

NEA NUCLEAR INNOVATION 2050

R&D COOPERATIVE PROGRAMME PROPOSAL

Advanced Fuel Cycle and P&T

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TITLE/SUBJECT: Advanced Fuel Cycle and P&T steps towards possible industrialisation from 2040 on

1. Justification of the selection (1/2 page)

Today, nuclear power produces 11 percent of the world's electricity. Nuclear power plants produce virtually no greenhouse gases or air pollutants during their operation. Emissions over their entire life cycle are very low. Nuclear energy's potential is essential to achieving a deeply decarbonized energy future in many regions of the world as of today and for decades to come, the main value of nuclear energy lies in its potential contribution to decarbonizing the power sector. Nuclear energy's future role, however, is highly uncertain for several reasons: chiefly, escalating costs and, the persistence of historical challenges such as spent fuel and radioactive waste management. In Europe, 2500 t of spent fuel are discharged annually from reactors containing Plutonium, and Minor Actinides, namely neptunium (Np), americium (Am) and curium (Cm), and long-lived fission products (LLFPs). The long-term management of waste at the back-end of the nuclear fuel cycle remains one of the most critical issues affecting the acceptance of nuclear power and consequently the challenges associated with the global expansion of nuclear power. Spent fuel can be reprocessed or not according to depending on national fuel cycle options and waste management policies. Currently the reference option is to dispose of these wastes in engineered disposal facilities located in suitable geological formations (ICRP 122, 2013 ;IAEA ,1997;OECD/NEA ,2008).

Advanced nuclear fuel recycling technologies can enable full use of natural energy resources while minimizing proliferation concerns as well as the volume and longevity of nuclear waste. Partitioning and Transmutation (P&T) has been pointed out in numerous studies as the strategy that can relax constraints on geological disposal, e.g. by reducing the waste radiotoxicity and the footprint of the underground facility.. Therefore, a special effort has been made to investigate the potential role of P&T and the related options for waste management all along the fuel cycle. Transmutation based on critical or sub-critical fast spectrum transmuters should be evaluated in order to assess its technical and economic feasibility and capacity, which could ease deep geological disposal implementation.

2. The issue to tackle and objectives to reach (1 page)

Explain what are the problems to be solved and the associated objectives to reach through innovation. The link with the grand Challenges/Opportunities defined by the Adv Panel, as given in annex, should be detailed.

In 2005, the research community on P&T within the EU in collaboration with the DG Research & Innovation of the European Commission started structuring its research towards a more integrated approach. This resulted in a European strategy based on the so-called four building blocks at engineering level for P&T as summarized below:

1. Demonstration of the capability to process a sizable amount of spent fuel from commercial LWRs in order to separate plutonium (Pu), uranium (U) and minor actinides (MA) from Pu based spent fuels,
2. Demonstration of the capability to fabricate at a semi-industrial level the dedicated fuel sub-assembly to be loaded in a dedicated transmuter,
3. Design and construction of one or more dedicated transmuters,
4. Provision of a specific installation for processing of the dedicated fuel unloaded from the transmuter, which can be of a different type than the one used to process the original spent fuel unloaded from the commercial power plants, together with the fabrication of new dedicated fuel.

The four building blocks must be consistently developed in parallel. This approach is applicable in NI2050 and will result in the identification of the costs and the benefits of P&T for closing the fuel cycle and solving the SNF legacy.

3. What is done/exist already, who is doing what, what are the means (resources and infrastructures) (2 or 3 pages)

In most cases, R&D and/or demonstration/validation/qualification programmes and infrastructures already exist and can be briefly described (more details or references can be given in annex). Main actors and resources (financial and infrastructures) have to be detailed to fix the level playing field. The reason why more is necessary, where are the difficulties, delays and bottlenecks, should be explained.

3.1 Advanced Partitioning

Japan

JAEA is developing separation process for MAs (Am and Cm) using new innovative extractants to improve the partitioning process from the viewpoints of the economy and the reduction of secondary wastes. The MA separation process consists of three solvent extraction steps: co-extraction of MA and rare earth (RE), MA/RE separation and Am/Cm separation. Phosphorus-free compounds consisting of carbon, hydrogen, oxygen and nitrogen (CHON principle) were applied to all separation steps. Co-extraction of MA and RE and MA/RE separation were verified by continuous extraction test using simulated HLLW containing MA tracers. The hot test using genuine HLLW for principle demonstration has been started in NUCEF facility. Radiolysis effect for extractants are tested in cooperation with INL. JAEA also develops models and simulation codes to develop separation process.

Russia

The existing Russian NFC infrastructure for the HLW partitioning development includes SNF reprocessing plants - Mayak (RT-1) plant, and the Experimental Demonstration Center MCC. Russia has industrial experience in HLW partitioning. Since 1996, the Mayak (RT-1) plant has operated the HLW partitioning facility. During the operation, more than 1200 m³ of HLW was processed with Cs-Sr recovering with chlorinated cobalt dicarbollyde and polyethylene glycol as an extractant system.

At present, the process with N, N, N', N'-dioctyl diamide diglycolic acid (TODGA) as an extractant in a polar diluent meta-nitrobenzotrifluoride, followed by chromatographic separation of americium and curium has developed. This technology includes both liquid extraction and sorption processes. The "hot" dynamic tests of the technology for recovering Am and Cm from real HLW by the TODGA system were carried out. The recovery of americium is more than 99.9%. Technology for the separation of americium and curium was demonstrated in the pilot plant at Mayak PA. About 14 grams of ²⁴⁴Cm were isolated, of which 9 g is a fraction of enriched curium with an americium content of less than 6% in activity. The mixed fraction of curium-americium contained about 4.6 g of ²⁴⁴Cm and about 40 g of ^{241,243}Am. In the enriched fraction of americium, the curium content was less than 0.8% by weight, and the ^{154,155}Eu content was less than 0.1% by activity.

France

Since 1991 and the first French Act on nuclear waste management, France has carried out extensive studies on advanced partitioning strategies, often completed through European collaborations. The aim of these studies has been to develop process options based on the best scientific knowledge, allowing the implementation of different P&T strategies (heterogeneous recycling or homogeneous recycling). Most of the designed processes were hot-tested in the ATALANTE facility of the CEA Marcoule.

The first concept was a three step approach to recover and separate americium and curium from the PUREX raffinate. The DIAMEX process (1998), based on the DMDOHEMA extractant allows the separation of minor actinides together with lanthanides from the other fission products. The SANEX process (2001), based on a tri-azinyl-pyridine extractant, allows the separation of the minor actinides from the lanthanides. For the separation of americium and curium, the initial process idea, SESAME, was based on an electrochemical oxidation of Am(III) up to Am(VI) (2000). This concept was too complex and finally another option, DIAMEX2 (2002) based on DMDOHEMA allowed the Am/Cm separation.

The following studies were oriented to a simplification of this reference route, by developing the DIAMEX-SANEX process (2005) or the SANEX-TODGA process (2009), and then a one-step approach with the EXAM process (2010) allowing the direct recovery of the americium alone from a PUREX raffinate. DIAMEX-SANEX was also based on DMDOHEMA. It is worth to mention that a EXAM hot-test allowed the recovery of more than 2 g of americium for minor actinide bearing blanket fuel material fabrication.

In parallel, the GANEX process (2008) was developed for the group actinide separation. In a first step, uranium is selectively and quantitatively recovered from the spent fuel dissolution liquor with a monoamide and in a second step, all the transuranic elements are recovered together from this solution with an adaptation of the DIAMEX-SANEX process.

In recent years, and following the efficiency of the SANEX-TODGA process, the use of TODGA extractant developed in Japan allowed a simplification of the different process options and became the reference extractant in most of the flowsheets that were further developed mainly in the frame of European collaboration (ACSEPT, SACSESS, GENIORS). One can mention the 1c-SANEX process (based on a TODGA/BTBP mixture, 2010, Juelich), the EURO-GANEX process (2012, NNL), the EURO-EXAM (based on the TPAEN stripping molecule, 2014, CEA) and more recently, the AMSEL process (KIT, Juelich) for Am alone recovery, still under development and to be hot-tested.

The methodology applied to all these processes is the same: first, a co-extraction of An/Ln with TODGA, and second a selective stripping of An, Am, or TRUs from the Ln, based on a hydrophilic complexing molecule in the stripping aqueous phase.

The work currently carried out focuses on two axes: 1) substituting the TODGA by another DGA derivative which does not present the drawbacks of TODGA (too efficient complexation, low capacity, third phase formation) and 2) optimising the stripping molecule. The current reference molecule is a sulphonyl-BTP, with a sulphur atom that can causes troubles at the conversion step. Teams are working at the development of a CHON molecule with the same efficiency and the best candidate today is the py-tri-diol molecule.

3.2 MA Fuel Production

Japan

In JAEA, the MA-bearing nitride fuel in conjunction with the pyrochemical reprocessing is considered as the prior concept for MA recycling with the ADS. MAs and Pu nitrides are diluted with an inert matrix of ZrN, which forms single phase solid solution, (MA, Pu, Zr)N. JAEA demonstrated formation of a (Np, Pu, Am, Cm)N solid solution from the oxide mixture containing MA separated from HLLW of commercial fuel reprocessing for LWR. JAEA also demonstrated formation of PuN and AmN from Cd-Pu and Cd-Am system, respectively, which are considered in pyrochemical reprocessing for dedicated fuel. Then, small pellets and disk specimens with numerous compositions with ZrN were fabricated. The solid solubility of lanthanide and actinide nitrides into ZrN matrix was experimentally evaluated, and single-phase solid solution formalism with ZrN for various MA compositions can be now simulated using lattice parameter of each component. Essential thermal properties of MA nitrides

and (MA, Pu, Zr)N have been measured in this decade, and a material properties database has been constructed in JAEA. These tests are mainly done in the iron-cell in NUCEF facility in which 10 gram of ²⁴¹Am can maximally be handled in Ar atmosphere.

Irradiation test of Am-bearing nitride fuel pellets with a small-pin scale in JOYO is under planning to understand the irradiation behavior. Development of a fuel behavior analysis code for ADS nitride fuel is also ongoing based on the FEMAXI-7 code.

France

In the context of MA transmutation, the CEA (Commissariat à l'Énergie Atomique), has become a leader in P&T research, as defined by a French program Act [28].

The technical feasibility of P&T in fast neutron reactor was notably demonstrated through the EFTTRA (Experimental Feasibility of Targets for TRANsmutation) and the Superfact experimental irradiations early in the 90's.

Now two P&T strategies are proposed: homogenous and heterogeneous modes. In homogeneous recycling, MA are added (1 up to 5 %) in MOX driving fuels in the core. In the case of heterogeneous recycling, MA are separated from the U–Pu flow and concentrated in larger amounts into dedicated assemblies located at the periphery of the core, in order to lower their impact on the core safety parameters. Several types of Am-based heterogeneous fuels have been considered, including IMF (inert matrix fuels): targets containing AmOx or Pu_{1–y}Am_yOx diluted in an inert matrix (yttria-stabilized zirconia or magnesia for example) and MABB (minor actinide-bearing blankets): uranium–americium mixed oxides.

In recent years, interest in U-based MABB has risen, especially in those integrating only Am. Although there is up to now no industrial-scale experience in fabrication of high MA content fuels specifically designed for transmutation, experimental fuel fabrications are currently carried out at the laboratory scale. Several processes, based on powder metallurgy or wet-chemical techniques have been investigated and developed for the fabrication of MA-bearing fuels.

3.3 Transmutation

Japan

JAEA is developing pool-type ADS, where LBE is used as both the primary coolant and the spallation target. Items of R&D are divided into three technical areas peculiar to the ADS: superconducting linear accelerator (SC-LINAC), LBE as spallation target and core coolant, and subcritical core design and technology. For these technical areas, various R&D activities are progressing.

JAEA fabricated a prototype high beta elliptical cryomodule to test the performance of the electric field and the helium cooling. In addition to the development of the cryomodule, the system study of the SC-LINAC with high conversion efficiency was also performed.

For the spallation target, engineering feasibilities of the beam window were shown assuming unirradiated condition. To estimate irradiation effect, JAEA joins MEGAPIE project and performing PIE. For R&D of ADS target, JAEA plans ADS target test facility (TEF-T) which can accept a maximum 400 MeV-250 kW proton beam from J-PARC linac on a LBE spallation target. To construct TEF-T, JAEA is operating mock-up loop of TEF-T and other loops for material corrosion test and development of LBE technologies (O₂ control, sensors, ...).

For subcritical core design and technology, JAEA has developed subcriticality monitoring technique and validated nuclear data for lead, bismuth and MAs using FCA, KUCA and other critical facilities. Moreover, JAEA plans TEF-P facility to study reactor physics aspects of the subcritical core driven, to demonstrate the controllability of the subcritical core, and to investigate the transmutation

performance using considerable amount of MA. JAEA continues design of TEF-P including licensing aspect and remote handling of MA fuel.

Russia

The technology of MA transmutation is planned for studying using both solid-fuel fast reactors (like BN-800 type) and molten salt reactor (MSR).

The decision to build the power unit with the BN-800 reactor was made taking into account the need for advanced development of Russian CNFC (closed nuclear fuel cycle) innovative technologies. In the initial stage of BN-800 operation, uranium fuel and partly mixed oxide uranium-plutonium fuel (MOX) are mainly used. The first step towards CNFC is the transfer to the full core MOX load with fuel based on the separated plutonium from power reactor SNF. This step is planned for 2019-2020. To achieve this goal, industrial production facility of MOX fuel was created at the MCC. In the "Breakthrough" project, homogeneous and heterogeneous loading of MA to the core are under investigation (See Annex A). It was revealed that the necessary efficiency can be achieved in both.

As to MSR, R&D has mainly focused on fast-spectrum MSR options combining the generic advantages of fast neutron reactors (extended resource utilisation, waste minimisation) with those related to molten salt fluorides as both fluid fuel and coolant including Np, Pu, Am and Cm. The NRC "Kurchatov Institute" carried out complex studies on the MOSART project since 2000, which included for configurations selected: neutronic, thermal hydraulic and safety analysis; experiments concerned the main physical and chemical properties of fuel/coolant salts; compatibility of structural materials and fuel/coolant salts with its chemistry control. The accumulated experience now allows us to move from studying the calculated and experimental capabilities of MOSART concept to obtaining specific technical and technological solutions (See Annex B).

France

Experimental data on MABB fuels are scarce, with the unique irradiation SUPERFACT, performed in the PHENIX reactor in the 80's. In this experiment, the irradiation of $U_{0.6}Am_{0.2}Np_{0.2}O_{1.926}$ pellets irradiated at rather high linear heat rate led to a complete release of helium during irradiation and a highly porous fuel microstructure, consistent with high temperature operating conditions. In addition, the occurrence of a weak Fuel-Cladding Mechanical Interaction (FCMI) and the absence of a central hole were observed.

Complementarily, a comprehensive R&D program aiming at qualifying MABB fuel started in 2008, including, as a first stage, two separate-effect irradiation tests: MARIOS, irradiated within FAIRFUELS and examined within PELGRIMM FP7 European projects, and DIAMINO, implemented within the French national nuclear program. They were completed by MARINE, the first semi-integral irradiation of AmBB fuels, carried out within PELGRIMM FP7 European project.

3.4 MA Fuel Reprocessing

Japan

JAEA has developed a basic concept of the main process for pyrochemical reprocessing of spent nitride fuels based on the pyrochemical processing of metal fuels. Basic studies on the dissolution behavior of pure actinide nitrides have been carried out. The actinides recovered in the liquid Cd cathode by the electrorefining process are converted to nitrides to be used as recycled fuels.

These tests are also done mainly in the iron cell in NUCEF. The component for chopping of fuel pin, electrolysis and chemical dissolution are developed without using radioactive isotopes.

Central Research Institute of Electric Power Industry (CRIEPI) has been developing the technology of pyroprocessing for both metal fast reactor and oxide fuels. Fundamental properties of actinides and lanthanides in the molten chloride salt / liquid metal systems were investigated in-house and in collaboration with domestic universities. The feasibility of the major steps in the pyroprocessing has been demonstrated at laboratory scale in collaborations with JAEA (using U, Pu, Am) and with JRC-Karlsruhe (using irradiated metal fuel). With funding from MEXT, the engineering scale tests simulating oxide/metal fuel processing were carried out with unirradiated uranium (5 kg-U/batch scale) were successfully carried out with sufficient processing rate and favourable material mass balance. Transfer technologies for high temperature liquids (molten chloride and liquid Cd) have been also developed in prototype-scale. Recent activities also include waste treatment process development, investigation of innovative actinide recovery methods and application of pyroprocessing to corium treatment.

4. What can be done to improve/accelerate, ia through cooperation (1 page)

Explain conceptually how to go beyond what is done under 3, to improve and accelerate innovation, using enablers and game changers to overcome the bottlenecks and difficulties. Explain why cooperation seems most effective or necessary to progress and achieve timely and/or cost effective results. If possible, provide a preliminary list of actors already keen to enter the cooperation.

4.1 Advanced Partitioning

- To develop common database of extractants. On this topic, an initiative initiated by the SACSESS project in 2014 is now continued by the OECD/NEA.
- To further develop the modelling/simulation approach in order to optimise the use of the experimental results, orient the R&D needs and reduce the need of expensive pilot test
- benchmark process simulation codes.
- To share hot facilities: (CEA-ATALANTE, CEA-MARCEL-PROUST irradiation loop, NUCEF, Gamma irradiation facility in INL)

4.2 MA Fuel Production

- To develop common database of material property including irradiation behaviour
- To benchmark simulation codes for fuel behaviour
- To share hot facilities and irradiation reactor. Sequential steps are necessary to develop the dedicated fuel, i.e., supply of raw MA, fabrication of irradiation pin, irradiation in nuclear reactor, PIE and reprocessing of irradiated pin. Because it becomes difficult to have a set of these facilities in single country, international cooperation is vitally needed.
- To fabricate, irradiate and PIE of MA containing fuel for FR

4.3 Transmutation

ADS

Reliability of accelerator:

- To assess and confirm the performance criteria for the accelerator (value and/or stability of beam energy, beam intensity, beam size). ADS design requests rather challenging values because of thermal stress induced in (and amplified by!) connected subcritical system. So-called "reliability" has to be high, in other words the unwanted interruptions of the beam have to be small.

Material in LBE:

- To exchange of irradiation/corrosion data
- To exchange of test pieces for irradiation/corrosion test to develop new material
- To jointly develop Design Codes and Norms

Reactor physics and MA nuclear data:

- To share operation technique of ADS
- To share reactors and neutron sources to verify nuclear data of MA and coolant (Link to Expert Group on Improvement of Integral Experiments Data for Minor Actinide Management (EGIEMAM-II))

Fast reactor : reliability of fuel elements loaded with minor actinides

- Simulation & modelling:

To improve fuel performance codes towards the effect of minor actinides, several issues must be especially considered in the models:

- the high helium production during irradiation (and during inter-cycle or cooling) possibly resulting in swelling and/or release with a possible Fuel-Cladding Mechanical Interaction (FCMI) interaction or a higher internal pressure of the pin,
 - the actinides and fission products redistribution associated with higher oxygen potential potentially resulting in Fuel-Cladding Chemical Interaction (FCCI),
 - the lower melting temperature and thermal conductivity affecting the margin to melt.
- Performances
 - Transmutation rates :
to justify neutron-physical characteristics of core with MA and efficiency of transmutation of MA in FR with homogeneous and heterogeneous loading
 - Impact of fuel cycle on efficiency:
to evaluate the effect of reprocessing residue (Fission Products, others...) on fuel properties and fuel behavior
 - Safety analysis of fuel loaded with minor actinides during transients or severe accidents: behavior of fuel element, fuel compatibility with the coolant
 - Innovative concepts
 - New options for dedicated fuels by reconsidering requirements towards transmutation

MSR

- To construct demonstrator

4.4 MA Fuel Reprocessing

- To develop database for fabrication technology and reprocessing technology.
- To share hot facilities for reprocessing of irradiated pin as well as MA Fuel Production.

5. Plan of Actions and necessary means (resources and infrastructures) (5 to 6 pages)

Provide a Plan of Actions to implement the concepts described in 4. This is to be made of concrete projects scope, sequence and timeline, and associated necessary means in terms of resources (estimated budgets and competences) and infrastructures (access to existing or to be new built). An example of timeline is given in annex.

5.1 Advanced Partitioning

It is effective for development of MA separation to stepwise increase amount of MA. Now, JAEA is handling small amount less than 1 gram of MA to develop extractants and construct basic separation process. As the next step, we must handle several grams of MA to estimate effect of high concentrated MA and perform comprehensive test of the selected separation process. JAEA plans to manufacture glove box with effective shielding to treat such amount of ²⁴¹Am and ²⁴³Am. JAEA also plans to perform comprehensive test in hot-cell in NUCEF. Obtained results can be served for common database of extractants and code benchmark. In France, in the CEA ATALANTE facility, the CBP hot cell line allows the testing of advanced partitioning processes on several grams of minor actinides.

In Europe, today, the main challenge is the simplification of the reference flowsheets developed at the CEA or in the frame of the European collaboration, this simplification is driven not only by the efficiency and the performances of the processes (higher selectivity, higher capacity loading, higher solubility, more concentrated flowsheet) but also by their safety as a whole. In particular, the behaviour of the chemical systems in but also out of the reference process conditions have to be further studied. As well, the impact of the degradation products has to be studied not only in terms of process performances but also in terms of gas generation, specific fission product behaviour, impact on other process steps, impact on final product quality... at each step, the modelling either as a predictive tool or a support to the analytic technics has to be further developed. In the longer terms, all the data should contribute to a comprehensive process simulation tool.

Finally an integrated approach, considering the link of the different steps dissolution/separation/conversion/fabrication is of paramount importance to ensure the relevance of the proposed fuel cycle. It covers not only engineering studies but also global safety, plant design simulation and technico-economic considerations.

Development of advanced technologies for HLW partitioning in Russia

- Maturing of HLW partitioning technology (with Am, Cm, RE, Cs-Sr recovering from HLLW and its separation) including modernization partitioning facility at Mayak plant – (up to 2022)
- Developing and deployment facility of HLW partitioning facility at the MCC (up to 2025)
- Developing and maturing the technologies of Am, Cm oxides and mixed U-TPE oxides precipitation and production, the facility deployment (up to 2025)
- Developing and maturing the technologies for MA-bearing fuel fabrication (up to 2023)
- Fabrication, irradiation, PIE, recycling experimental Am and Np-bearing fuel (up to 2034)
- Complex database for fuel characteristics and codes for MA recycling (up to 2030)

The scale-up of MA treatment is essential to supply raw MA to the following development of dedicated fuel.

5.2 MA Fuel Production

It is also effective for development of dedicated fuel including reprocessing after irradiation in transmuter to stepwise increase amount of MA. JAEA plans to fabricate MA nitride sample from genuine HLLW following MA separation, reprocess it, and re-fabricate to MA nitride. The Next step is to fabricate irradiation pin of ^{241}Am separated from raw plutonium, irradiate it in Joyo, examine irradiated pin and reprocess it. Irradiation in other reactor can be considered. Fabrication of irradiation pin of actual MA including rare earth element as impurities is the following step, but supply of raw materials and hot facility to handle them in inert atmosphere is not available in JAEA. Construction of international structure will be discussed for supply of raw MA, fabrication of irradiation pin, irradiation in nuclear reactor, PIE and reprocessing of irradiated pin.

5.3 Transmutation

ADS

The design of the MYRRHA proton accelerator contains a number of provisions to insure highest reliability. Redundancy and fault-tolerance, conceptually possible by a super-conducting LINAC in a unique way, should permit to reach the desired performance. The anticipated learning curve is now started with the launched construction of the 100 MeV part. Indeed, this accelerator allows the required tests at full scale.

To develop and qualify materials able to withstand corrosion in HLM, high temperatures, and irradiation, JAEA plans to irradiate candidate materials in JOYO, BR-2, TIARA and SINQ (PSI).

Fast reactor

The “Breakthrough” program in Russia for technologies for transmutation of MA in FR includes:

1. Justification of neutron-physical characteristics of core with MA and efficiency of transmutation of MA in FR, development of requirements for fuel with of minor actinides. (Incl. the experiments with MA- FAs in BFS (Obninsk), in BOR-60 (Dimitrovgrad) and BN-800)(up to 2032)
2. Studies in support of homogeneous recycling of Np (designing, fabrication, irradiation and post-irradiation examination of mixed oxide and nitride uranium-plutonium fuel with Np in amounts of 0.1% to 1%) (up to 2031)
3. Studies in support of homogeneous and heterogeneous recycling of Am (designing, fabrication, irradiation and post-irradiation examination); homogeneous recycling of Am with content from 0.4% to 1.2% (mixed nitride and oxide fuel, uranium nitride and oxide fuel) (up to 2031), and, heterogeneous recycling of Am with a content of up to 10-12% (mixed nitride and oxide fuel, uranium nitride and oxide fuel)(up to 2026).

In Europe, the challenge is to bring robustness, accuracy and predictability to Fuel Performance Codes, and specifically investigate Am-bearing fuel safety-related behaviour, essential to support safety assessment of Gen IV fast reactors, thus contributing to the development of safe, long term solutions for the management of nuclear wastes.

The main objectives are :

- Extend the validation database of models and simulation codes through the generation of data related to the production and behaviour under irradiation of helium, fission gases and fission products and to the specific properties of fuels containing Am. This will be done by capitalising on the latest European irradiation campaigns and by using a combination of state-of-the art experimental techniques and modelling methods. The complete knowledge base will be made secure by synthesizing data and properties.
 - Investigation of gas release, swelling, redistribution/migration of actinides and FP, chemical/isotopic compositions and thermo-chemical properties of irradiated Am-bearing fuel, as they can affect the risk of cladding failure.
 - In-depth characterization of thermal and thermo-chemical properties of Am-bearing fuels (both fresh and corresponding irradiated fuels), in particular to determine the fuel melting temperature and margin to melt as a function of all the relevant parameters.
- Improve the prediction capabilities of FPCs by upgrading and implementing more reliable mechanistic models for the simulation of normal and off-normal conditions.
 - Improvement of the models and operational tools by deriving new correlations for model parameters, based on transfer and integration of the knowledge gained from the experimental and modelling activities.
 - Benchmark of FPCs on the evolution of physical properties of MA-bearing fuels in normal conditions with the objective to compare the results obtained between the various codes but also with post-irradiation examinations, leading to an extension of the validation domain.

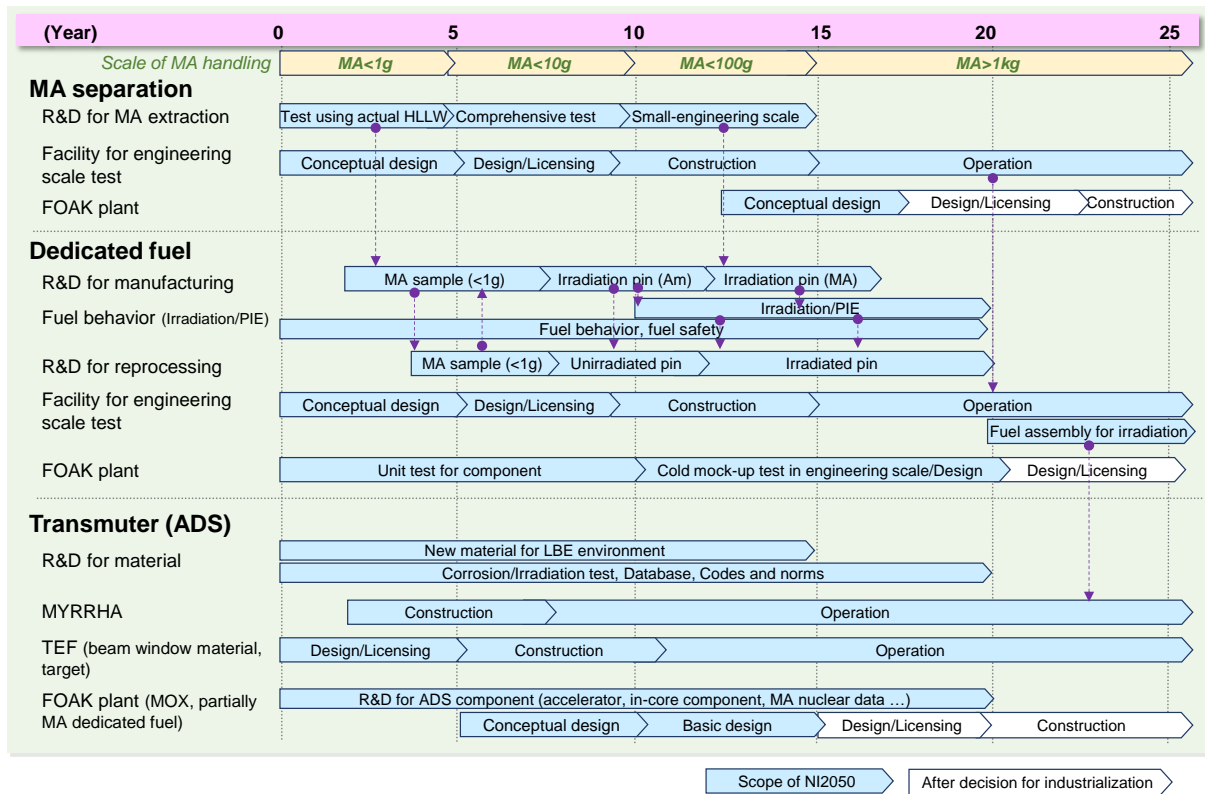
MSR

The construction of a large power MOSART is proposed to be preceded by the construction of 5-10 MWt Demo MOSART unit to demonstrate the control of the reactor and fuel salt management with its volatile and fission products with different TRU loadings for startup, transition to equilibrium, drain-out, shut down etc. (See Fig. B.2 in Annex B). There are opportunities to further improve the efficiency of burning minor actinides in MOSART, which will be justified by the results of the experimental setup.

5.4 MA Fuel Reprocessing

It is effective to develop reprocessing technology increasing amount of MA coupled with R&D on MA fuel production and its irradiation test as mentioned above.

Timeline sequence for P&T (Very preliminary need in-depth upgrade but the final date goal is the targeted one 2040)



Annex A: Transmutation of MA in fast neutron reactors (BN-800, BN-1200)

At the initial stage of the transition to a two-component nuclear power system in Russia, separated plutonium from LWR SNF are used in the fuel for the initial loading of fast reactors and for feeding up to the FR transition to self-sustainment in closed nuclear fuel cycle. The FRs' physics shows relatively low sensitivity to the content in fuel MA at 1-2%, which is a favorable factor from the point of view of nuclear safety requirements. This is sufficient both for the transmutation of MA accumulated in SNF of LWR reactors and for MA produced in the fast reactor. However, this amount can be distributed throughout the main fuel (homogeneous approach) or concentrated in a relatively small amount of fuel elements and fuel assemblies (heterogeneous approach). Currently, both schemes are under investigation within the framework of R & D for the "Breakthrough" project.

In power reactors, MA transmutation is carried out in parallel with the production of electricity, which is the main advantage of the approach in comparison with alternative approaches (eg, accelerated systems).

The following scenarios are considered when analyzing the concept of handling MA in FR:

- Homogeneous transmutation of MA in the fuel
- Heterogeneous transmutation of MA in special assemblies (example for BN-1200 in the Fig. A.1)

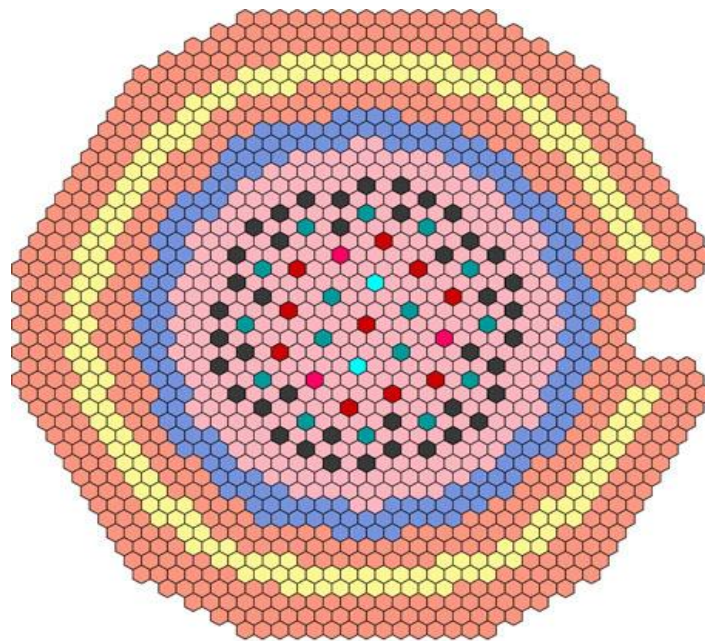


Fig.A.1 Heterogeneous transmutation of MA in special assemblies for BN-1200

Calculations of the MA transmutation efficiency have shown that FR can be an effective burner for MA. More than 2 tons of MA can be burned in the reactor during the reactor operation, i.e. the decrease in the amount of MA due to burning out is approximately the order (see Table A.1) The best option (for FR) is the option of transmutation of Np and Am with isolation and storage of Cm. The necessary efficiency can be achieved both in the homogeneous approach and in the heterogeneous.

Table A.1. The total balance of MA for various methods of burning them (kg)

Method of burning out MA	Without burning	homogeneous			heterogeneous		
		All MA	Without Cm	Without Cm+Am	All MA	without Cm	Without Cm+Am
MA Balance	2520	310	500	1150	315	500	1140

Homogeneous recycle of MA

The most optimal within the framework of closed nuclear fuel cycle is homogeneous transmutation of MA, included in the composition of fuel, with the following features:

- Np recycle is together with U and Pu (co-separation of the U-Pu-Np fraction),
- Am fraction is separated from Cm and added to the fuel with the correct concentration,
- Cm is not recycled (in the basic version), but is stored for decayed (until the main part decayed to Pu).

The small concentration of MA does not affect the nuclear safety of the reactor, which is operated, in fact, without noticing the presence of MA. Moreover, at the initial stage of the transition, the MA involvement can be used to stabilize the reactivity, i.e. contributes to its improvement

The main problem with MA transmutation is the deterioration of the radiation characteristics of the fuel. For this reason, the transmutation of Cm, which leads to an increase in the heat release of the fuel (by a factor of 7) and a high neutron dose rate (hundreds of times), is not considered. The recycling of Np and Am leads to an increase in the dose rate of the regenerated fuel by approximately an order of magnitude.

Heterogeneous transmutation of MA

The series of assemblies for heterogeneous transmutation of MA with concentrations up to 10% in the core and 20% in the lateral zones were developed. Calculation studies have shown the possibility of achieving a complete burning of MA, as well as for a homogeneous option, but MA is concentrated in a small part of the fuel element (intra cassette heterogeneity) or FAs of the order of 5-7% of the total amount.

The heterogeneous option allows to divide the flows of "clean" and "dirty" fuel and potentially to solve problems with radiation safety in relatively small robotized fuel production with MA.

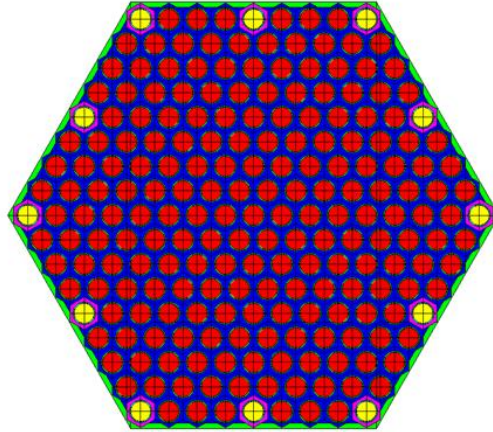


Figure A.2 The scheme of heterogeneous transmutation of MA (highlighted in yellow)

In addition, the production of homogeneous mixed U-Pu fuel with MA is associated with the problem of increased volatility A_m during sintering of pellets, contamination of equipment and problems with radiation safety.

One more aspect of heterogeneous burning is the more significant, also local, effect of heterogeneous elements on the physics of FR, primarily the inhomogeneity of energy release and the reconversion rate when using inert matrices.

Annex B: Studies on MA transmutation in molten-salt reactors

The introduction of minor actinides (MA) into traditional fast neutron reactors will complicate the design of these reactors, will require the development of new fuels, will complicate and lead to an increase in the cost of its fabrication. The scientific and engineering issues of manufacturing a fuel pellet with significant additions of minor actinides, as well as issues of justifying the safety of a fast reactor with such fuel, are nowhere in the world. In addition, for solid fuel reactors with a limited burnup, the loss of TRU to waste stream in multiple recycling will be comparable to the amount of MA burned over the same time.

From the outset molten salt reactors were thermal-neutron-spectrum graphite-moderated designs. The first experimental studies and design developments of MSR were performed in the 60-70s of the last century in the US ORNL. The 8 MWt MSRE reactor was built and successfully operated from 1964 to 1969. The success of MSRE stimulated the development of a thorium-uranium 1 GWe MSBR design with thermal neutron spectrum. In the Russian Federation, the MSR studies began at the NRC "Kurchatov Institute" in the second half of the 1970s.

Today in the Russian Federation MSR R&D has mainly focused on fast-spectrum MSR options combining the generic advantages of fast neutron reactors (extended resource utilisation, waste minimisation) with those related to molten salt fluorides as both fluid fuel and coolant (low pressure; high boiling temperature; good compatibility with high Ni-alloys, SiC ceramics and graphite; no exothermic reactions with water, air and, optical transparency).

The main attractive features of advanced MSR designs under consideration are as follows:

- Minimum number of parasitic absorbers and as a consequence less number of fissile materials in the core;
- Non limited fuel burn-up with minimal losses of actinides to waste in multiple recycling;
- Flexibility of the fuel cycle - the ability to work with fuels of various nuclide composition without reactor shutdown and special modifications of the core;
- On-site fuel processing - no temporary storage is required to hold SNF, transportation of SNF and fuel loading for the next transmutation cycle;
- High thermal efficiency, due to the high temperature of the fuel salt (>700C).
- Operation in load follow mode.

Chloride salts have been also considered as an alternative fluid fuel to obtain a fast neutron spectrum due to high solubility of TRU in the melt. Severe problems related to structural material corrosion (particularly at the high end of the temperature range), chemical stability of such systems and poor separation ability between some representatives of actinides and lanthanide's groups have been pointed out. Also, during irradiation ^{35}Cl transmuted to ^{36}Cl with $T_{1/2}=300000\text{yr}$. Binary and ternary systems of fluorides fuels still remain an interesting way out.

MSR developments in the Russian Federation on the 2.4 Gwt MOSART design address the concept of large power units with a fast neutron spectrum in the core without graphite moderator. The main characteristics of the MOSART design are given in Table B.1.

Table B.1: Main characteristics of the MOSART design

Fuel circuit	MOSART
Fuel salt, mole%	LiF-BeF ₂ +1TRUF ₃ LiF-BeF ₂ +5ThF ₄ +1UF ₄
Temperature, °C	620-720
Core radius/height, m	1.4/2.8
Core specific power, W/cm ³	130
Container material	kHN80MTY alloy
Removal time for soluble FPs, yrs	1-3

Since year 2000, the NRC “Kurchatov Institute” carried out complex studies on the MOSART project, which included for configurations selected: neutronic, thermal hydraulic and safety analysis; experiments concerned the main physical and chemical properties of fuel/coolant salts; compatibility of structural materials and fuel/coolant salts with its chemistry control. The accumulated experience now allows us to move from studying the calculated and experimental capabilities of MOSART concept to obtaining specific technical and technological solutions. The MOSART feasibility at present is beyond doubt. In this paper focus is placed on MOSART system without Th support with main design objective to close nuclear fuel cycle for all actinides, including Np, Pu, Am and Cm.

The effective flux of such system is near $1 \times 10^{15} \text{ n cm}^{-2} \text{ s}^{-1}$. The possibility of creating a high neutron flux and the lack of structural materials in the liquid homogeneous core, leads to optimization of the neutron balance, as well as the possibility to change the fuel salt composition without core modification and reactor shutdown, creates favorable conditions for the TRU utilization. The MA burning rate is directly proportional to the core specific power. When choosing this parameter, it is advisable to be within technical limits.

Summary times and possible methods for fission product removal and actinides recycling in MOSART system are presented in the Table B.2.

Table B.2: Summary times for fission product removal and actinides recycling for MOSART

Element	Time
Kr, Xe	50 s
Zn, Ga, Ge, As, Se, Nb, Mo, Ru, Rh, Pd, Ag, Tc, Cd, In, Sn, Sb, Te	2-4 hrs
Zr	1-3 yrs
Ni, Fe, Cr	1-3 yrs
Pu, Am, Cm, Np, U	1-3 yrs
Y, La, Ce, Pr, Nd, Pm, Gd, Tb, Dy, Ho, Er	1-3 yrs
Sm, Eu	1-3 yrs
Sr, Ba, Rb, Cs	5-10 yrs
Li, Be, Th	30 yrs

Even in the homogeneous core, where removal times for soluble fission products are long enough, taking away of neutronic poisons is, of course, the primary purpose of fuel processing. All actinides are immediately returned to fuel circuit. The consideration done demonstrated the potential of the MOSART as systems with flexible configurations and fuel cycle scenarios which can operate within technical limits with different loadings and make up based on TRUs (from spent VVER fuel with MA/TRU ratio up to 0.45) as dedicated actinide transmuter, as self-sustainable system (CR=1) or even as a breeder (CR>1).

The main advantages of MOSART are the ability to vary widely the MA content in fuel salt without losing the inherent safety and the absence of stages related to the fuel fabrication and re-fabrication in multiple actinides recycling. The molten salt fluoride mixtures, due to the high separation coefficients between actinides and lanthanides, make it possible to organize an effective removal of soluble fission products, based on the reductive extraction, to substantially reduce the time of the external fuel cycle for actinides and its losses in waste stream in multiple recycling in comparison with solid fuel reactors.

Thus, the MOSART concept, using the advantages associated with the liquid fuel structure, opens the prospect of a significant improvement in nuclear power technology with regard to the closure of the fuel cycle for all actinides. It is proposed to use the technical and technological capabilities of the MCC site to place MOSART in the immediate vicinity of SNF reprocessing facilities, linking it to the EDC infrastructure. It is assumed that the fuel cycle of this complex will be organized as follows (see Figure B.1): the bulk of the removed uranium and plutonium return to thermal and fast solid fuel reactors, and the remaining TRU are transferred for utilization in the MOSART system. The co-location of MOSART and SNF reprocessing plants, will provide the complex and the surrounding by electricity, facilitates the problems of nuclear materials transport and radwaste management.

The creation of a full-scale MOSART is proposed to be preceded by the construction of an experimental small power Demo unit demonstrate the joint operation of the reactor loaded by different TRU compositions with fuel salt processing unit. The configuration, materials and characteristics of the MOSART fuel circuit were chosen primarily for reasons of technological validity.

In the MOSART, a well-established molten LiF-BeF₂ salt mixture is chosen as a solvent for TRU trifluorides fuel addition. The molten fluoride chemistry (solubility, redox chemistry, chemical activity etc.) for the LiF-BeF₂ system is well established and can be applied with great confidence, if TRU based fuels are to be used in the LiF-BeF₂ solvent. The solubility of TRU trifluorides in molten 73LiF-27BeF₂ (in mole %) salt mixture with decreased beryllium difluoride fraction of 0.27 and the minimum temperature in the fuel circuit of 600 ° C is more than 2 mole %. The structural material selected for the MOSART container is the special Ni-Mo alloy kH80MTY with a low concentration of Cr alloyed by 1% of Al. The composition of the alloy was optimized by Kurchatov Institute researchers for corrosion resistance (both in a low oxygen gas atmosphere and in molten fluorides), irradiation resistance and high temperature mechanical properties.

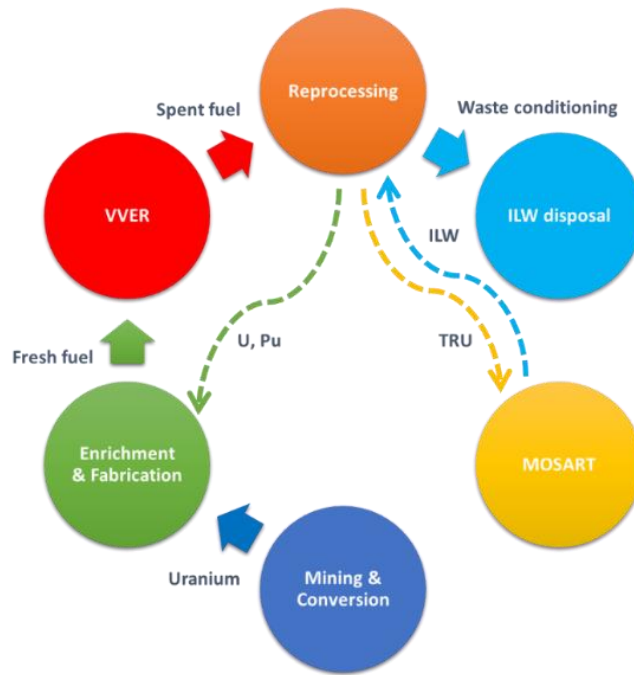


Figure B.1: Nuclear fuel cycle with MOSART

The performed calculations show that the Li,Be/F MOSART, starting at TRU from SNF of VVER with the ratio of MA to (Pu + MA) equal 0.1, without core modification and changing temperature in the fuel circuit, can use any TRU make up with the MA to (Pu + MA) ratio up to 0.33. At equilibrium ^{245}Cm fission contribute 28% to the core reactivity. This allows 2.4 GWt MOSART with a fuel salt of the selected composition to utilize up to 250 kg of MA per year.

Interior Ni-base reflectors/shielding are employed to reduce the radiation damage to the reactor vessel and fuel salt chemistry control is employed to substantially limit oxidizing the container alloy constituents. For those systems that include graphite, the salt is maintained in a redox window oxidizing enough to prevent damage to the graphite while reducing enough to prevent alloy oxidative corrosion. Those systems without graphite can maintain the salt in a more reducing condition.

The radiation resistance of reflectors/shielding material determines the upper limit core specific power in the MOSART design. If the damage caused by fast neutrons is critical for graphite, then for high-nickel alloys, the reduction in plasticity at a temperature above 500°C, associated with the formation of helium along the grain boundaries, is the most important process, caused by both fast and thermal neutrons. To obtain an acceptable service life for reflectors (> 5 years), the core specific power should not exceed 130-150 W/cm³. Otherwise, frequent stops to replace the reflector will result in a reduction in the reactor load factor and an unjustified increase in operating costs. In addition, with this limitation for the core specific power, there is no problem of heat removal from the fuel circuit. In this case the service life for the reactor vessel, made of the KHN80MTY alloy, will be about 50 years.

For 2.4 GWt MOSART, taking into account the adopted limits, the primary circuit will contain 50 m³ of fuel salt, of which only half is in the core. It is obvious that the demonstration reactor should have significantly less specific power and thermal capacity to test the MSR.

MSR fuel maintenance

In general, to achieve fuel maintenance, (1) the fuel must be delivered to and into the reactor in a proper state of purity and homogeneity, (2) the fuel must be sufficiently protected from extraneous impurities, and (3) sound procedures must exist for addition and recycling of the actinides required and (4) provision of the required redox potential in the system.

For MOSART that propose chemical reprocessing to remove fission products (see Table B.2), the required fuel maintenance operations also include (1) continuous removal (by the purging and stripping section of the reactor) of fission-product krypton and xenon, (2) addition of U and TRUs to replace those lost by burnup, (3) in situ production of UF_3 to keep the redox potential of the fuel at the desired level, (4) recycling of all actinides, (5) removal of soluble fission products (principally rare earths); they probably also include (6) removal of inadvertent oxide contaminants from the fuel; in addition, they may include (7) removal of a portion of the insoluble noble and semi noble fission products. Each of these is discussed briefly below.

Preparation of initial fuel: Initial purification procedures for the MSR present no formidable problems. Nuclear poisons (e.g., boron, cadmium, or lanthanides) are not common contaminants of the constituent raw materials. All the pertinent compounds contain at least small amounts of water, and all are readily hydrolyzed to oxides and oxyfluorides at elevated temperatures. The compounds LiF and BeF_2 generally contain a small quantity of sulfur as sulfate ion. Uranium tetrafluoride commonly contains small amounts of UO_2 , UF_5 , and UO_2F_2

Purification procedures used to prepare materials in many laboratories and engineering experiments have treated the mixed materials at high temperature (usually at 600 °C) with gaseous H_2 -HF mixtures and then with pure H_2 in equipment of nickel or copper. The HF- H_2 treatment serves to (1) reduce the U^{5+} and U^{6+} to U^{4+} , (2) reduce sulfate to sulfide and remove it as H_2S , (3) remove Cl^- as HCl , and (4) convert the oxides and oxyfluorides to fluorides. Final treatment with H_2 serves to reduce FeF_3 and FeF_2 to insoluble iron and to remove NiF_2 that may have been produced during hydrofluorination. To date, all preparations have been performed in batch equipment, but continuous equipment has been partially developed. Such a purification procedure can provide a sufficiently pure and completely homogeneous fuel material for initial operation of the reactor.

Addition of actinides: It will apparently be necessary, assuming the fuel volume changes from these additions or other causes do not require removal of any fuel to storage. These will be inherently more complex (and radioactively dirty), and stating which of the options would be preferred is not presently possible. If making a few additions of plutonium and minor actinides to the reactor fuel during its lifetime is necessary, then adding it e.g. as a liquid containing 7LiF - PuF_3 mixture should be possible. A possibility would be a melts containing about 80 LiF -20 PuF_3 (in mole %) melting near 740°C. Alternatively, a procedures presumably could be developed for addition of solid $TRUF_3$.

Maintaining the desired UF_3/UF_4 ratio: Operation of the MSRE demonstrated that in situ production of UF_3 could be accomplished readily and conveniently by permitting the circulating fuel to react in the pump bowl with a rod of metallic beryllium suspended in a cage of HN80MTY. This technique could be adapted for use in other MSR designs; beryllium reduction would be desirable if the fissionable and fertile uranium additions are to be made as Li_3UF_7

Removal of fission-product krypton and xenon: Stripping of krypton and xenon makes possible their continuous removal from the reactor circuit by the purely physical means of stripping with helium. Such a stripping circuit would remove an appreciable (but not a major) fraction of the tritium and a small (perhaps very small) fraction of the noble and semi noble fission products as gas-borne particulates. In addition, the stripper would remove BF_3 if leaks of secondary coolant into the fuel were to occur. None of these removals (except possibly the last) appreciably affects the chemical behavior of the fuel system.

Partial removal of noble and seminoble metals: The behavior of these insoluble fission-product species, as indicated previously, is not understood in detail. If they precipitate as adherent deposits on the MOSART heat exchanger, they would cause no particularly difficult problems. However, should they form only loosely adherent deposits that break away and circulate with the fuel, they would be responsible for appreciable parasitic neutron captures. To the extent that they circulate as particulate material in the fuel, insoluble fission-product species could probably be usefully removed by a small bypass flow through a relatively simple Ni based-wool filter system. Presumably, such a system would need to have a reasonably low pressure drop and probably would need to consist of sections in parallel so that units whose capacity was exhausted could be reasonably replaced.

Fuel chemical processing: In MSR, from which xenon and krypton are effectively removed, the most important fission products poisons are among lanthanides which are soluble in the fuel. Also, the trifluoride species of actinides and the rare earth's are known to form solid solutions so, that in effect, all the LnF_3 and AnF_3 act essentially as a single element. In combination of all trifluorides, actinides solubility in the melt is decreased by lanthanides accumulation. Since actinides must be removed from the fuel solvent before rare earth's fission products the MSR must contain a system that provides for removal of all actinides from the fuel salt and their reintroduction to the fresh or purified solvent. This fuel processing system can be based principally on three types of operations: removal of actinides, rare earths, and other fission products from the salt by extraction into molten bismuth. The chemical basis on which the processing system is founded is well established (the coefficients of the distribution of actinides and lanthanides in the Li,Be/F - liquid bismuth system with respect to plutonium at $T = 873 \text{ K}$ are respectively 6 for curium, 3.000 for neodymium and 25.000 for lanthanum); however, only small engineering experiments have been carried out to date, and a considerable engineering effort remains.

In fuel salt, approximately 10% of the initial amount of lanthanides remains (mainly cerium). The purified salt is then transferred to the actinide recycling. The lanthanide precipitate with salt residues is sent for vacuum distillation of the salt components. The lanthanide salts remaining after the distillation are sent to the EDC for conditioning and subsequent near-surface disposal.

Engineered safety features of MSR

The main feature of the MSR which sets it apart from the solid fuel reactors is that the nuclear fuel is in fluid form (molten fluoride salt) and is circulated throughout the primary coolant system, becoming critical only in the core. Thus, for an MSR to have equivalent overall containment, greater requirements must be placed on the containment barriers from the fuel salt outward.

The possible problems and engineered safety features associated with this type of reactor will be quite different from those of the present day solid fuel designs. In the MSR, the primary system coolant serves the dual role of being the medium in which heat is generated within the reactor core and the medium which transfers heat from the core to the primary heat exchangers. Thus the entire primary system will be subjected to both high temperatures ($>700^\circ\text{C}$ at core outlet) and high levels of radiation by a fluid containing most of the daughter products of the fission process. On the other hand,

the fuel-coolant barrier in a solid-fuel reactor, interposed between the heat source and the cooling fluid, is the barrier most vulnerable to damage in a nuclear excursion so that its protection and the consequences of its failure tend to impose more restrictive nuclear safety requirements on a solid-fuel reactor. Because of the low fuel salt vapor pressure, however, the primary system design pressure will be low, as in an liquid metal cooled designs. The entire primary coolant system is analogous, in terms of level of confinement, to the cladding in a solid fuel reactor. Although much larger, it will not be subjected to the rapid thermal transients with melting associated with accident scenarios for VVER and liquid metal cooled designs. Two additional levels of confinement will be provided in the MSR in accordance with present practice. The problem of developing a primary coolant system which will be reliable, maintainable (under remote conditions), inspectable, and structurally sound over the plant's lifetime will probably be the key factor in demonstrating ultimate safety and licenceability.

It is the breach of the primary coolant system boundary, resulting in a large spill of radioactive salt into the primary containment, which will provide the design basis accident. The analogous level of occurrence in a solid fuel reactor would be from major cladding failure (min) to core meltdown (max). Possible initiators of this accident include pipe failure missiles, and pressure or temperature transients in the primary coolant system, failure of the boundary between the primary and secondary salt in the intermediate heat exchanger could be especially damaging. In the event of a salt spill, a possibly redundant system of drains would be activated to channel the salt to the cooled drain tank. The primary system containment, defined as the set of vertically sealed, concrete-shielded equipment cell, would probably not be threatened by such a spill, but cleanup operations would be difficult.

A unique safety feature of the MSR is that, under accident shutdown conditions, the fuel material would be led to the emergency core cooling system (represented by drain tank cooling), rather than vice versa. The reactor and containment must be designed so that the decay heated fuel salt reaches the drain tank under any credible accident conditions. In any case, the decay heat is associated with a very large mass of fuel salt, so that melt through does not appear to be a problem.

The safety philosophy for accidents involving the reactor core is very different for fluid-fueled reactors and for solid-fueled ones because the heat source is (mainly) in the liquid and not in a solid, which requires continuous cooling to avoid melting. An LMR, for example, has a tremendous amount of stored energy in the fuel pins which must be removed under any accident conditions. Dry out, which leads to almost immediate meltdown in an LMR, would not be nearly as severe in the MSR because the heat source would be removed along with the heat sink capability.

For 2.4 GWt MOSART severe accident with the rupture of the primary circuit and fuel discharged on the reactor box bottom was estimated. The model based on mass transfer theory describing main radionuclides distribution between the fuel salt, metallic surfaces of the primary circuit, graphite and the gas purging system was applied for calculation releases to the containment atmosphere. A great deal of practical information on the disposition of these different fission products groups was provided by operation of the MSRE.

As a criteria characterizing an isotope yield from the fuel salt is accepted the ratio of this isotope activity changed into a gas phase of a containment (A_g) to its full activity built up in a reactor by the moment of the accident (A_0). For a molten salt fuel there are three broad classes of fission products: whose fluorides are stable in the salt at its redox potential (soluble fission products), the noble gases and the noble metals. The major soluble fission products are rare earths (including Y), Zr, Ba and Sr, Rb and Cs, I and Br. The noble gases have very low solubility in molten salt and take first opportunity to migrate to any gas phase in contact with fuel. The competitive sinks for noble gases are the pore spaces in the graphite and the circulating bubbles of the cover gas. Another major group of fission

products, consisting largely of Nb, Mo, Tc, Ru and Te, does not form fluorides that stable at the redox potential of the fuel salt and is therefore called noble. They are not wet by the salt and also tend to migrate to the salt surfaces. For noble metals it was estimated that about 50% on the metal surfaces and 50% would go into the off-gas system with the bubbles.

After accident considered all noble gases and metals available should move to the gas phase ($A_g/A_s = 1$, where A_g/A_s -the ratio of isotope activity in the gas phase of the containment after an accident to its activity concentrated in the fuel salt by the moment of the accident). However, already as it noted before during the normal operation these nuclides are almost completely leave the fuel salt. As can see from the Table B.3 only from 0,1 to 3,4 % of them is remained in the fuel. Therefore, the release for the noble metals is not so big. Only for Te129 it amounts to 25%. This isotope has a sufficiently small half - life period ($T_{1/2}=69$ min) to leave the fuel salt completely, for example in comparison with Te132 ($T_{1/2}=78$ h).

Alkaline and alkaline earth metals and rare earth form in the fuel salt stable and well soluble fluorides which have enough high temperature of melting (above 1200 °C). For this group in normal operation A_s/A_o ratio is about 0,95. As a result for isotopes of these group the relative activity yield A_g/A_o comes up 0,1 to 2,5 % . Note, that such isotopes as Sr89 and Cs137 have gas precursors Kr89 and Xe137 with a low half life. But because of low concentration of noble gases in the fuel salt an escape of Sr89 and Cs137 into the gas phase due to gas precursors is insignificant and relative activity yield for these isotopes is not different for others.

Table B.3: Activity releases into the MOSART containment for the accident with the fuel circuit failure

Isotope	A_s/A_o	A_g/A_s	A_g/A_o
	0,25	1	0,25
Te132	0,005	1	0,005
Ru103	0,01	1	0,01
Ru106	0,001	1	0,001
Nb95	0,034	1	0,034
Zr95	0,99	0,0011	0,0011
Sr89	0,99	0,00046	0,00046
Sr90	0,98	0,00046	0,00046
Ba140	0,97	0,006	0,006
La140	0,98	0,026	0,025
Ce141	0,99	0,0024	0,023
Ce144	0,96	0,0024	0,023
I131	0,62	0,43	0,27
I133	0,94	0,43	0,43
Cs137	0,7	0,016	0,011

The molten salts have many desirable properties for such applications, and it seems likely that – given sufficient development time and money - a successful burner system could be developed.

Extraction of long-lived actinides from high-level radwaste with their further utilization in a dedicated reactor system will allow to reduce the volume of high-level waste and radiotoxicity of reprocessing for spent nuclear fuel and exclude the costs of long-term storage and subsequent disposal of selected minor actinides, thereby increasing the public acceptability and commercial attractiveness of SNF reprocessing.

The advantages of MOSART as a TRU burner from SNF reprocessing before solid-fuel reactor systems are primarily due to the lack of the need to manufacture a fuel pellet and the possibility of widely varying the content of long-lived actinides in fuel salt.

MOSART will allow to burn all the produced MA and approximately 500 kg/year of recycled reactor grade plutonium, while producing 1 GWe of electricity consumed by the EDC and surrounding companies. The construction of a large power MOSART is proposed to be preceded by the construction of 5-10 MWt Demo MOSART unit to demonstrate the control of the reactor and fuel salt management with its volatile and fission products with different TRU loadings for start up, transition to equilibrium, drain-out, shut down etc.. There are opportunities to further improve the efficiency of burning minor actinides in MOSART, which will be justified by the results of the experimental setup.

The industrial site of the MCC has all the necessary engineering communications, automobile and railway access roads, areas for the expansion of storage facilities, heat and power supply systems, water supply, electric networks. MOSART plants can use the existing radiochemical infrastructure of the MCC, which should somewhat reduce investment costs through the use of existing mine workings, the absence of long arms to transport SNF and processed products, and the availability of qualified personnel.

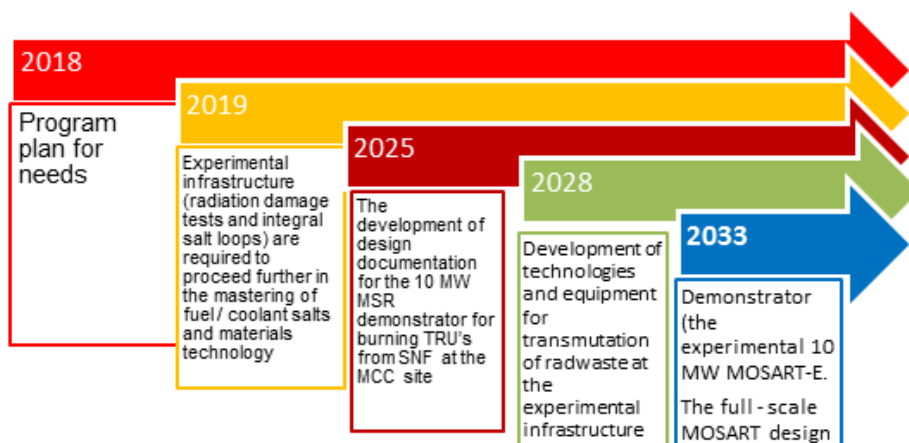


Fig. B.2 The road map of Demo transmuter creation