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MINISTERO DELLO SVILUPPO ECONOMICO

Ricerca di Sistema elettrico

Determinazione dei parametri di sicurezza del core e dell'andamento del burnup di un reattore veloce refrigerato a metallo liquido

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CIRTEN - CONSORZIO INTERUNIVERSITARIO PER LA RICERCA TECNOLOGICA NUCLEARE

Report RdS/2013/038

DETERMINAZIONE DEI PARAMETRI DI SICUREZZA DEL CORE E DELL'ANDAMENTO DEL BURNUP DI UN REATTORE VELOCE REFRIGERATO A METALLO LIQUIDO

C. Paris (ENEA), F. Giannetti, A. Naviglio (CIRTEN)

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Report Ricerca di Sistema Elettrico

Accordo di Programma Ministero dello Sviluppo Economico - ENEA
Piano Annuale di Realizzazione 2012
Area: Produzione di energia elettrica e protezione dell'ambiente
Progetto: Sviluppo competenze scientifiche nel campo della sicurezza nucleare e collaborazione ai programmi internazionali per il nucleare di IV Generazione
Obiettivo: Sviluppo competenze scientifiche nel campo della sicurezza nucleare
Responsabile del Progetto: Mariano Tarantino, ENEA

Il presente documento descrive le attività di ricerca svolte all'interno dell'Accordo di collaborazione "Sviluppo competenze scientifiche nel campo della sicurezza nucleare e collaborazione ai programmi internazionali per il nucleare di IV generazione"
Responsabile scientifico ENEA: Mariano Tarantino
Responsabile scientifico CIRTEN: Giuseppe Forasassi

Titolo

Determinazione dei parametri di sicurezza del core e dell'andamento del burnup di un reattore veloce refrigerato a metallo liquido

Descrittori

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Sommario

In questo rapporto vengono documentate le attività svolte per lo sviluppo e la validazione di un modello Monte Carlo basato sul codice MCNPX v.2.7 per la simulazione neutronica di un reattore veloce raffreddato a sodio. Tale modello è stato sviluppato seguendo le specifiche dell'OECD/NEA Sodium Task Force. I principali parametri nucleari (coefficienti di reattività, parametri nucleari, distribuzione tridimensionale della potenza e concentrazione degli attinidi) del core, sono stati calcolati per un nocciolo all'equilibrio, ad inizio e fine del ciclo. I risultati ottenuti sono stati sottomessi all'OECD/NEA ed il confronto preliminare con altre soluzioni indipendenti dimostra il buon accordo dei dati calcolati. I risultati di questo lavoro verranno utilizzati per lo sviluppo di un modello termoidraulico di un SFR utilizzando il codice RELAP5-3D©.

Note

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Sommario

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Abstract

This report describes the activities concerning the development and validation of a detailed Monte Carlo model for the simulation of a sodium-cooled fast neutron reactor. The model specification were set up by an ad-hoc OECD/NEA Sodium Task Force. Core reactivity coefficients, nuclear parameters, assembly-wise power distributions and actinides concentrations were calculated at BOEC and EOEC by MCNPX v.2.7 code. Results were submitted to the OECD/NEA and preliminary comparison shows good agreement with the other independent solutions. Results of this work are being used for setting up a RELAP5-3D© TH model of an SFR.

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Acronyms

ANL	Argonne National Laboratory
BEOC	Beginning of Equilibrium Cycle
CEA	Commissariat à l'Energie Atomique et aux énergies alternatives
CR	Control Rods
EFPD	Effective Full Power Day
EOEC	End of Equilibrium Cycle
LFR	Lead Fast Reactor
MCNPX	Monte Carlo n-Particle Code X
OECD/NEA	Nuclear Energy Agency of the OECD
ODS	Oxide Strengthened Steel
NK	Neutron Kinetics
NPP	Nuclear Power Plant
SFR	Sodium Fast Reactor
TF	Task Force
TH	Thermal-hydraulics

1. Introduction

In Nuclear Technology it is of a paramount importance achieving the proper level of qualification for a computational model devoted to the R&D of a new nuclear power plant (NPP) concept. This requirement is directly linked to the “defence in depth” philosophy principles, which recommend the highest possible quality levels for the techniques used during the design phases, in order to minimize the likelihood of abnormal operations and accidents.

Therefore the assessment of codes limitations and their uncertainties quantification has to be carried out, using as far as possible, experimental and operational data. Up to now, Gen. IV reactor concepts are being developed with limited (e.g., for Sodium Fast Reactor, SFR) or totally missing (e.g., for Lead Fast Reactor, LFR) operating experience. Thus preliminary qualification of codes and data has to be achieved by analytical and code-to-code benchmarks.

Recently, the OECD/NEA constituted, inside the Working Party on the Reactor Systems (WPRS) a Sodium Fast Reactor Task Force (SFR-TF) with the objective of performing a comparative analysis of the safety characteristics of different fuels types as metal, oxide, nitride and carbide cores being analyzed for SFR Gen. IV concept.

The activities reported hereafter were developed in such framework and constitute a mean for obtaining the needed qualification for the chain-of-codes being employed by ENEA for its Gen. IV R&D activities.

2. Reactor Core Description

2.1 General parameters

Neutronic calculation are related to the characterization of a large oxide SFR Core, at Beginning of the Equilibrium Cycle (BEOC) and at the End of Equilibrium Cycle (EOEC). Detailed core description has been provided by CEA and ANL [1]. Core has a medium power density that result in low reactivity swing during the equilibrium burn cycle. The core concept is based on the use of Oxide Strengthened Steel (ODS) cladding with helium bond. The fuel pellet is based on the “fat pin with small wire” concept that enables to reach self-breeding without fertile blanket.

The resulting core exhibits an average burnup around 100 GWd/tHM for a corresponding cycle length of 410 equivalent full power days with one fifth reloading scheme.

Table 1. Reactor Core Nominal Conditions

Parameter	Value
Reactor power (MWth)	3600
Core inlet temperature (°C)	395
Core outlet temperature (°C)	545
Average core structure temperature (°C) (structure, absorber and coolant medium)	470
Average Fuel Temperature (°C)	1227

The core layout is presented in Figure 1. The core consists of 453 fuel, 270 radial reflector and 27 control subassemblies. The core is divided into inner and outer core zones, which are composed of 225 and 228 fuel assemblies, respectively.

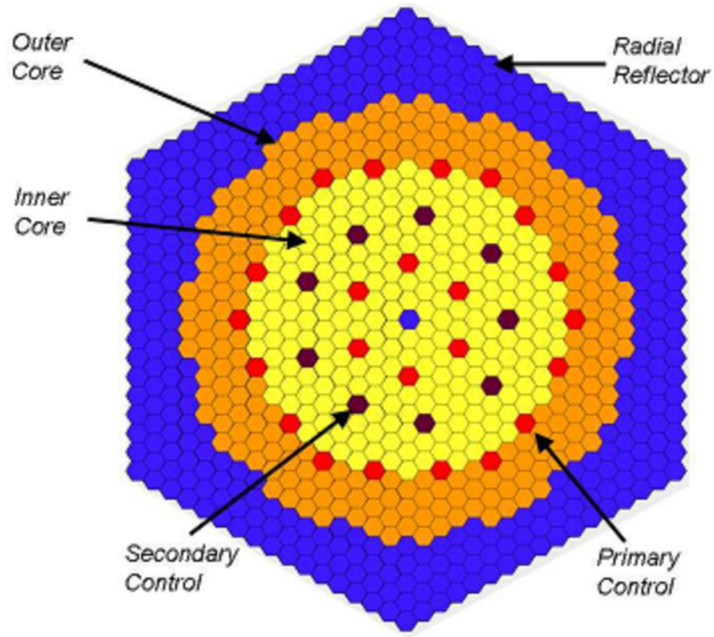

Figure 1. 3600 MWth SFR core model

Two independent safety-grade reactivity control sub-systems are used. The primary control system consists of 6 control subassemblies in the inner core and 12 control subassemblies at the interface between the inner and the outer zones. The secondary system contains 9 control subassemblies located in the 7th row. Although the core is surrounded by various materials, a vacuum boundary condition (i.e., no-return current) is specified for neutronic modelling.

The fuel sub-assembly consists of a hexagonal wrapper tube that contains a triangular arrangement of helium bonded fuel pins with helical wire wrap spacers. The hexagonal wrapper tube and the wire wrap spacers are made of EM10-like steel. The volume of wire wrap spacers is included in the cladding volume by means of radius increase in order to simplify the pin description.

The fuel pin consists of $(U,Pu)O_2$ pellets with ODS cladding.

The axial pin design is based on a central 1 meter active zone surrounded by two gas plenum. The upper one accounts for the top of the pin and has a limited dimension. The axial reflector at the bottom of the active zone is composed of steel pellets located in the pin. The same composition is used also for upper axial reflector for simplicity. In Figure 2, a sketch of the axial core and of the fuel assembly is given.

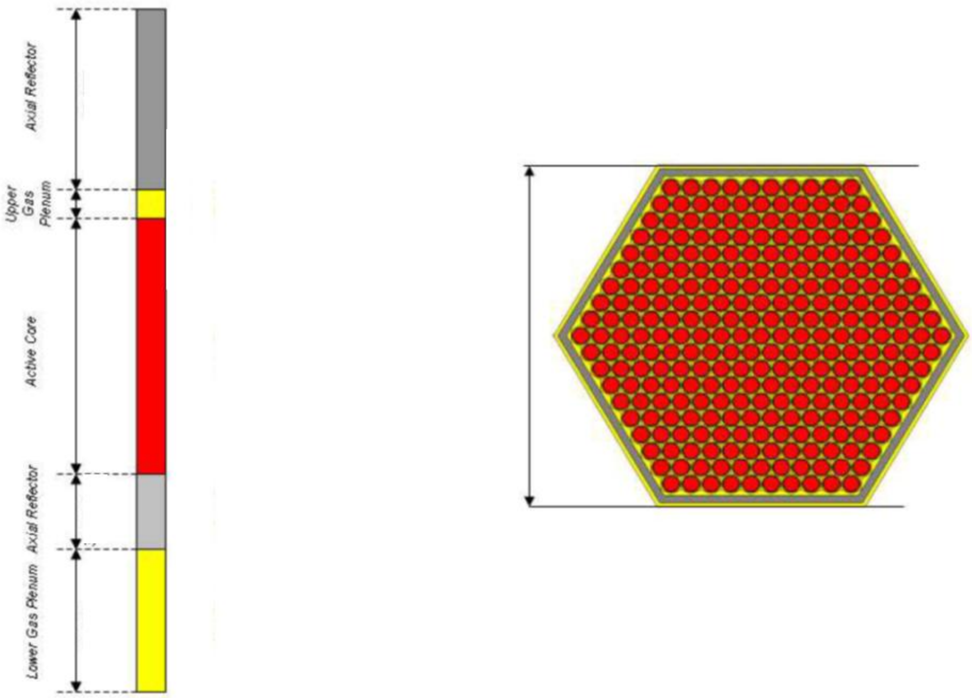

Figure 2. Axial Core and Fuel Assembly structures

2.2 Control Systems

Primary and secondary control systems consist of hexagonal lattice of sodium bonded boron carbide pins of wire wrap spacers inside several ducts.

The volume of wire wrap spacers was included in the cladding volume by means of radius increase in order to simplify the pin description. Due to a low reactivity swing, the primary system uses natural boron carbide, while the secondary system uses enriched $_{10}$B boron carbide. Duct and Cladding structure used the EM10 material. The Primary and Secondary Control subassembly geometry and materials are shown in Figure 3.

For both control rod systems, the absorber height is the same as the active zone. In all calculations executed and reported hereafter, they remain located at the top of the active core zone (interface between active core zone and upper gas plenum). The other part of the sub-assembly consists of an empty duct filled with sodium.

For radial reflector, a single unique homogeneous medium is used. This medium spread along the overall length of the corresponding sub-assembly. The associated volume fractions are 26% for sodium and 74% for steel (EM10 material).

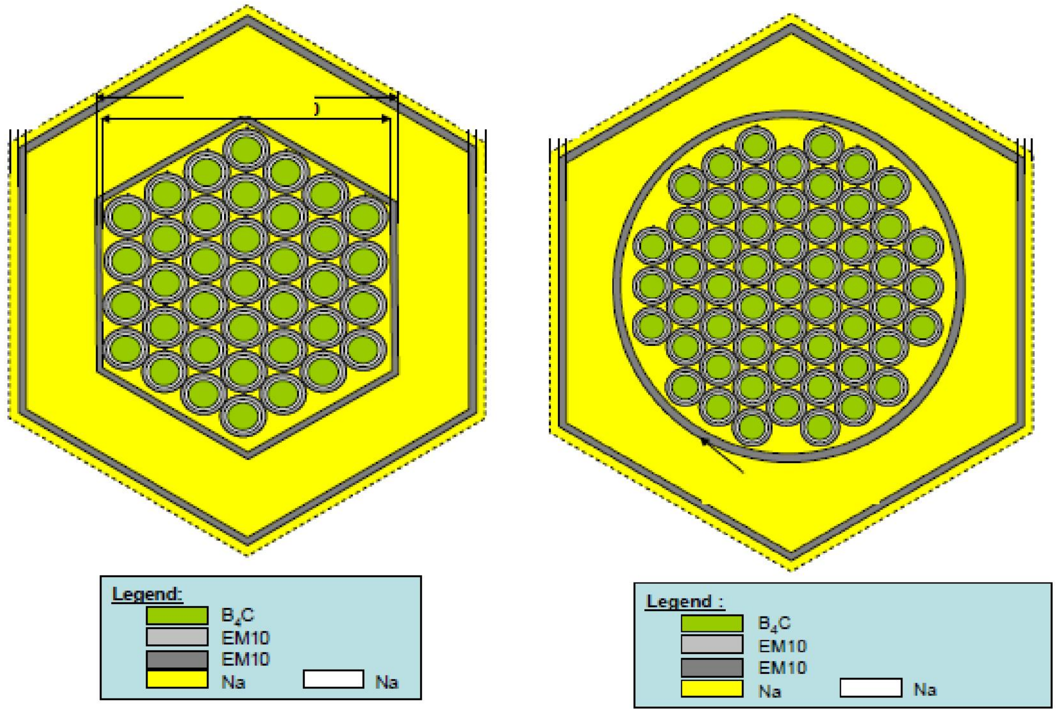

Figure 3. Primary (Left) and Secondary (Right) Control Subassembly

2.3 Fuel

Data for the nominal operating condition were supplied by CEA and they were calculated by accounting for the effects of thermal expansion and irradiation swelling from the fuel fabrication state. The subassembly was divided into five axial concentration sets for each different initial Pu content (inner core and outer core). Fission product isotopes were replaced by a representative isotope (Molybdenum, Mo) in terms of equivalent absorption, and only one averaged value was available for each active zone (inner and outer).

BOEC number densities were given. Number densities lower than 10^{-10} atoms/barn were omitted.

2.4 Structure, Coolant and Absorber materials

Number densities for Structures, Cool and Absorber materials were also supplied by benchmark specifications [1].

3. Computational Tools and Calculation Parameters

Calculations were performed using the Monte Carlo code MCNPX v.2.7. [2]. ENDF/B-VII.0 continuous-energy nuclear data cross sections were employed and temperature corrections was obtained by the MAKXFS code [3].

The keff and fission source distribution convergence was checked and achieved using at least 10 million neutron histories.

The CINDER90 depletion module [4], fully integrated in MCNPX, was used for performing depletion calculations, using a classical predictor/corrector scheme.

Basic fission products ($_{93}$Zr, $_{95}$Mo, $_{99m}$Tc, $_{101}$Ru, $_{131}$Xe, $_{134}$Xe, $_{133}$Cs, $_{137}$Cs, $_{138}$Ba, $_{141}$Pr, $_{143}$Nd, $_{145}$Nd) and immediate daughters of the burned fuel nuclides are tracked and used for transport calculations with continuous energy data. Moreover CINDER90 tracks also all the 3400 fission products of its database using a 63 energy groups cross section library. 17 depletion steps were used, with the maximum time interval being of 45 days.

Calculations were executed on ENEA CRESCO supercomputer [5], exploiting its massive parallel calculation resources. Up to 512 cores were used for detailed power distribution calculations.

4. MCNPX Developed Model

MCNPX input deck based on a fully heterogeneous core description was developed. All the main core components (fuel pins, wrappers, CR) were modeled, without performing any homogenization. Vacuum boundary conditions were imposed on the reflector periphery. 5 different materials were used for the inner and outer core in order to simulate the variation of the fuel depletion in the axial and radial directions. The Sketches of the developed model are given from Figure 4 to Figure 8.

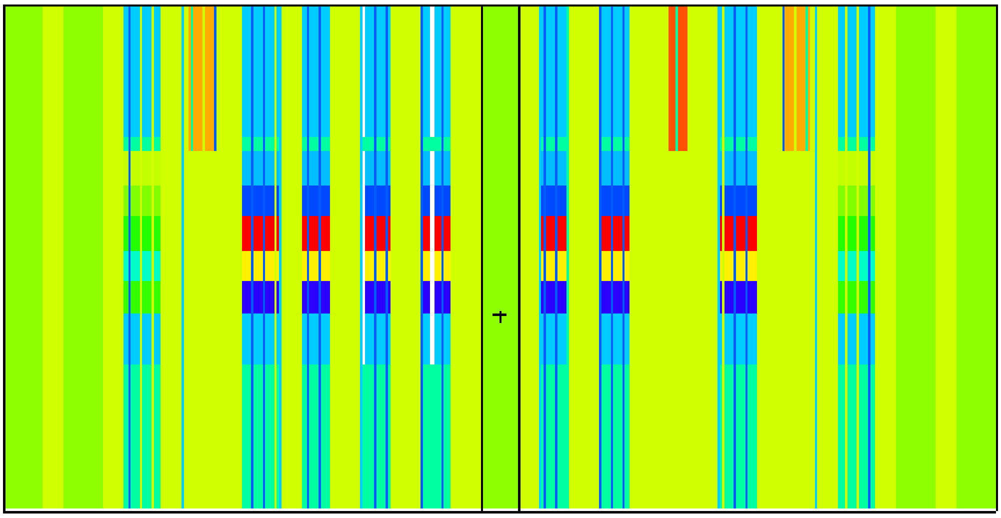

Figure 4. x-z view

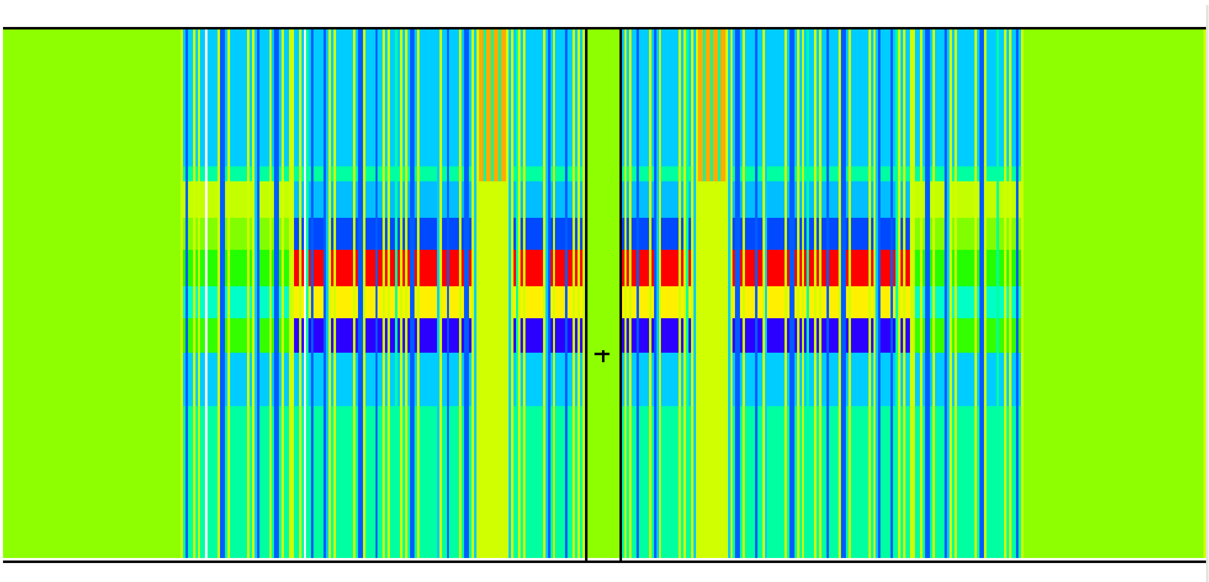

Figure 5. y-z view

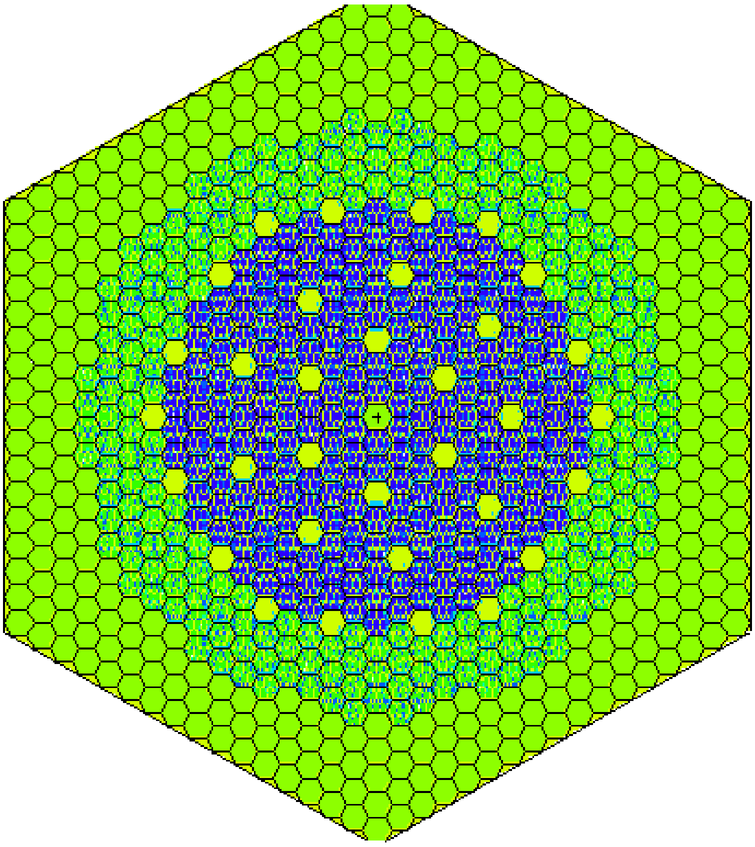

Figure 6. x-y view

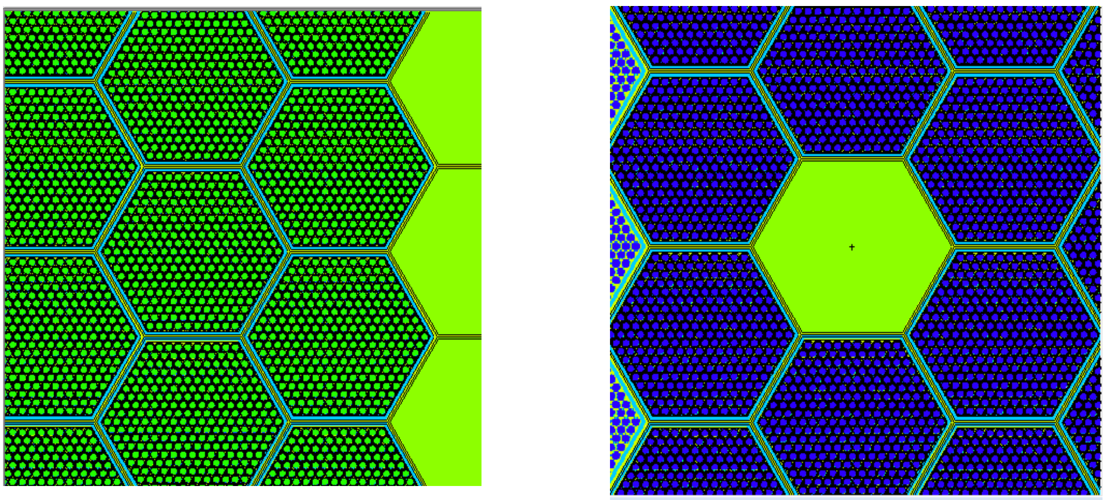

Figure 7. Outer Fuel and Radial Reflector (Left), Central Fuel (Right)

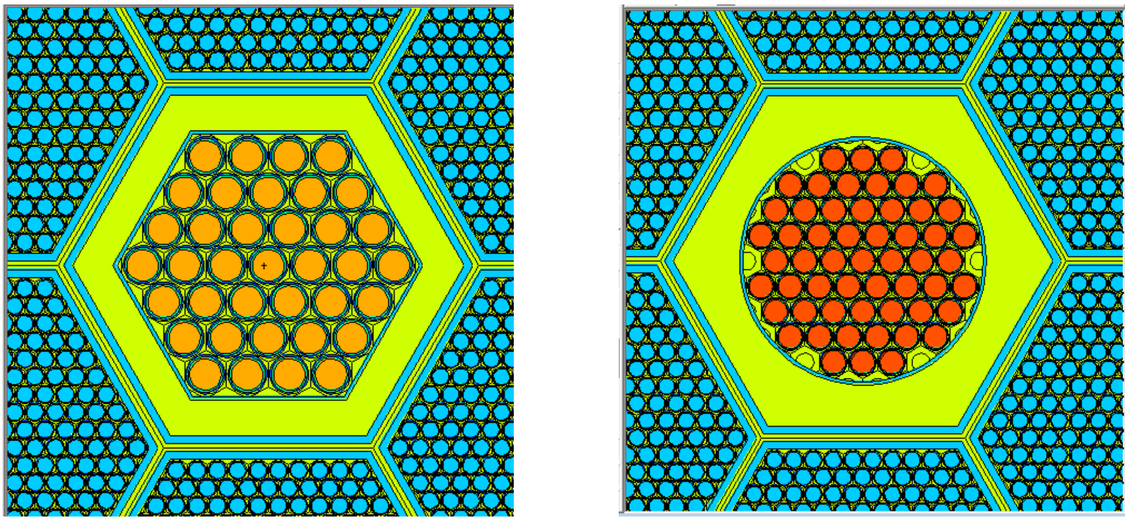

Figure 8. Control Rods. Primary (Left) and Secondary (Right) System

5. Results

Calculations for core characterization at BOEC and EOEC are reported hereafter. Depletion calculations was executed for 410 Effective Full Power Day (EFPD), reaching an average fuel burnup of 410 GWd/tHM.

5.1 BOEC

Main neutronic parameters were obtained performing criticality calculations and sensitivities on the corresponding physical parameters of the reactivity coefficient. Results are summarized in Table 2. They were compared with other independent solutions obtained by other Monte Carlo (MVP) and Transport solution codes (ERANOS/VARIANT, MARBLE) using also different cross sections database (JENDL4.0, JEFF3.1). The greater deviation was found in the estimation of the CR worth. All the parameters are generally in good agreement, considering the differences of data and methods used.

Table 2. BOEC Parameters

Parameter	Value	Deviation from the average of other independent solutions, pcm
k_{eff}	1.01080	-196
β_{eff} (pcm)	+352	-15
Δr Na (pcm)	+1940	-43
Δr Doppler (pcm)	-866	+62
Δr Control Rods (pcm)	-5530	-482

5.2 EOEC

Main neutronic parameters were calculated for the core at EOEC, after performing depletion calculations. Results are summarized in Table 3. They were compared with other independent solutions obtained by other Monte Carlo (MVP) and Transport solution codes (ERANOS/VARIANT, MARBLE) using also different cross sections database (JENDL4.0, JEFF3.1). Also in this case, the greater deviation was found in the estimation of the CR worth. All the parameters are generally in good agreement, considering the differences of data and methods used.

Table 3. EOEC Parameters

Parameter	Value	Deviation from the average of other independent solutions, pcm
k_{eff}	1.01070	+16
β_{eff} (pcm)	353	~0
Δr Na (pcm)	+2033	+3
Δr Doppler (pcm)	-798	+75
Δr Control Rods (pcm)	-5913	-412

5.3 Depletion Calculations

Calculation of Actinides concentration was performed in order to determine the core inventory. The Inner and the Outer core materials depletion (10 independent materials) were simulated. Results summaries are given in Table 4, Table 5 and Table 6, while detailed nuclide distribution given in Table 7. Comparison with the average prediction of other independent solutions shows a very good agreement in nuclide mass calculation. Variation of actinides concentration from BEOC to EOC is instead given in Table 8. Core keff trend, reported in Figure 9, shows the expected slight reactivity swing.

Table 4. Actinides Masses in the Inner Core at EOEC

		Inner Core				
		Upper boundary from active core bottom (cm)				
		20.11	40.22	60.33	80.44	100.55
mass (kg)	U	5796.92	5647.82	5597.44	5687.05	5851.58
	Pu	1134.78	1139.82	1141.37	1137.33	1127.10
	Am	24.66	24.37	24.39	24.28	24.19
	Cm	3.45	4.42	4.91	4.11	2.72

Table 5. Actinides Masses in the Outer Core at EOEC

		Outer Core				
		Upper boundary from active core bottom (cm)				
		20.11	40.22	60.33	80.44	100.55
mass (kg)	U	5805.89	5671.77	5624.35	5681.05	5843.19
	Pu	1260.71	1252.76	1252.54	1254.27	1258.79
	Am	28.63	28.54	28.64	28.54	28.38
	Cm	2.68	3.64	4.13	3.48	2.34

Table 6. Actinides Core Inventory at EOEC

Nuclide	Inventory @ EOEC (kg)	Deviation from the average of other independent solutions, %
U	57207.0	-0.02
Pu	11959.5	0.09
Am	264.6	-3.75
Cm	35.9	0.56

Table 7. Detailed Actinides distribution in the Inner and Outer Core at EOEC

		Inner Core					Outer Core				
		Upper boundary from active core bottom (cm)					*Upper boundary from active core bottom (cm)*				
		20.11	40.22	60.33	80.44	100.55	20.11	40.22	60.33	80.44	100.55
Isotopic number density (atoms/barn-cm)	U234	2.20E-06	2.06E-06	2.00E-06	2.10E-06	2.30E-06	2.69E-06	2.52E-06	2.45E-06	2.55E-06	2.74E-06
	U235	2.48E-05	2.12E-05	1.99E-05	2.20E-05	2.72E-05	2.76E-05	2.39E-05	2.25E-05	2.44E-05	2.87E-05
	U236	5.20E-06	5.66E-06	5.88E-06	5.51E-06	4.66E-06	4.37E-06	4.91E-06	5.14E-06	4.82E-06	4.11E-06
	U238	1.69E-02	1.65E-02	1.63E-02	1.66E-02	1.71E-02	1.67E-02	1.63E-02	1.62E-02	1.63E-02	1.68E-02
	Np237	5.30E-06	7.11E-06	7.54E-06	6.85E-06	4.85E-06	4.77E-06	6.49E-06	6.93E-06	6.34E-06	4.51E-06
	Pu238	7.84E-05	7.16E-05	6.94E-05	7.30E-05	8.19E-05	9.69E-05	8.91E-05	8.67E-05	9.02E-05	9.89E-05
	Pu239	1.77E-03	1.82E-03	1.83E-03	1.82E-03	1.75E-03	1.85E-03	1.88E-03	1.89E-03	1.88E-03	1.85E-03
	Pu240	9.60E-04	9.49E-04	9.50E-04	9.47E-04	9.48E-04	1.09E-03	1.07E-03	1.07E-03	1.07E-03	1.08E-03
	Pu241	1.87E-04	1.76E-04	1.74E-04	1.76E-04	1.89E-04	2.19E-04	2.06E-04	2.03E-04	2.07E-04	2.20E-04
	Pu242	2.94E-04	2.85E-04	2.82E-04	2.86E-04	2.97E-04	3.47E-04	3.37E-04	3.33E-04	3.39E-04	3.49E-04
	Am241	3.96E-05	3.52E-05	3.36E-05	3.61E-05	4.20E-05	5.08E-05	4.57E-05	4.38E-05	4.64E-05	5.22E-05
	Am242g	1.61E-08	1.77E-08	1.84E-08	1.72E-08	1.49E-08	1.26E-08	1.43E-08	1.51E-08	1.41E-08	1.21E-08
	Am242m	9.99E-07	9.50E-07	9.30E-07	9.51E-07	9.62E-07	1.14E-06	1.15E-06	1.15E-06	1.15E-06	1.11E-06
	Am243	3.02E-05	3.38E-05	3.55E-05	3.27E-05	2.66E-05	2.92E-05	3.41E-05	3.62E-05	3.34E-05	2.72E-05
	Cm242	2.91E-06	3.25E-06	3.38E-06	3.16E-06	2.67E-06	2.31E-06	2.66E-06	2.80E-06	2.62E-06	2.22E-06
	Cm243	1.51E-07	1.78E-07	1.95E-07	1.65E-07	1.15E-07	1.15E-07	1.47E-07	1.65E-07	1.40E-07	9.88E-08
	Cm244	6.37E-06	8.53E-06	9.66E-06	7.84E-06	4.72E-06	4.85E-06	6.99E-06	8.08E-06	6.61E-06	4.09E-06
	Cm245	4.02E-07	6.14E-07	7.37E-07	5.40E-07	2.56E-07	2.58E-07	4.40E-07	5.45E-07	4.04E-07	1.97E-07
	Cm246	1.56E-08	3.09E-08	4.02E-08	2.58E-08	8.57E-09	8.14E-09	1.84E-08	2.50E-08	1.63E-08	5.61E-09

Table 8. Actinide Inventory variation from BEOC to EOEC

Nuclide	Inventory @ BOEC (kg)	Inventory @ EOEC (kg)	Variation (%) BEOC-EOEC
U	58860.4	57207.0	-2.8
Pu	11874.0	11959.5	+0.7
Am	230.1	264.6	+15.0
Cm	21.6	35.9	+66.2

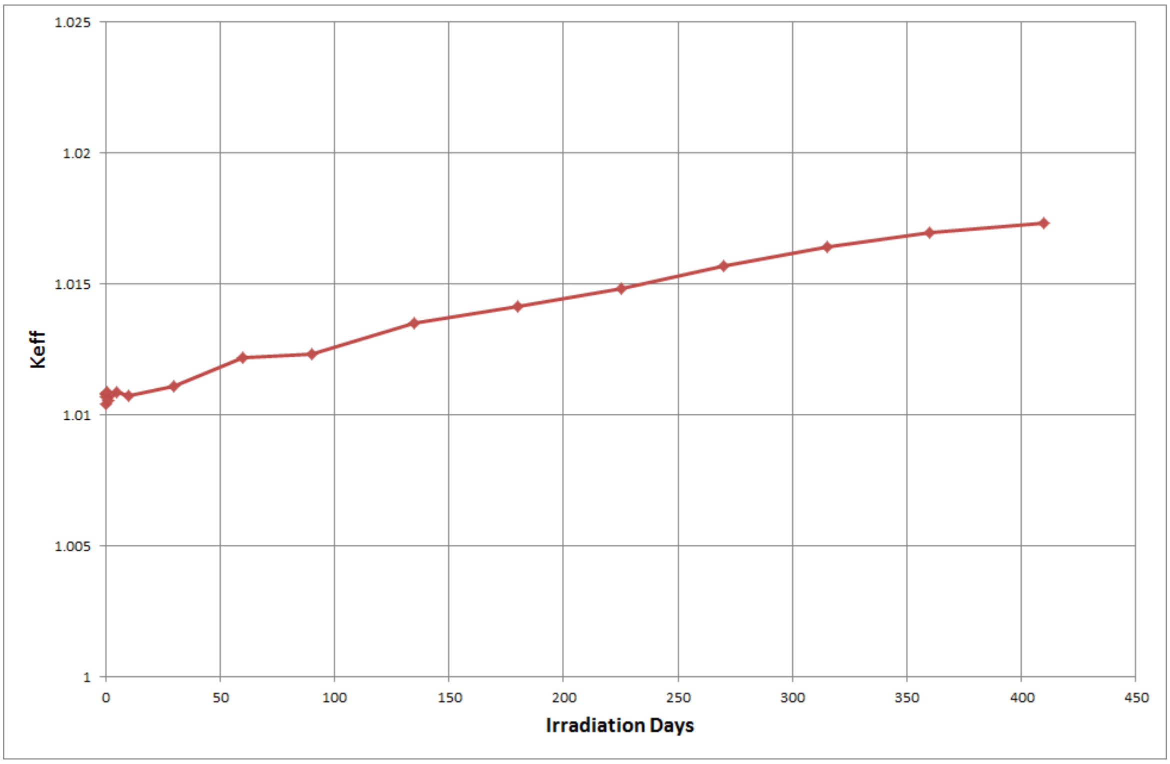

Figure 9. Core reactivity during irradiation

5.4 Power Distribution

Using MCNPX tallying tools, fission power distribution was calculated at BOEC and EOEC assembly-by-assembly, plane-by-plane. Thus radial and axial power distributions were obtained in order to get the peaking factors useful for the thermal-hydraulic code calculations. Results are summarized in Figure 10 and Figure 11. Detailed power distribution is also given in Table 9. Assembly are numerated starting from the central fuel assembly (position number 1, ring 0), then moving counter-clockwise on each ring, beginning from the assembly below (south) the central one.

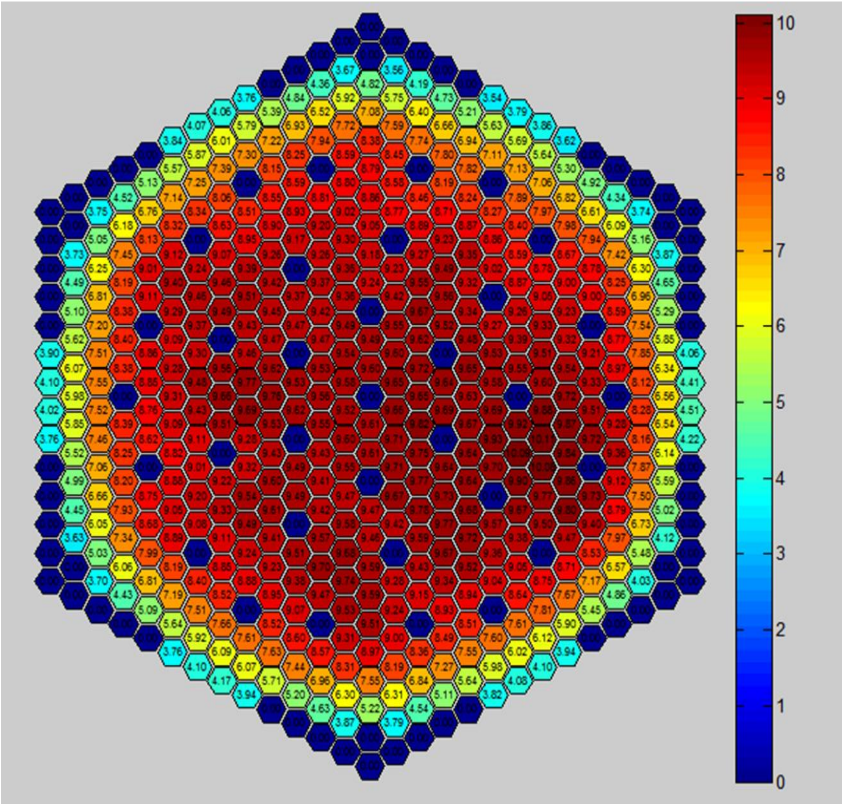

Figure 10. BOEC Radial Power Distribution

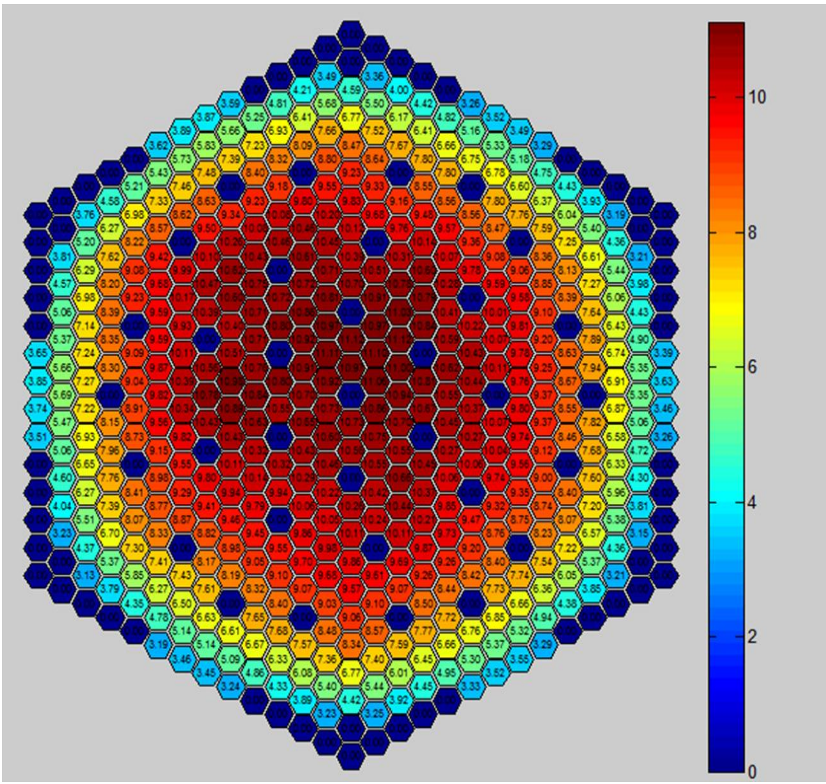

Figure 11. EOEC Radial Power Distribution

Table 9. Detailed Power Distributions

Sub-assembly position		Power (MW)	
Ring number	position number	BOEC	EOEC
0	1	0.00	0.00
1	2	9.88	11.41
1	3	9.86	11.95
1	4	9.85	12.02
1	5	10.15	11.53
1	6	10.12	11.48
1	7	10.02	11.54
2	8	9.95	10.68
2	9	9.48	11.39
2	10	9.36	11.65
2	11	9.70	11.95
2	12	9.62	11.93
2	13	10.11	11.59
2	14	10.31	11.35
2	15	10.24	11.08
2	16	10.64	11.21
2	17	10.17	11.40
2	18	9.66	11.27
2	19	9.67	10.83
3	20	0.00	0.00
3	21	10.19	11.05
3	22	9.84	11.41
3	23	0.00	0.00
3	24	9.75	11.37
3	25	9.69	11.38
3	26	0.00	0.00
3	27	9.74	11.96
3	28	10.05	11.85
3	29	0.00	0.00
3	30	10.43	11.30
3	31	10.50	11.13
3	32	0.00	0.00
3	33	10.43	11.41
3	34	9.81	11.23
3	35	0.00	0.00
3	36	9.47	10.52
3	37	9.47	10.49
4	38	10.20	9.48
4	39	10.59	10.49

4	40	10.43	10.86
4	41	9.49	10.87
4	42	9.64	10.81
4	43	9.60	11.32
4	44	9.49	11.40
4	45	9.35	11.38
4	46	9.16	11.33
4	47	9.10	11.82
4	48	9.59	12.46
4	49	10.00	12.09
4	50	10.22	11.54
4	51	10.74	11.18
4	52	10.56	11.00
4	53	10.35	11.26
4	54	10.24	11.43
4	55	10.17	11.35
4	56	10.05	11.53
4	57	9.33	10.86
4	58	9.13	10.17
4	59	9.16	10.02
4	60	9.28	9.94
4	61	9.63	9.54
5	62	10.13	9.29
5	63	10.49	9.67
5	64	10.85	10.40
5	65	10.15	10.67
5	66	9.34	10.95
5	67	8.97	10.78
5	68	9.44	11.13
5	69	9.02	11.08
5	70	8.97	11.11
5	71	8.68	11.14
5	72	8.51	11.18
5	73	8.85	11.73
5	74	9.38	12.19
5	75	9.86	12.16
5	76	9.97	11.64
5	77	10.37	11.08
5	78	11.01	11.31
5	79	10.84	11.20
5	80	10.80	11.14
5	81	10.83	11.21
5	82	10.62	10.99

5	83	10.16	10.69
5	84	10.10	10.89
5	85	9.67	10.96
5	86	9.20	9.97
5	87	9.15	9.71
5	88	8.96	9.99
5	89	9.20	9.33
5	90	9.20	9.32
5	91	9.88	8.94
6	92	10.49	8.79
6	93	0.00	0.00
6	94	10.70	9.74
6	95	10.94	10.31
6	96	10.16	10.32
6	97	0.00	0.00
6	98	8.93	10.44
6	99	9.12	10.92
6	100	9.06	10.69
6	101	0.00	0.00
6	102	8.48	10.78
6	103	8.12	11.05
6	104	7.82	10.85
6	105	0.00	0.00
6	106	8.75	11.66
6	107	9.82	11.93
6	108	10.14	11.67
6	109	0.00	0.00
6	110	10.38	11.35
6	111	10.40	11.21
6	112	10.78	11.07
6	113	0.00	0.00
6	114	11.17	11.08
6	115	11.32	11.02
6	116	10.83	10.48
6	117	0.00	0.00
6	118	10.01	10.50
6	119	10.01	10.54
6	120	9.11	10.19
6	121	0.00	0.00
7	122	8.46	9.05
7	123	8.76	9.31
7	124	9.01	9.03
7	125	0.00	0.00

7	126	9.56	8.61
7	127	10.05	8.66
7	128	10.76	7.96
7	129	10.75	8.32
7	130	10.38	8.80
7	131	10.98	9.40
7	132	10.70	9.64
7	133	9.42	9.62
7	134	8.90	9.82
7	135	8.87	10.03
7	136	8.84	10.61
7	137	8.73	10.51
7	138	8.56	10.23
7	139	8.27	10.21
7	140	7.93	10.51
7	141	7.60	10.53
7	142	7.53	10.30
7	143	7.74	10.37
7	144	8.32	10.59
7	145	9.20	10.90
7	146	9.58	10.98
7	147	9.85	10.89
7	148	10.34	11.04
7	149	11.04	11.15
7	150	10.75	11.14
7	151	10.61	10.78
7	152	10.61	10.68
7	153	11.05	10.94
7	154	11.24	10.96
7	155	11.21	10.56
7	156	10.92	9.86
7	157	10.32	9.96
7	158	10.17	9.83
7	159	9.75	10.14
7	160	9.35	9.64
7	161	8.42	9.04
7	162	8.12	8.75
7	163	7.95	8.58
8	164	8.08	8.87
8	165	8.25	8.99
8	166	8.64	8.58
8	167	9.00	8.29
8	168	9.69	8.32

8	169	10.10	8.10
8	170	10.23	7.12
8	171	10.60	7.22
8	172	10.53	7.71
8	173	10.56	8.38
8	174	10.68	8.76
8	175	9.90	9.08
8	176	9.04	8.91
8	177	8.91	9.33
8	178	8.35	9.64
8	179	8.21	9.97
8	180	8.17	9.85
8	181	8.15	9.80
8	182	8.01	9.66
8	183	7.83	9.55
8	184	7.54	9.16
8	185	7.45	9.45
8	186	7.28	9.24
8	187	7.30	9.81
8	188	7.86	9.70
8	189	8.54	10.10
8	190	9.28	9.84
8	191	9.66	9.72
8	192	9.79	10.20
8	193	10.70	10.44
8	194	10.89	10.47
8	195	10.62	10.85
8	196	10.57	10.51
8	197	10.61	10.49
8	198	10.55	10.46
8	199	10.81	10.40
8	200	10.79	10.36
8	201	10.77	9.78
8	202	10.34	9.62
8	203	9.95	9.48
8	204	9.58	9.39
8	205	9.42	9.37
8	206	8.97	8.93
8	207	8.59	8.68
8	208	8.38	8.16
8	209	7.68	7.93
8	210	7.63	7.80
8	211	7.64	8.29

9	212	7.92	8.35
9	213	8.22	8.41
9	214	8.34	8.08
9	215	9.14	7.72
9	216	9.63	7.39
9	217	9.88	7.30
9	218	9.55	6.68
9	219	9.90	6.81
9	220	0.00	0.00
9	221	10.04	7.23
9	222	9.66	7.36
9	223	9.63	7.99
9	224	9.21	8.32
9	225	0.00	0.00
9	226	8.43	8.54
9	227	7.64	8.12
9	228	7.64	8.65
9	229	0.00	0.00
9	230	7.47	9.06
9	231	7.47	8.63
9	232	7.44	8.60
9	233	7.24	8.38
9	234	0.00	0.00
9	235	7.21	8.84
9	236	7.19	8.05
9	237	7.27	8.65
9	238	0.00	0.00
9	239	7.85	9.04
9	240	8.72	9.00
9	241	9.19	8.74
9	242	9.45	9.18
9	243	0.00	0.00
9	244	10.31	9.75
9	245	10.37	9.84
9	246	10.42	9.87
9	247	0.00	0.00
9	248	10.02	10.09
9	249	10.29	9.80
9	250	10.38	9.77
9	251	10.32	9.82
9	252	0.00	0.00
9	253	10.38	9.18
9	254	9.92	8.81

9	255	9.67	9.07
9	256	0.00	0.00
9	257	9.00	8.62
9	258	8.62	8.41
9	259	8.33	8.15
9	260	8.14	7.76
9	261	0.00	0.00
9	262	7.63	7.47
9	263	7.54	7.28
9	264	7.73	7.50
9	265	0.00	0.00
10	266	7.62	7.86
10	267	7.57	7.57
10	268	8.23	7.31
10	269	8.61	7.28
10	270	0.00	0.00
10	271	9.37	6.98
10	272	7.83	5.55
10	273	9.15	5.99
10	274	9.42	5.88
10	275	9.55	6.27
10	276	9.22	6.30
10	277	0.00	0.00
10	278	8.70	7.46
10	279	8.52	7.82
10	280	7.99	7.97
10	281	7.20	7.66
10	282	6.17	6.53
10	283	6.56	7.48
10	284	6.74	7.87
10	285	6.53	8.13
10	286	6.65	8.19
10	287	0.00	0.00
10	288	6.70	7.65
10	289	6.85	7.57
10	290	6.63	7.77
10	291	6.44	7.46
10	292	6.13	6.62
10	293	6.74	7.15
10	294	7.18	7.63
10	295	7.56	7.71
10	296	8.17	7.83
10	297	0.00	0.00

10	298	8.53	7.94
10	299	8.90	8.61
10	300	9.02	8.70
10	301	9.49	8.73
10	302	8.85	8.22
10	303	9.68	9.12
10	304	9.81	9.46
10	305	9.73	9.69
10	306	9.65	9.31
10	307	0.00	0.00
10	308	10.34	9.02
10	309	10.22	8.66
10	310	10.01	8.23
10	311	9.72	7.83
10	312	8.63	6.93
10	313	8.71	7.54
10	314	8.93	7.98
10	315	8.59	8.17
10	316	8.29	7.56
10	317	0.00	0.00
10	318	7.94	7.17
10	319	7.50	6.68
10	320	7.22	6.81
10	321	7.00	6.57
10	322	6.56	5.86
10	323	6.86	6.33
10	324	6.82	6.88
10	325	6.98	7.20
10	326	7.21	7.14
10	327	0.00	0.00
10	328	8.00	6.92
10	329	8.16	6.53
10	330	8.51	6.27
10	331	8.45	6.38
11	332	4.89	3.81
11	333	6.64	4.55
11	334	7.63	4.91
11	335	7.98	4.92
11	336	8.19	5.35
11	337	7.66	5.54
11	338	7.58	6.14
11	339	7.22	6.67
11	340	6.81	6.50

11	341	6.18	6.25
11	342	5.20	5.67
11	343	4.35	4.59
11	344	5.02	5.35
11	345	5.37	6.17
11	346	5.54	6.80
11	347	5.52	6.90
11	348	6.00	6.90
11	349	5.79	6.66
11	350	5.66	6.52
11	351	5.74	6.59
11	352	5.40	6.41
11	353	4.93	5.50
11	354	4.27	4.16
11	355	4.96	5.22
11	356	5.89	5.98
11	357	6.51	6.45
11	358	7.49	6.53
11	359	7.71	7.06
11	360	7.54	7.28
11	361	7.81	7.19
11	362	7.56	7.21
11	363	7.74	7.15
11	364	7.17	6.55
11	365	5.97	5.28
11	366	7.46	6.70
11	367	8.20	7.45
11	368	8.26	8.01
11	369	8.92	8.09
11	370	8.95	8.02
11	371	8.87	7.70
11	372	9.30	7.55
11	373	9.21	7.22
11	374	8.46	6.78
11	375	7.38	5.74
11	376	5.82	4.81
11	377	7.15	5.84
11	378	7.37	6.13
11	379	7.56	6.65
11	380	7.22	6.83
11	381	7.42	6.74
11	382	7.30	6.50
11	383	7.07	5.84

11	384	6.51	5.67
11	385	5.85	5.32
11	386	5.35	4.84
11	387	4.68	4.10
11	388	5.23	4.67
11	389	5.64	5.27
11	390	5.79	5.75
11	391	6.42	6.09
11	392	6.71	6.21
11	393	6.95	6.32
11	394	7.35	6.08
11	395	7.21	5.44
11	396	6.95	5.30
11	397	6.24	4.86
12	398	0.00	0.00
12	399	3.88	2.50
12	400	4.81	3.08
12	401	5.68	3.49
12	402	6.24	3.76
12	403	6.14	4.13
12	404	6.23	4.64
12	405	5.74	4.89
12	406	5.27	5.06
12	407	4.80	4.59
12	408	4.11	3.93
12	409	3.21	3.25
12	410	0.00	0.00
12	411	3.21	3.22
12	412	3.68	3.98
12	413	4.02	4.60
12	414	4.21	5.26
12	415	4.42	5.31
12	416	4.58	5.45
12	417	4.30	5.22
12	418	4.32	5.02
12	419	4.09	4.78
12	420	3.55	4.12
12	421	3.20	3.53
12	422	0.00	0.00
12	423	3.11	3.10
12	424	3.78	3.84
12	425	4.58	4.32
12	426	5.80	4.73

12	427	6.16	5.27
12	428	6.24	5.66
12	429	6.28	5.46
12	430	5.97	5.23
12	431	5.60	5.04
12	432	5.16	4.35
12	433	4.35	3.83
12	434	0.00	0.00
12	435	4.70	4.06
12	436	5.43	4.99
12	437	5.94	5.65
12	438	6.39	5.78
12	439	7.02	6.10
12	440	7.02	6.18
12	441	7.20	5.91
12	442	6.92	5.38
12	443	6.46	4.93
12	444	5.44	4.28
12	445	4.42	3.35
12	446	0.00	0.00
12	447	4.15	3.54
12	448	5.00	4.48
12	449	5.32	4.66
12	450	5.45	4.94
12	451	5.65	5.07
12	452	5.68	5.20
12	453	5.57	4.94
12	454	5.23	4.36
12	455	4.50	3.84
12	456	3.77	3.57
12	457	3.45	2.99
12	458	0.00	0.00
12	459	3.25	2.87
12	460	3.62	3.32
12	461	4.14	3.97
12	462	4.66	4.22
12	463	5.22	4.60
12	464	5.44	4.83
12	465	5.51	5.02
12	466	5.39	4.41
12	467	5.26	3.99
12	468	4.67	3.33
12	469	3.81	2.75

13	470	0.00	0.00
13	471	0.00	0.00
13	472	0.00	0.00
13	473	0.00	0.00
13	474	0.00	0.00
13	475	3.94	2.65
13	476	4.18	2.86
13	477	3.92	3.14
13	478	3.72	3.38
13	479	0.00	0.00
13	480	0.00	0.00
13	481	0.00	0.00
13	482	0.00	0.00
13	483	0.00	0.00
13	484	0.00	0.00
13	485	0.00	0.00
13	486	0.00	0.00
13	487	0.00	0.00
13	488	2.85	3.39
13	489	3.00	3.75
13	490	3.24	3.71
13	491	2.85	3.50
13	492	0.00	0.00
13	493	0.00	0.00
13	494	0.00	0.00
13	495	0.00	0.00
13	496	0.00	0.00
13	497	0.00	0.00
13	498	0.00	0.00
13	499	0.00	0.00
13	500	0.00	0.00
13	501	3.94	3.28
13	502	4.27	3.71
13	503	4.39	3.77
13	504	4.04	3.54
13	505	0.00	0.00
13	506	0.00	0.00
13	507	0.00	0.00
13	508	0.00	0.00
13	509	0.00	0.00
13	510	0.00	0.00
13	511	0.00	0.00
13	512	0.00	0.00

13	513	0.00	0.00
13	514	4.41	3.99
13	515	4.67	4.14
13	516	4.73	3.95
13	517	4.58	3.85
13	518	0.00	0.00
13	519	0.00	0.00
13	520	0.00	0.00
13	521	0.00	0.00
13	522	0.00	0.00
13	523	0.00	0.00
13	524	0.00	0.00
13	525	0.00	0.00
13	526	0.00	0.00
13	527	3.84	3.19
13	528	4.10	3.38
13	529	4.10	3.21
13	530	3.62	2.97
13	531	0.00	0.00
13	532	0.00	0.00
13	533	0.00	0.00
13	534	0.00	0.00
13	535	0.00	0.00
13	536	0.00	0.00
13	537	0.00	0.00
13	538	0.00	0.00
13	539	0.00	0.00
13	540	3.27	2.89
13	541	3.53	3.20
13	542	3.63	3.25
13	543	3.67	3.07
13	544	0.00	0.00
13	545	0.00	0.00
13	546	0.00	0.00
13	547	0.00	0.00

6. Conclusions

A complete Monte Carlo MCNPX model for an SFR core was developed in the framework of the activities of an SFR-TF organized by the OECD/NEA. The scope of this work was to set up and validate the MCNPX model in order to understand the main calculation uncertainties. Preliminary comparisons with other totally independent solutions show a good agreement for all the calculated parameters. This allows to have a quantitative confidence in the calculation output, that will be then used for setting up detailed RELAP5-3D© thermal-hydraulic models for SFR transient calculations.

7. References

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