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Preparatory Studies for Qualification of Lead Test Assemblies with Gadolinium as Burnable Absorber in the BR2 Reactor

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Abstract - *A preliminary feasibility neutronics analyses have been performed for HEU (UAlx) and LEU (UMo-dispersed) fuels with various combinations of burnable poisons (within-meat and outside fuel meat). Reactivity and experimental performances were compared in order to assess the fuel utilization. Former analyses have been focused on discrete absorbers (wires) located outside the fuel meat. Limitations to manufacture very thin wires (if needed) is the motivation of recent studies to investigate within-meat absorbers, similarly to the ones used in the standard BR2 fuel element. The purpose in this paper will be to compare the reactivity performances of three LTA's (Leading Test Assemblies) with HEU (Highly Enriched Uranium) fuel utilizing Gd_2O_3 or B_4C & Sm_2O_3 as burnable absorbers inside fuel meat. Given the differences in the absorption cross sections of the burnable absorbers and the resulting different burn-up evolutions, the present paper discusses the loading strategies of the LTA's during several consecutive irradiation cycles foreseen for this test irradiation.*

I. INTRODUCTION

The feasibility to convert the BR2 reactor from HEU to LEU fuel has been actively studied during the last decade. An important feature of the BR2 standard HEU fuel elements is the use of burnable absorbers, homogeneously mixed with the fuel meat. Therefore, the preliminary neutronic feasibility analyses for HEU-LEU conversion of the BR2 reactor always needed to take into account the choice of a suitable/compatible burnable absorber.

Earlier feasibility studies for various fuel types using burnable absorbers preserved the current fuel element design in order to avoid significant changes in the thermal-hydraulics characteristics of the reactor core. For instance, the neutronic analyses of burnable absorbers in the form of wires in the aluminum side plates of the fuel elements have shown that for this configuration cadmium has the best burn-up characteristics.

Recently, neutronic feasibility studies have been executed for a modified (so-called 'COBRA') geometry of the BR2 fuel element. In this modified geometry, the fuel meat thickness is slightly increased, while the aluminum cladding is slightly reduced. Neutronic analyses for COBRA-HEU and COBRA-LEU fuel types have been performed for different burnable absorbers: cadmium wires outside fuel meat; B_4C and Sm_2O_3 homogeneously mixed in the fuel meat as for the standard BR2 HEU fuel elements; and the newly considered burnable absorber Gd_2O_3 also homogeneously mixed in the fuel meat. Comparison shows that a COBRA-HEU fuel element with boron and samarium as burnable absorbers behaves similarly to a standard BR2 HEU fuel element, while a COBRA-HEU element with gadolinium as burnable absorber has longer cycle length. At the same time, the COBRA-LEU fuel type with gadolinium absorber has also significantly longer cycle length compared to cadmium wires or other absorbers.

Therefore, a decision has been taken to fabricate and test LTA's with the new COBRA geometry with gadolinium as burnable absorber inside the fuel meat. In total three LTA's will be irradiated in the BR2 reactor in the years 2017 and 2018: two COBRA-HEU LTA's with Gd_2O_3 as burnable absorber and one COBRA-HEU LTA with B_4C and Sm_2O_3 as burnable absorber. The purpose will be to compare the reactivity performances of the LTA's during several consecutive reactor cycles. Given the differences in the absorption cross sections of the burnable absorbers and the resulting different burn-up evolutions, the present paper discusses the loading strategies of the LTA's during several consecutive irradiation cycles foreseen for this test irradiation.

II. SUMMARY OF PREVIOUS FEASIBILITY CONVERSION STUDIES

1. Optimization of Burnable Absorber Nature & Geometry

A. Outside Fuel Meat

During 2008-2015 a detailed comparative analysis has been performed for the efficiency and absorption capabilities of 3 major candidates as burnable absorber for the new LEU BR2 fuel: Cd, Gd_2O_3, and B_4C. It was shown that the most favorable absorber, used outside of the fuel meat was Cadmium. This was due to the fact that for the minimum wire diameter, which could be fabricated (Ø=0.3-0.4 mm) cadmium had the highest burn up rate. Gadolinium, boron and other considered absorbers are self-shielded for such diameters and therefore they need extremely thin wire diameters (Ø << 0.1 mm) in order to have high burn up rate.

Therefore, a preliminary choice of new burnable absorber has been made: 12 or 6 Cd-wires per each one of the three Al-side plates in the fuel element. The results of these studies have been discussed in joint BR2-ANL meetings and reported at the RERTR & RRFM conferences [1-5].

B. Inside Fuel Meat

Later (current) studies involve analysis of various fuel types (LEU and HEU) with various burnable poisons, homogeneously mixed with the fuel meat: B_4C & Sm_2O_3 as in the standard BR2 fuel, and Gd_2O_3. The highest burn up rate has gadolinium due to its 2 major isotopes ^{155}Gd and ^{157}Gd which have very high thermal absorption cross sections and deplete very fast with the fuel burn-up [6].

2. Comparison of Burnable Absorber Capabilities

The burn up rates of the major burnable isotopes have been calculated by MCNP6 [7] during one operation cycle with duration ~ 24 days. As it is seen from the graphs in Fig. 1, ^{157}Gd acts similarly to ^{149}Sm, burning almost totally in the first 5 days. The burn up rate of ^{155}Gd, compared to ^{157}Gd is slower, but after 20 days is also totally burnt. The burn up rate of the major cadmium isotope ^{113}Cd strongly depends on the wire diameter, being higher for smaller diameters. The major boron isotope ^{10}B has the slowest burn up rate, burning almost linearly with time.

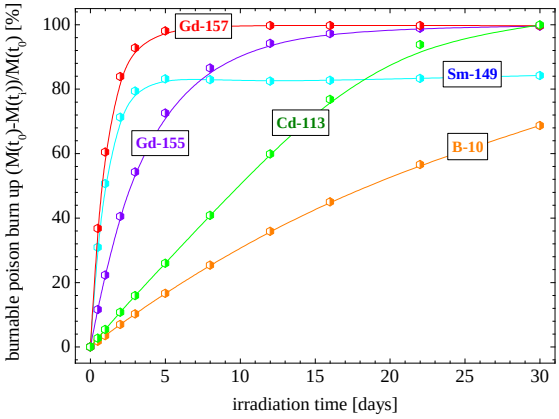

Fig. 1. Comparison of burn up rate of different burnable poisons used in the HEU fuel assemblies.

3. Reactivity Performances

Previous analyses [6] of the reactivity performances have shown that the fuel types with gadolinium in the fuel meat have significantly longer cycle length by about 5-6 days for the HEU fuel type and 6-7 days for the LEU fuel types compared to the standard HEU fuel with standard boron and samarium poisons (see Fig. 2). The optimum amount of the gadolinium poison for the HEU and UMo fuels is about 2.5 grams Gd in a fresh fuel element. The experimental performances of HEU fuel type with gadolinium in the fuel meat were comparable (or even better) to the standard HEU fuel with standard boron and samarium poisons. The results reported in [6] allowed to conclude that fuel types with gadolinium absorber used in a form of homogeneous mixture with the fuel meat have significant economic advantages. Therefore, a decision has been taken to fabricate and test three LTA's with HEU fuel type: two with gadolinium and one with boron & samarium as burnable absorbers inside the fuel meat.

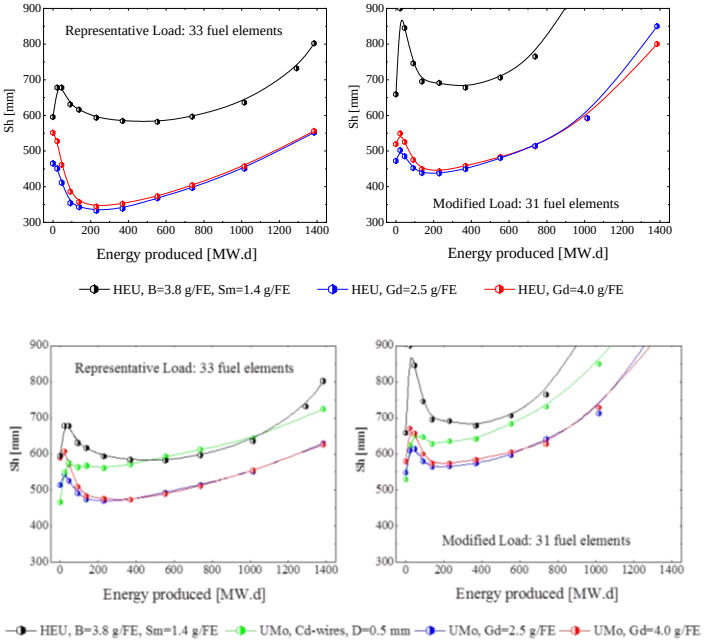

Figure 2. Critical position of the control rod bank vs. produced energy in one BR2 operation cycle with average power P_{BR2}=59 MW and cycle length 24 days for representative load, containing 33 fuel elements (left); modified load, containing 31 fuel elements (right): *(top) HEU core, (bottom) LEU core*.

III. DESCRIPTION OF LTA'S

Three Lead Test Assemblies (LTA) with HEU fuel in the new 'COBRA' geometry will be irradiated in several BR2 operation cycles during 2017-2018. Two equal LTA's will have Gd as burnable absorber inside the fuel meat, and one LTA will have boron and samarium absorber inside fuel meat as for the standard BR2 HEU fuel type. The fuel system parameters of the LTA's are given in Table I.

Table I. Description of LTA's fuel system parameters.

BR2 operation	Standard FE	COBRA-HEU-BSM	COBRA-HEU-GD
^{235}U enr. [%]	93.0	93.0	93.0
g U_{tot}/cm^3	1.3	1.04	1.04
^{235}U [g]	400	400	400
^{238}U [g]	30	30	30
Fuel meat [mm]	0.51	0.63	0.63
Al clad [mm]	0.38	0.36	0.36
Water gap [mm]	3.0	3.0	3.0
Boron (B_4C)	3.8 g/FE	3.8 g/FE	-
Sm (Sm_2O_3)	1.4 g/FE	1.4 g/FE	-
Gd (Gd_2O_3)	-	-	2.5 g/FE

IV. BR2 REACTOR CORE EVOLUTION SIMULATION

1. Description of the BR2 Reactor

The Belgian Material Test reactor (MTR) BR2 is a strongly heterogeneous high flux engineering test reactor operated by SCK•CEN at the Mol site in Belgium. This tank-in-pool reactor is cooled by light water in a compact HEU core (93% ^{235}U), positioned in and reflected by a beryllium matrix. The beryllium matrix is an assembly of a big number of irregular hexagonal prisms which are skew and form a twisted hyperbolic bundle around the central 200 mm channel H1 containing beryllium plugs. The reactor can be operated at the power level of 50÷100 MW, currently about 130 to 150 full power days per year with thermal neutron flux 1.2×10^{15} $cm^{-2}.s^{-1}$ and fast neutron flux 1.0×10^{15} $cm^{-2}.s^{-1}$ at power 60 MW.

2. 3-D Geometry Model of the BR2 Reactor

A 3-D geometry and burn-up model of the BR2 core is developed by the SCK•CEN team using the latest versions of the Monte Carlo transport code MCNP6 [7]. The model is a complete 3-D description of BR2's one sheet hyperboloid reactor core composed of twisted and inclined reactor channels and represents each channel separately, with its individual position and inclination (see Fig. 3). The fuel assemblies, beryllium plugs, experimental devices or control rods loaded in the channels are modeled with the same level of details. The fuel region of each of the 6 fuel rings of every fuel element is axially divided into 10 material cells of 6 cm height and 2 extreme cells of 8.1 cm height, and into 72 azimuth meshes in needed axial segments.

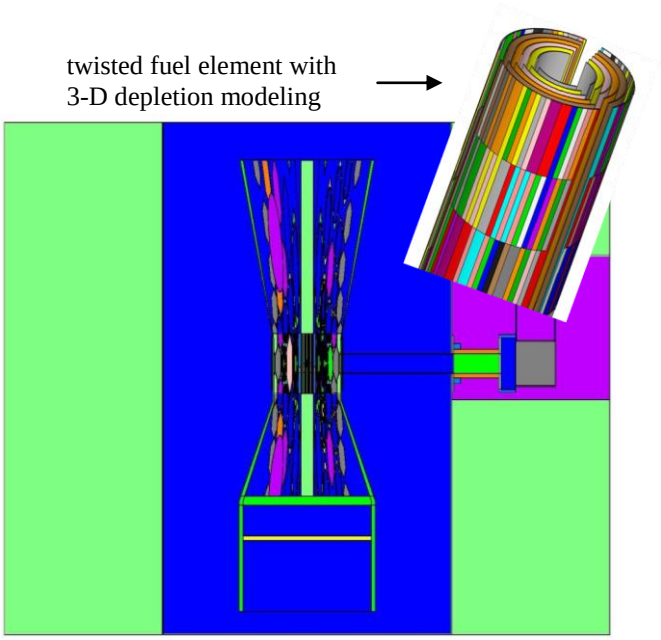

Fig. 3. MCNP model of the BR2 reactor including the vessel and the bio-shield (embedded in the picture inclined fuel element).

3. MCNP Methodology

The MCNP6 simulation is coupled for the isotope evolution part with CINDER90, which is included in the code MCNP. The credibility of the MCNP model was demonstrated by multiple comparisons of code predictions with available experimental data, such as control rod worth's, neutron fluxes, gamma heating and linear power [8-13].

The preferred approach for the whole core automatic depletion calculations by MCNP6 is to take advantage of the existing symmetries of the burned fuel material distribution in the core, i.e., the axial, radial and azimuth symmetries around the core center, in order to significantly reduce the computation time. In this case it is not necessary to give a unique material number to each burn-up cell. Cells having similar burn-up and power will experience the same composition evolution and can therefore be given the same material number. The MCNP model developed for this approach has a total of 2304 burn-up cells (12 axial zones x 6 fuel rings x 32 fuel elements + 16 beryllium matrix regions) with 56 unique materials (40 materials for the fuel and 16 materials for the beryllium matrix). Increasing the number of unique burn up materials improves in general the accuracy of whole core criticality calculations. In this case MCNP6 burns each unique material at the specific flux in the spatial cell, occupied by the material, generating automatically unique spatial flux solutions and unique spatial fuel compositions. However, burning a unique material will increase the statistical uncertainty in power/depletion determination due to the reduced

calculation volume of the unique burn up cell. Therefore, a compromise between the used NPS per depletion step and the number of unique burn up materials was made based on the existing symmetries of the burned fuel material distribution in the core as described above.

V. VALIDATION OF THE MCNP METHODOLOGY

The validation is performed using experimental data and cross-code comparison. The MCNP methodology has been verified on various experimental data, including routine measurements of critical heights of control rods motion during each operating cycle, measurements of control rod worth, reactivity effects of experiments, on dosimetry measurements of neutron fluxes, gamma heating and linear power in different fuel types.

1. Critical Height

The most straight-forward way to evaluate the results of the core load management calculations using the MCNP-based method is comparison of the predicted and measured value for the critical height whenever a reactor load is assembled.

Table II. Comparison of the critical heights of a sample of different BR2 reactor loads, see Ref. [10].

BR2 operation Cycle #	Critical height Sh_{crit} Predicted by MCNP [mm]	Critical height Sh_{crit} Measured [mm]	Reactivity difference Δρ Measured – Predicted [pcm]
01/2011.A3	610	617	-108
06/2010.A2	468	453	+317
06/2010.A4	465	471	-130
05/2010.A2	570	555	+266
05/2010.A3	555	547	-144
04/2010.A4	456	456	0
03/2010.M6	465	442	+410
03/2010.M7	455	460	-86
Average deviation (sum of absolute values)			≈200

Table II shows the critical height of a sample of different reactor loads assembled during measurement campaigns and/or in preparation of a suitable load for a reactor cycle at nominal power. The measured critical heights as well as the predicted critical height by MCNP are listed. As can be seen in this table, the corresponding reactivity difference between the two critical heights remains limited.

2. Reactivity Evolution During a Reactor Cycle

The calculation of the evolution of the reactivity for the reactor core load management as a function of produced energy is one of the main results of the reactor core simulations. It, firstly, permits to evaluate the minimum available negative reactivity (safety margin) at any moment of the cycle as prescribed by the Technical Specifications to be respected. Secondly, it permits to evaluate operational parameters like, the maximum cycle length and the reactivity effect of the ^{135}Xe – ^{149}Sm transient at the start-up of the reactor. Figure A.1 in Appendix A show the evolution of the reactivity of the BR2 reactor as a function of produced energy for the cycles of the years 2012 and 2013 as measured and as predicted by the MCNP-based simulations. To quantify the differences between the reactivity evolution ρ_{MCNP} predicted by MCNP and the measured reactivity evolution ρ_{meas}, six evaluation indicators have been defined:

$$\Delta_{\max} = \max \left| \rho_{\mathrm{MCNP}} - \rho_{\mathrm{meas}} \right|, \qquad (1)$$

the maximum difference between both reactivity evolutions,

$$\Delta_{\mathrm{average}} = \frac{\int_0^{E_{\mathrm{produced}}} dE \left(\rho_{MCNP} - \rho_{\mathrm{meas}} \right)}{E_{\mathrm{produced}}}, \qquad (2)$$

$$\Delta_{|average|} = \frac{\int_0^{E_{\mathrm{produced}}} dE \left| \rho_{MCNP} - \rho_{meas} \right|}{E_{produced}}, \qquad (3)$$

$$\Delta_{\text{at minimum}} = \rho_{\mathrm{MCNP}}(\text{at min.}) - \rho_{\mathrm{meas}}(\text{at min.}), \quad (4)$$

$$\Delta_{\text{at Xe peak}} = \rho_{\mathrm{MCNP}}(\text{at Xe peak}) - \rho_{\mathrm{meas}}(\text{at Xe peak}), \quad (5)$$

$$\Delta_{\text{at EOC}} = \rho_{\mathrm{MCNP}}(\text{at EOC}) - \rho_{\mathrm{meas}}(\text{at EOC}). \quad (6)$$

The first three indicators, $\Delta_{\max}$, $\Delta_{\mathrm{average}}$, $\Delta_{|\mathrm{average}|}$ give an overall evaluation of the difference between the two reactivity evolutions. The last three indicators, $\Delta_{\text{at minimum}}$, $\Delta_{\text{at Xe peak}}$, $\Delta_{\text{at EOC}}$ give a detailed evaluation of three key moments in the reactor cycle: at the minimum of the rods (important for safety), at the ^{135}Xe –^{149}Sm peak (important for the start-up) and at EOC (important for the cycle duration). The results of the indicators for the reactor cycles of the years 2012 till 2015 are listed in Table III.

Figure A.1 in Appendix A and the results listed in Table III for the different indicators show that the difference between values predicted by the MCNP-based method and measured values remains limited. Since predicting the reactivity evolution as a function of energy produced

involves several iterative operations in a detailed three-dimensional mesh model (in total about 2500 fuel and beryllium burn-up meshes):

- spatial flux distribution calculations over the whole reactor core,
- reaction rates calculations in hundreds of material zones (fuel, beryllium, experiments, ...)
- burn-up calculations for all these material zones,

making a good prediction of the reactivity curve implicitly means that all these different operations had to be performed adequately.

Table III. Evaluation indicators of the reactivity evolution as a function of produced energy of the MCNP based predictions compared to the measured evolution for cycles 01/2012 to 01/2015, see Figure A.1 in Appendix A.

year	cycle	Δ_{max} [$]	$\Delta_{average}$ [$]	$\Delta\|_{average}\|$ [$]	$\Delta_{at\ minimum}$ [$]	$\Delta_{at\ Xe\ peak}$ [$]	$\Delta_{at\ EOC}$ [$]
2012	1	0.34	-0.11	0.13	-0.02	0.05	-0.05
2012	2	1.27	0.32	0.34	0.17	-0.15	1.27
2012	3	0.79	0.63	0.64	0.72	0.23	0.73
2012	4	0.42	0.20	0.22	0.39	0.24	0.06
2012	5	-0.51	-0.28	0.29	-0.50	-0.02	0.23
2013	1	0.63	0.10	0.11	0.07	0.47	0.33
2013	2	-1.00	-0.43	0.48	-0.06	-0.09	0.09
2013	3	-0.55	-0.33	0.33	0.01	0.10	-0.08
2013	4	0.41	0.02	0.11	0.13	0.10	0.27
2013	5	-0.53	0.03	0.11	0.04	-0.09	0.53
2013	6	0.44	0.09	0.19	0.33	0.38	0.40
2014	1	1.41	-0.09	0.50	-0.11	0.23	1.36
2014	2	-0.72	-0.44	0.45	-0.05	-0.11	-0.38
2014	3	-0.89	-0.47	0.50	-0.37	0.06	0.28
2014	5	0.96	0.48	0.49	0.34	0.06	0.96
2014	6	0.68	0.15	0.17	0.20	0.01	0.68
2015	1	0.98	0.52	0.52	0.49	0.38	0.98
range	$\min\{\Delta\}$	-1.00	-0.47	0.11	-0.50	-0.15	-0.38
	$\max\{\Delta\}$	1.41	0.63	0.64	0.72	0.47	1.36
mean	$\overline{\Delta}$	0.24± 0.78	0.02± 0.34	0.33±0 .18	0.11± 0.30	0.11± 0.19	0.45± 0.49
	$\overline{\lvert\Delta\rvert}$	0.74± 0.31	0.28± 0.19	0.33±0 .18	0.24± 0.21	0.16± 0.14	0.51± 0.42

3. Burn-up Determination

The burn-up determination based on MCNP calculations is firstly evaluated by comparison with results of burn-up measurements for three sets of fuel irradiation experiments performed in the BR2 reactor. Two different experimental burn-up measurement methods, viz. γ – spectrometry and radiochemistry, have been used.

A. MCNP vs. γ - Spectrometry

The first fuel irradiation burn-up evaluation is based on results from an irradiation program [1] whereby the fission density of three fuel plates was determined using MCNP calculation results as well as measured by γ – spectrometry. The results for these fission densities are listed in Table IV. As can be seen from this table the results correspond very well. The maximum difference is < 5%.

Table IV. Mean fission density of three fuel plates as determined based on MCNP calculations and as measured by γ – spectrometry.

Method	Mean fission density [fissions/cm³]		
	Plate 1	Plate 2	Plate 3
MCNP based	3.38×10^{21}	4.98×10^{21}	4.89×10^{21}
γ-spectrometry	3.51×10^{21}	5.12×10^{21}	5.10×10^{21}
Difference [%]	3.8	2.8	4.3

B. MCNP vs. Radiochemistry

The second fuel irradiation burn-up evaluation is based on results from an irradiation program [2] whereby the Fissions per Initial Metal Atom (FIMA) of a fuel plate was determined using MCNP calculation results as well as measured by radiochemistry. The results for these fission densities are listed in Table V.

Table V. Fissions per Initial Metal Atom (FIMA) of a fuel plate as determined based on MCNPX calculations and as measured by radiochemistry (see supra 1).

Method	FIMA [%]
MCNP based	11.2
Radiochemistry	10.68
Difference [%]	4.8

The small differences between the calculated and measured values of the burn-up as reported in Table IV and Table V show that the MCNP based calculations are capable of determining fuel burn-up in the BR2 reactor.

[1] The reports of this irradiation program have a restricted status, therefore, no further details of this program can be provided.

[2] See supra 1.

C. MCNP & On-Line Thermal Balance Measurement vs. γ-Spectrometry

The third direct fuel irradiation burn-up evaluation is based on the comparison of the combination of on-line thermal balance measurement and MCNP-based calculations versus γ-spectrometry measurements for the GERONIMO fuel irradiation campaign, see Ref. [11-12]. During this campaign fuel irradiation transient tests have been performed for MOX fuel rod in the PWC/CCD device loaded in BR2 reactor channel. The activity of certain isotopes of suitable half-life (e.g. ^{140}Ba, ^{140}La) can be measured by γ-spectrometry and, taking into account the appropriate factors, be converted into linear power, see Ref. [11-12]. Table V presents the results of these comparison for four fuel rods irradiated in the PWC/CCD device during the GERONIMO fuel irradiation campaign.

Table V. Comparison of the on-line power determination (based on combination of thermal balance measurements and MCNP-calculations) with the γ-spectrometry measurements, see Ref. [11-12].

Fuel rod	BR2 power	Fission rate	Total fuel power	Average linear fuel power after the transient		
		Measured by γ-spectrometry	Measured on-line by thermal balance & MCNP	Measured by γ-spectrometry	Measured on-line by thermal balance & MCNP	Difference
	[MW]	[s^{-1}]	[W]	[W/cm]	[W/cm]	[%]
1	14.0	3.924×10^{14}	12510	301.3	297	+1.4
2	20.5	5.797×10^{14}	18482	446.3	425	+4.8
3	20.7	5.537×10^{14}	17653	423.5	426	-0.6
4	16.6	5.068×10^{14}	16158	388.7	369	+5.1

D. Reactivity Effects of Fuel Elements with Different Burn-Up

Next to the first evaluation of the burn-up determination performance of the MCNP-based method, viz. by comparison with three series of experimental burn-up measurements using γ spectrometry or radiochemistry, the evaluation of the burn-up can be performed indirectly by comparison of the reactivity effect of loading fuel elements with various burn-up in the same BR2 reactor channel as measured and as calculated by MCNP. Only if the burn-up is modelled correctly, the measured and calculated reactivity effects will correspond.

In the framework of the conversion of the BR2 reactor from HEU to LEU fuel system, an irradiation campaign was performed with two HEU fuel elements not having the standard burnable poisons in the fuel meat (B_4C and Sm_2O_3) but instead having burnable poisons in the stiffeners (cadmium wires), see Refs. [4-5] and Figure 4.

The two fuel elements with cadmium stiffeners were irradiated for 5 reactor cycles and the reactivity effect of the burn-up of the fuel element was measured experimentally. These results were compared with the calculated reactivity effects with the MCNP-based method, see Ref. [5]. Figure 5 shows the results of this comparison. The evolutions of the reactivity vs. the burn-up as measured experimentally and as calculated by the MCNP-based method correspond. The differences were somewhat larger at the end-of-life of the fuel element which was due to change of the foreseen surrounding conditions (N.B. The MCNP calculations were performed in advance to the measurement for slightly different surrounding conditions of the tested FE).

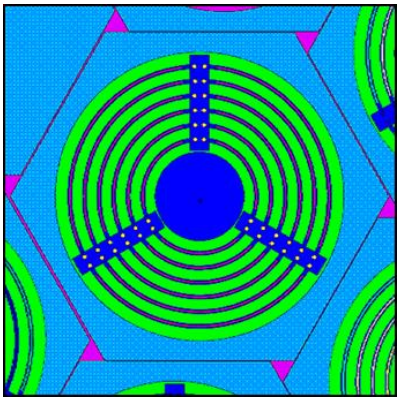

Figure 4. Cross-section of the MCNP model of a HEU fuel element with Cd wires as burnable absorber, see Ref. [5].

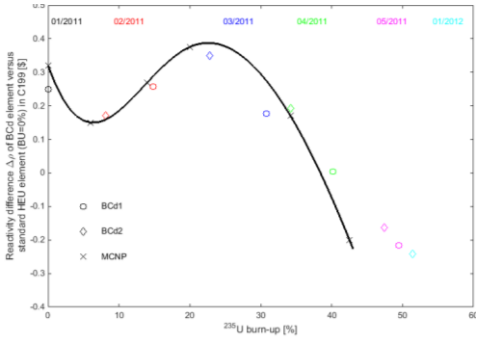

Figure 5. Reactivity effects as a function of burn-up of a HEU fuel element with cadmium wires as burnable absorber, see Ref. [5].

4. Comparison with Dosimetry Results

For thermal flux dosimetry, measurements have been made during cycle 02/2001.A using ^{59}Co foils, see Ref. [10,13]. The central reactor channel H1 and two more

peripherally situated reactor channels, viz. G60 and G300, were monitored. Figure 6 shows the axial distribution of the fission power as calculated and as measured (by γ-spectrometry) in the fuel plate located in channel G300. The measured and axial distributions are for each case in good agreement.

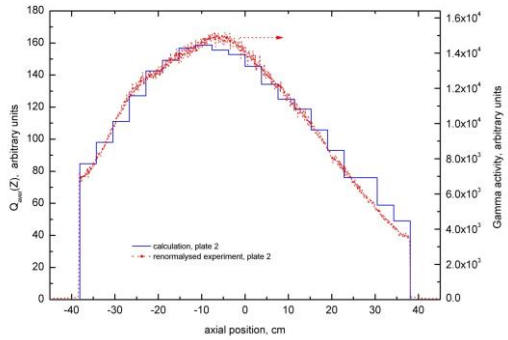

Figure 6. Axial distribution of the calculated and measured fission power for the fuel plate in channel G300 during cycle 02/2001.A, see Ref. [10,13].

5. Cross-code Comparison of Isotopic Fuel Densities

The calculated with MCNPX/CINDER90 time evolutions of fuel isotopes have been cross-checked with the ALEPH code, see Ref. [14]. The time evolutions of major fuel isotopic densities during continuous irradiation in an infinite BR2 lattice cell are compared in Fig. 7, which shows a good agreement between both codes.

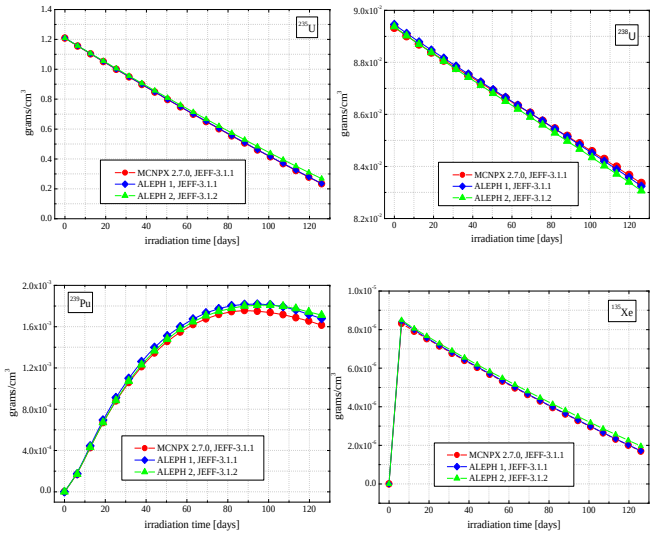

Fig. 7. Comparison of major fuel isotopic evolutions between MCNPX and ALEPH during continuous irradiation in an infinite lattice.

VI. CONSIDERED IRRADIATION SCENARIOS FOR LTA'S

1. Loading Map of Cycle 02/2016A

The load for the BR2 cycle 02/2016A, which was operated from 06 September to 28 September 2016 was used as a reference for the considered irradiation scenarios of the LTA's. The loading map of the cycle 02/2016A is given in Fig. 8. The major purpose of the irradiation will be to determine the reactivity effects of each LTA vs. ^{235}U burn-up in per cent. The typical load of a BR2 irradiation cycle contains 5-6 fresh elements loaded in channels C, and the rest of the channels are loaded with 2nd, 3rd and 4th or 5th batch fuel elements, having average ^{235}U burn-up between 10% and 50%. The fuel elements loaded in the strongest channels A, B and H1/Central undergo higher burn-up toward EOC than fuel elements loaded in channels C, D and peripheral channels F, G, H.

With a purpose to obtain more points on the LTA reactivity performance curve the following candidate channels for irradiation of the LTA's have been chosen: 'C', 'D', 'F' and 'G'. It should be noted, that the BR2 core load is very flexible and the loading channels can be adjusted depending on the reactor core load of the real irradiation cycle for the LTA's.

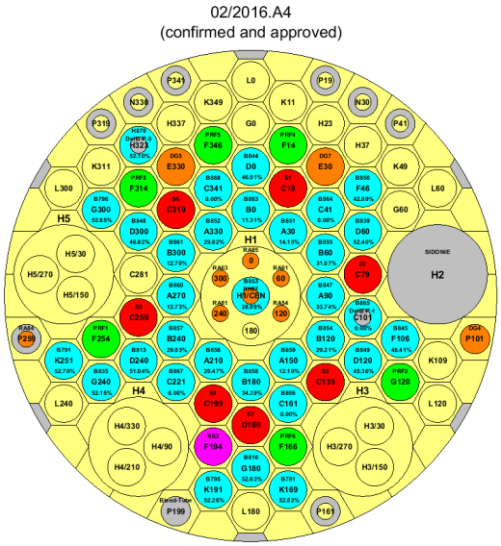

Fig. 8. Possible irradiation channels C, D, F, G for the LTA's in the loading map of cycle 02/2016A (blue color – fuel elements, red color – control rods, green & gray – irradiation experiments, yellow – beryllium matrix).

2. ^{235}U burn-up Evolution in Different Scenarios

Several irradiation scenarios have been considered which are described in Table VI and illustrated in Fig. 8. The motivation of the selected scenarios is to provide as much as possible similar irradiation conditions for the LTA's. It is

seen from Table VI that the final LTA's ^{235}U burn-ups [%] after 5 irradiation cycles in channels 'C' (Scenario 1) are similar to those after 6 irradiation cycles in channels 'D' (Scenario 3). Because the aim is also to obtain as much as possible points on the reactivity performance graph, it seems that the preferable scenario would be Scenario 3. The loading of the second LTA-COBRA-GD2 in the peripheral channel F46 is aimed to obtain intermediate points on the reactivity performance curve.

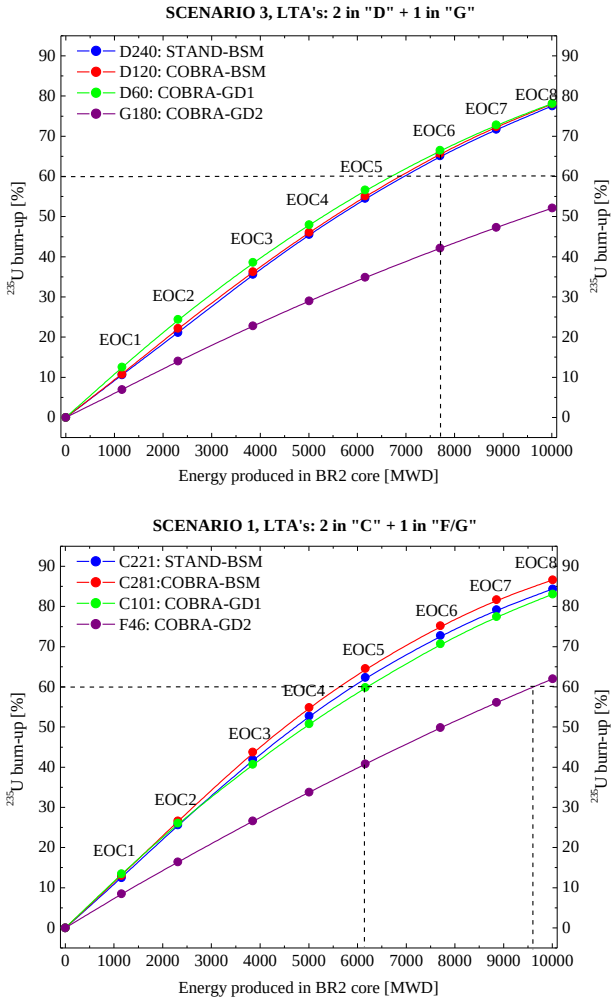

Fig. 8. Evolution of ^{235}U burn-up [%] as function of total produced energy [MWD] in the BR2 reactor core during irradiation in different scenarios.

The fuel burn-up evolution for the different scenarios was followed (calculated by MCNP6) during 8 irradiation cycles. The comparison of the ^{235}U burn-ups after each EOC_i (i=1,2,..,8 cycle) for Scenario 1 and for Scenario 3 is given in Fig. 8.

Table VI. Considered irradiation scenarios for the LTA's.

Scenario N°		Standard FE	LTA		
			COBRA BSM	COBRA GD1	COBRA GD2[3]
1	Irr. channel	C221	C281	C101	F46
	N° cycles	5	5	5	7
	U5 burn-up [%] at discharge	62.5	64.6	60.0	57
2	Irr. channel	C221	C281	D60	F46
	N° cycles	5	5	6	7
	U5 burn-up [%] at discharge	61.6	64.2	65.4	57
3	Irr. channel	D240	D120	D60	F46
	N° cycles	6	6	6	7
	U5 burn-up [%] at discharge	65.0	65.0	66.5	57

3. Reactivity Performances of LTA's

The final purpose of the LTA's irradiation is to compare the reactivity effects of fresh and burnt LTA's relatively to a standard fresh BR2 fuel element. The calculated/"measured" reactivity effects should be always compared in one and the same channel. In the present paper all calculated reactivity effects of the LTA's with different ^{235}U burn-up are compared in the channel C101.

A. Reactivity Performance of LTA-COBRA-GD

The reactivity performance curves of the LTA-COBRA-GD acquired during all irradiation cycles in a given (same for all cycles) channel A, C, D or G are presented in Fig. 9. The calculated/"measured" reactivity effects are performed always in the same channel C101. It is interesting to note that the reactivity effects of an LTA with a given ^{235}U burn-up are different for irradiation history acquired in different channels. For instance, LTA with one and the same ^{235}U burn-up of 6% has reactivity effect of 1.38 $, if the LTA is irradiated in all cycles in a channel 'A', 1.1 $ if irradiated in a channel 'C', 1.0 $ if irradiated in a channel 'D' and 0.95 $ in a peripheral channel 'F'.

The reason for the lower reactivity values of the LTA acquired in peripheral channels is due to higher fission products accumulation in more thermalized neutron spectrum compared to the irradiation channels close to the core centre.

The "ideal" irradiation of the LTA's would be in one and the same channel, which is not possible to accomplish in one cycle, but at least the irradiation of the LTA-COBRA-GD1 and LTA-COBRA-BSM should be performed in one and the *same type* of channel 'C' or 'D'.

[3] Two alternative channels F46 or G300 can be considered.

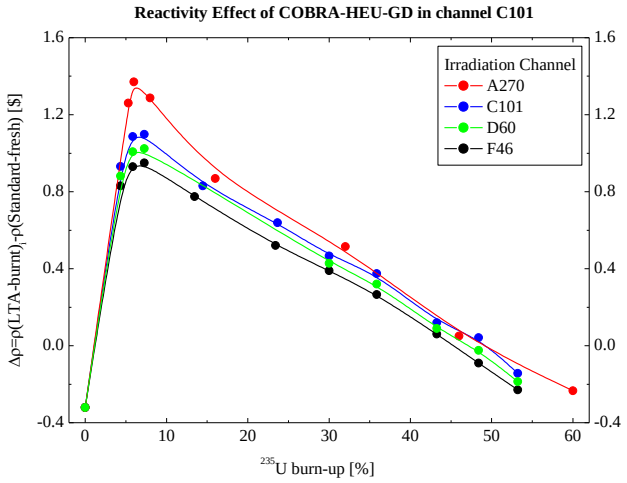

Fig. 9. Calculated reactivity effects of LTA-COBRA-GD in channel C101 relatively to standard fresh BR2 fuel element.

B. Comparison of Reactivity Performance of LTA-COBRA-GD vs. LTA-COBRA-BSM

The reactivity effects of LTA-COBRA-GD1 and LTA-COBRA-BSM acquired in a same type channel 'C' are compared relatively to standard fresh HEU fuel element. As before, the reactivity effects are calculated/"measured" in a same channel C101. It is seen from Fig. 10 that the fresh LTA-COBRA-GD has a negative reactivity effect compared to standard fresh fuel element, which is related to the high absorption cross section of gadolinium. However, after the first couple of days, the reactivity effect of LTA-COBRA-GD quickly increases due to the high burn-up rate of ^{157}Gd (see Fig. 1).

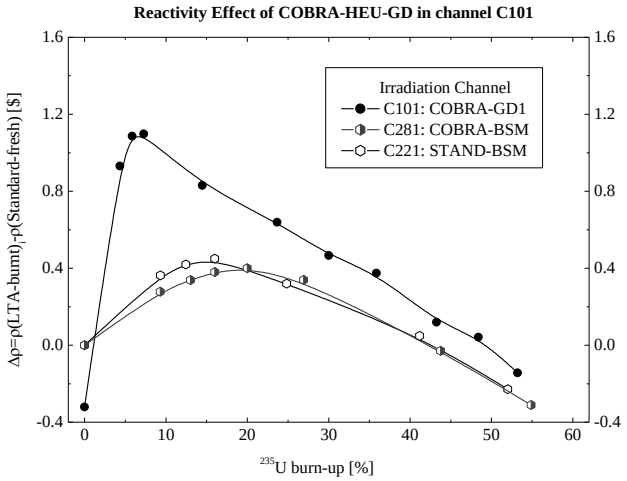

Fig. 10. Comparison of reactivity effects of LTA-COBRA-GD and LTA-COBRA-BSM vs. standard BR2 HEU fuel element.

The reactivity effects of burnt LTA-COBRA-GD are always significantly higher compared to standard burnt fuel elements, which is explained with the higher burn-up rates of Gd-isotopes compared to boron and samarium isotopes in the standard BR2 HEU fuel. From the other hand, the reactivity performances of LTA-COBRA-BSM and standard BR2 HEU fuel element are similar due to similar fuel system parameters (see Table I).

4. Impact of LTA's on Core Reactivity

Up to now we were interested providing best irradiation conditions for the LTA's, however we always aim to optimize the fuel utilization. For this purpose, we can use the LTA's irradiation to make savings of our standard HEU fuel elements. For instance, the three LTA's can replace 2 or 3 standard BR2 HEU fresh fuel elements and in addition to this we gain reactivity excess during the first three irradiation cycles according with Fig. 11. At the end of the presumed 5th or 6th irradiation cycle the reactivity loss due to the highly burnt LTA's is less than 1 $ in all three considered scenarios which can be easily compensated with the BR2 flexible reactor core loadings.

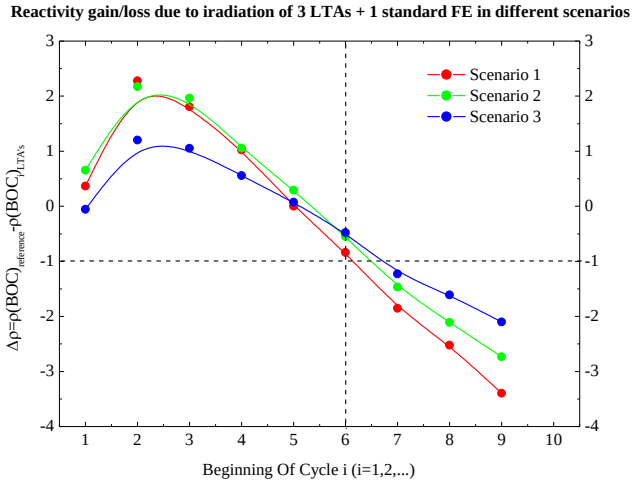

Fig. 11. Impact of LTA's reactivity evolution on the core reactivity.

5. Fuel Economics: Fuel Utilization

The economy advantage of the gadolinium absorber compared to other burnable absorbers was reported in previous studies [6]. In this section we present an "ideal" scenario for irradiation of a given LTA in one and the same channel 'C' or same channel 'D'. In practice we cannot perform this type of experiment, because the LTA's will occupy in the very best case same *type* channels 'C' or 'D' (even two same type channels may have different irradiation conditions).

The correlation between the ^{235}U burn-up [%] and the consumed energy [GWD/MTU] by LTA has been calculated by MCNP6 assuming continuous irradiation in a same channel C281 (see Fig. 12). The discharge of the burnt fuel

assembly is done at 450 GWD/MTU absorbed by the LTA. From Fig. 12 we see that at one and the same consumed discharge energy, the ^{235}U burn-up in the LTA-COBRA-BSM is about 54%, while in the LTA-COBRA-GD is about 62%, which means that about 32 grams more are utilized in the latter fuel type. So, we may conclude that fuel type with Gd absorber has some economy advantages among the considered fuel types in this paper.

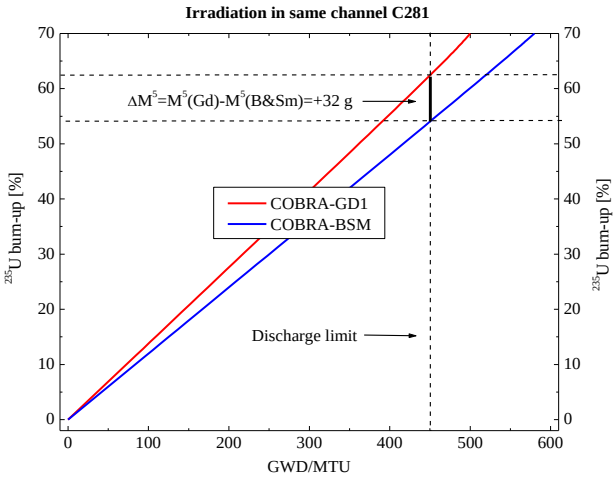

Fig. 12. Comparison of fuel utilization by LTA-COBRA-HEU-GD and LTA-COBRA-HEU-BSM irradiated in same BR2 fuel channel.

VII. CONCLUSION

Previous studies [6] have shown that the economy of the fuel cycle can be significantly improved by using gadolinium poison in a form of homogeneous mixture with the fuel meat. At the same time, the experimental performances for gadolinium were similar as for the standard poisons (boron and samarium) used in the standard BR2 HEU fuel type. In this paper we have compared the reactivity performances of LTA-COBRA-HEU-GD, LTA-COBRA-HEU-BSM vs. standard BR2 HEU fuel element. Several "ideal" scenarios for irradiation of LTA's were considered, assuming compromises in regard to the real BR2 operation needs. The newly performed analyses for the LTA's irradiation confirmed previous conclusions about the economy advantage of Gd absorber.

The reactivity effects of LTA-HEU fuel element with gadolinium absorber are significantly higher compared to boron & samarium. The reactivity effects calculated/ "measured" in same channel 'C' are dependent on the irradiation history acquired in a given channel. Proposed are following scenarios and irradiation channels:

- LTA-COBRA-GD1 & LTA-COBRA-BSM 5-6 cycles, each LTA irradiated in its (same) channel D;
- LTA-GD2 5-6 cycles in a channel F, G or H (used for intermediate points).

APPENDIX A: VALIDATION OF THE MCNP METHODOLOGY

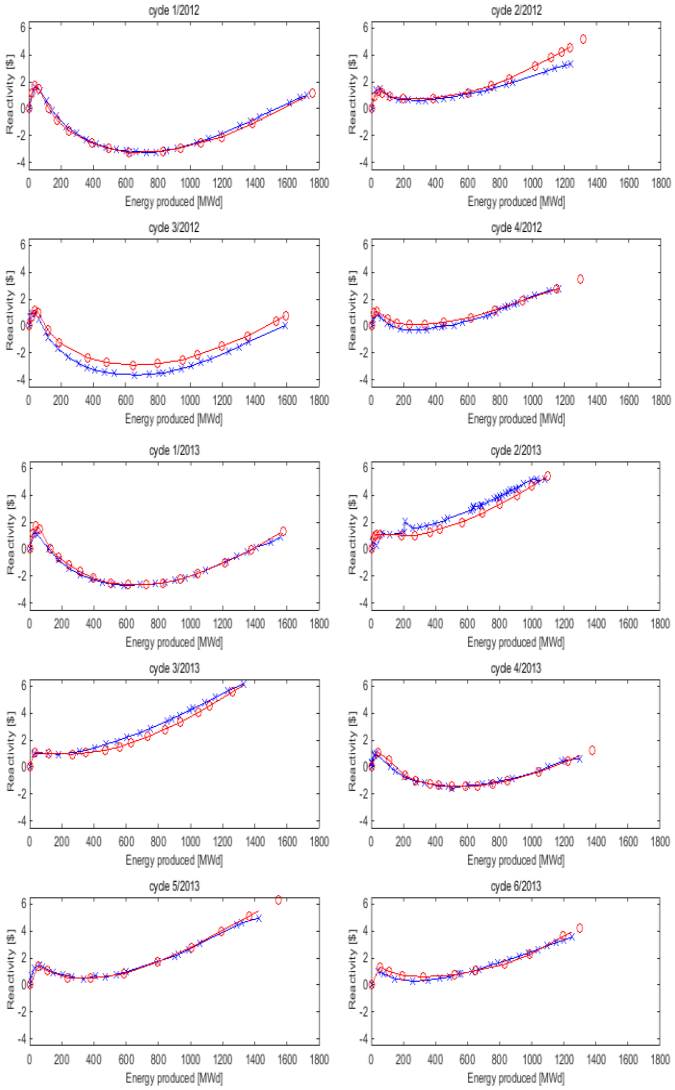

Figure A.1. Examples of reactivity evolution of the BR2 reactor as a function of produced energy for the cycles of the years: 2012 and 2013: as measured (blue), as predicted by the MCNP based simulations (red).

NOMENCLATURE

BOC – Beginning Of Cycle
BR2 – Belgian Reactor 2
COBRA – BR2 fuel element with increased fuel meat thickness
COBRA-BSM – COBRA fuel element with boron and samarium as burnable poison inside the fuel meat
COBRA-GD1 – COBRA fuel element with gadolinium as burnable poison inside the fuel meat
COBRA-GD2 – same as COBRA-GD1
CR – Control Rod

EOC – End Of Cycle
FE – Fuel Element
FIMA – Fissions per Initial Metal Atom
GERONIMO – fuel irradiation program for testing of MOX fuel rods in the BR2 reactor
H1 – beryllium channel (D=200 mm) in the reactor core center of BR2
H1/C – Central channel (D=84 mm) in H1 channel
HEU – High Enriched Uranium
LEU – Low Enriched Uranium
LTA – Lead Test Assembly
MOC – Middle Of Cycle
MOX – Mixed Oxide Fuel
PWC/CCD – Pressurized Water Capsule/Cycling and Calibration Device
refl. – reflector
SCK•CEN – Belgian Nuclear Research Center
STAND-BSM – standard BR2 HEU fuel element with boron and samarium as burnable poison
UMo – Uranium-Molybdenum fuel
RERTR – Reduced Enrichment for Research and Test Reactors
RRFM – Research Reactors Fuel Management
ρ - reactivity
$\Delta\rho$ - reactivity difference

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