

## Report on Intermediate Results of the IAEA CRP on 'Studies of Advanced Reactor Technology Options for Effective Incineration of Radioactive Waste'

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### ABSTRACT

In 2003 the IAEA has initiated a Coordinated Research Project (CRP) on “Studies of Advanced Reactor Technology Options for Effective Incineration of Radioactive Waste”. Major intermediate results have been obtained and will be reported here. The overall objective of the CRP, performed within the framework of IAEA’s Nuclear Energy’s Department *Technical Working Group on Fast Reactors*, is to increase the capability of Member States in developing and applying advanced technologies in the area of long-lived radioactive waste utilization and transmutation. Sixteen institutions from 12 member states and one international organization participated in this CRP. The CRP concentrated on the assessment of the dynamic behaviour of various transmutation systems. The reactor systems investigated comprise critical reactors, subcritical accelerator driven systems with heavy liquid metal and gas cooling, critical molten salt systems and hybride fusion/fission systems. Both fertile and fertile-free fuel options have been investigated. For a deep assessment of the transient and safety behaviour, the analytical capabilities have to be qualified. A major effort of the CRP consisted in the benchmarking of steady state core configurations and performing transient/accident simulations. For a general assessment and comparison, the safety coefficients were determined for the individual systems. In a second step transient analyses were performed which reflected the generic behaviour of the various reactors types. In addition the transmutation potential, burn-up behaviour and decay heat of minor actinide bearing fuels were investigated.

**KEYWORDS:** IAEA, Coordinated Research Project, transmutation systems, kinetics, dynamics, transients, benchmark

## 1 I. INTRODUCTION

The IAEA has initiated a Coordinated Research Project (CRP) on “Studies of Advanced Reactor Technology Options for Effective Incineration of Radioactive Waste” /1/ The overall objective of the CRP, performed within the framework of IAEA’s *Technical Working Group on Fast Reactors (TWG-FR)*, is to increase the capability of interested Member States in developing and applying advanced technologies in the area of long-lived radioactive waste utilization and transmutation. The final goal of the CRP is to deepen the understanding of the dynamics of various transmutation systems (e.g., the accelerator driven system), in particular of systems with large minor actinide load and deteriorated safety parameters, to qualify the available methods, specify the range of validity of these methods, and formulate requirements for future theoretical developments. Based on the results, the CRP will conclude on the potential need of transient experiments and make appropriate proposals for experimental programs. Sixteen institutions from 12 Member States and one international organization participated in this CRP.

For a sound assessment of the transient and accident behaviour, neutron kinetics and dynamics methods and codes have to be qualified, even more so as the safety relevant neutronics parameters are becoming unfavourable in transmutation systems, especially in transmuters with fertile-free fuels. Hence, the availability of adequate and qualified methods for the analysis of the various systems is an important point of the exercise. A benchmarking effort between the codes and nuclear data used for the analyses has been performed, which will eventually substantiate the methodology validity range assumptions, and also identify requirements for future theoretical and experimental research. The inter-comparisons performed within the framework of the CRP are not merely a comparison exercise between codes, but should reflect the overall status of methods and data-bases used by the individual participants. Therefore, individual responsibility is given to the participants to use their methods and data-bases. The main thrust of the benchmarking work is on ‘long timescale’ effects of transients in the ms to s and longer range, initiated by strong perturbations of the core and/or the external neutron source. This means that changes of the flux-shape and power distribution caused, e.g., by a strong reactivity perturbation are in the centre of interest.

The comparative investigations cover burner reactors and transmuters both containing fertile and fertile-free, so-called ‘dedicated’ fuels. These reactors are loaded with differing amounts of minor actinides (MAs). The systems are designed either as neutronically critical or sub-critical (hybrid) driven by an external neutron source. The neutron spectra of the reactors extend from low thermal to fusion neutron energy levels. Further, both systems with solid fuels and molten salt fuels are compared. The solid fuel systems investigated range from ordinary MOX to advanced dedicated fuels and cover also the impact of various coolants from sodium to heavy liquid metals and gas.

Specifically, the systems investigated are allocated to eight different domains, which comprise in detail:

- Critical transmuters with fertile fuel
- Critical transmuters with fertile-free fuel
- Hybrid systems (ADS) with fertile fuel
- Hybrid systems (ADS) with fertile-free fuels
- Molten salt reactor systems with fertile fuel
- Molten salt reactor systems with fertile-free fuels
- Gas cooled reactor systems (ADS)
- Hybrid Fusion/Fission systems

For a general assessment and comparison, the safety coefficients (prompt feedback effects like the Doppler effect, thermal fuel expansion, the delayed feedback from clad, coolant and other core constituents, and finally the kinetics parameters) are determined for the individual systems. In a second step, transient analyses are performed which reflect the generic behaviour of the reactor types and should allow a comparative assessment of, e.g., fertile versus fertile-free cores, critical versus sub-critical source driven systems, and solid fuel versus molten salt fuels. Issues as the transmutation potential, burn-up behaviour and decay heat of MA bearing fuels were investigated. According to the topics of the conference, the paper will focus on the most advanced designs to be realized in the future.

## ***II. Results of Analyses for the Different Domains***

### ***II.1 Domain I: Critical Transmuters with Fertile Fuel***

A critical sodium cooled fast reactor (FR) with solid fuel and a fertile material component have been proposed and investigated for this benchmark by IGCAR & BARC. The dominant producer of MAs in India is the PHWRs, the potential burner would be a FR. The benchmark for the CRP is a modification of the PFBR reactor whose design has been frozen /2, 3/. The MAs considered have the composition corresponding to the Indian PHWRs, and the benchmark fuel composition considers 5 % of these MAs. The axial blanket is  $\text{UO}_2$ , the radial blanket is made of  $\text{ThO}_2$  in order to minimize this concept's MA production. The maximum fuel burnup is 100 GWd/t. Static and transient analyses have been performed. The static neutronic analyses concentrated on the influence of different nuclear data bases as ENDF/B-VI, JENDL-3.3 and JEFF-3.1. The spread maximum to minimum k-eff was 200 pcm, overall material worth was in the range of +/- 10%, while the fuel Doppler values could be off by a factor of 3. For the transient analyses an unprotected loss of flow (ULOF) and an unprotected overpower (UTOP) have been simulated. It could be shown that the limited amount of MAs in the core did not compromise the safety of the reactor. A 10 % reduction in the long-lived MAs is attained during an equilibrium cycle, however accompanied by a substantial production of  $^{233}\text{U}$ . The second concept, a benchmark model proposed by IPPE, is a BN-800 type reactor /4, 5/ with a  $(\text{Pu-Th})\text{O}_2$  fuel containing MAs of VVER origin. In this benchmark, the MAs produced during the burnup are recycled, hence added to those of VVER origin. The self-production of MAs is reduced and the bred  $\text{U233}$  could be used to start a thorium cycle. The benchmark model is based on the BN-800 reactor /5/, except for the fertile component in the fuel ( $^{232}\text{Th}$  instead of  $^{238}\text{U}$ ). The static calculations show that a negative sodium void reactivity effect (SVRE) can be achieved in this BN-800 type reactor with  $\text{PuO}_2\text{-ThO}_2$  in case an upper sodium plenum is provided. No more than 26 kg of external VVER MAs per year should be added. In case of allowing a moderate positive void worth of 0.7 %dk/k the MA mass could increase to 104 kg/year. In this case 7 VVER-1000 reactors could be supported.

### ***II.2. Domain II: Critical Transmuters with Fertile-Free Fuel***

Two 600  $\text{MW}_e$  reactors, a lead and a sodium cooled one, are compared regarding safety relevant reactivity coefficients, waste-burning capabilities and reactivity swings during burn-up /6, 7/. The reactors analyzed are fuelled with a CERMET fuel ( $^{92}\text{Mo}$  matrix). The LFR core is larger due to the larger pitch-to-diameter of 1.5 compared to 1.2 for SFR. By

introducing additional pins with BeO moderator both cores show a negative Doppler that is larger than the positive coolant reactivity coefficient (LFR: Doppler  $-0.50$  pcm/K and  $0.38$  pcm/K coolant reactivity increase, SFR:  $-0.54$  and  $0.36$  pcm/K). As transients an Unprotected Loss-of-Flow (ULOF) and Loss-of-Heat Sink (ULOHS) have been simulated. The relatively large LFR overcomes the unprotected Loss-of-Flow due to its superior natural circulation whereas the SFR gets into sodium boiling. The latter may be avoided if fast negative structural feedbacks could be proven to be large enough. In the unprotected Loss-of-Heat Sink accident the grace time of the LFR is considerably larger but for these longer times the lower grid expansion, which was not considered, could terminate the accidents. The LFR burner can annually transmuted  $\sim 300$  kg of plutonium and MAs corresponding roughly to the annual production of the transuranics of a  $1.1$  GW<sub>e</sub> LWR. Annual TRU consumption in the SFR burner is slightly less and equal to  $263$  kg. However, due to lower actinide inventory in the SFR, the actinide burn-up rate is larger than in the LFR.

### ***II.3. Domain III: Hybrid Systems (ADS) with Fertile Fuels***

The Pb/Bi cooled MYRRHA /8/ ADS has been chosen as the fertile hybrid benchmark case and was supplied by SCK.CEN. The MYRRHA sub-critical reactor with a thermal power of  $50$  MW has been developed within the 5<sup>th</sup> Framework Programme (FP5) of the European Commission. The basic sub-critical core configuration consists of  $45$  fuel assemblies containing  $30$  wt% Pu-enriched MOX fuel. In a second step a core with  $24$  U-free MA and  $48$  MOX assemblies has been defined. The U-free pins contain a MgO matrix. The static neutronic analyses were performed by different laboratories using the MCNPX.2.5.0 code based on the JEFF3.1 data file /9/. The ALEPH code (coupling of MCNPX with ORIGEN2.2) /10/ was used for burn-up calculations. The static results show that the Doppler is slightly lower and the neutron generation time is by an order of magnitude less than in a sodium cooled FR. The mixed core showed a reduced burn-up swing. The core void worth is negative. For transient studies, the RELAP /11/ and the SITHER /12/ codes were used. The transients to be investigated were subdivided into two categories, namely the protected transients and the unprotected transients. The transient overpower (TOP), , LOF , LOHS and blockage accident have been analyzed. The MYRRHA core can survive most of the protected accidents without melting. Only the blockage accident leads to limited damage. For the unprotected accidents, clad failure will occur for the ULOF and the ULOHS. In case of a beam cut-off the transients could be coped with.

### ***II.4. Domain IV: Hybrid Systems (ADS) with Fertile-Free Fuels***

FZK and Kyushu University contributed two benchmark cases. Two fertile-free ADS systems, both with a  $580$  MW<sub>th</sub> power with three core zones and varying fuel/matrix and Pu/MA ratios have been developed and investigated. The general reactor concept and a radial core layout are displayed in Fig. II.1 and II.2.

The models are based on past FZK studies and on experience gained through FZK participation in the FP5 of the EU, namely the FUTURE project /13/. The fuels are based on composites and especially the CERCER fuels with a ZrO<sub>2</sub> and MgO inert matrix have been chosen. In addition a Mo based CERMET fuel (especially with Mo-92) has been investigated /14/. Assessment and analyses showed that the ZrO<sub>2</sub> fuel represents only a back-up solution if aqueous processing is a requirement. Currently the Mo fuel is favored as it shows superior

behavior for all transients. With the MgO matrix fuel a better transmutation ratio can be achieved and this fuel currently chosen as reference in the EUROTRANS project /15/. The interesting features of the fertile-free cores is their lack of a prompt stabilizing Doppler effect, a very small beta-eff and large reactivity potentials from coolant, clad and fuel (delayed reactivity effects) . Therefore these reactors can only be realized in a subcritical manner. The void worth (see below) of these cores is generally much higher than the in-built subcriticality of  $k_{eff} = 0.97$ . Voiding would not be introduced via coolant boiling (T-boil ~ 2000 K), but via pin failures and release of the fission gases and helium from the transmutation process /16/ or from a steam generator tube rupture /17/.

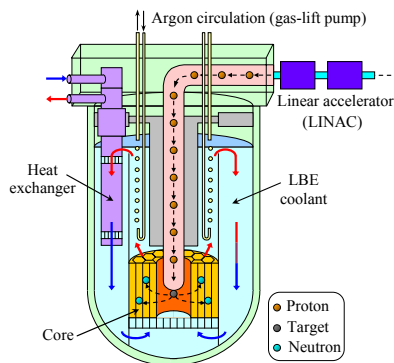


Fig. II.2 General sketch of an ADS

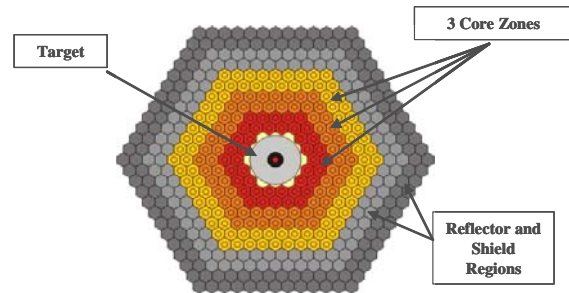


Fig. II.1 Radial core layout for an ADT with fertile-free fuel (CERCER MgO matrix fuel)

The fuels of the cores coming from a MOX LWR, contain 40 % Pu, 50 % Am and 10% Cm. Matrix fractions are 50 % and higher to guarantee a continuous matrix and improve thermal conductivity. Both the static and transient calculations were performed with the help of the SIMMER-III code /18, 19/. The SIMMER neutronics results were benchmarked against MCNP /20/, ERANOS /21/ and DANTSYS /22/. SIMMER uses an 11-group data library based on FZK data for the static and transient analyses. Benchmarking was done against JEF2.2, JEFF3.0, JENDL3.3 and ENDF/B-IV.8. The data processing at FZK has been performed with the C4P code system /23/. The results show good agreement. LBE nuclear data were the main reason for deviations between the different data files. The safety coefficients as the MgO and ZrO<sub>2</sub> core void worth ranged from 6500 pcm (FZK-11 group, JEF2.2-30groups) to 8300 pcm / 7700 pcm (JEFF3.0, JENDL3.3, both 30 groups). For the transient analyses the Unprotected Transient Over Current (UTOC, characterized by beam over-power up to 150% and 200% at reactor hot full power condition), ULOF, UTOP, and the unprotected sub-assembly blockage accident were performed. In Figs. II.3 and II.4 the results of the ULOF calculation for the MgO core are displayed.

Recent analyses /14/ revealed safety problems of the MgO matrix above 2100K caused by a possible disintegration process in case of pin failures. The benchmark results show that these temperature levels are not reached in case of a ULOF.

Besides establishing the basic thermal-physical properties for the materials considered in the benchmarks (lead-bismuth eutectic, fertile-free fuel with inert matrix), the Kyushu University group contributed experimental confirmation for important accident modeling phenomena, in particular related to molten clad freezing in case the clad could melt after a subassembly blockage accident.

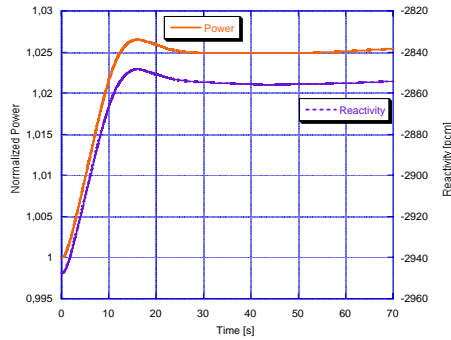


Fig. II.3 Power and reactivity trace for the ULOF in the CERCER MgO matrix core

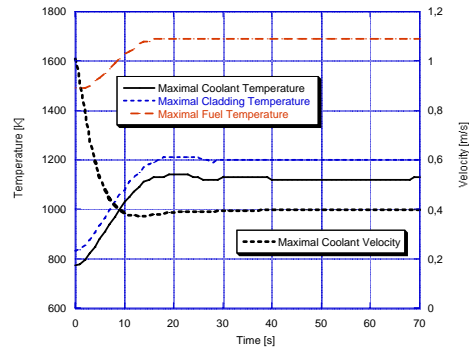


Fig. II.4 Temperatures and coolant flow velocity during a ULOF in the CERCER MgO matrix core

Besides establishing the basic thermal-physical properties for the materials considered in the benchmarks (lead-bismuth eutectic, fertile-free fuel with inert matrix), the Kyushu University group contributed experimental confirmation for important accident modeling phenomena, in particular related to molten clad freezing in case the clad could melt after a subassembly blockage accident.

### ***II.5. Domain V: Molten salt reactor systems with fertile fuel***

The benchmark proposed for Domain V is the Li/Be/Th-F AMSTER incinerator design /24/ as put forward by EdF during the 5<sup>th</sup> FP of the EU MOST Project /25/. AMSTER is based on the MSBR design, as initially proposed by the Oak Ridge National Laboratory (ORNL). For the benchmark analysis experimental results from the former Oak-Ridge MSRE tests were available. They have been used to benchmark various code systems as SimADS /26/, SIMMER /19/, DANAMOSS /27/ in pump startup and pump coast down tests. Because of the special dynamics of the molten salt reactors with moving neutron precursors, a pump coast-down leads to an increase in reactivity in the reactor. For the AMSTER core the safety coefficients have been calculated by APOLLO2 /28/ and WIMS8a /29/. Various transients have been analyzed as ULOF, ULOHS and UTOP. A special transient is the unprotected overcooling of the primary fuel because of secondary side malfunctioning. An important finding was that ER-167 has to be added to the graphite to obtain a negative graphite reactivity coefficient above 870 K and stabilize the reactor. Generally, the large thermal inertia of the systems damps its dynamics and allows significant grace times for reaction.

### ***II.6. Domain VI: Fertile-Free Molten Salt***

The benchmark for Domain VI is based on the Na/Li/Be-F MOSART concept /30, 31/ proposed by RRC-KI. This concept is developed in the ISTC 1606 project with various foreign collaborators. The design which is the basis of the benchmark model is displayed in Fig. II.5. The focus of the benchmark is placed on double component scenario, in which a Na,Li,Be/F MOSART is used as TRU burner system of the LWR long lived radioactive wastes. The start up and feed fuel material for MOSART critical core is a typical composition of TRU's from UOX spent fuel of a commercial PWR (60 GWd/tU - 4.9% <sup>235</sup>U/U; after 1 year cooling). The concept design is displayed in Figs. II.5 and II.6.

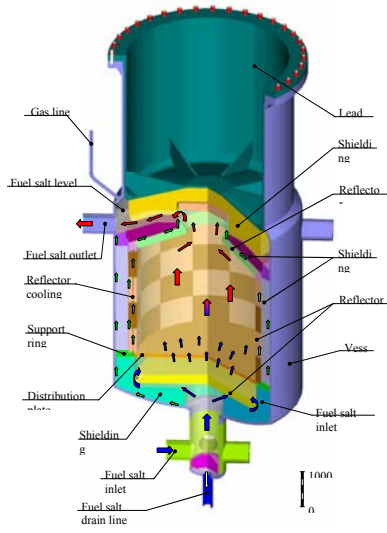


Figure 1: MOSART core

Fig. II.5 MOSART concept core set-up

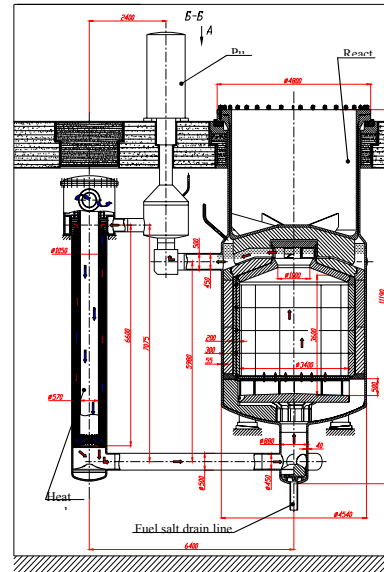


Figure 2: MOSART fuel circuit

Fig. II.6 MOSART concept circuit

The 2400MWth MOSART system has the cylindrical core having an intermediate to fast neutron spectrum of neutrons. No solid material is present in the core of this reactor as moderator, only as external reflector. Fuel salt is a molten  $58\text{NaF}-15\text{LiF}-27\text{BeF}_2$  (mole%) mixture with 752 K melting temperature and addition of about 1.05 mole% of  $(\text{TRUF}_3 + \text{LnF}_3)$  with mass proportion at equilibrium for the chosen fuel cycle scenario with a soluble fission product removal cycle of 300 epdf. The salt inlet temperature in core is assumed as 873K. At that temperature solubility of  $(\text{TRUF}_3 + \text{LnF}_3)$  is about 2 mole%. Fuel salt volume out of the core is  $V_{\text{loop}} = 18.40 \text{ m}^3$ .  $M_{\text{loop}} = 2140 * 18.4 = 39363 \text{ kg}$ . The fuel salt specific power is about  $47 \text{ W/cm}^3$ . The core salt mass flow rate  $G = 10000 \text{ kg/sec}$  and the average axial velocity of stream in the core is equal to 0.5m/s. The primary circuit material is Hastelloy NM, designed for an operation temperature of 1023 K. The melting point is at 1644K. For the static neutronic analyses both multigroup deterministic (DANTSYS /22/, SIMMER /19/, XSDRNPM /32/ and Monte-Carlo models (MCNP /20/ or MCNPX /9/, MCU /33/) were employed. Several nuclear data libraries (in particular ENDF/B-VI, JEF 2.2, JEFF 3.0, JEFF 3.1, JENDL 3.3) were used for performing the analyses. Participants made computations with different versions of the MCNP or MCNPX codes. The analyses demonstrated the essential differences in the data libraries concerning minor actinides. At the same time the results obtained with the different codes using the same data library were remarkably good with a  $k_{\text{eff}}$  deviation of less than 2.5 %. Reactivity coefficients for the molten salt and graphite reflector, kinetics data and power densities have been benchmarked. A steady state power density distribution is given in Fig. II.7. For molten salt reactors the delayed precursor movement is a key issue and was studied both with the SIMMER code /34/ and DYNAMOSS /35/. Preliminary evaluations of the effect of the precursor movement at steady-state show a relatively high reduction of the effective delayed neutron fraction(40-50%). This effect as well as the temperature distribution in the core strongly depend upon the velocity profile. This velocity profile depends on the design like the insertion of a distribution plate. For the MOSART design it becomes obvious that the redistribution of the individual precursor family within the core and the re-mixing in the outlet plays a significant role. Various transients as the UTOP (caused by a sudden reactivity addition because of an incidental MA fuel addition), ULOF, ULOHS and UOC are simulated in the benchmark

activity /36/. In Fig. II.7 the temperature redistribution and peaking based on a SIMMER calculation in case of a ULOF is displayed.

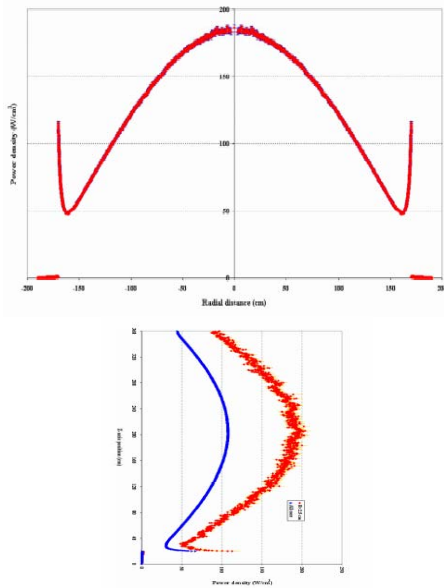


Fig. II.7 MOSART power density distribution in  $\text{W/cm}^3$ : Radial (top) and axial (bottom) profiles in cm; MCNPX JEFF3.1 system

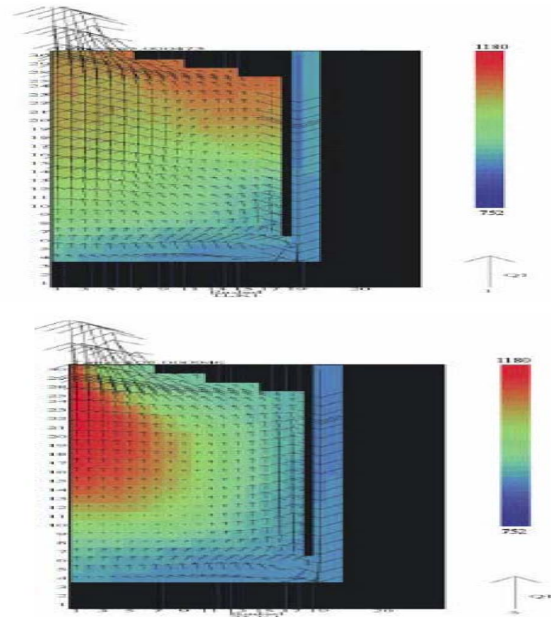


Fig. II.8 Temperature distribution after a ULOF calculated with SIMMER-III ( $T_{\text{max}} = 1180 \text{ K}$ )

In addition to the benchmark activity information is collected to describe the state of the art in molten salt fuel treatment /37/. Fuel processing and reprocessing technologies are divided from the point of view of their emplacement in the MSR-actinide burner fuel cycle to the “Front-end” and “Back-end” technologies. This categorization is important, because the LWR spent fuel reprocessing and the subsequent liquid molten salt fuel processing have quite different character in comparison to the “on-line” reprocessing (fuel salt cleaning up) of circulating MSR fuel. The final part of fuel processing technology has also to be placed in the reactor site and tightly attached to the “on-line” reprocessing, because the refilling of the fresh fuel into the reactor has to be carried out in connection with the removal of the burned out fuel and fission products from the primary fuel circuit.

## II.7. Domain VII: Gas cooled hybrid

CEA proposed a benchmark of the helium cooled ADS with dedicated MA fuels based on a  $400 \text{ MW}_{\text{th}}$  gas-cooled, inert (MgO) matrix fertile-free fuel with Pu/MA content such that  $\text{Pu}/(\text{Pu}+\text{MA})=36\%$ . The matrix/fuel ratio is around 40%. Helium pressure is 60 bar, the pressure drop 0.5 bars, while the inlet temperature is 473 K, and the outlet temperature 623 K. The average and peak volumetric power is  $44 \text{ W/cm}^3$  and  $94 \text{ W/cm}^3$ . The calculated sub-criticality level was  $k_{\text{eff}}=0.98$ . For this benchmark several combinations of Monte-Carlo and deterministic codes and four libraries (JEF2.2, JEFF3.1, JENDL3.3, ENDF/B-VI) with several variants both in the codes and in the data have been applied. The code list comprises ERANOS2.0 /21/, TRIPOLI4 /38/, MCNP4C /39/, OCTOPUS (MCNP4C3+FISPACT) /40/ MCNPX.2.5.0 /9/, DANTSYS/22/ + C4P /23/. At the present status of the CRP only static

neutronic results are available as the transient codes have to be updated and adjusted. The main comparison has been carried out for the following parameters: reactivity, Doppler effect, coolant depressurisation, core compaction, kinetic parameter  $\beta_{\text{eff}}$ , mass inventory with transmutation rate. Initially a large discrepancy was observed between the results of the participants for the initial reactivity, especially the ERANOS results. After several studies the reason was found. The recommendation for ERANOS is now to treat both Mg and O and Minor Actinides at the fine group level and with JEFF3.1 when calculating cores with large content of MgO and Minor Actinides. JEFF3.1 is the only library used by ERANOS to include both nuclides in fine groups and therefore allows a correct treatment of the overlapping of Mg and O resonances. It includes also the most up-to-date evaluations for Minor Actinides. End of cycle reactivities are particularly sensitive to the branching ratios used in the TRU chains and to the fission product treatment. It seems necessary to improve the ERANOS result here. A more precise burn-up calculation must be performed based upon the use of 87 explicit fission products with ERANOS. Concerning the mass balance prediction, a good agreement between the deterministic codes and the Monte-Carlo codes has been reached. It is hoped that at the end of the CRP also transient studies for benchmarking will be available.

### ***II.8. Domain VIII: Fusion/fission hybrid***

The fusion/fission hybrid benchmark has been proposed by ASIPP /41, 42/ and AGH University of Science and Technology /43, 44, /. The fusion/fission hybrid benchmark proposed by ASIPP is based on a preliminary tokamak fusion/fission hybrid concept called FDS-I, whose missions are plutonium breeding, as well as incineration and transmutation of minor actinides and long-lived fission products /42/. The AGH proposed a Tandem Mirror Concept with the goal of minor actinide incineration. In Figs. II.9 and II.10 the conceptual designs are displayed.

One advantage of subcritical systems lies in their neutronic distance to delayed and super-prompt criticality. A second advantage of that option is the prospect of a radical reduction of necessary plasma energy gain  $Q$  (because of the energy release from fissions) to levels achievable in much smaller i.e. much more economic devices. The coupling of fission and fusion with the objective of transmutation/incineration appears to be advantageous, since the fission process is energy rich and neutron poor (80 MeV/n), while the fusion process is neutron rich and energy poor (17.6 MeV/n). Therefore, the fission fraction of the energy released in a fusion-driven sub-critical system can be very high (up to 90%) already at low values of  $k_{\text{eff}}$  (as low as e.g. 0.6). For such a system also the value of the plasma energy gain ( $Q$ ) can be low, depending upon the efficiency of the electricity self-consumption and its fraction of the total energy produced by the fusion-driven sub-critical system. It can be concluded that the requirements regarding the plasma  $Q$  can be significantly relaxed in the case of a fusion-driven sub-critical system, down to levels achievable in small Tandem Mirror Concepts. Another interesting advantage of the system is the reduced load of the first wall with 14 MeV neutrons and homogeneous heating distribution. In Fig. II.11 the radiation damage effects in the first wall are compared between a pure fusion and a fusion-fission device of the same power class. As can be seen the radiation damage of the fusion-fission hybrid proves much less intense. The gas production is less by 50 and the dpa by ca. 15 times.

For the AGH proposed 500 MWth Tandem Mirror Concept with a  $k_{\text{eff}}$  of 0.84, neutronic analyses have been performed to calculate e.g. the neutron wall load with approximately  $0.5 \text{ MW/m}^2$ , the uniform distribution of the nuclear heating due to the introduction of fission

power and the transmutation performance. In addition the worst credible accident scenario (collapse of the Tandem Mirror System, Fig. II.9) has been simulated and showed that criticality is not reached.

The FDS-I plasma core (Fig. 10) has a fusion power of approximately 150 MWth, major and minor radii of about 4 and 1 m, respectively, and an elongation factor of 1.7. The neutron wall load is approximately 0.5 MW/m<sup>2</sup>. The sub-critical “waste” blanket is cooled by helium and lithium-lead eutectic [dual-cooled waste (DWT) blanket] and contains the various zones (incineration of MAs, transmutation of long-lived fission products, plutonium breeding). The neutron static calculation benchmarking has been performed with the VisualBUS multifunctional neutronics analysis code system, containing both S<sub>N</sub> and Monte Carlo codes. The main data library was HENDL. For this benchmark exercise, the following main transient scenarios were analyzed: unprotected plasma over power (UPUP), unprotected overpower (UTOP), loss of flow accidents (ULOF), loss of cooling accidents (ULOC), loss of heat sink accident (LOHS) and collapse accident.

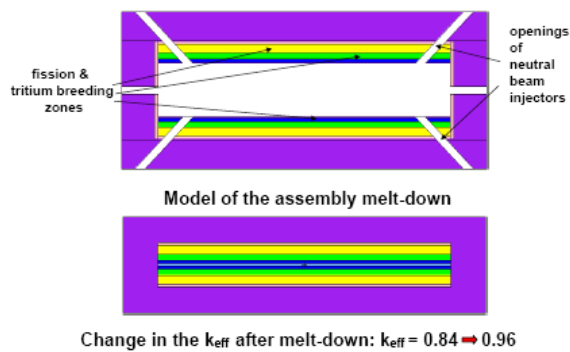


Fig. II.9 MCNP model of a Tandem Mirror system and calculation of a collapse scenario

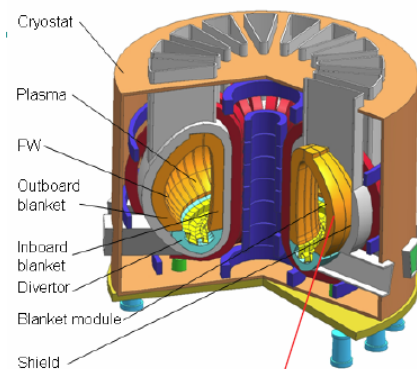


Fig. II.10 Model of the FDS-I Tokamak design

Effect Option	energy [MeV/sr.n <sup>+</sup> ]	tot. H prod. [nucl./(s.n.* cm <sup>3</sup> *s.n.enrg)]	tot. He prod. [nucl./(s.n.* cm <sup>3</sup> *s.n.enrg)]	dpa [events/(s.n.* s.n.enrg)]
fusion	22.4	2.69E-07	5.08E-08	6.21E-07
fus-fis	1100	5.94E-09	1.06E-10	4.05E-08
fu-fi/fu ratio	49.1	0.0225	0.0208	0.0652

Fig. II.11 Comparison of radiation damage effects in the first wall (same power of fusion and fusion-fission)

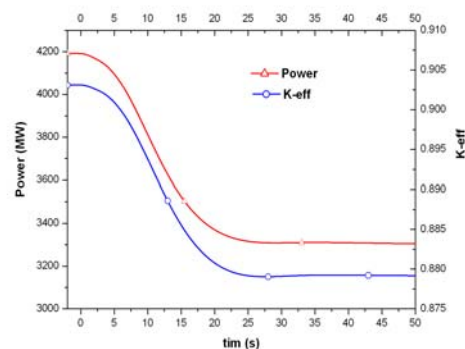


Fig. II.12 ULOF simulation for the FDS-I

The static neutronic results show a significant influence of the geometrical models (1D, 2D and 3D) chosen for the analyses.

For the FDS-I the ULOF simulation describing a coast down of the pumps and the loss of forced LiPb coolant circulation is displayed in Fig. II.12. The remaining natural convection of

the coolant would prevent the fuel particles from melting and also a melting of the structural steel for the first 30 seconds. Results of the collapse scenario for the FDS-I show that up to a collapse of three blanket modules subcriticality is maintained.

For the transient analyses the negative temperature reactivity coefficient is important caused by the fuel inventory decrease inside the blanket while the coolant is expanding.

### ***III. CONCLUSIONS AND FINAL REMARKS***

In 2003 the IAEA has initiated a Coordinated Research Project (CRP) on “Studies of Advanced Reactor Technology Options for Effective Incineration of Radioactive Waste”. The CRP concentrated on the assessment of the dynamic behaviour of various transmutation systems. Major intermediate results have been obtained and have been reported here. The reactor systems investigated were categorized in 8 domains and comprise critical reactors, subcritical accelerator driven systems with heavy liquid metal and gas cooling, critical molten salt systems and hybride fusion/fission systems. For all reactor systems a fertile and a fertile-free fuel option have been investigated. The present paper presented an overview of all systems, with a special focus on the most advanced systems as an ADS with fertile-free fuels, a fast spectrum Molten Salt Reactor and a Fission-Fusion Hybride System.

A major effort of the CRP consisted in the benchmarking of steady state core configurations and performing transient/accident simulations. For a general assessment and comparison, the relevant safety coefficients were determined for the individual systems. In a second step transient analyses were performed which reflected the generic behaviour of the various reactors types. Issues as the transmutation potential, burn-up behaviour and decay heat of minor actinide bearing fuels were investigated. The intermediate results of the CRP showed that for steady state analyses the neutronic tools in each domain are advanced enough to provide good agreement. This holds for both mechanistic  $S_N$  and Monte Carlo codes. Larger spreading of results is generally caused by the different nuclear data libraries used. These deviations may not only be caused by the minor actinide data but also by data of other constituents as e.g. the treatment of matrix material in inert fuels and the fission products. For all domains transient calculations have been or will be performed. Very different code systems were employed from point-kinetics to space-time kinetics, and also different levels in the sophistication of the thermal-hydraulic modelling. The benchmarking shows that code systems of different level are currently needed to cover all the timescales of the different systems and transients. The very detailed codes have difficulties in their running times e.g. for a long-lasting loss of heat sink accidents, while the less detailed codes naturally neglect important phenomena. A need for an intermediate class of codes becomes obvious. With one exception, the benchmarking has exclusively been performed in the range of transients and accidents without core disruption. In case one would allow severe transients in each domain, the benchmarking effort would become even more difficult, but might be of interest in a future CRP. Within the current CRP also a comparison of the dynamic behaviour of the different systems is intended. This comparison has to be balanced in the sense that a large knowledge base exists e.g. for the critical fast reactors whereas less is known e.g. on fusion-fission hybrid systems. Nevertheless characteristic transients, phenomena and time scales can already be identified.

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