Specifications of the Calculation Benchmarks on Low Void SFR Burner and MOSART Cores

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Introduction

- In the framework of the activities of the NEA Expert Group on Improvement of Integral Experiments Data for Minor Actinide Management (EGIEMAM-II), two U&S calculation benchmarks have been launched.
- The 1200 MWth low void Sodium Fast Reactor (SFR) MA burner* and the 2400 MWth Molten Salt Advanced Reactor Transmuter (MOSART) critical reactor concept with fertile-free fuel developed at NRC-KI** have been considered.
- The aim of the benchmarks was evaluating the performance of different computational codes, nuclear data libraries, and calculation procedures to analyze the impact of the MAs nuclear data uncertainties on selected design parameters of the two systems.

*F. Gabrielli, A. Rineiski, B. Vezzoni, W. Maschek, C. Fazio, M. Salvatores, ASTRID-like Fast Reactor Cores for Burning Plutonium and Minor Actinides, Energy Procedia 71 (2015) 130 – 139.
**V. Ignatiev et al., Integrated Study of Molten Na,Li,Be/F Salts for LWR Waste Burning in Accelerator Driven and Critical Systems, GLOBAL 2005, Tsukuba, Japan, Oct 9-13 (2005).

The Low Void SFR Burner Core Model

Benchmark Description

- **Aim: Evaluation of the impact of the uncertainties of MAs, U, and Na nuclear data on selected design parameters of a low void SFR MA burner model**

  - criticality level;
  - kinetics parameters (β_{eff} and Λ);
  - Doppler constant;
  - Sodium void effect.

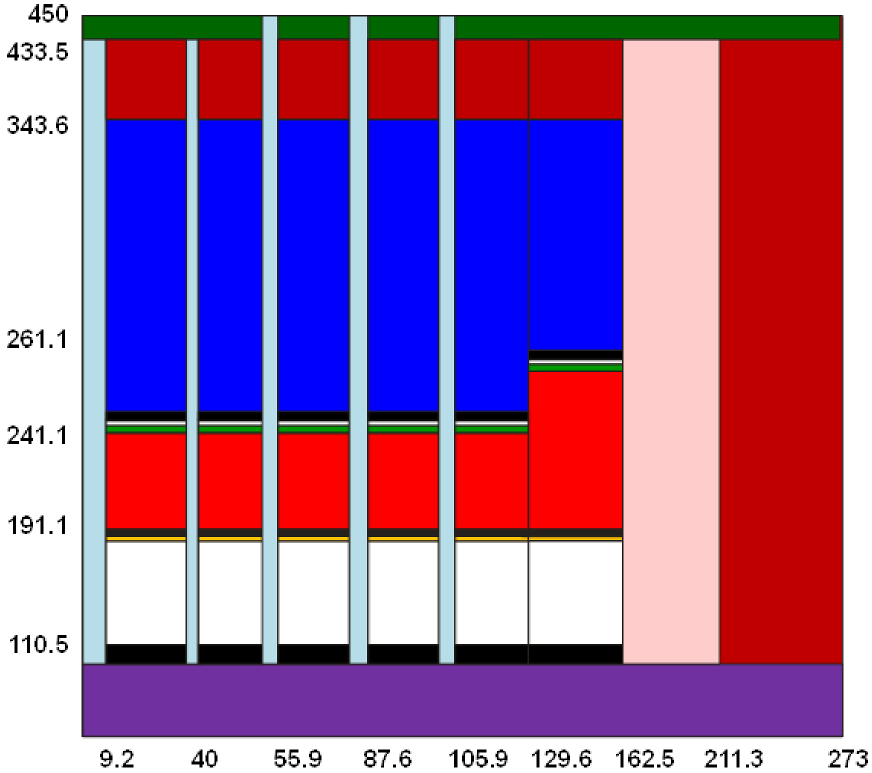
- Studies have been performed in the past providing a preliminary evaluation of the impact of nuclear data uncertainties on the design parameters of SFRs*.

- **A further step is done here by considering a low coolant void SFR burner with low CR (~0.5-0.6) loaded with a significant amount of MAs (MA:Pu=1:2).**

- The SFR MAs burner model is based on the 1500 MWth French ASTRID CFV-V1 (Coeur à Faible effet de Vide, i.e. low sodium void core) concept developed by CEA with support of AREVA and EdF.

*G. Palmiotti, M. Salvatores, G. Aliberti, et al., A Global Approach to the Physics Validation of Simulation Codes for Future Nuclear Systems, PHYSOR 2008, Interlaken, Switzerland, September 14-19, 2009.

2D (RZ) Geometrical Arrangement



#SAs	Type of SA	R_{ext} (cm)
1	Dummy	9.2
15	Inner core	36.7
3	Dummy	40.0
15	Inner core	53.5
3	Control Rods	55.9
48	Inner core	84.7
6	Safety Rods	87.6
33	Inner core	102.3
9	Control Rods	105.9
66	Inner core	129.6
114	Outer core	162.5
216	Reflector	211.3
354	Radial Shielding	273.0

F. Gabrielli, A. Rineiski, B. Vezzoni, W. Maschek, C. Fazio, M. Salvatores, ASTRID-like Fast Reactor Cores for Burning Plutonium and Minor Actinides, Energy Procedia 71 (2015) 130 – 139.

Characteristics of the Low Void SFR Burner

Parameter	Value
Inner/Outer SAs[1]	177/114
Reflector SAs[1]	216
Radial shielding SAs[1]	334
Dummy SAs[1]	4
Control/Safety SAs[1]	12/6
Inner/Outer core Fuel material[2]	$(U\text{-}TRU)O_2$
Power (MWth) [2]	1200
MA:Pu ratio[2]	**1:2**
Inner/Outer Pu enr. (wt.%) [2]	22.5/24.5
Blanket material[2]	(Depleted U-MA)
MA enrich in the LB (wt.%)[2]	10

[1]MS. Chenaud, N. Devictor, G. Mignot, et al., Status of the ASTRID core at the end of the pre-conceptual design phase 1, Proc. of Int. Conf. on Fast Reactors and related Fuel Cycles: Safe Technologies and Sustainable Scenarios, Paris, France, 4-7 March; 2013.

[2]F. Gabrielli, A. Rineiski, B. Vezzoni, W. Maschek, C. Fazio, M. Salvatores, ASTRID-like Fast Reactor Cores for Burning Plutonium and Minor Actinides, Energy Procedia 71 (2015) 130 – 139.

Geometrical Data

Parameter	Value
SA pitch (mm)[1]	175.0
Sodium gap width inter assembly (mm)[2]	4.5
Wrapper tube outer flat-to-flat width (mm)[2]	171.5
Wrapper tube thickness (mm)[2]	4.5
Wrapper tube material[3]	EM10
Wire wrap spacer diameter (mm) [2]	1.0
Wrapper tube material[3]	EM10
Number of fuel pins[1]	217
Fuel pellet diameter (mm)[1]	8.45
Inner clad diameter (mm) (from ESFR)	8.72
Outer clad diameter (mm) (from ESFR)	9.72
Cladding material[3]	15-15Ti mod
Fuel average density(porosity and central hole considered)[2]	89% TD

[1]MS. Chenaud, et al., Status of the ASTRID core at the end of the pre-conceptual design phase 1, Proc. of Int. Conf. on Fast Reactors and related Fuel Cycles: Safe Technologies and Sustainable Scenarios, Paris, France, 4-7 March; 2013.
[2]Fiorini, G.L. The Collaborative Project on European Sodium Fast Reactor (CP ESFR). In Proceedings of the Int. Conf. on Euratom Research and Training in Reactor Systems (FISA2009), Prague, Czech Republic, 22–24 June 2009.
[3]IAEA, Structural materials for liquid metal cooled fast reactor fuel assemblies - Operational behavior, Vienna, 2012, IAEA nuclear energy series, ISSN 1995–7807, no. NF-T-4.3.
Note: same ratio between the inner clad diameter and the outer fuel pin diameter as in ESFR

Fuel Vector

- Mixed U-TRU oxide (MOX) fuel with MA:PU=1:2.
- The Pu and MA isotopic vectors corresponding to those in MOX SNF reprocessed after 30 years after irradiation in a PWR, i.e. with a burn-up of about 45 MWd/kg (used in the past for EFIT design studies)*.

Isotope	wt.%
PU238	3.737006
PU239	46.4457
PU240	34.1214
PU241	3.844722
PU242	11.85002
PU244	1.154E-3

Isotope	wt.%
NP237	3.8844
AM241	75.51
AM242M	0.2536
AM243	16.054
CM243	0.0662
CM244	3.0014
CM245	1.13922
CM246	0.0885
CM247	1.67630E-3
CM248	1.0089E-4

*C. Artioli, et al., Minor actinide transmutation in ADS: the EFIT core design, Proc. Int. Conf. PHYSOR 2008, Interlaken, Switzerland, September 14-19; 2008.

Expected Results (1/2)

1. **Criticality level**

2. **Kinetics parameters:**
   2.1 Effective delayed neutron fraction (β_{eff});
   2.2 Mean Neutron generation time (Λ).

3. **Doppler constant**
   3.1 For the whole core: it is calculated for **ΔT** =1500 K-2500 K and **ΔT** =900 K-1500 K for fuel and fertile, respectively.
   3.2 Region-wise (fuel and blanket) Doppler effect can be optionally evaluated.

4. **Spatial distribution of the sodium void reactivity effect**: evaluation of the reactivity effect induced by removing the coolant only inside the wrapper.
   4.1 **Sodium void reactivity effect in the core region**: the Na density has to be reduced by 82.4% in the inner (or outer) fuel region and in the above Block, the Gas Plenum, and the upper Plug regions.
   4.2 **Extended (full core+Na plenum) sodium void reactivity effect**: the Na density has to be reduced by 82.4% in the inner and outer fuel regions and in the above Block, the Gas Plenum, and the upper Plug regions and by 94.4% in the Na plenum region.

Expected Results (2/2)

5. **Integrated (with or w/o group splitting) uncertainty results should be given**
   - Elastic, inelastic, nu-bar, capture and fission cross-sections for fuel isotopes: ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{241}Am, ^{242m}Am, ^{243}Am, ^{237}Np, ^{243}Cm, ^{244}Cm, and ^{245}Cm.
   - Elastic and inelastic cross-sections for ^{23}Na and ^{56}Fe.

- Each parameter is proposed to be computed by employing one or several nuclear data libraries.

- **Uncertainty results may be evaluated by each participant by employing any calculation procedure, i.e. perturbation/sensitivity method or direct calculation (i. e. numerical differentiation).**

- **Moreover, as far as possible participants should use actual covariance data (e.g. documented by WPEC Subgroup 39).**

Available List of Sensitivity Coefficients

- **33 energy-group sets of sensitivity coefficients are available (JEFF3.1+COMMARA2.0 employed).**

Isotopes	Neutron Cross-section
^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{241}Am, ^{242m}Am, ^{243}Am, ^{237}Np, ^{243}Cm, ^{244}Cm, and ^{245}Cm	Elastic, inelastic, nu-bar, capture, and fission
^{23}Na and ^{56}Fe	Elastic, inelastic, and capture

- **Sets of sensitivity coefficients on:**
  - Criticality level
  - Fuel Doppler in the core
  - Fuel Doppler in the Blanket
  - Fuel Doppler in the Core+Blanket
  - Inner Core void
  - Outer Core void
  - Inner+Outer core void
  - Full core+Na plenum void

The MOSART Core Model

Aim of the benchmark

- **Evaluation of the impact of the uncertainties of MAs, U, and Na nuclear data on selected design parameters of the 2400 MWth (1100 MWe) Molten Salt Advanced Reactor Transmuter (MOSART) critical reactor concept with fertile-free fuel developed at NRC-KI (Kurchatov Institute", Russia)*.**

  - criticality level;
  - kinetics parameters (β_{eff} and Λ);
  - Temperature reactivity effects.

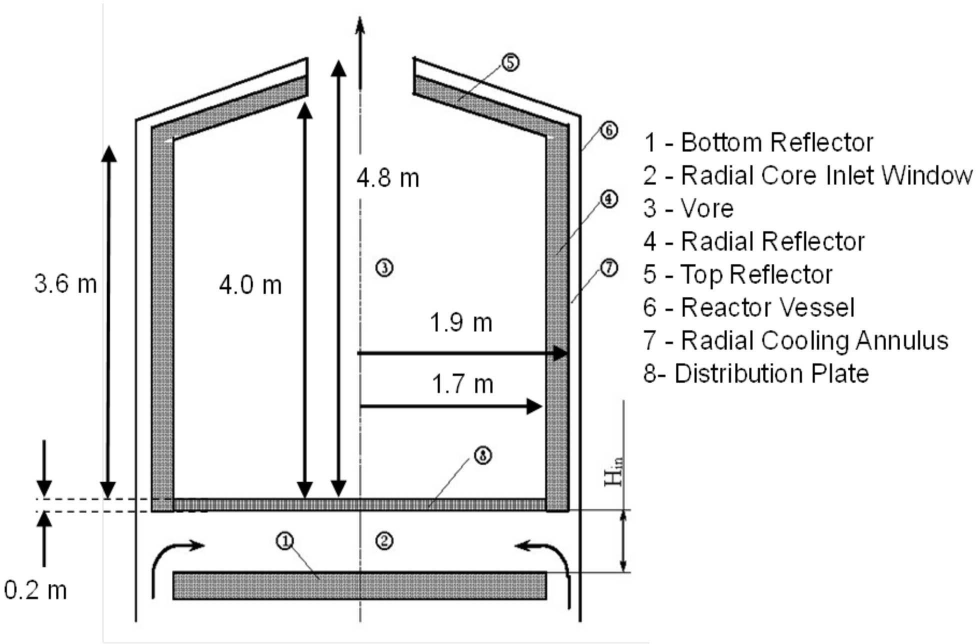
- The systems has been considered in the framework of the Coordinated Research Project (CRP) on "Studies of Advanced Reactor Technology Options for Effective Incineration of Radioactive Waste" (2003-2007) (16 institutions from 14 member states and 3 international organizations)**

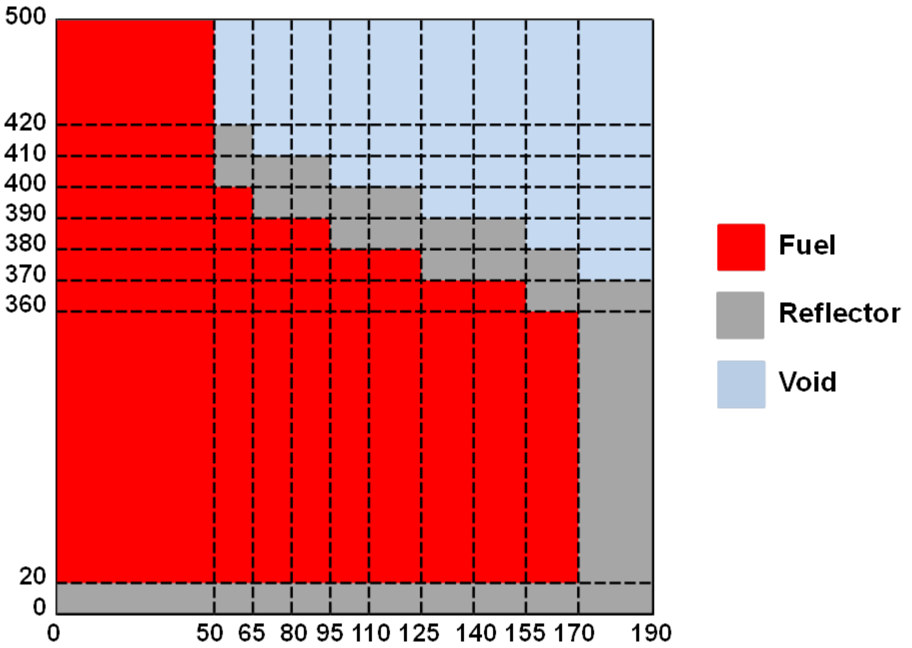
- The benchmark aims providing additional results for two MOSART TRU burner models loaded with **5%, 10%, and 15 % of MA content in the TRU vector in the fuel salt** and **two different reflectors (graphite** and the **HN80MTY Ni-based alloy).**

- **EFIT TRU vector is considered instead of the originally proposed feed material.**

*V. Ignatiev et al., Integrated Study of Molten Na,Li,Be/F Salts for LWR Waste Burning in Accelerator Driven and Critical Systems, GLOBAL 2005, Tsukuba, Japan, Oct 9-13 (2005).
**Advanced Reactor Technology Options for Utilization and Transmutation of Actinides in Spent Nuclear Fuel, IAEA, VIENNA, 2009, IAEA-TECDOC-1626.

2D (RZ) Geometrical model





*Advanced Reactor Technology Options for Utilization and Transmutation of Actinides in Spent Nuclear Fuel, IAEA, VIENNA, 2009, IAEA-TECDOC-1626.

Characteristics of the MOSART model

Power [MWth]/[MWe]	2400/1100	
Fuel Solvent System	^{58}NaF-^{15}LiF-$^{27}BeF_2$	
Fuel (Pu and MA)[1]	MOX spent fuel (45 MWd/kg reprocessed after 30 ys)	
Max. Solubility Limit (mole %)	~ 2	
Core Volume [m^3]	~ 32	
Fuel Salt Average T [K]	900	
Reflector Operative T [K]	Graphite	950
	HN80MTY Ni-alloy[2]	1073
Fuel Salt Flow Rate [kg/s]	10000	
Fuel Salt Inlet/Outlet T [K]	873/983	
Core Circulation Time, s	3.94	
Out Core Circulation Time, s	6.99	

[1]C. Artioli, et al., Minor actinide transmutation in ADS: the EFIT core design, Proc. Int. Conf. PHYSOR 2008, Interlaken, Switzerland, September 14-19; 2008.
[2]V. Fedulov, V.Ignatiev, A. Surenkov, e.a., “Development of Materials as Applied to Molten Salt Reactor: Problems and Ways of Solution”, IAE–5678, Moscow, 1993.

Fuel Vector

- In the original MOSART model, 10% MAs are loaded (UOX spent fuel of a commercial PWR (60 GWd/tU – 4.9% ^{235}U/U, after 1 year cooling).

- In the benchmark the TRU composition of MOX SNF reprocessed after 30 years after irradiation in a PWR (45 MWd/kg) is proposed*.

Isotope	MOSART (IAEA, wt%)	EFIT (wt.%)
PU238	3.51	3.737006
PU239	48.77	46.4457
PU240	23.71	34.1214
PU241	**15.14**	**3.844722**
PU242	**8.87**	**11.85002**
PU244	**3.51**	**1.154E-3**

Isotope	MOSART (IAEA, wt%)	EFIT (wt.%)
NP237	62.239	3.8844
AM241	**5.41538**	**75.51**
AM242M	0.11020	0.253684
AM243	**23.18856**	**16.0544**
CM242	0.00058	
CM243	0.06645	0.0662083
CM244	7.84877	3.001467
CM245	1.04101	1.13922
CM246	0.08878	8.852421E-2
CM247	0.00145	1.67630E-3
CM248		1.0089E-4

Fuel Salt Composition

Isotope	MA:Pu=1:20	MA:Pu=1:10	MA:Pu=1:5.6666	Original
Li6	2.14604E-07	2.14432E-07	2.14159E-07	2.15000E-07
Li7	4.29308E-03	4.28963E-03	4.28419E-03	4.30100E-03
Na23	1.77073E-02	1.76931E-02	1.76706E-02	1.77400E-02
Be9	8.01521E-03	8.00878E-03	7.99861E-03	8.03000E-03
F19	3.84890E-02	3.84581E-02	3.84093E-02	3.85600E-02
Np237	5.13872E-07	1.26904E-06	2.37901E-06	8.67550E-06
Pu238	9.35357E-06	1.07014E-05	1.29149E-05	4.28920E-06
Pu239	1.15764E-04	1.32445E-04	1.59841E-04	5.93330E-05
Pu240	8.46914E-05	9.68950E-05	1.16938E-04	2.87320E-05
Pu241	9.50313E-06	1.08725E-05	1.31214E-05	1.82710E-05
Pu242	2.91689E-05	3.33719E-05	4.02749E-05	1.06520E-05
Pu244	2.81723E-09	3.22318E-09	3.88989E-09	
Am241	9.82319E-06	2.42590E-05	4.54771E-05	7.42300E-07
Am242M	3.28658E-08	8.11642E-08	1.52154E-07	1.50430E-08
Am243	2.07131E-06	5.11524E-06	9.58928E-06	3.15230E-06
Cm243	8.54209E-09	2.10953E-08	3.95462E-08	9.03330E-09
Cm244	3.85656E-07	9.52402E-07	1.78542E-06	1.06260E-06
Cm245	1.45778E-07	3.60009E-07	6.74891E-07	1.40360E-07
Cm246	1.12817E-08	2.78610E-08	5.22296E-08	1.19210E-08
Cm247	2.12764E-10	5.25435E-10	9.85007E-10	1.93760E-10
Cm248	1.27537E-11	3.14962E-11	5.90443E-11	
Sum	0.0687663	0.0687662	0.0687662	0.0687662

Reflector

- By employing each feed fuel material, two MOSART models with different reflectors are proposed for benchmark calculations:

- **graphite** according to the study performed in the IAEA CRP*

- one more recent model, which does employ the **HN80MTY Ni-based alloy** which is the recommended material according to dedicated material compatibility tests**,***.

- The **HN80MTY (EK-50) alloy** developed in 1984 at NRC-KI jointly with the Central Research Institute of Ferrous Metallurgy contains such admixtures as Ti (0.98%) and Al (1.12%), giving the alloy high resistance against the tellurium intergranular corrosion and against the radiation destruction.

- Compositions of graphite and HN80MTY Ni-based alloy are provided in the benchmark.

*Advanced Reactor Technology Options for Utilization and Transmutation of Actinides in Spent Nuclear Fuel, IAEA, VIENNA, 2009, IAEA-TECDOC-1626.
V. Ignatiev, et al., Molten salt actinide recycler and transforming system without and with Th-U support: Fuel cycle flexibility and key material properties, *Annals of Nucl. En.* **64 (2014) 408-420.
***V. Fedulov, V.Ignatiev, A. Surenkov, e.a., “Development of Materials as Applied to Molten Salt Reactor: Problems and Ways of Solution”, IAE–5678, Moscow, 1993.

Expected Results (1/2)

1. **Criticality level**

2. **Kinetics parameters:**
   2.1 Effective delayed neutron fraction (β_{eff}).
   2.2 Effective delayed neutron fraction for each family k ($\beta_{eff,k}$).
   2.3 Mean Neutron generation time (Λ).

3. **Temperature reactivity effects:** α-coefficients (pcm/K) have to be computed as ratio of the reactivity variation to the corresponding temperature variation
   3.1 **α-total (pcm/K)**: obtained by heating up both the fuel and the reflector by 600 K.
   3.2 **α-core (pcm/K)**: obtained by heating up the fuel by 600 K.
   - The density of Fuel, graphite, and HN80MTY is assumed to be temperature-independent.

Expected Results (2/2)

4. **Integrated (with or w/o group splitting) uncertainty results should be given**
   - Elastic, inelastic, nu-bar, capture and fission cross-sections for fuel isotopes: ^{239}Pu, ^{241}Am, ^{242m}Am, ^{243}Am, ^{237}Np, ^{243}Cm, ^{244}Cm, and ^{245}Cm.
   - Elastic and inelastic neutron cross-sections for ^{23}Na.
   - Elastic neutron cross-sections for ^{7}Li and ^{19}F.
   - Elastic and (n,2n) neutron cross-section for ^{9}Be.

- Each parameter is proposed to be computed by employing one or several nuclear data libraries.

- **Uncertainty results may be evaluated by each participant by employing any calculation procedure, i.e. perturbation/sensitivity method or direct calculation (i. e. numerical differentiation).**

- **Moreover, as far as possible participants should use actual covariance data (e.g. documented by WPEC Subgroup 39).**

Available List of Sensitivity Coefficients

- **33 energy-group sets of sensitivity coefficients are available (JEFF3.1+COMMARA2.0 employed).**

Isotopes	Neutron Cross-section
^{239}Pu, ^{241}Am, ^{242m}Am, ^{243}Am, ^{237}Np, ^{243}Cm, ^{244}Cm, and ^{245}Cm	Elastic, inelastic, nu-bar, capture, (n,xn), and fission
^{23}Na, ^{7}Li, ^{19}F, and ^{9}Be.	Elastic, inelastic, captur, and (n,xn).

- **For each of the 6 models (3 MA:Pu ratios + 2 different reflectors), sensitivity coefficients on:**
  - Criticality level
  - Fuel temperature reactivity effect
  - Fuel+reflector temperature reactivity effect