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TOUTATIS : ILL CONVERSION FEASIBILITY STUDY

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ABSTRACT

The Institute Laue-Langevin (ILL) has launched conversion studies in order to evaluate the impact of operation with LEU UMo fuel instead of HEU. The Réacteur à Haut Flux (RHF) based in Grenoble, France, is a research reactor designed primarily for neutron beam experiments for fundamental science. It delivers one of the most intense neutron fluxes worldwide. The RHF has a single fuel element with 280 involute-shaped fuel plates.

Neutronic and thermal hydraulic studies have been carried out in close collaboration between ILL and Argonne National Laboratory (ANL). Preliminary studies in which the HEU fuel meat was simply replaced by LEU UMo dispersion indicated that such a direct replacement would cause unacceptable performance penalties for core lifetime and experimental brightness.

ILL and ANL have collaborated to investigate alternative LEU designs. A promising candidate has been selected and studied, which increases the total amount of fuel without changing the external plate dimensions by relocating the burnable poison. This proposed LEU design has been called TOUTATIS and is predicted to provide cycle length comparable and brightness well within 10% of current operation. Preliminary thermal hydraulic safety margins have been evaluated with favorable results, as described in a companion paper.

This paper will describe the general conversion analysis project and the neutronic results of feasibility analyses.

INTRODUCTION

The Institute Laue-Langevin (ILL) in Grenoble (France) produces one of the most intense thermal neutron flux in the world (being $1.5.10^{15}$ n/s/cm2) thanks to its Réacteur à Haut Flux (RHF). The reactor has only one fuel element, which is made out of curved plates of Highly Enriched Uranium (HEU), and is moderated by heavy water.

The reactor has been conceived to operate continuously for 46 day cycles with a thermal power of 58.3 MW (nuclear power 57 MW). However, in order to improve the fuel cycle efficiency, ILL has recently decreased the power of the reactor to a thermal power of 53.3 MW (nuclear power 52 MW) so that the reactor can operate for 49-50 day cycles. Under current normal circumstances, there are 4 cycles each year providing up to 200 days of neutrons for scientific applications. Figure 1 is a scheme representing a 3D side view of the reactor.

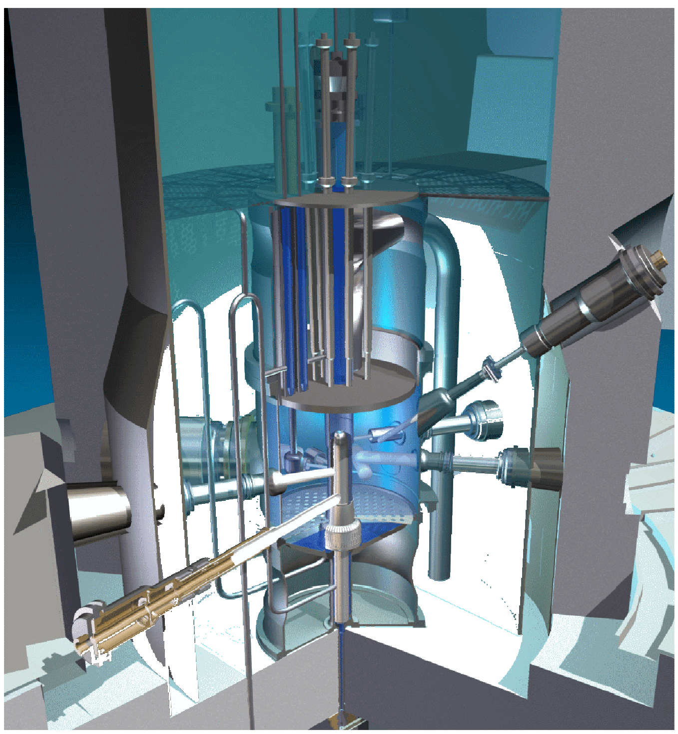

Figure 1 : 3D side view of the RHF reactor

The neutrons are extracted from the pile using beam tubes and sent to the ILL instruments. The reactor is cooled, moderated and reflected by heavy water. It has only one fuel element, based upon the Oak Ridge National Laboratory High Flux Isotope Reactor (HFIR) design. RHF is composed of three concentric regions. The heavy water tank contains the fuel element and has a diameter of 2.50 m. It is placed in the bottom part of a light water pool which has a diameter of 6 m and a height of 14 m. This pool is placed at the center of a 60 m diameter cylindrical building. The reactor is mainly used for fundamental research, employing 13 horizontal and 4 inclined beam tubes which extract neutrons. Hot and cold neutrons are produced by graphite and liquid deuterium volumes set up in the heavy water tank and linked to beam tubes. Several beam tubes extend through the building's heavy concrete walls.

PROJECT OVERVIEW

The ILL conversion study into LEU was launched several years ago. Unfortunately, previous studies had shown that, due to its compact core design and high power density, the RHF could only be converted to an LEU fuel (enriched at <20%) if the fuel density was in a range of 7-9 g/cm3 [1] which was not possible with the existing fuel. However, the recent development of a new kind of high density (7.5-8.5 g/cm3) dispersed fuel based on a mixture of uranium – molybdenum (UMo) could allow the RHF conversion. ANL and ILL have collaborated to investigate alternative designs of the LEU fuel to maintain safety margin, to minimize the fuel element design changes, and to maintain the RHF performance in terms of both neutron flux and cycle length.

Nevertheless, the UMo dispersed fuel is not yet experimentally validated. The current HEU fuel is a UAlx powder dispersed in an aluminum matrix. The new LEU UMo fuel proposed is also based on a dispersed technology, a UMo powder mixed in an aluminum matrix. Prior experiments have shown that dispersion UMo fuel can suffer a problem of swelling under irradiation [2]. Fortunately, the swelling problem is controlled well by the addition of Silicon to the aluminum matrix. That is the reason why the ILL decided to participate to the LEONIDAS (Low Enriched Optimized fuel Nuclear Irradiation group between DOE And European Structure) initiative. The initiative LEONIDAS is in charge of dispersion UMo development and qualification in Europe. It includes the French Atomic Energy Commission CEA, the French company AREVA, the European laboratory ILL and the Belgian laboratory SCK-CEN. It is scheduled to qualify the ILL plates in the SCK-CEN reactor BR2 [3]. The ILL conversion is thus strongly dependent upon the success of the LEONIDAS irradiation experiments.

Thanks to a huge amount of work, ILL and ANL were able to assess and minimize the performance losses due to RHF conversion (see below). Because the losses seem acceptable for the proposed LEU fuel design, the ILL has decided to launch a project to prepare for conversion into LEU, provided that the UMo fuel is experimentally validated and that LEU operations are licensed by the French Nuclear Safety Authority.

Performance Evaluation

The impact of a possible conversion on the reactor performances has been assessed by detailed neutronic calculations. Two criteria were used to evaluate the performance losses due to the conversion: the neutron flux/ brightness; and the cycle length. The neutron brightness is a flux divided by an energy range and a solid angle and was recorded at the entrance of the ILL neutron guides, within the beam tubes. Thus, the Figure of Merit which serves as a basis for our performance evaluation is based upon variation of both the neutron fluxes and brightness in key locations. A normalized weight, which corresponds to the number of allocated beam time days at each location, has been applied to the calculation results. The nuclear reactor power, related to the cycle length, is taken into account in the flux / brightness normalization factor as a proportional value (in first approximation).

Thermal hydraulics calculations have also been carried out in order to check the feasibility of the conversion and its impact on the steady state safety margin. This has been detailed in another RERTR 2010 paper [4]

CREDIBILITY OF THE HEU NEUTRONICS AND T-H MODELS

The ILL team in charge of the calculations, the Projects and Calculations Office (Bureau de Projets et Calculs in French, BPC) mainly uses the codes MCNPX 2.6f (Monte Carlo) and VESTA 2.00g (depletion) [5],[6]. The ILL team utilizes ENDF/B7 cross-sections with a customized heavy water thermal neutron scattering kernel, $S(\alpha,\beta)$. The ANL Global Threat Reduction Initiative team carried out parallel studies in support of the ILL BPC, using MCNP5 1.51 (Monte Carlo) and REBUS-MCNP (depletion) with the libraries ENDF/B6 and/or ENDF/B7 [7], [8]. Depletion calculations at ANL were performed in REBUS-MCNP with a newly developed, explicit list of 90 fission products.

VESTA is a Monte Carlo depletion interface code written in C++ that is currently under development at IRSN (Institut de Radioprotection et de Sûreté Nucléaire). From its inception, VESTA is intended to be a "generic" interface code so that it will ultimately be capable of using any Monte Carlo (MC) code or depletion module and that can be tailored to the user's needs on practically all aspects of the code. In our case we chose to couple MCNPX 2.6f and ORIGEN 2.2.VESTA employs a scheme to reduce the time of simulation in continuous energy Monte-Carlo codes (like MCNPX) by using multigroup binning to calculate the reaction rates after the MCNPX is finished. For this scheme, only an ultra-fine multigroup spectrum has to be calculated by the Monte-Carlo code. The default group structure consists of 43000 groups. The reaction rates are then determined by collapsing multigroup cross sections to a single group using the calculated spectrum. Due to the nature of the ultra fine spectrum, this method takes into account the self shielding effects due to the resolved resonance range. The implementation of this alternative method for Monte Carlo depletion allows a significant reduction in calculation time in the MCNP simulations relative to more traditional approaches such as REBUS-MCNP that require detailed reaction rate tallies.

VESTA also offers users a wide array of general modeling features for a complete and detailed description of the irradiation environment. This includes temperature changes, material changes and even geometry changes that for instance allow a user to modify the position of a control rod during an irradiation cycle. This last option was extremely important for our calculations. IRSN and ILL collaborated to develop a procedure that can be used within VESTA to automatically adjust the position of a control rod in order to keep a target value of Keff [11]. At each irradiation step, the CR position is adjusted in order to obtain criticality (or another target value of the effective multiplication factor). The cycle is finished when criticality can no longer be achieved or when the lowest CR position is reached.

Neutronics

The RHF MCNP model has been benchmarked with series of measurements carried out by the ILL and validated in the International Handbook of Evaluated Reactor Physics

Benchmark Experiments (IRPhEP), published by the Nuclear Energy Agency (NEA) in 2009 [9]. In this model, all in-pile elements have been taken into account, i.e. all beam tubes, safety rods, cold and hot neutron sources, etc. Very few simplifications have been made but a small reactivity impact is expected for each of them. Furthermore, the current model results have been compared to the calculation results of the CEA / SERMA which uses the code French Monte-Carlo code TRIPOLI [10].

Nine subcritical approaches took place on June 24, 2006. At the beginning of each approach, the control rod was set in the initial position, (*i.e.* fully inserted). The studied safety rod or rods are in the 'operating' (*i.e.,* inserted) position while the other ones are in the withdrawn position. The subcritical approaches each began with a reactor set at standard operating conditions in the beginning of cycle. These conditions are the following: a ^{235}U standard load of 8.568 kg and a ^{10}B standard load of 5.8 g in the fuel element; heavy water tank filled by pure heavy water (99.9% mol) at 4 bar; hot neutron source not heated by radiation; cold neutron sources filled by 25 K liquid deuterium; safety rods in the withdrawn position (unless otherwise specified) and a control rod in the 'zero' position (fully inserted to begin approach to critical). The control rod was then withdrawn in each case until criticality was achieved. Table 1 lists the associated critical control rod positions. The overall uncertainty, based upon a detailed sensibility study, which includes manufacturing tolerances, temperatures, material composition and densities, control rod position and so on. For this model and this experiment was evaluated to ± 110 pcm at 1 sigma.

Table 1 : Configuration studied during the June 24th 2006 experiment with 99.9% heavy water in fuel zone

Case Number	**Configuration**	**Measured Critical Control Rod Position (cm withdrawn)**
1	All safety rods (SR) up	22.94 ± 0.2
2	SR1 inserted	52.06 ± 0.2
3	SR2 inserted	48.20 ± 0.2
4	SR3 inserted	48.88 ± 0.2
5	SR4 inserted	49.93 ± 0.2
6	SR5 inserted	45.73 ± 0.2
7	SR1 + SR2 inserted	67.95 ± 0.2
8	SR1 + SR5 inserted	87.05 ± 0.2
9	SR1 + SR4 inserted	No criticality

Table 2 lists the deviation from critical obtained for this series of experiments. Case 1 is a reference case with all safety rods fully withdrawn. For the cases 2 to 7, the k-effective obtained by the three studies is always higher than the reference k-effective, indicating that the safety rod worth is underestimated in each case. For the ILL and ANL, most calculation results are within the 2 (or 3) sigma range of overall uncertainties. This means that the accuracy of the MCNP models is very high for these benchmarks relevant of reactor operation.

Table 2 : Deviation from Critical for the different cases of the June 24th 2006 experiment with 99.9% heavy water in fuel zone

Case Number	**ILL MCNP (k-1)/k in pcm**	**ANL MCNP (k-1)/k in pcm**
1	-169	-341
2	369	-156
3	218	-284
4	176	-250
5	362	-152
6	255	-173
7	125	-272
8	-24	-312
RMS	238	-252

Depletion calculations and cycle length determination for the RHF were historically carried out by the French laboratory SERMA (depending on the French Atomic Energy Commission CEA), using the deterministic code APPOLO2 (transport and depletion). They have used the library APPOLIB for APPOLO2 and JEFF3.1 for TRIPOLI4. We were thus able to benchmark with success Monte-Carlo depletion calculations with both experiments and deterministic calculations.

Furthermore, a benchmark with a reference cycle (cycle 143) has been possible thanks to the automated determination of the control rod position in the code VESTA 200g. At the beginning of a new cycle, the control rod (CR) is fully inserted. The CR is then withdrawn until the first critical position is reached. The cycle starts at that time. Then the CR continues to be withdrawn to compensate the drop of reactivity as fission products build up and the fuel is depleted. The cycle is finished when the lowest CR position is reached (i.e., when the CR is fully withdrawn). During a cycle, the power and critical control rod position are continuously tracked by the ILL staff, so this information can be implemented as a benchmark of a depletion model. The cycle simulated is divided in several time steps and for each of them the real control position is set. Models and codes are validated if k-effective near 1.0 is calculated for each steady state step (plus or minus the uncertainties). Cycle 143 was chosen to benchmark the calculations with an HEU core because the control rod was new (i.e., no prior irradiation) and the history of power is well known.

Figure 2 shows the critical CR position as a function of time obtained by the ILL staff applying the VESTA code with the automatic criticality search. In this case, the fuel meat and borated zones were divided in 45 and 6 independent burn-up zones respectively in order to increase the accuracy of the calculations. The black line represents the fully withdrawn position of the control rod. The blue and red curves represent the measured critical CR position during the cycle 143 and the CR position obtained by simulation, respectively. The cycle is over when the CR has reached the minimal position. On the graph, this situation happens when the red line crosses the black line. The control rod position obtained by simulation stays very close to the measured position the beginning to the end of cycle. In consequence, the model and code were demonstrated capable for prediction of the CR motion and should be able to determine the cycle length with an LEU fuel.

These results also highlight that 50 more benchmarks may be added to the Beginning of Cycle benchmark detailed above. We can thus assess again that the RHF MCNP model is extremely reliable for the HEU. It is therefore reasonable to think that this will also be the case for the LEU configuration.

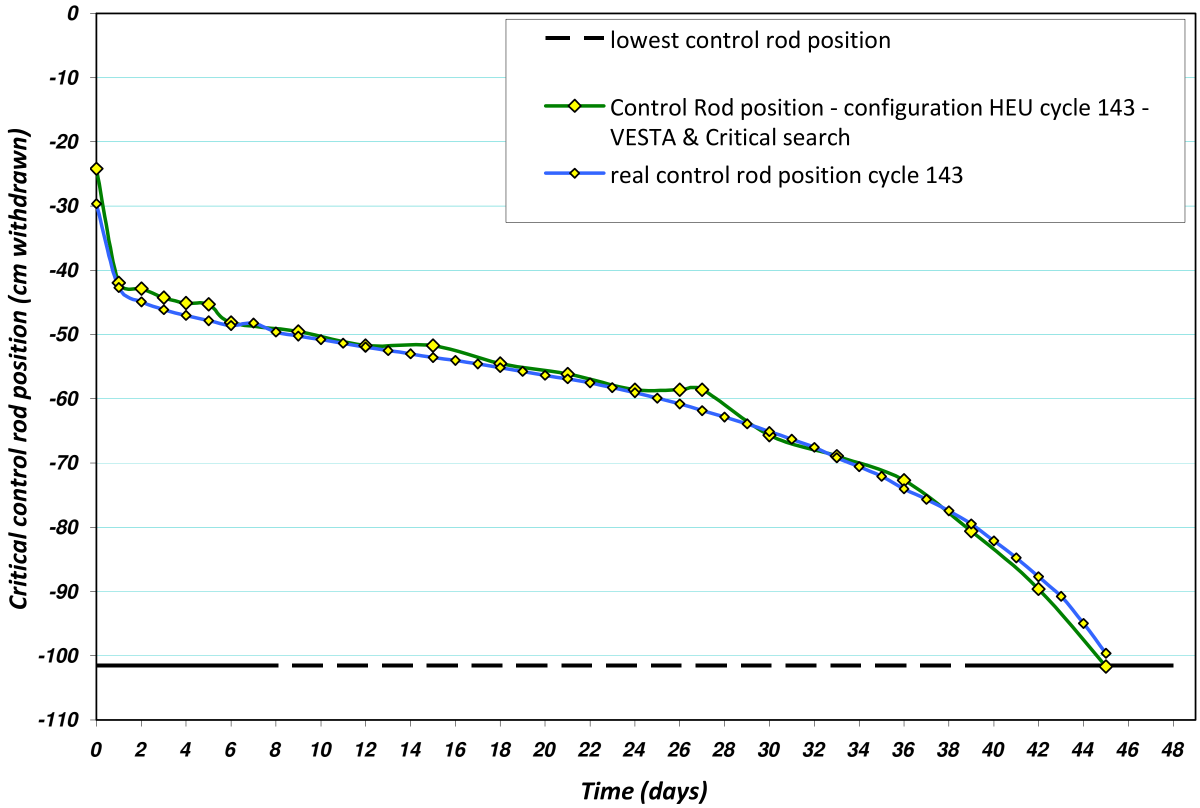

Figure 2 : Control rod motion evolution obtained by VESTA and the automatic critical search for an HEU core at 57 MW (cycle 143)

Thermal-Hydraulics

Thermal-Hydraulic calculations were carried out by the Computational Fluid Dynamics codes ANSYS/CFX (for the ILL) and STARCD (for the ANL). The models are based upon the 3D geometry of the fuel element. The heat loads are provided thanks to the 3D MCNP calculations. Historically, the Thermal-Hydraulic calculations were performed by 1D codes like FLIKA. A benchmark between all of these codes was carried out and showed coherent results. We can thus conclude that the Thermal-Hydraulic models are relevant for the real water channels within the RHF fuel element.

The details of this study are presented in another RERTR 2010 paper [4].

TOUTATIS : THE CHOSEN CONFIGURATION

The first step of the RHF conversion study was to consider the direct replacement of HEU fuel meat by LEU U-Mo fuel. We evaluated the cycle length and the thermal neutron flux/brightness for this configuration. The simulations indicated that the cycle length is strongly impacted by the simplistic change of fuel. Indeed, with an optimized amount of boron, the cycle length cannot exceed 39 days for the direct fuel replacement case-- a drop of 14% with respect to HEU core at 57 MW (nuclear). In addition, the thermal neutron flux would decrease by roughly 7%. These results were confirmed by a study carried out by the CEA / SERMA.

Under these conditions, ILL has assessed that the reactor performances would be too deeply reduced in order to seriously envisage the conversion. This is the reason why the ILL, in collaboration with the ANL, has developed alternative LEU fuel element designs.

Because of safety and economic reasons, the modifications to the present fuel configuration have to be as minor as possible. Therefore we have decided to make no modification on the external plate dimensions or the thickness of the meat. With these options, the whole core of the RHF does not need to be modified. In a first step, the modifications are mainly focused on the fuel plates. In addition, the fabrication of LEU plates with borated zones is not guaranteed by the manufacturer, the CERCA. Consequently, we have chosen to consider the presence of boron within the plates as impossible and to remove all boron from the LEU plates.

As the cycle length depends of the total mass of fuel, we have designed a configuration where the fuel length is extended by replacing both of the borated zones. The fuel volume has been extended by 4 cm on both the top and at the bottom. This new configuration with an 'extended meat' has been named 'TOUTATIS' (French acronym: Traitement Optimisé de l'Uranium et Thermique Améliorée pour une Technologie Intégrant la Sûreté) and is shown in Figure 3. The overall fuel volume increase in this configuration is roughly 9%.

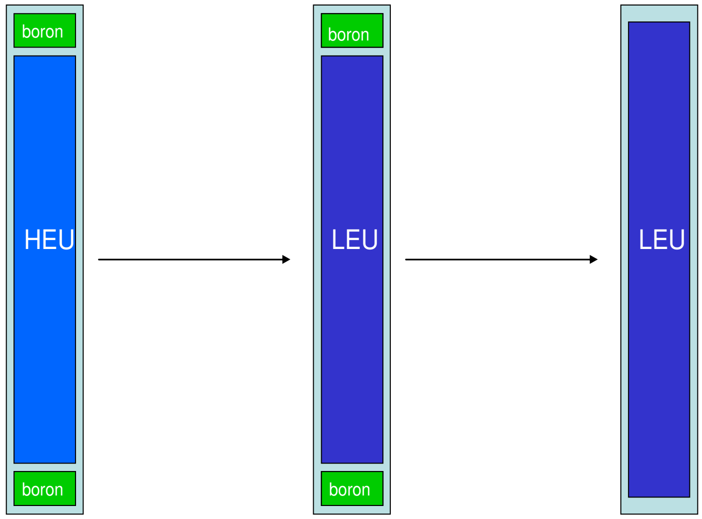

Figure 3 : Scheme of HEU plates, LEU plates as studied in a first step and TOUTATIS LEU plates. In the latter case, the boron is set above the fuel plate in the outer tube of the element

A first depletion calculation has been carried out for the TOUTATIS configuration, using the divided model (45 cells to describe the fuel meat) and the VESTA automatic critical search. Then, a second depletion has been performed with REBUS-MCNP applying the Control Rod (CR) positions found by VESTA.

Figure 4 shows the CR motion predicted by VESTA with the automatic critical search. The cycle is over when the CR has reached the minimal position (on the graph, this situation occurs when the green line crosses the black line). Thus, the cycle length is estimated to be 47.5 days (± 1 day at 1 sigma). In comparison of the typical cycle length obtained with a HEU core at the currently used nuclear power (52 MW) that represents a decrease of 2 days, *i.e.* 4%. Thus, in order to reach the present 50 days cycle length, the nuclear reactor power of the TOUTATIS configuration has to be reduced by 4% by comparison to the nominal value, that is 55MW.

Figure 5 illustrates the k-effective versus time obtained by REBUS-MCNP using the CR position obtained by VESTA. The k-effective obtained at t = 0 (first calculation, no depletion) is a k-effective of reference. As can be seen, throughout the cycle the k-effective stays stable around the reference k-effective, thus confirming the control rod position search result obtained by VESTA, and the depletion of materials by VESTA.

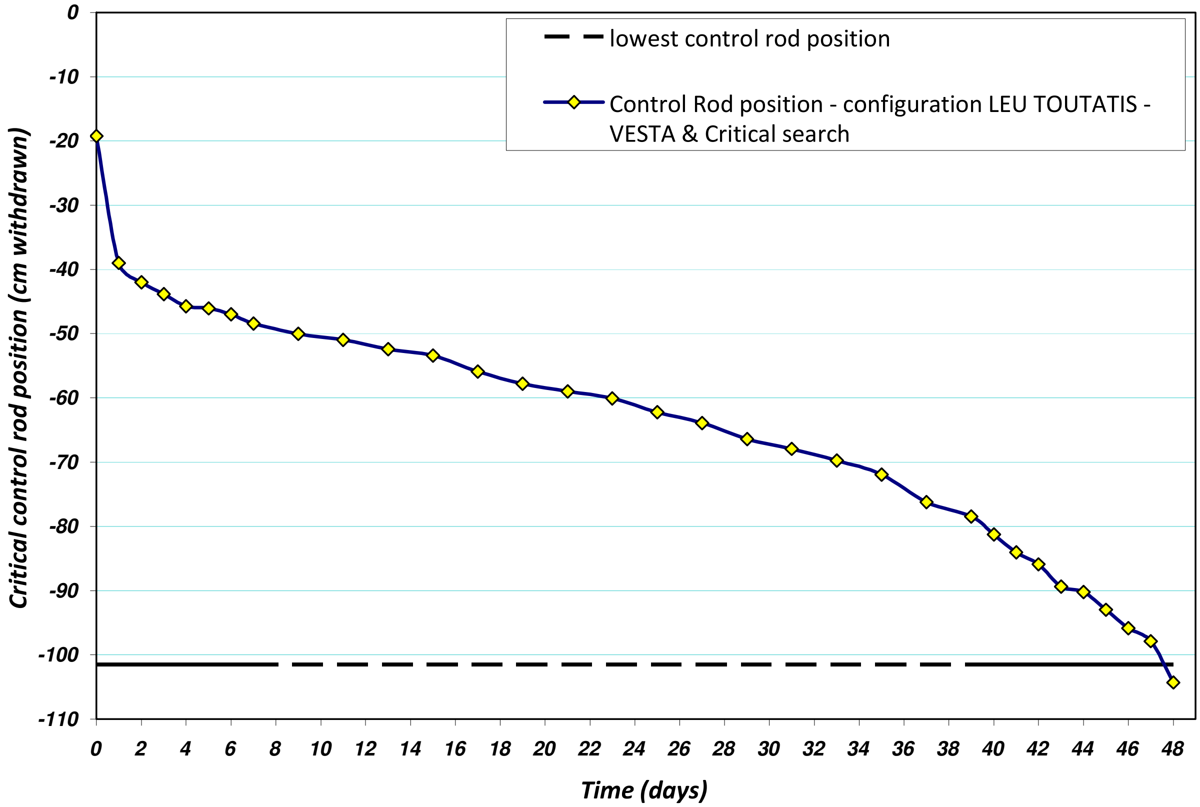

Figure 4 : Control rod motion evolution obtained by VESTA with the automatic critical search, TOUTATIS configuration at 57 MW, divided model

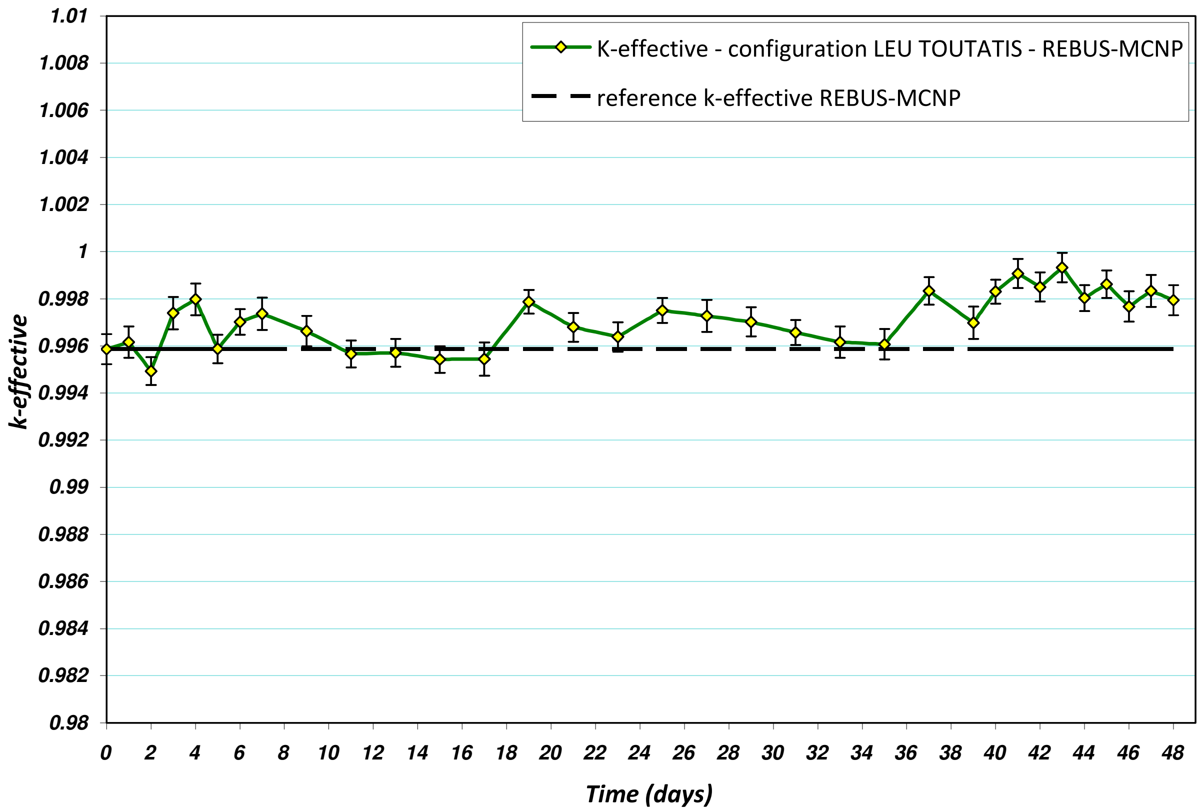

Figure 5 : k-effective evolution obtained by REBUS-MCNP using the VESTA control rod positions, TOUTATIS configuration at 57 MW, divided model.

Performance evaluation

The determination of the neutron flux and brightness losses have been carried out by MCNP DXTRAN spheres at the extremity of a beam tube in the heavy water tank and at the entrance of the neutron guides, 2 to 3 meters away, respectively. In order to compare rigorous cases we benchmarked the HEU core to the TOUTATIS LEU core both being at nominal reactor power, 57MW. The normalized weight corresponding to the number of allocated beam time days for each instrument and described above has then been applied in order to take into account the real importance of all ILL instruments. A Factor Of Merit (FOM) has been defined to balance the importance of weighted brightness and beam time (cycle length). We could thus conclude that the change in neutron flux / brightness between the present HEU core and the TOUTATIS core, both at nominal reactor power 57MW, is -13±3 % at 2 sigma with an increase of cycle length of 2.5days ± 0.5 which lead to a FOM of -7% ± 4%. The typical operating power of the RHF has been recently decreased to 52MW. To have a better idea of the difference of performance between HEU and LEU configuration, the FOM has also been used to compare the HEU operating case at 52MW with the TOUTATIS case with a power of 55MW and 57MW. Figure 6 presents the relative weighted brightness versus cycle length for the different configurations described above. The associated changes in FOM for different fuel and operating powers is presented in Table 3. As we can see, for different power levels, the change in FOM due to conversion to LEU stays around -7% ± 4% in comparison to either nominal or typical HEU operation..

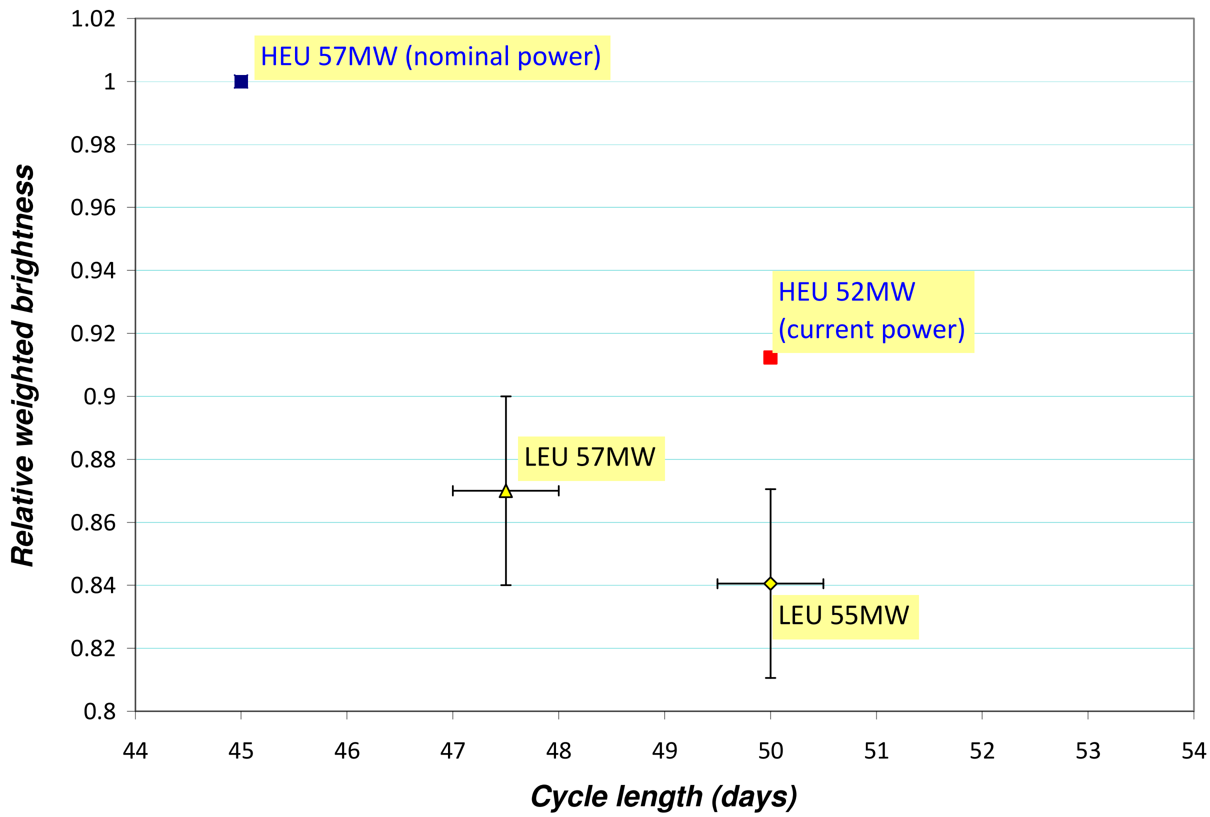

Figure 6 : Relative weighted brightness versus cycle length for HEU and TOUTATIS

	Reference for Comparison	
Candidate	HEU 57MW	HEU 52MW
HEU 52MW	+2.3%	0.00%
TOUTATIS 57MW	-7 % +/- 4%	-7 % +/- 4%
TOUTATIS 55MW	-7 % +/- 4%	-7 % +/- 4%

The thermal hydraulics calculations are detailed in another RERTR 2010 paper [4]. Nevertheless the conclusions can be recalled here. The benchmark between CFX and STARCD on the RHF core is excellent. The maximum cladding temperature and the margins to saturation, ONB (Onset of Nuclear Boiling) and FNB (Full Nuclear Boiling) for the TOUTATIS configuration were determined and compared to the HEU core. The results show a slight decrease of the margin in the case of the TOUTATIS configuration, but the margin for the LEU case remains compliant with technical specifications.

CONCLUSION

The ILL and the ANL have collaborated to investigate LEU designs for the RHF in the framework of a conversion study into LEU. A promising candidate has been selected and studied, which increases the total amount of fuel without changing the external plate dimensions by relocating the burnable poison. This proposed LEU design has been called TOUTATIS. Preliminary thermal hydraulic safety margins have been evaluated with favorable results, as described in a companion paper.

The Factor of Merit performance losses induced by the conversion were evaluated at 7±4 % for the neutron flux available for the ILL. This means that the impact on the RHF performances should be between 5 and 10%, which seems to be an acceptable value. Obviously a significant number of calculations must still be carried out to confirm this result.

Nevertheless there are still a number of caveats which are conditions for the RHF possible conversion, like success of the LEONIDAS experiments, the commercialization of the UMo, the experimental validation of the UMo in normal and transient operating conditions and eventually the approval of the French Safety Authority.

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