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NEUTRONIC CALCULATIONS FOR THE CONVERSION TO LEU OF A RESEARCH REACTOR CORE

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ABSTRACT

For a five-year transitional period the Greek Research Reactor (GRR-1) was operating with a mixed core, containing both Low Enrichment (LEU) and High Enrichment (HEU) Uranium MTR-type fuel assemblies. The neutronic study of the GRR-1 conversion to LEU has been performed using a code system comprising the core-analysis code CITATION-LDI2 and the cell-calculation modules XSDRNPM and NITAWL-II of the SCALE code. A conceptual LEU core configuration was defined and analyzed with respect to the three dimensional multi-group neutron fluxes, the power distribution, the control-rod worth and the compliance with pre-defined Operation Limiting Conditions. Perturbation calculations and reactivity feedback computations were also carried out to provide input to a subsequent thermal-hydraulic study.

1. Introduction

GRR-1 is a 5 MW pool type, light water moderated and cooled Reactor, using beryllium reflector. The GRR-1 core consists of standard and special MTR fuel assemblies. A standard fuel assembly contains eighteen aluminium-clad fuel elements (plates), distributed so as to leave equal spaces between them for the water circulation. A special fuel assembly consists of ten fuel plates, five at each side, with a central space. A control assembly is a special one with a control rod plunged in its central space. Cross sectional drawings of both standard and special assemblies as well as their dimensions and meat composition can be found in [1].

In the GRR-1 core configuration, a special assembly should be placed adjacent to the core centre for material irradiation in high neutron flux and five control assembly locations must be defined where shim/safety rods are to be placed. The active part of the shim control rod has an elliptical horizontal cross-section with a central hollow and its length is equal to the active length of the fuel elements (i.e. 62.55 cm). The absorbing part composed of Ag-107, Ag-109, In-113, In-115 and Cd-113, encircles the hollow with a stainless steel cladding at both edges. Details about the control rod dimensions and composition can be found also in [1]. The later configuration of the mixed GRR-1 core contained 34 fuel assemblies, i.e. 17 standard and 4 control LEU (19.75% U235), and 11 standard, 1 control and 1 special HEU (93% U235), for a normal Reactor operation at 5 MW.

The new conceptual GRR-1 LEU core is analysed using the above-described type of LEU fuel assemblies and control rods. In the new core configuration, the number of control rods and the special fuel assembly adjacent to the core centre (flux trap) are maintained. Since the new LEU core is substantially smaller than the mixed one, due to the utilization of slightly burned and fresh fuel assemblies, the maintenance of 5 MW operation power level induces higher power densities and higher neutron fluxes. However the Reactor power is held at its nominal value of 5 MW, since the thermal-hydraulic calculations indicate compatibility with the operation safety limits[10]. For the fuelling of the new core, 22 already irradiated fuel assemblies (18 standard, 3 control and 1 special) and 5 fresh fuel assemblies (3 standard and 2 control) are used (Table I).

Table I: Irradiated Fuel Assemblies – '6xx' corresponds to standard and '6Cxx' to special/control assemblies

Assembly ID	Burnup (%)	Assembly ID	Burnup (%)
601	9.5	612	10.03
602	14.92	613	10.41
603	1.37	614	6.98
604	15.45	615	6.09
605	13.14	616	5.34
606	15.02	617	1.41
607	11.96	618	1.22
608	12.68	6C01	27.6
609	11.04	6C02	14.63
610	10.73	6C03	12.06
611	12.4	6C04	8.19

2. Neutronic Calculations

The neutronic analysis was performed based on the definition of an operational scheme of the Reactor and its utilization. The main requirements and conditions for the new core configuration include:

a. eight hours of operation daily
b. conservation of the existing beryllium reflector blocks (see Figure 1)
c. core volume as large as possible
d. five control rods for safety and efficient control purposes
e. peripheral water holes as material irradiation positions (at least two grid locations empty)
f. position of most irradiated fuel assemblies near the core center (locations of higher neutron fluxes); position of slightly irradiated and fresh fuel assemblies at the core boundaries
g. presence of fuel assemblies in grid locations A2, A3, A4, A5, F2, F3, F4 and F5 due to adjacent experimental neutron tubes
h. safety margins for simultaneous operation of the six experimental neutron beam tubes
i. "flux trap" (special fuel assembly) at a central core position for material irradiation with the highest possible neutron flux
j. satisfaction of the OLCs registered in [2]

Table II: Definition of Homogenized Zones

Zone Number	Core Constituent	Zone Number	Core Constituent
1	Fresh LEU Standard Fuel Assembly (18 fuel plates)	26	Control Rod
2	Fresh LEU Special Fuel Assembly (Active Part of 5 fuel plates)	27	Beryllium
3-20	Irradiated LEU Standard Fuel Assemblies	28	Water
21 - 24	Irradiated LEU Special Fuel Assembly (Active Part)	29	cladding layer of Standard Fuel Assembly
25	Central Water Box of Special Assembly	30	cladding layer of Active Part of Special Assembly

Based on the above requirements, the configuration of the GRR-1 LEU core shown in Figure 1 was determined after repeated criticality calculations for several tentative fuel-assembly distributions and control rod locations and insertions, until a critical core was obtained.

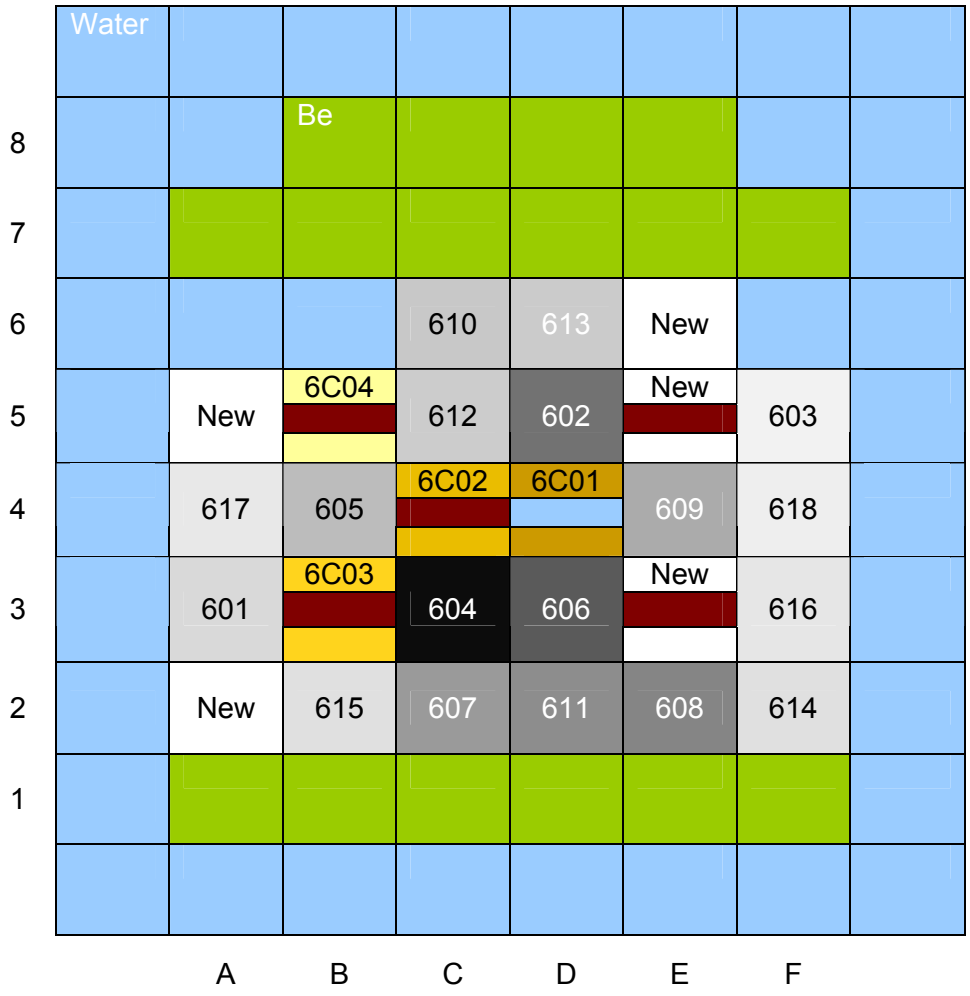

Figure 1: Conceptual LEU Core Pattern. "New" indicates fresh fuel assemblies (white). Grey-scale and yellow-scale variations indicate lower (lighter) to higher (darker) burnup. Dark red indicates the control rod positions. Green and blue indicate respectively beryllium blocks (same or double external dimensions as the fuel assemblies) and water.

Three-dimensional diffusion calculations for the core of Figure 1 were performed with the numerical code system XSDRNPM and CITATION-LDI2 [3, 4], using the 238-group NDF5 library. XSDRNPM provided the microscopic cross sections for each homogenized zone defined for the GRR-1 core (Table II). Resonance calculations for the fertile nuclides cross sections were performed with NITAWL-II [5].

The "operational core" calculations mentioned throughout the text correspond to a fuel temperature of 400K and moderator temperature of 300K. Six axial meshes were used to simulate the core in the vertical direction, including: above- and below- core pool water; upper and lower cladding of fuel elements; active core with control rods inserted; part of active core below control rods. The simulated core was surrounded by a 20cm-thick water layer. Five neutron energy

groups were defined for the calculations, with their upper bounds (in eV) being respectively at 5.3158E-01, 1.4450E+00, 4.3074E+03, 7.8082E+05, 1.9640E+07.

2a. Criticality and Reactivity Calculations

Criticality calculations for the operational core with the five control rods fully inserted gave the effective multiplication factor $k_{eff} = 0.929024$, i.e., $\frac{\Delta k}{k} = \frac{k_{eff} - 1}{k_{eff}} \cong -0.0764$, which (based on the calculated fraction of delayed neutrons, β_{eff}), corresponds to a reactivity $\rho \cong$ -8.97\$. Calculations with the control rods 100% withdrawn provided $k_{eff} = 1.065091$, i.e. $\frac{\Delta k}{k} \cong 0.0611 \cong 7.17\$$. The core becomes critical (k_{eff} = 1.000046) with all the control rods inserted at ~53% of their active length.

To account for the reactivity requirements for the core normal operation, calculations were performed concerning the two main processes that introduce negative reactivity, i.e. the xenon build-up after shutdown and the partial substitution of lateral pool water with air, assuming that all neutron beam tubes contain experimental devices instead of water.

If xenon and iodine have reached equilibrium prior to shutdown, the reactivity equivalent of the xenon is given as function of the daily Reactor shutdown duration [6]. Assuming eight hours of continuous Reactor operation per day and using the calculated average thermal flux for the core of Figure 1, $\overline{\Phi}_{th} = 3.21\text{x}10^{13}$ n/(cm^{2}s), the reactivity equivalent of the xenon was found $\rho_{xe} = -0.0366 \cong -4.30\$$. Adopting the value of $\rho = 0.0270\,(or \cong 3.17\$)$ for the worth of equilibrium xenon in GRR-1 [7], and taking into consideration that this quantity is included in the cycle calculations of CITATION-LDI2, the additional reactivity needed for the xenon override is 1.13\$.

The negative reactivity due to the air-filled experimental tubes was estimated from criticality calculations for the core shown in Figure 1, in which air was substituted for pool water contained in the tubes at both core sides (i.e. left of row A and right of row F). The water channel of different widths between the tubes and the core was also taken into account, together with the inclination of each tube with respect to the core side. The induced negative reactivity was found $\frac{\Delta k}{k} \cong -0.0074 \cong -0.87\$$.

Based on the above, the sum of reactivity allowances is found $\Delta\rho = 1.13\$ + 0.87\$ = 2.00\$$, which is well below the calculated excess reactivity of 7.17\$.

2b. Thermal Neutron Flux and Power Distribution

Figures 2a,b show two representative profiles of the calculated thermal neutron flux (energy group 5) that include the thermal neutron flux maximum. The latter is found in grid position D4, very close to the boundary with C4, about 8 cm below the control rods. In the above graphs, the surrounding pool water is not included. Figures 3 and 4 show respectively the horizontal

distribution of the thermal neutron flux at the level of maximum and the average flux distribution per fuel assembly in the Reactor core (beryllium blocks and pool water are not shown). Both Figures include only the active part of the core. The values shown in Figures 3 and 4 correspond to a critical Reactor core operating at 5 MW.

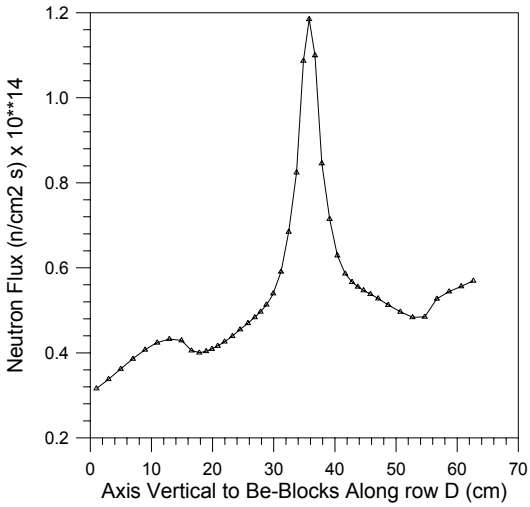

Figure 2a: Profile of thermal neutron flux along row D, passing from the thermal flux maximum

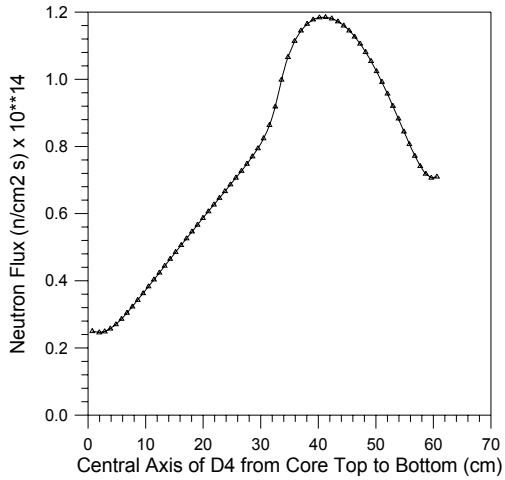

Figure 2b: Vertical profile of thermal neutron flux, close to the boundary of D4 channel with C4, passing from the thermal flux maximum

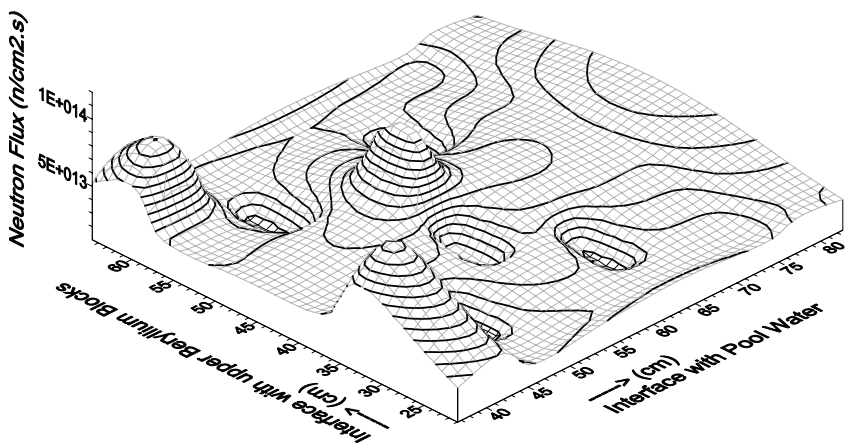

Figure 3: Horizontal distribution of the thermal neutron flux at a representative level with all rods deepened, for 5 MW Reactor power. Equal flux contours from 1.5×10^{13} n/cm^{2}s to 1.15×10^{14} n/cm^{2}s are depicted per 0.5×10^{13} n/cm^{2}s

Figure 5 shows the distribution per fuel assembly of the calculated power in the Reactor core. Only the active core part is included in Figure 5, while the values correspond to a critical Reactor core operating at 5 MW.

	A	B	C	D	E	F
6			3.25 9.03	3.07 10.28	2.94 8.50	
5	2.16 5.36	3.27 8.01	3.34 11.15	3.65 12.06	3.24 10.16	2.70 6.93
4	2.34 7.18	3.00 12.73	4.08 11.85	5.15 6.97 12.63	3.52 11.66	2.72 8.38
3	2.54 7.40	3.32 10.37	3.85 12.88	4.04 13.32	3.51 11.25	2.75 8.16
2	2.36 6.68	2.95 9.90	3.58 11.86	3.64 12.04	3.14 10.28	2.53 7.01

Figure 4: Average neutron fluxes distribution per fuel assembly [n/(cm^{2}s)x10^{13}] for 5 MW Reactor power: Thermal neutron flux (upper value), epithermal neutron flux (lower value), thermal neutron flux in D4 flux-trap

	A	B	C	D	E	F
6			208.6	199.8	205.4	
5	149.2	119.5	217.8	227.7	128.0	184.7
4	162.1	190.9	142.8	157.4	226.9	188.4
3	164.7	119.5	239.4	251.9	138.8	184.2
2	164.4	198.5	229.5	232.9	200.0	167.0

Figure 5: Average power distribution in kW per fuel assembly for 5 MW Reactor power

2c. Perturbation Calculations

Three dimensional perturbation calculations were performed for the operational core using the five neutron energy groups defined in Table IV. CITATION-LDI2 code was used to obtain the effective delayed neutron fraction β_{eff} and the prompt neutron lifetime ℓ. The effective reciprocal velocity cross sections required by CITATION-LDI2 for the prompt neutron lifetime calculation were computed for each fuel assembly from XSDRNPM. The utilized relative yield β_i/β for six energy groups of delayed neutrons, as well as the total delayed neutron fraction β and the number of delayed neutrons per fission, consecutive to fission of U-235 by thermal neutrons, were taken from [8]. The mean generation time of prompt neutrons was estimated from $\Lambda = \ell/k_{eff}$ [8]. The delayed neutrons energy distribution fraction for each neutron energy group was determined based on the energy spectrum of delayed neutrons from thermal fission of U-235 found in [9]. The results obtained for the critical operational core are given in Table III.

Table III: Delayed Neutron Dependent Parameters

k_{eff}	β_{eff}	ℓ (s)	Λ (s)
1.000046	8.52x10^{-3}	37x10^{-6}	37x10^{-6}

2d. Calculations for Reactivity Feedback

The effective multiplication factor k_{eff} for the core configuration of Figure 1 was calculated by CITATION-LDI2 for varying fuel and moderator temperatures, as well as for various moderator densities. The fuel temperature was modified keeping constant the moderator temperature at 300K. The calculations for increasing moderator temperature were performed with constant fuel temperature of 400K. The modifications of the moderator density in the whole core volume were made for the operational core, i.e. 400K for the fuel and 300K for the moderator. XSDRNPM and NITAWL-II were used in each case to compute the nuclides cross sections for the defined five neutron energy groups. The obtained results are shown in Figures 6 - 8.

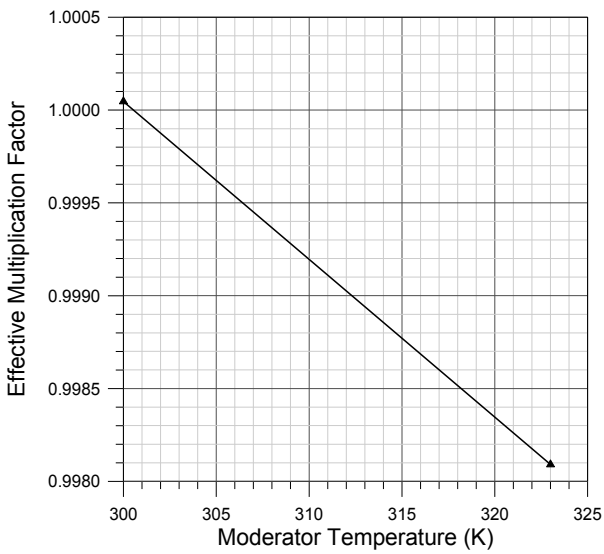

Figure 6: Effective multiplication factor as function of the moderator temperature

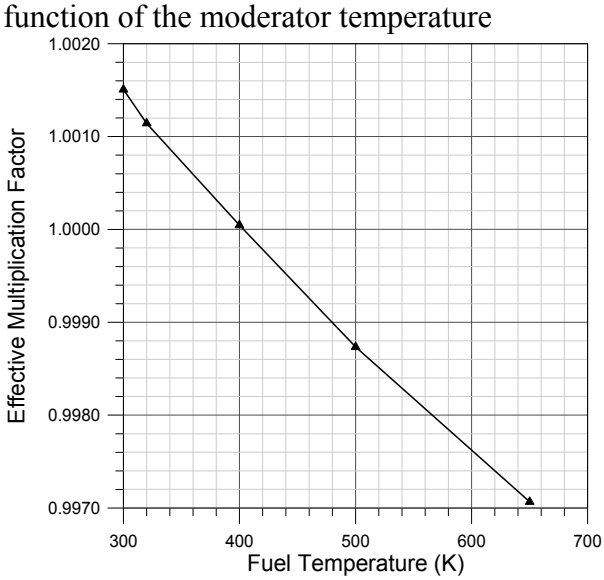

Figure 7: Effective multiplication factor as function of the fuel temperature

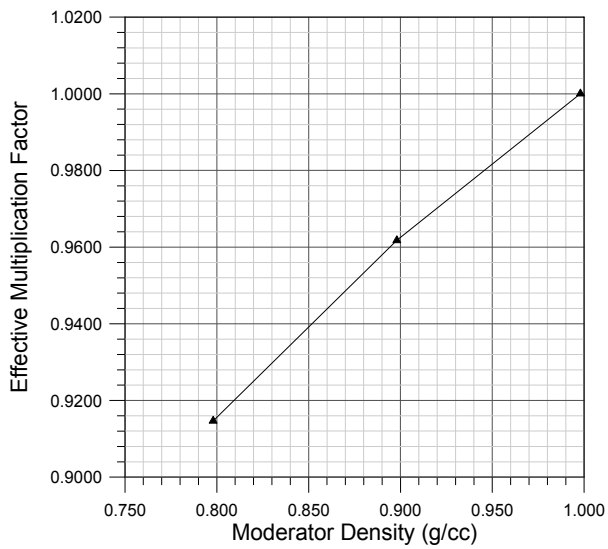

Figure 8: Effective multiplication factor as function of the moderator density

2e. Calculations of Control Rods Worth

Three dimensional calculations were performed with the CITATION-LDI2 code, using the operational core data, to estimate the worth of the five control rods of the GRR-1 conceptual core (Figure 1).

The control rods numbering (with respect to the core grid) and their absolute worth are shown in Table IV. The total reactivity worth of the cluster of five control blades as a function of their common withdrawal is shown in Table V.

Table IV: Control Rods Numbering

Control Rod Number	Core Grid Position	Control Rod Absolute Worth (%Δk/k)
1	B3	2.35
2	C4	3.47
3	B5	1.61
4	E3	2.60
5	E5	2.17

Table V: Cluster of Five Control Rods Worth

Control Rods withdrawal	Effective multiplication factor	Δk/k (%)
0	0.929024	-7.64
47.2	1.000046	0.00
100	1.065091	6.11

3. Compliance with the Predefined OLC's

The compliance of the conceptual core with the Limiting Conditions for Operation (OLCs) appearing in the Safety Analysis Report (SAR) of GRR-1 [2] is shown in Table VI.

Table VI: Calculated OLCs Compared with SAR

Specifications	SAR OLCs (% Δk/k)	Conceptual LEU Core (% Δk/k)
overall core excess reactivity	6.5 maximum	6.1
shutdown margin (Rods fully inserted)	3.5 minimum	7.6
shutdown margin with the most reactive shim-safety rod withdrawn.	0.7 minimum	3.8 (No 2 Rod 100% withdrawn)
total reactivity worth of experiments	1.6 maximum	0.74
Sub-criticality during approach to criticality with least reactive rod withdrawn	2.0 minimum	11.8

4. Conclusions

A conceptual LEU core configuration for GRR-1 was defined and analyzed with respect to the three dimensional multi-group neutron fluxes, the power distribution, the control-rod worth and the compliance with pre-defined Operation Limiting Conditions. The core contains fresh as well as already irradiated MTR-type fuel assemblies of various burnup levels. Requirements on the utilization schedule, the use of Be blocks, the availability of beam tubes and the existence of central and peripheral sample irradiation positions are fulfilled. Perturbation calculations and reactivity feedback computations were also carried out to provide input to a subsequent thermal-hydraulic study [10].

5. References

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