# On the Use of Existing High Enriched MOX Fuel in an Experimental ADS

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**Abstract** – *The work presented is part of the FZK and RIT contributions to the investigations for the assessment of the possibility to use existing highly-enriched MOX fuel in an ANSALDO-ENEA design for an experimental ADS (XADS) with lead bismuth (LBE) coolant, studied in the PDS-XADS Project of the 5. Framework Program of the European Community. Use of slightly modified existing fuel assemblies, of fuel pins, of fuel pellets and of reprocessed fuel is considered. Neutron physics, including burn-up analysis are presented. Use of SNR300 fuel assemblies appears feasible if technical and logistical constraints can be solved. SPX fuel pins or pellets also may be utilized, but due to the lower fissile enrichment a larger core is required. Both reprocessed SPX and SNR300 fuel may be utilized as demonstrated in the ANSALDO reference design with reprocessed SPX fuel. An interesting alternative is the utilization of both SPX and SNR300 pellets in fuel assemblies with identical outer dimensions, but with different lattice parameters. The preliminary burn-up analysis for a reference core with SNR300 fuel assemblies with Monte Carlo and deterministic methods show a burn-up reactivity loss of 6 to 7 pcm per full power day, corresponding to a reactivity loss of about 2% per year at 70 to 80% duty cycle.*

## I. INTRODUCTION

The work presented is part of the FZK and RIT contributions to the investigations for the assessment of the possibility to use existing fresh highly-enriched MOX fuel in an ANSALDO-ENEA design for an experimental ADS (XADS) with lead bismuth (LBE) coolant, studied in the PDS-XADS Project of the 5. Framework Program of the European Community. A first very preliminary assessment to utilize existing SNR300 fuel assemblies in an experimental ADS was presented in [1]. In Appendix 5 of the European Roadmap for ADS Development [2] a more detailed analysis was performed, considering available fuel assemblies (FA) both from the fast breeder programs in Germany (SNR300) and France (SPX). Some main results in [2] are:

- From technological point of view it is feasible to use SNR300 and SPX FA.
- For SPX fuel a preliminary assessment was made to disassemble the FA and to utilize the fuel either in the rods or as pellets. Both options are feasible if facilities to handle this fuel stay available. The high Americium content in this old MOX fuel is not a major problem.
- Both SNR300 and SPX fuel pins contain axial blankets in bottom and top areas.
- SNR300 FA have, because of the smaller core, higher enrichments than the SPX FA.
- Fabrication periods: SNR300 FA between 1978 and 1985. SPX first core between 1978 and 1984, second core between 1985 and 1988.

From these considerations no other significant differences than the enrichments are to be expected in the utilization of SNR300 and SPX fuel. After a short characterization of the existing fuels in section 2, in section 3 the neutron physics analysis is described for the available fuel options. First burn-up analyses for the reference design of a LBE cooled XADS with SNR300 FA are presented in section 4. More comprehensive information may be found in reference [3].

## II. CHARACTERIZAZION OF EXISTING FUEL

A good characterization of the SNR300 and SPX fuels may be found in reference [2]. A summary of construction details of SNR300 fuel is given in TABLE 1. TABLE 2 shows the fuel compositions as predicted for the year 2010 [4] of the three FA types, lower enriched fuel from MAGNOX Plutonium (C1_MAG) and from LWR Plutonium (C1_LWR) and higher enriched fuel from LWR Plutonium (C2_LWR). A summary of construction details of SPX fuel is given in TABLE 3. TABLE 4 gives the fuel masses per FA of four types of SPX FA, predicted for the year 2010 on the basis of specifications in reference [5] for the year 1995. The depletion calculations have been performed with the code KORIGEN [6]. The available fresh FA have fissile enrichments varying from 12.2% to 15.8% for the SPX fuel and from 18.8% to 26.4% for the SNR300 fuel. These enrichments may be a good basis for the construction of an experimental ADS with conventional MOX fuel in a first operation stage.

## III. NEUTRON PHYSICS CORE SIMULATIONS:

In the next sections exploratory neutron physics simulations of XADS core configurations with Lead Bismuth Eutectic (LBE) cooling will be discussed.

### *III.A. SNR-300 FA cooled with Lead Bismuth Eutectic*

For the SNR-300 the existing FA have been considered. The characteristics data appears in TABLE 1. For the use of this fuel with LBE cooling in a XADS system the core layout with 66 FA , shown in FIGURE 2, is proposed. The first three rings (the central element is considered as the first ring) are filled with LBE mixed with structure material of the target (zone 1). The next two rings (zone 2, which contains 42 FA) are loaded with the C1_MAG fuel; the next ring (zone 3) is loaded with 24 subassemblies of type C2_LWR. This means that 6 FA are missing in the far edge of zone 3. Isotope compositions of fuels C1_MAG and C2_LWR are presented in the TABLE 2. The radial reflector (zone 4) and the inner tube are filled with homogeneous mixture of LBE and structure material in proportion depending on the specific core design. The multi group cross sections were prepared within the KAPROS/KARBUS modular code system starting from a 69 group master library, mainly based on the JEF2.2 evaluated data library. In most cases standard collapsing from 69 to 12 groups was applied. The flux calculations were carried out with the codes CITATION and TWODANT. The results, with 80% LBE and 20% structure material in the reflector and beam pipe are given in TABLE 5. To confirm the results obtained by the deterministic codes, the core simulation was also done by the MCNP code with the same homogenized cylindrical model. The criticality obtained was $K_{eff}$=1.001 with standard deviation of 7 $10^{-4}$ for fuel, coolant and clad material at temperature of 300 K. Comparing the above result to the 300 K transport cylindrical TWODANT value for 12 energy groups 1.006, shown in TABLE 5, one observes a difference of about 0.5%, acceptable for these investigations. The amount of the structure material in the reflector and in the guiding tube was analyzed by means of hexagonal based calculations with CITATION code as well as with TWODANT code using cylindrical transport option. The main results are presented in TABLE 6. The range of criticality level values in TABLE 6 is almost 5% for the extreme cases. The extremely small core enhances the importance of the reflector dimensions in addition to the material composition shown above. The influence of the reflector size on the criticality is presented for the reference design in FIGURE 1. It can be seen that about 70 cm is needed before reaching an asymptotic behavior. It should be noted here that the exploratory investigations in this section applied the same canning material as in the ANSALDO design. In the burn-up calculations of section IV the actual canning materials of the SNR300 FA (WN1.4981) was utilized, leading to lower criticality values. These results indicate that the SNR300 fuel is well suited for use in the envisaged XADS project. The original FA, including axial blanket zones, may be used from the neutron physics point of view. However, the specific construction details must be taken into account in the XADS design.

### *III.B. SPX fuel cooled by Lead Bismuth Eutectic (LBE)*

The existing Super Phenix (SPX) fuel is considered intensively in the XADS project. The ANSALDO XADS core design is based on reprocessed SPX fuel with increased fissile fraction, compared to the existing SPX fuel composition. This chapter describes the possibilities to use SPX rods or pellets for the ANSALDO XADS design proposals with LBE coolant. In this case the fuel re-fabrication for the XADS could be avoided. The benefit of handling SPX fuel on the pellet level is in the wrapper thickness. The original SPX rods are very long and thereafter require thick wrapper (about 4.6 mm) whereas the fuel rods of the ANSALDO design are considerably shorter and allow for only 2 mm thickness leading to significant reduction in parasitic neutron absorption within the core. The proposed core design is shown in FIGURE 3. The number of FA is 198 in 6 full fuel rings, whereas the ANSALDO model consists of only 120 FA.. Calculations were carried out using the hexagonal option of the diffusion code CITATION. The average temperature of the fuel is 1273 K and the coolant and cladding are taken at 773 K. The criticality obtained was $k_{eff}$=0.962 The CITATION results were verified by corresponding (R-Z) calculations using the (R-Z) transport version of TWODANT. For this case a cylindrical ring with equal volume as the 6 hexagonal fuel rings (198 FA) resulted in $k_{eff}$=0.971. These calculations show that the SPX fuel is not so well suited for direct use in an XADS. The dimensions of the existing FA and the relatively low enrichments lead to the conclusion that the SPX pellets may be used in an XADS core with ≈ 200 FA.

### *III.C. ANSALDO core design with two existing enriched fuel types*

It is of interest to check the feasibility of using the existing SNR300 and SPX fuels to avoid the reprocessing costs of fuel fabrication. In the following three constraints are satisfied: the criticality level is around 0.97, the number of subassemblies is 120 and the subassembly geometry is identical with the original ANSALDO design. From several possibilities, two representative configurations are discussed to emphasize the balance between lower costs, improved neutron physics characteristics and thermal–hydraulics aspects. The first option is shown in FIGURE 4. The inner rings consisting of 78 subassemblies are loaded with original SPX pellets (in the fuel zones). The rest outer 42 subassemblies are filled with highly enriched MOX with the composition of the C2_LWR SNR pellets of TABLE 2, embedded in the ANSALDO coating cladding. The second option is shown in FIGURE 5. Here only 30 subassemblies with original SPX pellets are introduced and the rest 90 subassemblies contain MOX with composition of the C1_MAG SNR fuel pellets of TABLE 2. These assumptions imply for both cases reprocessing of the SNR300 fuel. The hexagonal configurations of the two cores were analyzed using the CITATION diffusion code. The criticality values of the CITATION results lay within 0.5 % of the MCNP reference calculations. In TABLE 7 the relevant neutron physics and thermal hydraulic features are presented. The first core, which uses more SPX fuel, exhibits worse thermal hydraulic behavior. The two enrichment zones differ by about 25% leading to peak power factors of about 1.6 and to differences in power densities between two adjacent fuel elements (differently enriched) larger than 55%, which is very high from thermal hydraulic and fuel mechanics point of view. The second design utilizes more SNR300 fuel. Due to the lower enrichment of the C1_MAG fuel the power distribution is smoother leading to lower peak factors and significantly lower power ratio of two adjacent FA, 1.2 instead of 1.55 in the former case. Furthermore, both configurations are practically insensitive to source shut down in a sense of the multiplicity value. This first analysis of these two core designs clearly shows that the available MOX fuel pellets may be used for the design of an acceptable XADS core. A very interesting solution could be the use of SNR fuel pellets with 5mm outer diameter in an assembly with the same outer dimensions as the ANSALDO FA design for SPX pellets with 7.14mm outer diameter. This option could be investigated more detailed in a next stage.

## IV. BURN-UP INVESTIGATIONS FOR THE CORE WITH SNR300 FUEL

First preliminary burn-up investigations were peformed for the reference design of a LBE cooled XADS with SNR300 FA. For this purpose, two alternative methods were applied:

- Full Monte Carlo burn-up calculations with the MCB1C Code, performed by RIT Stockholm
- Deterministic multi-group depletion calculations, performed by FZK Karlsruhe.

### *IV.A. Burn-up investigations with the Monte Carlo code MCB1C*

Although a burn-up simulation with the Monte-Carlo method MCB [8] can be performed on a high level of geometry complexity, the main goal in the current stage of analysis is to produce results in a relatively simple model in order to allow direct comparisons with deterministic burn-up simulation methods. The model simplifications concern mainly the system geometry and the definition of a spallation neutron source.

#### *IV.A.1. Calculation Model and Methods*

The burn-up calculation was preformed by using the MCB code in a geometry model of XADS reactor using the adapted version of core model prepared by FZK as follows:

a) Target and core geometry in appropriate (R-Z) geometry

b) The neutron source is represented by a surface source file (RSSA) that was obtained by a collection of all neutrons leaking from the LBE target during an MCNPX simulation of a 600MeV proton beam interaction with the target in a standalone bare target model, thus avoiding an overlapping of core neutrons contributions. This source is more detailed than in the corresponding FZK calculations.

c) The core is divided into 24 burnable zones: 3 radial sections times 8 axial sections.

In order to assess the required reactor core parameters, the Monte Carlo method of neutron transport was applied using the MCNP based codes MCNPX and MCB1C, with temperature dependent cross section libraries of the MCB system. Two modes of calculations were applied: Firstly, the fixed source mode of MCNPX in a bare target system to produce the neutron source for burn-up calculations in MCB and to obtain core power distribution and radiation damage. Secondly, the burn-up mode of MCB to calculate the system evolution with time over 3 years of the 80MWth thermal power irradiation. In general, the JEF2.2 cross section libraries were used in the current calculations, whereas for nuclides of lead that are missing from JEF2.2 the JENDL3.2 data were used. The cross section at power temperatures 1200K for fuel and 900K for target were used. For the calculation of $k_s$ an integral approach has been applied. This leads us to following definitions:

$$k_s = \frac{N_f}{N_f + 1}$$

where $N_f$ is the number of fission neutrons generated per one external source neutron. For the source effectiveness the following formula is applied:

$$I = \frac{(1 - k_{eff}) \cdot k_s}{(1 - k_s) \cdot k_{eff}}$$

with $k_{eff}$ the criticality from the eigenmode calculation.

### *IV.A.2 Power Normalization*

MCB calculates the thermal power released per one source particle and adjusts the source intensity to keep power at the user specified level. The core actual thermal power consists of direct radiation heating, including neutron KERMA heating and gamma heating, as well as decay heating – mostly from decays of short lived fission products. In our case the decay heating accounts for about 3.0 $MW_{th}$ while the remaining 77.0 $MW_{th}$ comes from prompt radiation interaction with matter. The total power, however, neglects the energy release into the target in the high-energy particle transport.

### *IV.A.3. Results*

The results obtained in the MCB burn-up calculation presented in TABLE 8 summarize the burn-up performance of the XADS core with SNR300 fuel. For the purpose of comparison with calculation systems that are normalized to fission energy we have assessed the energy release per one fission event by dividing the total heating by the number of fission events as follows.

| | |
|---|---|
| Heating per source neutron | 1.692GeV |
| Fissions per source neutron | 7.817 |
| Energy/number of fissions | 216.5MeV |

The obtained value should not be understood as the energy of one fission event since it includes all energy releasing processes taking place in the system. The external source multiplication factor – $k_s$ is slightly higher than the actual reactivity or effective neutron multiplication factor - $k_{eff}$ implying the source effectiveness greater than one.

### *IV.B. Burn-up investigations with the deterministic multi-group code system KAPROS/KARBUS*

The modular code system KAPROS contains several options to perform burn-up investigations. The basic module BURNUP is based on the formalisms of the KORIGEN code [6]. The required one-group data for the evolution calculations can be provided within the KAPROS system with the help of best estimate weighting spectra, applied to the macroscopic multi-group zone dependant cross sections. The current 69 group master data library contains multi-group cross sections for activation analyses for about 800 isotopes, including more than 160 isotopes with data for full multi group transport calculations. For this first XADS burn-up investigations the same calculation scheme was applied as for the IAEA ADS benchmark for $Th/U^{233}$ fuel [7]. The applied spallation neutron source was similar as in the IAEA benchmark: constant source in the central fuel assembly position with total extension of 30cm symmetric to the center. The energy spectrum of the source was the same as in [9] . The reactor model is the same (R-Z) geometry as in the MCB1C case. Calculations were performed for 12 and 69 groups based on JEF2.2 and JEFF3.0 evaluated data. The power generation in this deterministic burn-up calculations is based on 210 MeV energy release per fission. The results for the reactivity for the system, driven by the external source (source on), $K_s$, are based on consistent calculations of the ratio of neutron production to neutron losses. Up till now the time behavior of $k_{eff}$, $k_s$ , $dk_{eff}/dt$ and the heavy metal masses have been evaluated.

### *IV.C. First comparisons of RIT and FZK results*

In this section first comparisons of the RIT results with Monte Carlo methods and of FZK results with deterministic methods are presented. The comparisons are performed with a 12 group constant library derived from a 69 group master library; mainly based on the JEF2.2 evaluated data file. A comparison of the time evolution of the criticality for source on/off states is presented in FIGURE 6. The “source on” results are obtained from the calculations with external source, whereas the results for “source off” come from homogenous eigenvalue calculations. We may observe a satisfactory agreement between the RIT and FZK results. The discrepancies decrease with increasing burn-up. Although the source representation is quite different in the RIT and FZK investigations, the behaviour of the reactivity for source on/off is similar in both cases: the value for source on is slightly higher than for source off and the difference has a tendency to increase with increasing burn-up. The data for the reactivity loss in pcm per full power day show agreement better than 10%. The reactivity loss per full power day amounts 6 to 7 pcm and is slowly decreasing with burn-up. These values are rather high and lead to a reactivity decrease of $\Delta k_{eff} \approx 2\%$ per year at 70..80% duty cycle. A measure for the absolute fluxes are the required source strengths, compared in FIGURE 7. We may here also observe good agreement. In order to explain possible differences in the burn-up evolution obtained with the different simulation methods one need to compare the evolution of the nuclide densities as well as their influence on the system criticality. More detailed information may be found in reference 3. As an example FIGURE 8 shows comparisons between RIT and FZK results for the changes in the heavy metal inventories of the main isotopes in the core. Generally, the results show that nuclides that are being reduced in mass are $U^{238}$, $Pu^{239}$, $U^{235}$ and $Am^{241}$,other nuclides usually are being build-up. The differences in the heavy metal inventory changes in the RIT and FZK results increase if the build-up of the isotopes becomes more complicated by multiple transitions. The differences for the individual isotopes have to be analyzed in more detail.

## V. SUMMARY

Several options for the utilization of the existing high enriched fuel from the European fast breeder programs SNR300 and SPX in the core of an XADS have been investigated. In these studies the use of slightly modified fuel assemblies, of fuel rods and of pellets have been considered. From the neutron physics point of view the reference core with SNR300 fuel assemblies is feasible. The two level enrichment strategy allows for low peak factor and a flattened spatial flux distribution. For the SPX fuel based model an optimization procedure is necessary to favor either the option of fabricating new fuel (ANSALSO model) or to remain with the existing fuel and to deal with the consequences of a larger system. The combination of two fuel pellet types within fuel assemblies of the ANSALDO core design seems to be feasible and might be the best solution if technical and logistics problems can be solved. First preliminary burn-up calculations with Monte Carlo and deterministic methods for the SNR300 fuel reference core show reasonable agreement for important parameters. The differences in the evolution of minor actinides are larger and have to be analyzed in more detail. The reactivity loss per full power day amounts 6 to 7 pcm and is slowly decreasing with burn-up. These values are rather high and lead to a reactivity decrease of $\Delta k_{eff} \approx 2\%$ per year at 70..80% duty cycle.


## ACKNOWLEDGEMENTS

Part of the presented work was funded in the 5. Framework Program of the European Community, contract number FIKI-CT-2000-00033.


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TABLE 1: Main characteristics of the SNR-300 sub-assembly.

| **casing** | | **fuel bundle** | |
|---|---|---|---|
| Form | hexagonal | number of fuel pins | 166 |
| pads distance | 115.0 mm | number of structure pins | 3 |
| wrapper distance across flats | 110.25 mm | spacers | 14 grid spacers |
| wrapper thickness | 2.6 mm | pitch/diameter | 7.9/6.0 = 1.316 |
| **fuel pin** | | | |
| outer cladding diameter | 6.00 mm | fertile fuel length | 2 x 400 mm |
| cladding thickness | 0.38 mm | upper fission gas plenum length | 43.0 mm |
| fissile fuel length | 950.0 mm | lower fission gas plenum length | 645.0 mm |
| pellet diameter | 5.09 mm | | |

TABLE 2: SNR-300 fuel element material composition in the year 2010 for the three existing fuel types. The data corresponds to 166 fuel pins in one FA.

| isotope | mass of fuel isotopes per one FA for different fuel types, (g/FA) | | |
|---|---|---|---|
| | C1_MAG | C1_LWR | C2_LWR |
| $U^{234}$ | 2.6 | 11.3 | 18.8 |
| $U^{235}$ | 76.2 | 127.9 | 128.6 |
| $U^{236}$ | 4.8 | 5.3 | 6.9 |
| $U^{238}$ | 19848.5 | 19512.1 | 16755.3 |
| $Np^{237}$ | 6.70 | 12.2 | 17.5 |
| $Pu^{238}$ | 8.7 | 40.9 | 73.7 |
| $Pu^{239}$ | 4921.8 | 4486.5 | 6219.7 |
| $Pu^{240}$ | 1360.9 | 1638.2 | 2265.6 |
| $Pu^{241}$ | 49.2 | 114.8 | 206.5 |
| $Pu^{242}$ | 52.4 | 204.5 | 331.7 |
| $Am^{241}$ | 197.3 | 393 | 616.4 |
| Total fuel | 26529.10 | 26546.70 | 26640.70 |
| $Pu_{-fi}$+Am/HM | 0.194816 | 0.188133 | 0.264355 |

TABLE 3: Main characteristics of the SPX sub-assembly [5].

| casing | | fuel bundle | |
|---|---|---|---|
| Form | hexagonal | number of fuel pins | 271 |
| pads distance | | number of structure pins | - |
| wrapper distance across flats | 173 mm | spacers | 14 grid spacers |
| wrapper thickness | 4.6 mm | pitch/diameter | 10.5/8.5=1.235 |
| **fuel pin** | | | |
| outer cladding diameter | 8.50 mm | fertile fuel length | 2 x 300 mm |
| cladding thickness | 0.565 mm | upper fission gas plenum length | Total 852 mm |
| fissile fuel length | 1000.0 mm | lower fission gas plenum length | |
| pellet diameter | 7.14 mm | | |

TABLE 4: SPX Fuel element material composition in the year 2010 for the three existing fuel types. The data corresponds to 271 fuel pins in one FA.

| isotope | mass of fuel isotopes per one FA for different fuel types, (g/FA) | | | |
|---|---|---|---|---|
| | R1–inner core | R2–inner core | R1–external core | R2–external core |
| $U^{234}$ | 28.9 | 37.7 | 56.1 | 27.1 |
| $U^{235}$ | 408.7 | 404.7 | 381.0 | 383.6 |
| $U^{236}$ | 9.43 | 30.7 | 34.3 | 31.7 |
| $U^{238}$ | 75407 | 75060 | 70930 | 70944.3 |
| $Np^{237}$ | 17.5 | 21.1 | 29.9 | 22.5 |
| $Pu^{238}$ | 136.8 | 178.1 | 255.8 | 128.0 |
| $Pu^{239}$ | 9854.7 | 9842.7 | 11765.0 | 12733.5 |
| $Pu^{240}$ | 3702.5 | 3792.0 | 4832.4 | 4596.3 |
| $Pu^{241}$ | 394.1 | 508.2 | 658.7 | 542.2 |
| $Pu^{242}$ | 533.2 | 720.4 | 1056.0 | 686.2 |
| $Am^{241}$ | 913.0 | 1114.7 | 1554.0 | 1189.0 |
| Total fuel | 91406 | 91710 | 91553 | 91284 |
| $Pu_{fis}$ + Am/HM | 0.1221 | 0.1250 | 0.1527 | 0.1584 |

TABLE 5. SNR–300 fuel reference core parameters, with 80% LBE and 20% structure material in the reflector and the central core (zones 1 and 4).

| Case | | $K_{eff}$ TWODANT S8 12 groups | $K_{eff}$ CITATION 12 groups |
|---|---|---|---|
| Cylinder | 300 K | 1.006 | 0.996 |
| | 1273 K | 0.994 | 0.983 |
| Hexahedral | 300 K | | 0.994 |
| | 1273 K | | 0.981<br>4 meshes<br>0.983*) |

*) source on, otherwise source off

TABLE 6. Influence of the structure material in the target and the reflector on $K_{eff}$ , calculated with cylindrical transport model of TWODANT at fuel temperature 1273 K.

| Coolant/structure fraction, %vol | | $K_{eff}$ for the „source off“ option |
|---|---|---|
| Target | reflector | |
| 80/20 | 80/20 | 0.994 |
| 100/0 | 100/0 | 1.019 |
| 100/0 | 80/20 | 1.005 |
| 100/0 | 50/50 | 0.985 |
| 100/0 | 20/80 | 0.970 |

TABLE 7. Criticalities and power densities of two core designs (FIGURE 4,5) for a 80MW$_{th}$ ANSALDO based XADS core design. The power densities refers to the “source on” option.

| Core layout | FIGURE 4 | FIGURE 5 |
|---|---|---|
| Number of SPX assemblies | 78 | 30 |
| Number of C1_MAG SNR assemblies | ------ | 90 |
| Number of C2_LWR SNR assemblies | 42 | ------ |
| $K_{eff}$ | 0.9684 | 0.9717 |
| Multiplicity with source on | 0.9677 | 0.9722 |
| Peak linear power (W/cm) | 128.6 | 125.0 |
| Mean linear power in SPX assembly (W/cm) | 78.3 | 82.4 |
| Mean linear power in C1_MAG assembly (W/cm) | ------ | 89.7 |
| Mean linear power in C2_LWR assembly (W/cm) | 95.1 | ------ |

TABLE 8: Summary of MCB1C results for LBE cooled XADS with SNR300 fuel

| Time [year] | Power [MW] | Source Strength [1/s] | $k_{eff}$ | $k_s$ | H/S [MeV] | I | Decay Heat [W] |
|---|---|---|---|---|---|---|---|
| BOL | 80 | 3.04E+17 | 0.9592 | 0.9613 | 1642.93 | 1.0549 | 2.94E+06 |
| 0.25 | 80 | 3.29E+17 | 0.9523 | 0.9547 | 1515.12 | 1.0549 | 2.97E+06 |
| 0.50 | 80 | 3.71E+17 | 0.9471 | 0.9496 | 1346.43 | 1.0537 | 2.97E+06 |
| 0.75 | 80 | 4.15E+17 | 0.9408 | 0.9446 | 1201.79 | 1.0732 | 2.97E+06 |
| 1.00 | 80 | 4.58E+17 | 0.9350 | 0.9378 | 1090.56 | 1.0492 | 2.97E+06 |
| 1.25 | 80 | 5.04E+17 | 0.9292 | 0.9325 | 989.53 | 1.0534 | 2.97E+06 |
| 1.50 | 80 | 5.54E+17 | 0.9233 | 0.9273 | 900.64 | 1.0590 | 2.97E+06 |
| 1.75 | 80 | 6.16E+17 | 0.9168 | 0.9211 | 810.07 | 1.0584 | 2.97E+06 |
| 2.00 | 80 | 6.43E+17 | 0.9111 | 0.9164 | 775.79 | 1.0697 | 2.97E+06 |
| 2.25 | 80 | 6.90E+17 | 0.9061 | 0.9116 | 723.26 | 1.0694 | 2.97E+06 |
| 2.50 | 80 | 7.48E+17 | 0.9000 | 0.9059 | 667.16 | 1.0696 | 2.97E+06 |
| 2.75 | 80 | 7.70E+17 | 0.8946 | 0.9014 | 648.18 | 1.0773 | 2.97E+06 |
| 3.00 | 0.038 | 0 | 0.8891 | 0.8954 | 0 | 1.0687 | 3.76E+04 |

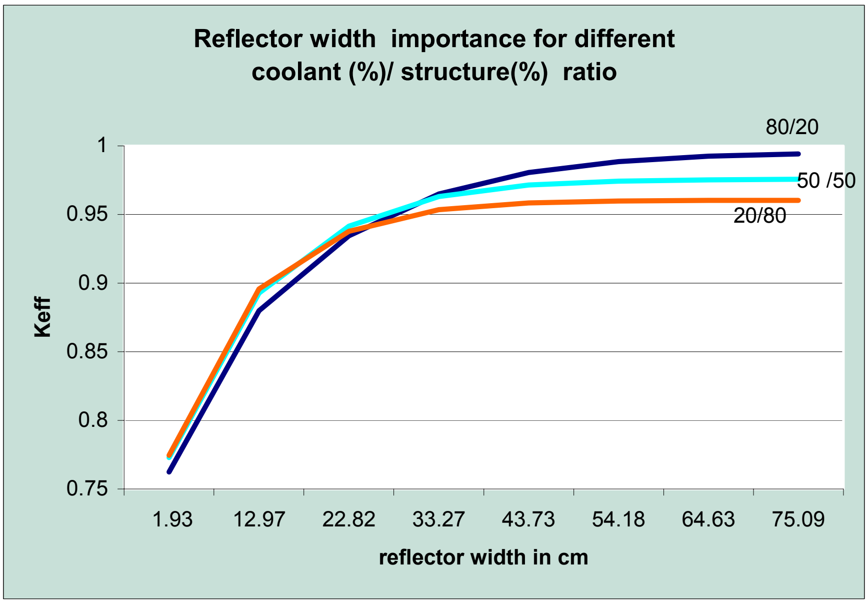


FIGURE 1. $k_{eff}$ as a function of reflector thickness. Three coolant to structure volume ratios are considered for the reflector. The target region for all cases is represented by 80% of LBE and 20% of structure material. The fuel temperature is 1273 K.

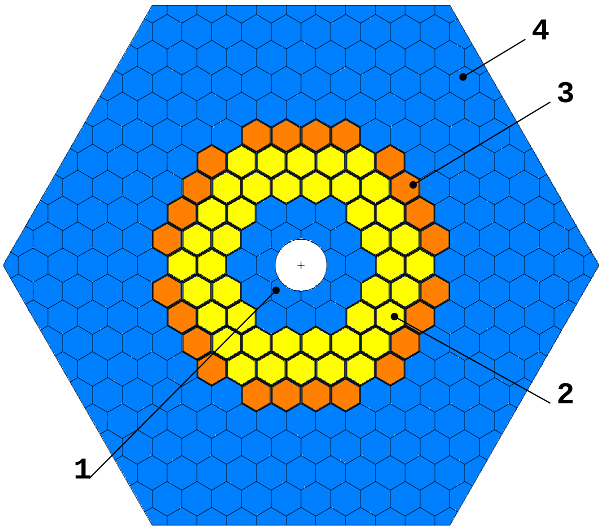


FIGURE 2. Reference SNR–300 core design. Low enriched C1_MAG assemblies (zone 2) are surrounded by high-enriched fuel assemblies of type C2_LWR (zone 3). Zone 1 target zone, represented by LBE mixed with structure material, zone 3 reflector, made of LBE - structure mixture

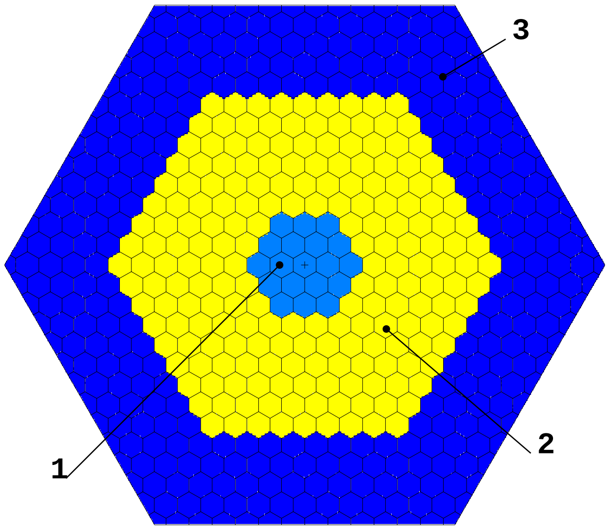


FIGURE 3 Core layout for the SPX fuel. Zone 1 – the target zone, represented by LBE mixed with structure material; zone 2 most enriched SPX fuel in ANSALDO sub-assembly configuration; zone 3 reflector, made of LBE - structure mixture.

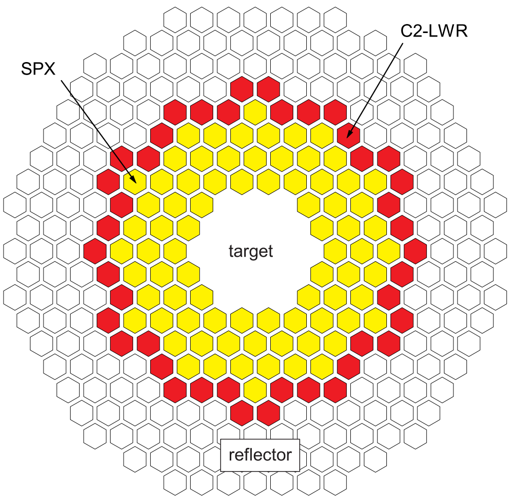


FIGURE 4 ANSALDO core layout for the two enrichment option. The target zone, represented by LBE mixed with structure material; SPX – most enriched SPX fuel; C2_LWR – most enriched SNR fuel, reflector is represented by homogenized LBE and structure mixture.

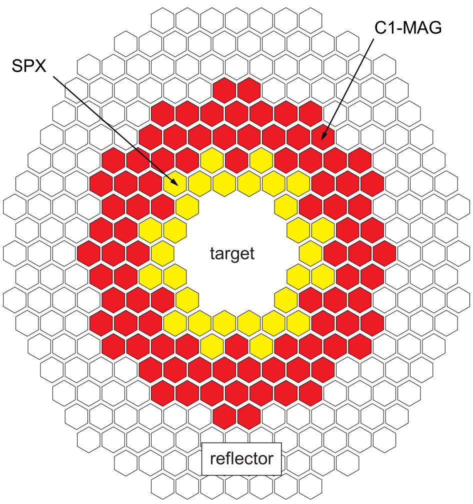


FIGURE 5. ANSALDO core layout for the two enrichment option. The target zone, represented by LBE mixed with structure material; SPX – most enriched SPX fuel; C1_MAG – SNR fuel (with lower enrichment compared with C2_LWR), reflector is represented by homogenized LBE and structure mixture.

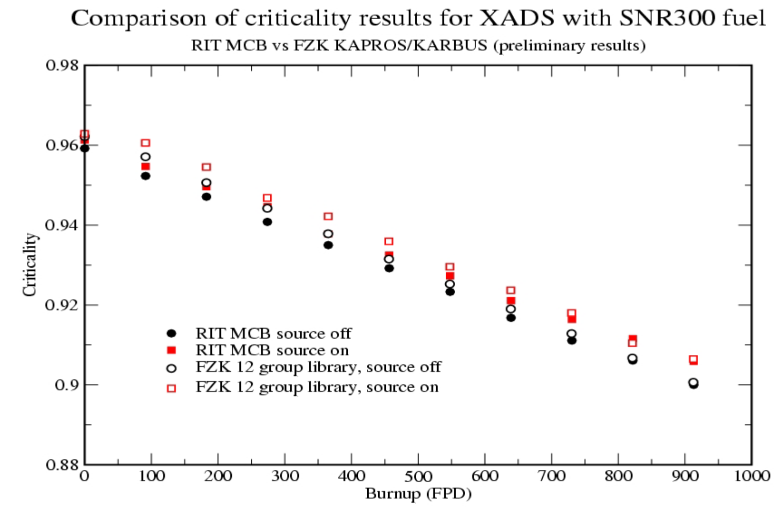


FIGURE 6. Comparison of RIT Monte Carlo and FZK deterministic results for burn up dependant criticality values for XADS with SNR300 fuel. Source on/off states are plotted.

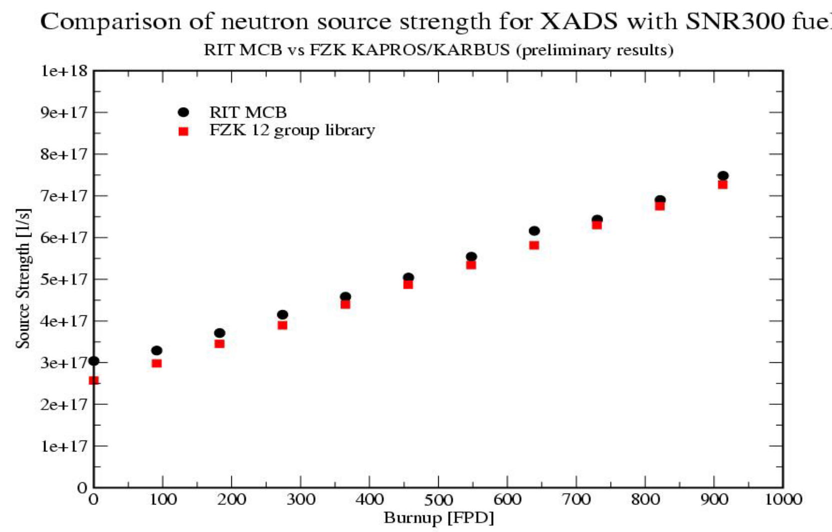


FIGURE 7. Comparison of RIT Monte Carlo and FZK deterministic results for burn up dependant source strength for XADS with SNR300 fuel

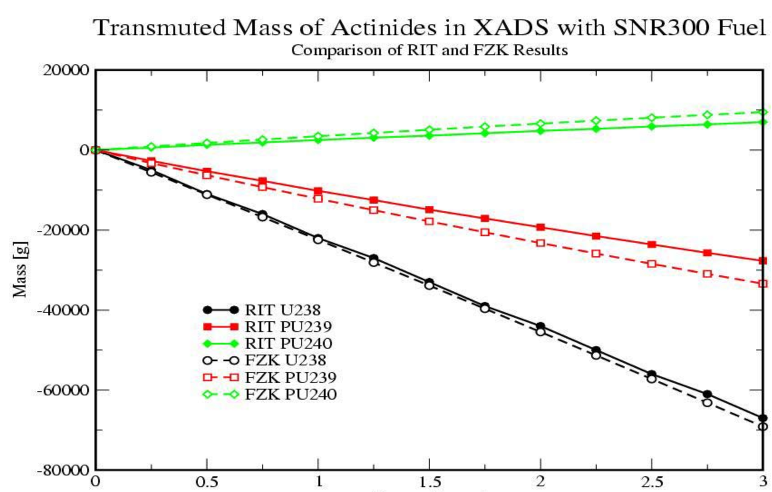


FIGURE 8. Comparison of RIT Monte Carlo and FZK deterministic results for burn up