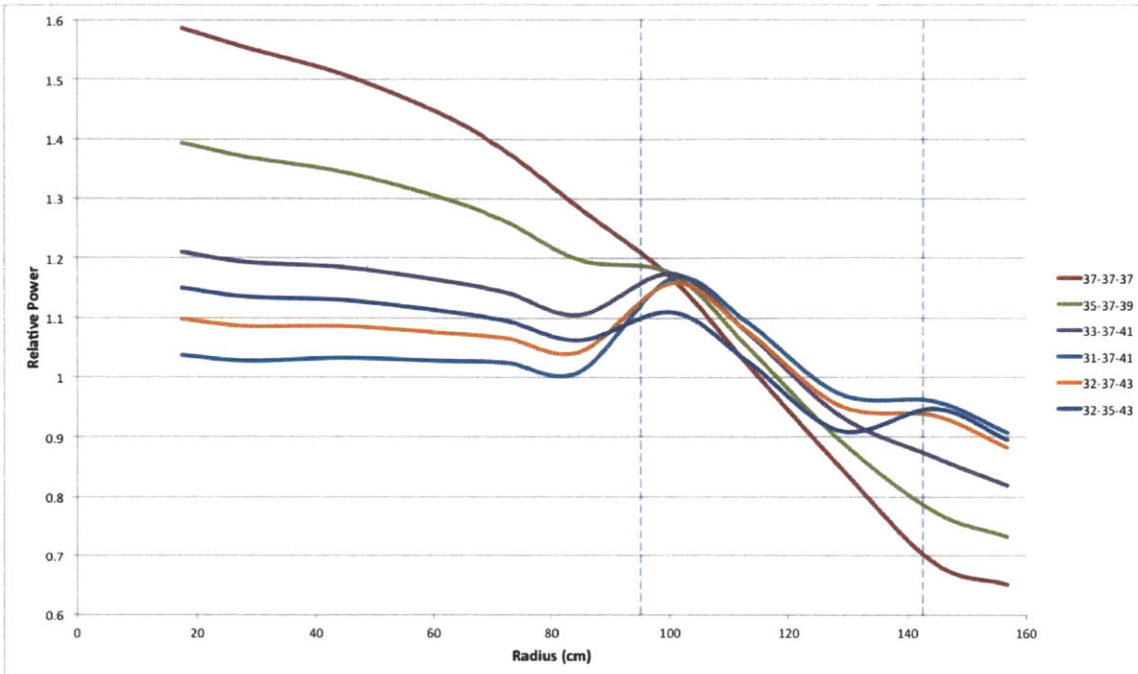

Figure 2: Normalized power along a radial axis at the beginning of the first cycle

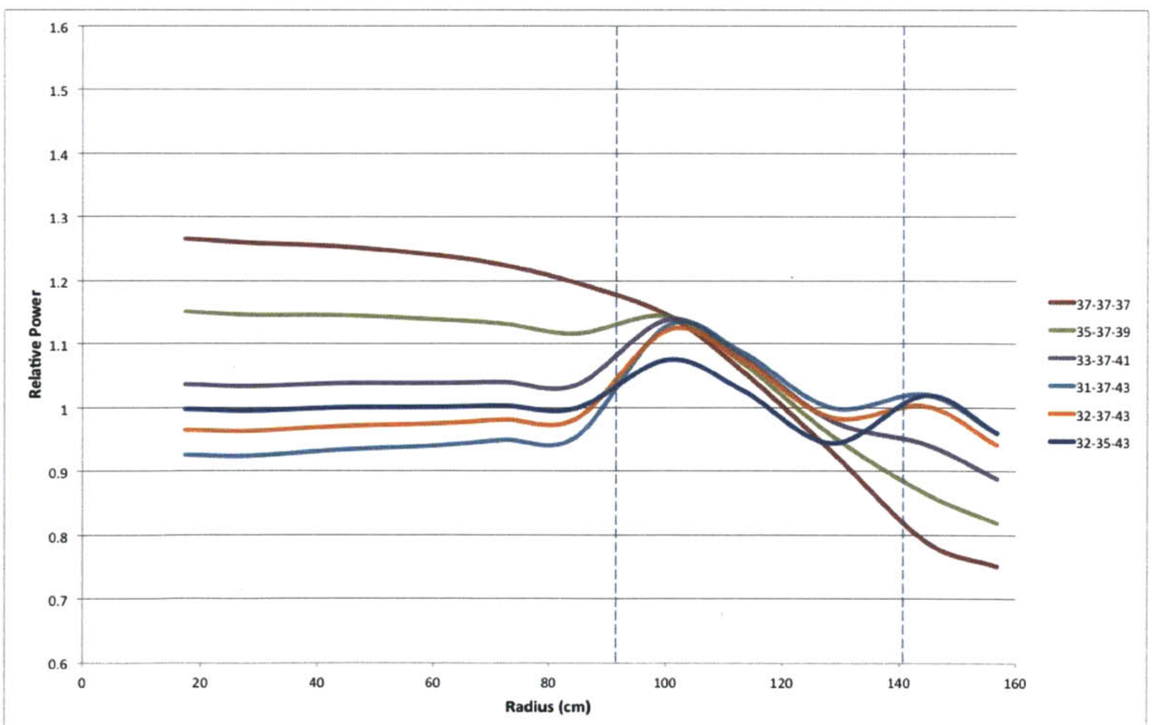

Figure 3: Normalized power along the radial axis at the end of the first cycle

Fuel	Maximum Normalized Power	Minimum Normalized Power	Relative Power Difference
37-37-37 (BOL)	1.59	0.65	0.94
35-37-39 (BOL)	1.40	0.67	0.73
33-37-41 (BOL)	1.21	0.75	0.46
31-37-43 (BOL)	1.17	0.84	0.33
32-37-43 (BOL)	1.16	0.89	0.27
37-37-37 (EOL)	1.27	0.75	0.52
35-37-39 (EOL)	1.15	0.77	0.38
33-37-41 (EOL)	1.14	0.84	0.30
31-37-43 (EOL)	1.13	0.91	0.22
32-37-43 (EOL)	1.12	0.94	0.18

Table 4: Comparison of specific power values

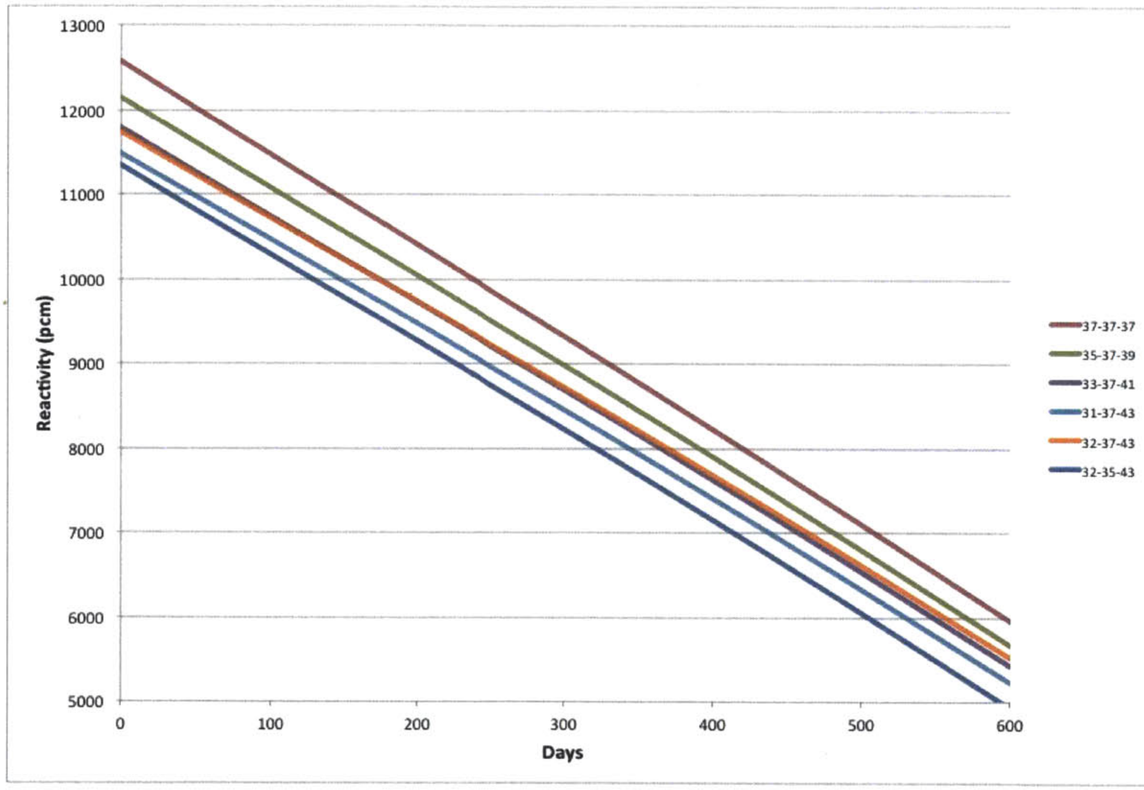

Figure 4: Comparison of reactivity over time for the initial cycle of the two fuel layouts

3.3 Heavy Metals

The quantity of heavy metals required for each core was of interest because of the cost of fuel for each core. A significant difference in the amount of heavy metals could outweigh the benefits of varying the fuel volume fraction. As shown in Table 5, the amount of heavy metals in the varied fuel is greater than the amount in the uniform fuel by only about 200 kg. There is a 1.2% increase in the mass of U-235 needed and 0.9% increase for total heavy metals mass, which should not significantly change the overall fuel cycle cost.

Fuel Type	Amount of Heavy Metal (kg)	Amount of U-235 (kg)
Uniform fuel volume fraction	29,631	1293.15
32%-37%-43% fuel volume fraction	29,898	1308.86

	Heavy Metal	U-235
Ratio of uniform to varied	1.009	1.012

Table 5: Comparison of fuel type and amount of heavy metal

4 Conclusions and Recommendations

This thesis addresses radial power flattening for a uranium startup sodium cooled fast reactor, with particular interest in how best to achieve this goal. Varying the fuel volume fraction was an effective way to flatten the radial power profile as shown by comparing the radial power profiles. The reactivity was slightly less throughout the first cycle of the varied fuel volume fraction cores, but not enough to affect the functioning of the reactor. The amount of heavy metals needed was higher by an insignificant amount and should not affect the fuel cycle cost. The best varied core had 32% fuel volume fraction in the center fuel region, 37% in the middle region, and 43% in the outer region, based on having one of the flattest radial power profiles and not having the lowest reactivity. The 32%-35%-43% also had a very flat radial power profile, but it had lost much more reactivity than the 32%-37%-42%. Further reducing the fuel volume fraction in the middle fuel section might produce minute changes in flattening, but would probably also further reduce the reactivity. For this reason, other ways to shape the power profile should also be explored.

In the future, further work can develop the annular pellet fuel for this reactor. Moderation could be added by putting MgO inside the pellet annuli to further shape local power. Staggered batch reloading can also be studied to further reduce the already low power peaking, reduce reload enrichment and BOC reactivity and extend burnup. Fei has already done this for the uniform core. Another aspect of interest is sodium void reactivity. Because power has been shifted to the periphery it is speculated that increased leakage will reduce void reactivity.

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5 Appendix A: Sample ERANOS code input

A disk containing all run cases has been filed with the thesis supervisor. This appendix contains the file used to run the calculations (run_calculation.data) and the cell descriptions for the homogenous core with varied fuel volume fraction (ecco_description.proc).

run_calculation.data

```
# include "BU_calculation.proc"
# include "flux_calculation.proc"
# include "output_ananlysis.proc"
# include "core_description.proc"
# include "ecco_description.proc"
# include "update_fuel.proc"
!*******************Fuel Volume Fractions and Enrichment***************************
->NA_IN_FRACTION  11.82 ; ! bond volume fraction (cannot be voided)
->FUEL_FRACTION   37.09 ; ! fuel volume fraction
->HT9_FRACTION    26.17 ; ! cladding+duct volume fraction
->NA_OUT_FRACTION 24.92 ; ! coolant volume fraction (can be voided)
! volume fractions are only used in homogeneous ecco cell calculation
->MEAN_ENRICHMENT 18.5 ;  ! enrichment of u2355
->FUEL_USED 'U235' ; ! 'UPU' or 'U235'
!*******************TO MODIFY***************************
->NG 33 ;
->TYPE_GEO '3D' ; ! for fluence calculation need to be 3D
->TRANSPORT 'YES' ;
->PTH 2.4E9 ;
->EXPENSION_CORE 'NO' ;
->ADJOINT 'NO' ;
->ZINT 1 150 40 80 ;
->MASS 37025 ; ! totale mass of HM in the REFERENCE core (used for BU calculation)
->PERT_NB 1 ; ! number of the perturbation, for the archives
->PERT_ITER 9 ; !nb of iteration of PASSE efpd before PERT calc
->DNAPERT 0.85 ; ! sodium density g/cm3
->TFUELPERT 1030 ; ! Tfuel celcius
!PERTURBATION_CALCULATION ; ! if we do this, then we can't do
                           ! the first ECCO_STD_CALCULATION
->T3M_EDITION 'YES' ;
->RAD_T_RZ_R 8.5 ;
->RAD_T_3D_XYMIN 30 30 ;
->RAD_T_3D_XYMAX 16 44 ;
->RAD_T_Z 151.0 ;
->AX_T_3D_XY 30 21 ;
->RAD_T_RZ_RMIN 0.0 ;
->RAD_T_RZ_RMAX 210.0 ;
->AX_T_Z_MIN 100.0 ;
->AX_T_Z_MAX 202.0 ;
!--------CYCLE 1----------------
->CYCLE 1 ;
->PASSE (100.0) ;
->ITER 6 ;
CORE_EVOLUTION_CALCULATION ;
```

```
! MATERIAL_RESULTS_EVOLUTION ;
* DAY ;
* MBUP ;
* 'Mean BU in MWd/HMKg' ;
* RHOV ;
* '----------------END OF CYCLE 1-----------------' ;
!
FUEL_UPDATE;
!
!---------------CYCLE 2---------------
->CYCLE 2;
->PASSE (100.0) ;
->ITER 6;
CORE_EVOLUTION_CALCULATION ;
! MATERIAL_RESULTS_EVOLUTION ;
* DAY ;
* MBUP ;
* 'Mean BU in MWd/HMKg' ;
* RHOV ;
* '----------------END OF CYCLE 2----------------'; !
FUEL_UPDATE;
!
!---------------CYCLE 3---------------
->CYCLE 3;
->PASSE (100.0) ;
->ITER 6;
CORE_EVOLUTION_CALCULATION ;
! MATERIAL_RESULTS_EVOLUTION ;
* DAY ;
* MBUP ;
* 'Mean BU in MWd/HMKg' ;
* RHOV ;
* '----------------END OF CYCLE 3----------------';
Fin;
```

ecco_description.proc (EXCERPTS)

```
 !--------fuel materials description------------
SI (CYCLE=1) ;
   ->FISSILE_MATERIAL_LIST
   SIMPLE_MATERIAL  'FUEL1'  FUEL
         WEIGHT_PERCENTAGE 10.97  ! Oxide Fuel
      ELEMENT CIP 88.135           ! CIP : masse percentage for each isotopes
      'Th232'    1.0E-10
      'Pa231'    1.0E-10
      'Pa233'    1.0E-10
      'U232'     1.0E-10
      'U233'     1.0E-10
      'U234'     1.0E-10
      'U236'     1.0E-10
      'U238'     (100-(E1))
      'U235'       (E1)
      ELEMENT CIP (1.0E-10)   ! Fuel contains no TRU Isotopes
      'Np237'    1.0E-10
      'Np238'    1.0E-10
```

```
     'Np239'    1.0E-10
     'Pu238'    1.0E-10
     'Pu239'    1.0E-10
     'Pu240'    1.0E-10
     'Pu241'    1.0E-10
     'Pu242'    1.0E-10
     'Am241'    1.0E-10
     'Am242g'   1.0E-10
     'Am242m'   1.0E-10
     'Am243'    1.0E-10
     'Cm242'    1.0E-10
     'Cm243'    1.0E-10
     'Cm244'    1.0E-10
     'Cm245'    1.0E-10
     'Cm246'    1.0E-10
     'Cm247'    1.0E-10
     'Cm248'    1.0E-10
     'Bk249'    1.0E-10
     'Cf249'    1.0E-10
     'Cf250'    1.0E-10
     'Cf251'    1.0E-10
     'Cf252'    1.0E-10
     'fpU234'   1.0E-10    ! fp stands for fission product stayed in the pin
     'fpU235'   1.0E-10    ! sfp are for fission products release to the plenum
     'fpU236'   1.0E-10
     'fpU238'   1.0E-10
     'fpNp237'  1.0E-10
     'fpPu238'  1.0E-10
     'fpPu239'  1.0E-10
     'fpPu240'  1.0E-10
     'fpPu241'  1.0E-10
     'fpPu242'  1.0E-10
     'fpAm241'  1.0E-10
     'fpAm242m' 1.0E-10
     'fpAm243'  1.0E-10
     'fpCm243'  1.0E-10
     'fpCm244'  1.0E-10
     'fpCm245'  1.0E-10
     ELEMENT CIP 11.865
     'O16' 100.00
      EXPANSION (EXP_FUEL)
! FUEL 2
   SIMPLE_MATERIAL  'FUEL2'  FUEL
        WEIGHT_PERCENTAGE 10.97  ! Oxide Fuel
     ELEMENT CIP 88.135           ! CIP : masse percentage for each isotopes
     'Th232'    1.0E-10
     'Pa231'    1.0E-10
     'Pa233'    1.0E-10
     'U232'     1.0E-10
     'U233'     1.0E-10
     'U234'     1.0E-10
     'U236'     1.0E-10
     'U238'     (100-(E1))
     'U235'       (E1)
```

```
      ELEMENT CIP (1.0E-10)    ! Fuel contains no TRU Isotopes
      'Np237'     1.0E-10
      'Np238'     1.0E-10
      'Np239'     1.0E-10
      'Pu238'     1.0E-10
      'Pu239'     1.0E-10
      'Pu240'     1.0E-10
      'Pu241'     1.0E-10
      'Pu242'     1.0E-10
      'Am241'     1.0E-10
      'Am242g'    1.0E-10
      'Am242m'    1.0E-10
      'Am243'     1.0E-10
      'Cm242'     1.0E-10
      'Cm243'     1.0E-10
      'Cm244'     1.0E-10
      'Cm245'     1.0E-10
      'Cm246'     1.0E-10
      'Cm247'     1.0E-10
      'Cm248'     1.0E-10
      'Bk249'     1.0E-10
      'Cf249'     1.0E-10
      'Cf250'     1.0E-10
      'Cf251'     1.0E-10
      'Cf252'     1.0E-10
      'fpU234'    1.0E-10    ! fp stands for fission product stayed in the pin
      'fpU235'    1.0E-10    ! sfp are for fission products release to the plenum
      'fpU236'    1.0E-10
      'fpU238'    1.0E-10
      'fpNp237'   1.0E-10
      'fpPu238'   1.0E-10
      'fpPu239'   1.0E-10
      'fpPu240'   1.0E-10
      'fpPu241'   1.0E-10
      'fpPu242'   1.0E-10
      'fpAm241'   1.0E-10
      'fpAm242m'  1.0E-10
      'fpAm243'   1.0E-10
      'fpCm243'   1.0E-10
      'fpCm244'   1.0E-10
      'fpCm245'   1.0E-10
      ELEMENT CIP 11.865
      'O16' 100.00
       EXPANSION (EXP_FUEL)
! FUEL 3
   SIMPLE_MATERIAL  'FUEL3'  FUEL
         WEIGHT_PERCENTAGE 10.97  ! Oxide Fuel
      ELEMENT CIP 88.135          ! CIP : masse percentage for each isotopes
      'Th232'     1.0E-10
      'Pa231'     1.0E-10
      'Pa233'     1.0E-10
      'U232'      1.0E-10
      'U233'      1.0E-10
      'U234'      1.0E-10
```

```
      'U236'     1.0E-10
      'U238'     (100-(E1))
      'U235'       (E1)
      ELEMENT CIP (1.0E-10)    ! Fuel contains no TRU Isotopes
      'Np237'    1.0E-10
      'Np238'    1.0E-10
      'Np239'    1.0E-10
      'Pu238'    1.0E-10
      'Pu239'    1.0E-10
      'Pu240'    1.0E-10
      'Pu241'    1.0E-10
      'Pu242'    1.0E-10
      'Am241'    1.0E-10
      'Am242g'   1.0E-10
      'Am242m'   1.0E-10
      'Am243'    1.0E-10
      'Cm242'    1.0E-10
      'Cm243'    1.0E-10
      'Cm244'    1.0E-10
      'Cm245'    1.0E-10
      'Cm246'    1.0E-10
      'Cm247'    1.0E-10
      'Cm248'    1.0E-10
      'Bk249'    1.0E-10
      'Cf249'    1.0E-10
      'Cf250'    1.0E-10
      'Cf251'    1.0E-10
      'Cf252'    1.0E-10
      'fpU234'   1.0E-10    ! fp stands for fission product stayed in the pin
      'fpU235'   1.0E-10    ! sfp are for fission products release to the plenum
      'fpU236'   1.0E-10
      'fpU238'   1.0E-10
      'fpNp237'  1.0E-10
      'fpPu238'  1.0E-10
      'fpPu239'  1.0E-10
      'fpPu240'  1.0E-10
      'fpPu241'  1.0E-10
      'fpPu242'  1.0E-10
      'fpAm241'  1.0E-10
      'fpAm242m' 1.0E-10
      'fpAm243'  1.0E-10
      'fpCm243'  1.0E-10
      'fpCm244'  1.0E-10
      'fpCm245'  1.0E-10
      ELEMENT CIP 11.865
      'O16' 100.00
       EXPANSION (EXP_FUEL)
;
FINSI;
SI (CYCLE>1) ;
  ->FISSILE_MATERIAL_LIST
   SIMPLE_MATERIAL  'FUEL2'  FUEL
         WEIGHT_PERCENTAGE 10.97  ! Oxide Fuel
      ELEMENT CIP 88.135          ! CIP : masse percentage for each isotopes
```

```
'Th232'    1.0E-10
'Pa231'    1.0E-10
'Pa233'    1.0E-10
'U232'     1.0E-10
'U233'     1.0E-10
'U234'     1.0E-10
'U236'     1.0E-10
'U238'     (100-(E1))
'U235'       (E1)
ELEMENT CIP (1.0E-10)   ! Fuel contains no TRU Isotopes
'Np237'    1.0E-10
'Np238'    1.0E-10
'Np239'    1.0E-10
'Pu238'    1.0E-10
'Pu239'    1.0E-10
'Pu240'    1.0E-10
'Pu241'    1.0E-10
'Pu242'    1.0E-10
'Am241'    1.0E-10
'Am242g'   1.0E-10
'Am242m'   1.0E-10
'Am243'    1.0E-10
'Cm242'    1.0E-10
'Cm243'    1.0E-10
'Cm244'    1.0E-10
'Cm245'    1.0E-10
'Cm246'    1.0E-10
'Cm247'    1.0E-10
'Cm248'    1.0E-10
'Bk249'    1.0E-10
'Cf249'    1.0E-10
'Cf250'    1.0E-10
'Cf251'    1.0E-10
'Cf252'    1.0E-10
'fpU234'   1.0E-10    ! fp stands for fission product stayed in the pin
'fpU235'   1.0E-10    ! sfp are for fission products release to the plenum
'fpU236'   1.0E-10
'fpU238'   1.0E-10
'fpNp237'  1.0E-10
'fpPu238'  1.0E-10
'fpPu239'  1.0E-10
'fpPu240'  1.0E-10
'fpPu241'  1.0E-10
'fpPu242'  1.0E-10
'fpAm241'  1.0E-10
'fpAm242m' 1.0E-10
'fpAm243'  1.0E-10
'fpCm243'  1.0E-10
'fpCm244'  1.0E-10
'fpCm245'  1.0E-10
ELEMENT CIP 11.865
'O16' 100.00
 EXPANSION (EXP_FUEL)
(FUEL1RELOAD) EXPANSION (EXP_FUEL)
```

```
   (FUEL3RELOAD) EXPANSION (EXP_FUEL)
   ;
FINSI ;
!---> list of materials :
->MEDIUM_LIST
! 3 fuel description for 3 zones in my core
! we describe the volume percentage in the fuel assembly
   MEDIUM 'FUEL1'
      'FUEL1'       32.0000
      'HELIUM'      11.4311
      'STEELHT9'    25.7316
      'SODIUMCR'    28.5440
   MEDIUM 'FUEL2'
      'FUEL2'       37.0000
      'HELIUM'      11.4311
      'STEELHT9'    25.7316
      'SODIUMCR'    28.5440
   MEDIUM 'FUEL3'
      'FUEL3'       43.0000
      'HELIUM'      11.4311
      'STEELHT9'    25.7316
      'SODIUMCR'    28.5440
! Lower Grid Plenium
   MEDIUM 'LRGP'
      'SODIUM'      37.9088
      'STEELHT9'    62.0912
! Lower Assembly Shielding
   MEDIUM 'LASH'
      'SODIUM'      (NA_OUT_FRACTION+5.2786)
      'STEELHT9'    (HT9_FRACTION)
      'B4C20'       (NA_IN_FRACTION+FUEL_FRACTION-5.2786)
! Upper Assembly Shielding
   MEDIUM 'UASH'
      'SODIUMCR'    (NA_OUT_FRACTION)  ! Top voided
      'SODIUM'     5.2786 ! (NA_IN_FRACTION)
      'STEELHT9'    (HT9_FRACTION)
      'B4C20'       (FUEL_FRACTION+NA_IN_FRACTION-5.2786)
! Lower Assembly Reflector
   MEDIUM 'LAREFL'
      'SODIUM'      (NA_OUT_FRACTION+5.2786)
      'STEELHT9'    (HT9_FRACTION)
      'AX_REFL'     (NA_IN_FRACTION+FUEL_FRACTION-5.2786)
! Upper Assembly Refelctor
   MEDIUM 'HAREFL'
      'SODIUMCR'    (NA_OUT_FRACTION)  ! Top voided
      'SODIUM'     5.2786 ! (NA_IN_FRACTION)
      'STEELHT9'    (HT9_FRACTION)
      'AX_REFL'     (FUEL_FRACTION+NA_IN_FRACTION-5.2786)
! Gas Plenium
   MEDIUM 'GPLN'
      'SODIUMCR'    (NA_OUT_FRACTION+5.2786)  ! Top voided
    !  'SODIUM'      (NA_IN_FRACTION)
      'STEELHT9'    (HT9_FRACTION)
      'HELIUM'      (FUEL_FRACTION+NA_IN_FRACTION-5.2786)
```

```
! control rods
   MEDIUM 'MROD'
   !    'RODB4C'      28.9996   ! 100% B10
   !    'RAD_REFL'     42.8 ! 28.9996
      'B4C20'        42.8 ! 28.9996   ! 20% B10
      'STEELHT9'     20.8 ! 32.7328
      'SODIUM'       36.4 ! 38.2676
! followers of the rods
   MEDIUM 'SUIV'
      'SODIUM'       50.4 !New VF Including Bond  ! Original VF = 28.9996%
      'STEELHT9'     20.8 ! 32.7328
      'SODIUM'       28.8 ! 38.2676
! Radial reflector
   MEDIUM 'REFL'
      'SODIUM'       17.762 ! 12.4977
      'RAD_REFL'     77.368
      'STEELHT9'     4.8701 ! 10.1343
! radial shielding
   MEDIUM 'SHIELD'
      'SODIUM'       30.2268
      'STEELHT9'     24.9926
      'B4C20'        44.7806
;
->CELLS_LIST
! cell liste
! these are defined at 20 celcius degrees, ECCO code will make them evoluate
! until the right temperature using the dilatations data we entered previously.
! case2 UC fuel pitch=1.9cm
   CELL 'CFUEL1'
      RANGEMENT_COMPOSITION  'FUEL1'
      GEOMETRY DATA
         HOMOGENEOUS
         REGION 1  'RFUEL1'   COMPOSITION 1  293.16
      END OF GEOMETRY DATA
   CELL 'CFUEL2'
      RANGEMENT_COMPOSITION  'FUEL2'
      GEOMETRY DATA
         HOMOGENeOUS
         REGION 1  'RFUEL2'   COMPOSITION 1  293.16
      END OF GEOMETRY DATA
   CELL 'CFUEL3'
      RANGEMENT_COMPOSITION  'FUEL3'
      GEOMETRY DATA
         HOMOGENeOUS
         REGION 1  'RFUEL3'   COMPOSITION 1  293.16
      END OF GEOMETRY DATA
   CELL 'CSUIV'
      RANGEMENT_COMPOSITION 'SUIV'
      GEOMETRY DATA
         HOMOGENEOUS
         REGION 1  'SUIV'  COMPOSITION 1  293.16
      END OF GEOMETRY DATA
   CELL 'CLRGP'
      RANGEMENT_COMPOSITION 'LRGP'
```

```
    GEOMETRY DATA
      HOMOGENEOUS
      REGION 1  'LRGP'  COMPOSITION 1  293.16
    END OF GEOMETRY DATA
  CELL 'CROD'
    RANGEMENT_COMPOSITION 'MROD'
    GEOMETRY DATA
      HOMOGENEOUS
      REGION 1  'MROD'  COMPOSITION 1  293.16
    END OF GEOMETRY DATA
  CELL 'CLASH'
    RANGEMENT_COMPOSITION 'LASH'
    GEOMETRY DATA
      HOMOGENEOUS
      REGION 1  'LASH'  COMPOSITION 1  293.16
    END OF GEOMETRY DATA
  CELL 'CLAREFL'
    RANGEMENT_COMPOSITION 'LAREFL'
    GEOMETRY DATA
      HOMOGENEOUS
      REGION 1  'LAREFL'  COMPOSITION 1  293.16
    END OF GEOMETRY DATA
  CELL 'CHAREFL'
    RANGEMENT_COMPOSITION 'HAREFL'
    GEOMETRY DATA
      HOMOGENEOUS
      REGION 1  'HAREFL'  COMPOSITION 1  293.16
    END OF GEOMETRY DATA
  CELL 'CGPLN'
    RANGEMENT_COMPOSITION 'GPLN'
    GEOMETRY DATA
      HOMOGENEOUS
      REGION 1  'GPLN'  COMPOSITION 1  293.16
    END OF GEOMETRY DATA
  CELL 'CUASH'
    RANGEMENT_COMPOSITION 'UASH'
    GEOMETRY DATA
      HOMOGENEOUS
      REGION 1  'UASH'  COMPOSITION 1  293.16
    END OF GEOMETRY DATA
  CELL 'CREFL'
    RANGEMENT_COMPOSITION 'REFL'
    GEOMETRY DATA
      HOMOGENEOUS
      REGION 1  'REFL'  COMPOSITION 1  293.16
    END OF GEOMETRY DATA
  CELL 'CSHIELD'
    RANGEMENT_COMPOSITION 'SHIELD'
    GEOMETRY DATA
      HOMOGENEOUS
      REGION 1  'SHIELD'  COMPOSITION 1  293.16
    END OF GEOMETRY DATA   ;
!
!
```

```
! steps of calculation used for each cells
! this is where we can decide to define some heterogeous/homogeneous cells
->STEP_1_FUEL
     GEOMETRY HOMOGENEOUS
     ELEMENTS ALL
     GROUP STRUCTURE OTHER 33
     INPUT LIBRARY 'BBL_33G'
     FLUX SOLUTION FM P1 CONSISTENT ORDER 1
     LEAKAGE NLFACT CELL BENOIST FLUXWT MEAN
     BSEARCH 1.0
     SELF SHIELDING NODBBSH ;
->STEP_2_FUEL
     GEOMETRY HOMOGENEOUS
     GROUP STRUCTURE FINE
     INPUT LIBRARY 'BBL_1968G'
     ELEMENTS 24
        U235  U238  Pu239 Pu240 Pu241 Pu242 Am241 Fe57  Fe58
        Fe54  Fe56  Cr50  Cr52  Cr53  Cr54  Ni58  Ni60  Ni61
        Ni62  Ni64  O16   Na23  Np237 Cm245
     FLUX SOLUTION FM P1 CONSISTENT ORDER 1
     LEAKAGE NLFACT CELL BENOIST FLUXWT MEAN
     BFROM 1
     SELF SHIELDING NODBBSH
     CONDENSE 33
           1   82  142  202  262  322  382  442  502  564
         624  686  746  808  868  928  988 1048 1108 1168
        1228 1288 1336 1422 1480 1516 1579 1648 1708 1768
        1837 1919 1952 ;
->STEP_STRU
     GEOMETRY HOMOGENEOUS
     GROUP STRUCTURE OTHER 33
     FLUX SOLUTION FM P1 CONSISTENT ORDER 1
     INPUT LIBRARY 'BBL_33G'
     LEAKAGE NLFACT CELL BENOIST FLUXWT MEAN
     SELF SHIELDING DBBSH ;
->CELLS_CALCULATION
   CELL 'CFUEL1'
      EDITION MINI
      'INTERNAL FUEL ASSEMBLY'
      TEMPERATURE   1 (TFUEL+273)
      STEPS 2
         STEP (STEP_1_FUEL)
         STEP (STEP_2_FUEL)
              OUTPUT LIBRARY 'FUEL1' CROSS SECTIONS FLUXES
      ENDSTEPS
!
   CELL 'CFUEL2'
      EDITION MINI
      'MEDIUM FUEL ASSEMBLY'
      TEMPERATURE   1 (TFUEL+273)
      STEPS 2
         STEP (STEP_1_FUEL)
         STEP (STEP_2_FUEL)
              OUTPUT LIBRARY 'FUEL2' CROSS SECTIONS FLUXES
```

```
      ENDSTEPS
!
   CELL 'CFUEL3'
      EDITION MINI
      'EXTERNAL FUEL ASSEMBLY'
      TEMPERATURE   1 (TFUEL+273)
      STEPS 2
         STEP (STEP_1_FUEL)
         STEP (STEP_2_FUEL)
              OUTPUT LIBRARY 'FUEL3' CROSS SECTIONS FLUXES
      ENDSTEPS
```

6 Appendix B: Python code

Selected sections of Python code used to extract power, reactivity, and flux:

```
import sys
fileName = sys.argv[1]
file1=open(fileName,'r')
file2=open(fileName+'-flux0.txt','w')
file3=open(fileName+'-powerTable0.txt','w')
timeStep=0        # extract power tables for certain time step. For BOL this number is 0
numSA=60 # number of assemblies
lines=file1.readlines()

data=[]

for i,line in enumerate(lines):
   if line[0:12]=='1TYPE  AFUEL':
       for j in range(10000):
           line2=lines[i+j+1]
           len2=len(line2)
           if len2>=6:
               if line2[6]=='3' or line2[6]=='4':
                   file2.write(line2[5:14]+line2[17:29]+line2[31:43]+'\n')
                   data.append(line2)
               elif line2[7:12]=='TOTAL':
                   break
avePower=[]
pTot=0.0
average=2400*1.0E6/360
for i in range((timeStep-1)*2*numSA,(timeStep-1)*2*numSA+60):
   line1=data[i]
   avePower.append(float(line1[17:29]))
   pTot=pTot+float(line1[17:29])
   aveP=float(line1[17:29])/average
   file3.write(line1[5:14]+str(aveP)+'\n')
file1.close()
file2.close()
file3.close()

# This python script is used to extract datas from 3 batch oxide fuel.
# inputs needed from math import ceil
```

```
import sys
timeStep=142.0
# time step for eranos depletion calculation.
(days) it=[5,5,5,5,5,5,5,5]
# number of depletion calculation steps in cycle 1,2,3,...
fileName=sys.argv[1]
# name of the file to be extracted
# Extract data
rhoLine=[]
file1=open(fileName,'r')
lines=file1.readlines()
k=0 for i,line in enumerate(lines):
   if line[2:21]=='Mean BU in MWd/HMKg':
     len1=lines[i+2]
     if len1[1:7]=='->RHOV':
        rhoLine.append(len1)
       numberRov=ceil((it[k]+1)/5.0)
       for j in range(int(numberRov)-1):
         rhoLine[k]=rhoLine[k]+lines[i+3+j]
       k=k+1
   else:
        continue
file1.close()
file2=open(fileName+'-reactivity.txt','w')
numCycle=k
day=0.0
tempR=[]
for k in range(numCycle):    tempR.append(rhoLine[k].split())
  l1=len(tempR[k])
  day=day-timeStep
  for i in range(l1-1):
     day=day+timeStep
     file2.write(str(day)+'  '+tempR[k][i+1]+'\n')
file2.close()
```

7 Appendix C: Assembly Radial Power Maps

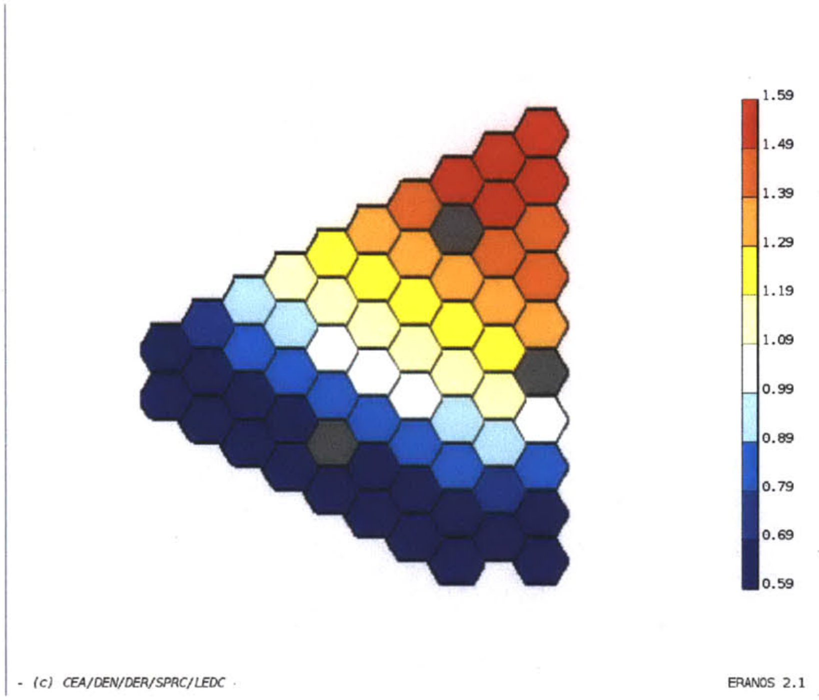

Figure 5: Radial Power for the uniform fuel volume fraction core at BOC

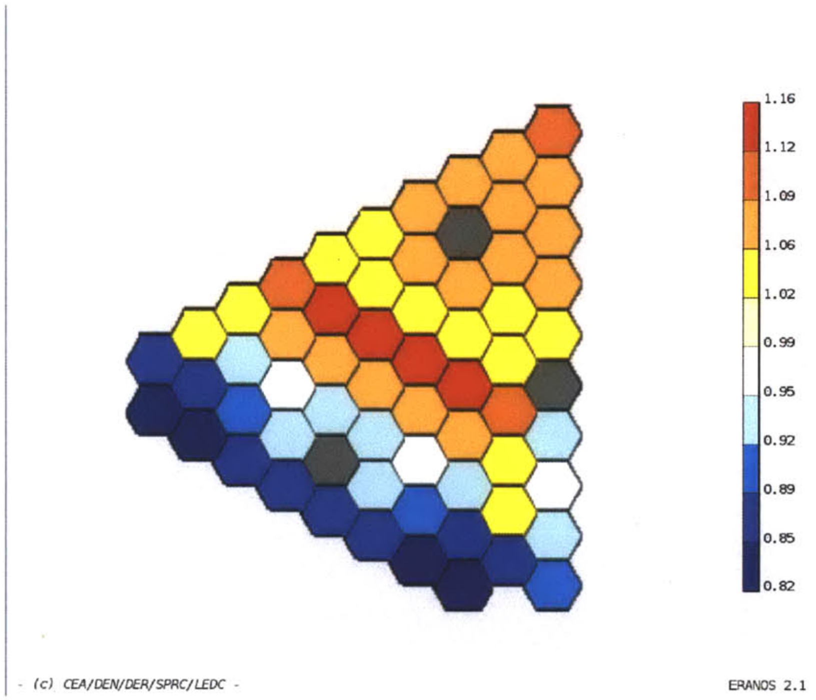

Figure 6: Radial Power for the varied fuel volume fraction core at BOC

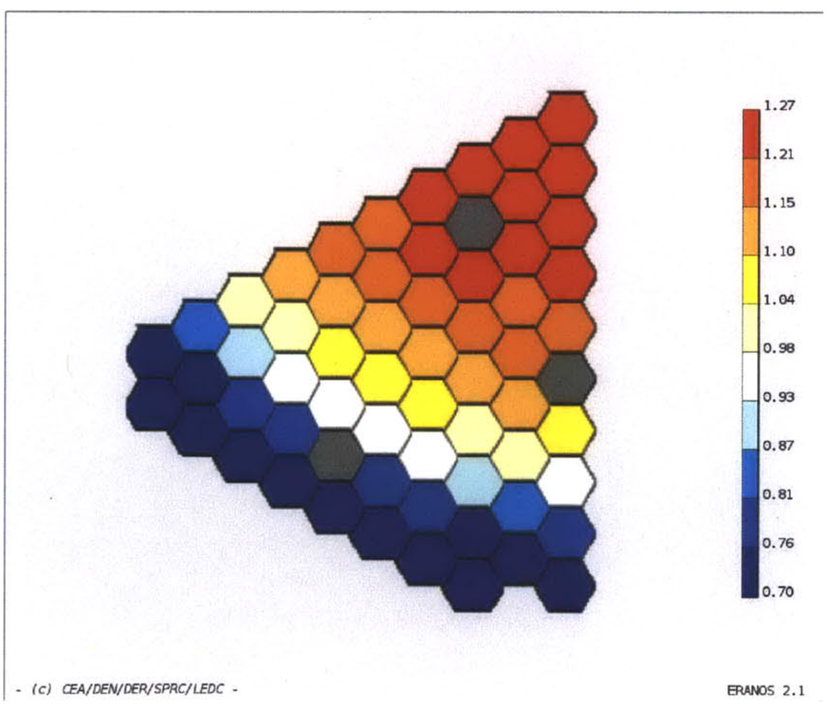

Figure 7: Radial Power for the uniform fuel volume fraction core at EOC

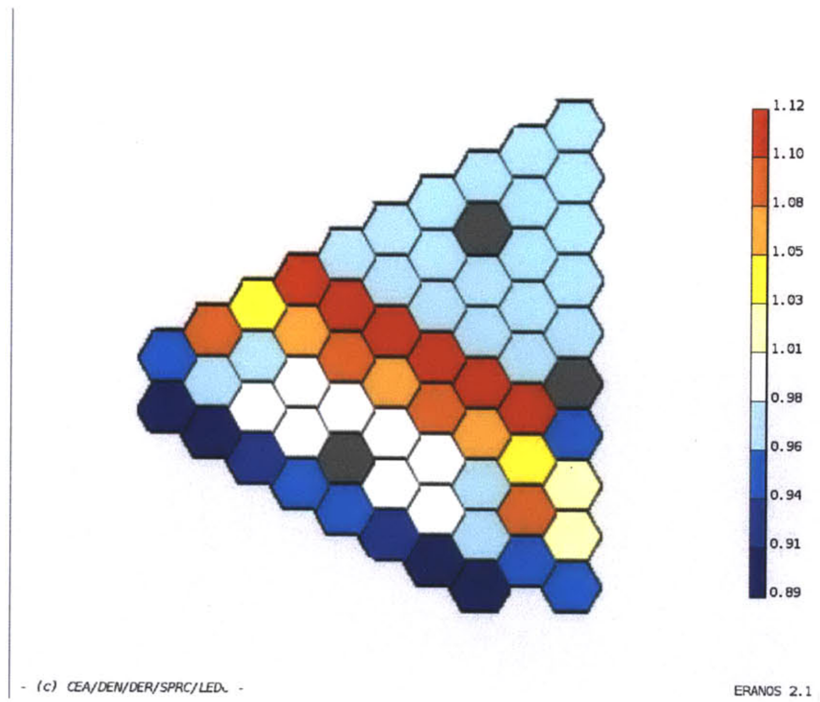

Figure 8: Radial Power for the varied fuel volume fraction core at EOC