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Review of national accelerator driven system programmes for partitioning and transmutation

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FOREWORD

One of the current important issues of nuclear power is the long lived radioactive waste toxicity problem. The sharpness of this problem could be considerably reduced if, during energy production, there was the possibility to incinerate at least the most toxic radioactive isotopes (long lived fission products and minor actinides). The combination of external intensive neutron sources with facilities containing nuclear fuel, so-called hybrid systems, are under investigation in several countries. The surplus of neutrons in such systems may be used to convert most of the long lived radioactive nuclides into isotopes having a shorter lifetime.

Currently, an increasing number of groups are entering this field of research. There is clearly a need for co-ordinating their efforts, and also for the exchange of information from nationally or internationally co-ordinated activities. Consideration of the advantages of hybrid systems, and the wide field of interdisciplinary areas of research involved, show the need for an international co-operation in this novel R&D area.

The International Atomic Energy Agency has maintained an active interest in advanced nuclear technology related to accelerator driven systems (ADS), and related activities have been carried out within the framework of its programme on emerging nuclear energy systems. After thorough analyses of the outcomes of several international forums and recommendations of the IAEA Technical Committee Meeting on Feasibility and Motivation for Hybrid Concepts for Nuclear Energy Generation and Transmutation (Madrid, Spain, 1997), the IAEA conducted an Advisory Group Meeting on Review of National Accelerator Driven System Programmes in Taejon, Republic of Korea, from 1 to 4 November 1999. The scope of the meeting included review of the current R&D programmes in the Member States and the assessment of the progress in the development of hybrid concepts.

The IAEA would like to express the most appreciative thanks to KAERI, Taejon, Republic of Korea, for the efficient organization of the Advisory Group meeting and the warm welcome, as well as to all lecturers and participants for their interest and personal contributions to the meeting. Thanks are due to V. Arkhipov, IAEA consultant for help in the organization of the meeting. The IAEA officer responsible for the preparation of this publication was A. Stanculescu of the Division of Nuclear Power.

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SUMMARY

1. INTRODUCTION

Nuclear power generation is an established part of the world's electricity mix providing 17% of world electricity. One of the current important issues of nuclear power is the long lived radioactive toxicity problem. The seriousness of this problem could be considerably reduced if, during energy production, there was the possibility to incinerate the most long lived radioactive isotopes (long lived fission products and minor actinides).

The combination of external intensive neutron sources with facilities containing nuclear fuel, so-called hybrid systems, are under investigation in several countries. The surplus of neutrons in such systems may be used to convert most of the long lived radioactive nuclides into isotopes having a shorter lifetime. Moreover, subcritical conditions in hybrid systems provide potential advantages from a safety point of view in comparison with conventional critical nuclear reactors. Several components of the proposed hybrid systems need specific investigations (e.g. high intensity particle accelerators, target materials, window features, etc.).

Nowadays, an increasing number of groups are entering this field of research. Some of these groups are not embedded in wider national activities nor funded via international projects. For these groups, there is the need for co-ordinating their efforts and also for information exchange in the frame of nationally or internationally co-ordinated activities. Consideration of the advantages of hybrid systems, and the wide field of interdisciplinary areas of research involved, clearly show the need for an international cooperation in this new area. The needs for strengthening international cooperation in the field of the R&D for hybrid systems were emphasized by participants in several international forums and IAEA meetings.

After a thorough analyses of the outcomes of several international forums and IAEA meetings on Hybrid Concepts for Nuclear Energy Generation and Transmutation, and as a follow-up action to recommendations made by participants in the IAEA Technical Committee Meeting on Feasibility and Motivation for Hybrid Concepts for Nuclear Energy Generation and Transmutation (Madrid, Spain, 1997), the IAEA decided to convene an Advisory Group Meeting (AGM) on Review of National Accelerator Driven System (ADS) Programmes.

The purpose of this AGM was to review the current R&D programmes in the Member States, and to assess the progress in the development of hybrid concepts, as well as their potential role relative to both the current status and the future direction of nuclear power worldwide. The AGM participants were expected to provide options and guidance for the IAEA activities in the ADS area. The AGM was hosted by KAERI in Taejon, Republic of Korea, from 1 to 4 November 1999. Attendance at the meeting was from 17 participants from 11 countries and two international organizations.

2. AGM TOPICS

The programme of the AGM included the following topics:

Comprehensive presentations from groups conducting the most active research and development projects (Prof. Rubbia (on behalf of the “3 countries initiative”: French-Italian-Spanish collaboration), CERN, CEA, JAERI, USA (LANL, BNL, LLNL, ANL, ONL), EC ADS-project (TARC, IABAT), Russian Federation.

The presentations centered on the following issues:

- General issues and motivations: national/international views, specifically:
  - Why is ADS needed for the country?
  - Role of ADS in future nuclear cycles and waste management
  - General safety issues and requirements of ADS
  - Public acceptance

- ADS Technology
  - Accelerators for ADTT (cyclotrons and linacs)
  - Specific requirements and features of ADS accelerators
    - Reliability of operation
    - Design efficiency

- Development of ADS-oriented codes and methods
  - Benchmark reports
  - Experiments and validation of codes
  - Deterministic and Monte-Carlo codes
  - Coupling of high and low energy transport
  - Static and dynamic methods for subcritical systems
  - Spallation and fission product modeling

- Targets and nuclear assemblies for ADS
  - ADS Targets (solid and liquid)
  - Damage to materials
    - Materials irradiation in:
      - Proton field
      - High energy neutron field
    - Temperature/shock effects, etc.
    - Experiments

- Technology of heavy liquid metals
  - Thermohydraulics
  - Corrosion
  - Experiments
  - Hg target experiments at BNL

- Subcritical cores
  - Fuels and fuel processes
  - Molten salt reactors
  - Coolants

3. SUMMARY OF THE SESSIONS

Presentations and/or statements with regard to the status of the respective national as well as internationally coordinated ADS R&D programmes were made by all participants. The surmised advantages of ADS — apart from their intrinsic low waste production, and transmutation capability — are also enhanced safety characteristics and better long term resources utilization (e.g. thorium fuels). It appeared that important R&D programmes are being undertaken by various institutions in many member states to substantiate these claims and advance the basic knowledge in this innovative area of nuclear energy development.

In Asia, ADS R&D studies are pursued with both goals in mind: energy production with reduced waste production and decreased proliferation hazard, on the one hand, and long lived waste transmutation, on the other hand. The R&D efforts are concentrated in China, India, Japan and the Republic of Korea. The programmes are presently conducted at national level, with some bilateral or multilateral co-operation agreements.

In Europe, the main driving force behind ADS is long lived waste transmutation, but the ADS capability to produce energy is also investigated. Driven by the establishment of the "European Industrial Partnership" (EIP) to advance the engineering design studies of a ~100 MW(th) ADS demonstration facility, the national programmes on ADS R&D are converging towards the demonstration of the basic aspects of the ADS concept. These R&D activities are conducted both nationally and as joint efforts within the fifth framework programme of the EU. Presently, the EIP consists of leading European nuclear industrial companies from six countries.

In the Russian Federation, there is considerable R&D effort dedicated to the development of the ADS technology. These studies are strongly coupled with advanced fuel cycle studies that aim at waste minimization and at a strong overall simplification of the nuclear fuel cycle (e.g. molten salt). The Russian R&D activities are mainly conducted in support of western ADS programmes (e.g. within the framework of ISTC projects). However, a Russian national ADS programme is currently being finalized.

In the USA, the ADS efforts are dedicated to "Accelerator Transmutation of Waste" (ATW). The roadmap study, mandated to the US DOE by the US Congress, has been completed. This study describes a recommended R&D programme for federal support. It identifies the challenges of ATW, the associated R&D and preliminary budgetary needs, as well as potential spin-offs of this R&D. Both the roadmap document and the indications put forward by the US Congress are emphasizing very strongly the need and importance of international collaboration.

The technical and scientific issues of the various R&D programmes can be summarized as follows:

A. Theoretical work

Nuclear data
The ongoing activities are very limited compared to the actual needs. The participants expressed the need to pursue nuclear data evaluation work taking into account the new data becoming available in the energy range between 20 and 200 MeV.

Codes
It appears that design codes could be used for ADS studies. However, the participants reported about coupled Monte-Carlo/burnup codes for ADS modeling. Methods development work is needed in the area of neutron kinetics codes dedicated to ADS.

Conceptual designs
Different conceptual designs considering different spectra, fuels, fuel cycles and coolants are being considered. There is a strong need to perform comparative assessments of the various options, in order to narrow down the technological choices. It was stressed that acceptable ADS concepts require compact accelerator designs. Finally, it was stressed that the paper studies need strong experimental backing (see also: B. experimental work).

Accelerators
Accelerators in the 30 MW power range are suggested to drive an ADS for energy production and/or waste transmutation. However, the highest power presently achieved is in the 1 MW range (worldwide two machines: the 800 MeV LINAC at LANL (1.3 MW), and the 590 MeV cyclotron at PSI (0.89 MW)). Detailed paper designs of accelerators in the 1–5 MW range exist (e.g. SNS in the US and ESS in Europe), and it is expected that such machine(s) will be built in the near future. However, designs in the 30 MW range still require important R&D efforts, especially in terms of beam availability and reliability, beam dynamics, activation of the structures, and, in particular, the choice of the technology for ADS applications (LINAC vs. cyclotron, super or normal conducting cavities). It is important to note that these R&D efforts will necessarily bring together the two communities pursuing research in the fields of high energy accelerators and spallation sources.

B. Experimental work

Nuclear data
Many measurements are being performed or are planned for the near future for nuclear cross sections and spallation products. These activities are mainly pursued in Europe and the Russian Federation

.
Codes
Integral experiments such as MUSE (fast spectrum), Minsk (thermal spectrum), PSI/SCK•CEN/SOREQ (spallation yields experiments in the range 300–590 MeV) are very important for code validation purposes. It was stressed, however, that the organizers of code benchmarking activities have to pay particular attention to the fact that the most important result of such exercises are the conclusions and recommendations to the participants.

Experiments for technological development
Various experiments are being performed or are planned for the near future in support for ADS conceptual designs (ISTC 559 (high energy target), design of a liquid Pb-Bi spallation module, windowless design experiments at SCK•CEN, Pb-Bi lifting experiments at CIRCE, KALLA at FZK,)

As regards the various national R&D programmes, as well as the international collaborations, some participants in the AGM provided the following written statements/summaries:

Sweden

Theoretical work
Data
Intermediate energy range cross-section Data File (20–150 MeV)
^{232}Th, ^{238}U, ^{239}Pu, ^{240}Pu (EC & Russia)

Codes
Monte-Carlo Burnup -MCB
MCNP4B
MCNPX
NYOK
ORIGEN
CINDER4O

Conceptual designs
Waste incinerator, fast system, Pb–Bi, nitride fuel, altering once-through cycle

Experimental work

Collaboration in spallation target:	ISTC # 554
MUSE-4, MUSE-5 experiments:	EC
Cross section measurements (n, f), (p, f) 20–150 MeV, Pb, Bi:	ISTC and bilateral with Russia
Material and radiation studies Pb/Bi (thermalhydraulics and corrosion studies):	EC
Fuel development (nitride fuels):	EC
Accelerator reliability:	Sweden and the USA
On-line k_{eff} monitoring:	EC

Italy

Theoretical work
Data
- Some work done and planned on nuclear data (intermediate energy 20–150 MeV and thorium); 2 pers./a.
- Extensive work on material and thermalhydraulics data for Pb/Bi and related structural materials; 7 pers./yr.
- Some work done and planned on both pre- and post-irradiation characterization of ThO_2 as “inert matrix”, and other (e.g. allumina, spinel); 1 pers./yr.
- Work planned on thermodynamical data for modeling pyrometallurgical processes; 1 pers./yr.

Codes
- Monte-Carlo and deterministic for neutronics design; Monte-Carlo for burnup calculations; 3D kinetics of the subcritical core; 5 pers./yr.
- Codes for plant transient analyses (thermalhydraulics (including the secondary circuit) coupled to neutronics; 3 pers./yr.
- ADS fuel evolution (planned); 1 pers./yr.
- Pyrometallurgical process modeling (planned); 1 pers./yr.

Conceptual designs
- Demonstration facility reference configuration (EA80, done) without optimization, best known technologies, some non validated codes, few extrapolations of known phenomenologies;
- EA80 preliminary design (basis for the PSAR of the plant, according to the US NRC definitions);
- with some confirmatory tests; participation in the confirmatory tests of other conceptual designs (planned for the next 18 months);
- Comparisons of the EA80 with other proposals for the Demo; concept decision; large funding within the 6^{th} EC framework programme (2002).

Experimental work
- Pb/Bi technology, material compatibility and irradiation effects (two facilities in Brasimone, beginning 2000);
- Pb/Bi thermal hydraulics integral testing facility CIRCE (beginning 2001); CIRCE is the international centre for testing components and systems for Pb/Bi ADS.

Structures

- National R&D programme TRASCO (ENEA-INFN & partners, formalized and funded by the Research Minister with ad hoc law, for 4 years);
- National industrial initiative (ANSALDO-CRS4 & partners, formalized and funded as above, for 1.5 years);
- European R&D network, and participation in the Technical Working Group chaired by C. Rubbia, for 4 years (presently an informal co-ordination effort, in the near future hopefully a formalized network funded by the EC);
- Many bilateral agreements on both industrial and R&D side (e.g. ENEA-CEA on ADS, participation in the MEGAPIE programme), for 2–5 years;
- The total funding of the Italian ADS programme for the period 1998–2002 amounts to USD 35 million;
- (of which 22 approved, 13 expected); industrial participation to this budget amounts to 50%.

European Industrial Partnership

Six leading European nuclear companies have established an industrial partnership (EIP) to promote and develop engineering design studies of a demonstration facility to assess the industrial feasibility of an accelerator driven subcritical system for high level, long lived waste transmutation. This industrial partnership will act both at national and at European Union level, in particular within the fifth framework programme for Research, Technological Development and Demonstration Activities.
The six European industrial partners are:

- Ansaldo Nucleare, Italy;
- Belgatom, Belgium;
- Empresarios Agrupados, Spain;
- Framatome, France;
- NNC Limited, United Kingdom;
- Siemens, Germany.

The rationale and background behind the decision to strongly support the ADS demonstration facility design studies can be summarized as follows:

- A common industrial interest already exists for the ADS technology and for the realization of a demonstration facility; in 1997 such interest has been witnessed by the document "An European Nuclear Industry Interest for the Accelerator Driven Systems Technology to Assess an European Reference System Configuration" underwritten by Ansaldo, Framatome, NNC and Siemens which has been transmitted to the European Commission.
- A long lasting experience of cooperation exists among the EIPs in all fields of the industrial nuclear applications.
- The recommendations of the Technical Working Group set by the Research Ministers of France, Italy and Spain clearly indicate the need to design and operate, in about a decade, an ADS demonstration facility at a "sufficiently large scale to become the precursor of the industrial, practical scale transmuter". Such a facility, for which a common European effort is deemed necessary, is considered "the fastest and most cost effective way to conclusively assess the potentialities and the feasibility of a full scale industrial programme based on ADS; it is also assessed that in such a perspective the creation of an industrial platform is highly recommended.
- A great amount of R&D and engineering activities have been and are being performed in Europe to demonstrate separate basic aspects of the ADS concept and to define conceptual plant reference configurations — several different technological and design options have

been considered and studied — it now clearly appears that starting from this acquired knowledge, it is worth while to concentrate efforts only on two design options: the lead-bismuth cooled and the gas cooled ADS; the purpose is to develop two reference configuration to a level sufficient to perform objective comparisons and to eventually choose the solution to be engineered in detail and realized.
- The process of studying and comparing the configurations of the two different alternatives will have fundamental inputs coming from the R&D activities performed by Universities, R&D Centres and other National Bodies; a strict co-operation among R&D institutions and nuclear industries is therefore crucial to obtain the needed relevant results in an efficient and cost effective way; basic research is still very necessary (e.g. nuclear data, materials compatibility), and several confirmatory experiments will be required to support engineering and technological choices; for the latter, the R&D effort shall be limited to the issues originated by the design studies of the demonstration facility.

The very innovative engineering and technological issues dictated by the ADS concepts call for a very focalized, 'problem solving' approach, avoiding effort duplications and finalizing the activities to the goal of ADS feasibility and safe operation demonstration. This can be achieved only with a well-balanced effort between basic R&D and design studies.

The contribution of the EIP to such a pragmatic approach is constituted by the draft proposal "Preliminary Design Studies of a Demonstration Facility of an ADS" which has been prepared to be discussed with R&D bodies, and eventually be submitted to the EU under the fifth framework programme call for proposals.

The draft proposal has the purpose to identify a minimum set of design activities which are considered mandatory by the EIP to assess the engineering feasibility of the two reference options and to perform the selection of one solution to be further developed in detail within the 6th EU framework programme.

The proposed activities, organized in similarity with the EFR Project, can be summarized as follows:
- General requirements and safety criteria;
- Preliminary design studies including: systems and components design, nuclear island layout, safety and plant performances;
- Identification of confirmatory experimental R&D;
- Engineered options technical evaluation and comparison;
- Preliminary cost evaluation and comparison;
- Selection of the reference configuration to be engineered in detail.

United States of America

Summary of "A Roadmap Developing Accelerator Transmutation of Waste (ATW) Technology – A Report to Congress".

In response to the terms of the 1999 appropriation bill for the US Department of Energy, on 1 November 1999, the Department delivered to the US Congress an ATW roadmap, which describes a recommended R&D programme for Federal support. The programme extends over a six-year period and would cost approximately USD 280 million. The roadmap identifies the critical technical issues that must be addressed in order to establish feasibility, and emphasizes the value of collaboration with foreign R&D programmes. In response to a specific request from Congress to identify institutional challenges to ATW implementation, the report notes the potential policy conflicts involving spent fuel treatment and problems of long term financing, regulatory and environmental implementation, and public acceptance.

The Congressional mandate also included an assessment of the potential impact of ATW on the current civilian spent fuel programme, as well as the cost and schedule to treat all such spent fuel. The report clearly states the importance of continuing the current repository programme, since it will be required whether ATW is successful or not. If implemented, ATW applied to the current fuel would only be effective in reducing potential long term radiation doses from repository wastes by a factor of about 10, since radioactivity released from defense waste would become the dominant source of exposure. The total cost of treating approximately 87000 t of civilian spent fuel would be approximately USD 250 billion (undiscounted 1999 USD), a portion of which could potentially be offset by power sales. Such a programme could require nearly 120 years.

United Kingdom

Accelerators with power of 30 MW are suggested as machines suitable to drive an ADS. However, present technology can only demonstrate two examples of near 1 MW machines: the 800 MeV LINAC at Los Alamos (1.3 MW) and the 590 MeV cyclotron at the PSI (890 kW).

Detailed paper designs of accelerators for neutron spallation sources exist in the 1–5 MW class, notably the US-SNS and the ESS. It is likely that examples of these machines will be built in the near future. The accelerator design has concentrated on low losses and the minimization of beam halo: much theoretical and calculational effort has been expended towards this.

Designs for ≥30 MW machines for ADS have independently appeared from a number of different countries. At the same time, a number of demonstration facilities (low energy high current RFQ accelerators) are being developed.

India

- ADS has a potential to develop reactor systems for nuclear waste incineration a well as to develop more safe nuclear power systems, as alternative to present day critical nuclear power reactors. ADS is particularly attractive to build the Th-^{233}U based systems, which is of particular interest to countries like India. Issues related to ADS need therefore to be examined from both the above possible applications;
- There is need to carry out further research (possibly CRP) to work out more innovative reactor systems with overall large energy gain, requiring not very large beam powers (of, say, only 1–2 MW) for a typical 750 MW(e) system. Such a system could become compact enough to be acceptable for a practical nuclear power plant. Proposals of booster-blanket combination concept need to be further examined from this point of view. Also issues related to fast and thermal neutron spectra based ADS need deeper study;
- Development of a nuclear data library for ADS applications can be addressed to IAEA Nuclear Data Section;
- IAEA's co-ordination in this effort can help give a proper direction to this activity, cutting down duplication of efforts and utilizing the strong points of expertise and the facilities available in the participating countries;
- The most critical component in the ADS efforts is the development of a high beam power, high energy accelerator with a good efficiency of 40–50%. Can there be an international effort in this direction, like ITER?

- The spallation target technology issues are also to be satisfactorily resolved;
- There is a need for research (CRP/TCM) to examine if a modular approach is possible, to integrate existing fast reactor/thermal reactor technologies which a separately developed accelerator/target technology to make a practical accelerator driven sub-critical systems.

Belarus

- Development of the method of γ-spectrometry on high radioactive targets (…, ^{129}I, ..., ^{135}Cs, …, ^{237}Np, Cm);
- Experimental benchmark performance on subcritical assemblies with different types of neutron spectrum (external sources: Cf, d+d $\rightarrow$ ^{3}He+n, d+t$\rightarrow$$^{4}He$+n; continuous and pulsed neutron sources);
- Experimental activities in the field of spallation reactions on heavy targets (A>200) and on nuclei of structural materials (20<A<100);
- Development of the theoretical models of spallation reactions; comparison between experimental and theoretical data.

Czech Republic

The current status of the efforts to develop the transmutation technology and to employ it in nuclear power programs is in the stage of national or at least regional programs covering a broad variety of approaches. Therefore, the role of the IAEA as organizer of the exchange of information on national ADS R&D programmes or projects is of the first order of importance. The instrument of AGM or TCM is a very useful platform for such a purpose as well as a preliminary co-ordination, database implementation, etc.

For the time being, there is only a limited number of domains or technical problems which have any desirable feature of a common denominator for being put forward as a basis of higher level co-ordination (like the CRP). One of them, according to the understanding of the state of the art developed in the Czech Republic within the framework of its national ADS programme, namely the time behavior (kinetics) of a coupled subcritical multiplying system with an external neutron source is felt to be an actual issue. This domain represents without any doubt also on of the most important safety aspects of the ADS. The proposal to establish a CRP for an intercomparison of the results of national or regional studies in this sense would be the subject of a more detailed elaboration, should there be a consensus amongst the interested Member States, as well as the IAEA.

The development of ADS will very likely soon reach the level of demonstration or pilot unit design, and respective preparation of realization. An international conference on the role of ADS in nuclear power in the 21st century, in identifying alternatives for a positive solution to a crucial issue of nuclear power production, would meet high appreciation amongst the professionals.

CERN

National programmes on ADS in Europe have mostly converged or restricted themselves to two common systems as a result of the European Industrial Partnership (EIP) to promote and develop engineering design studies of a demonstration facility to assess the industrial feasibility of an accelerator driven subcritical system for high level, long lived waste transmutation. This has resulted or will result in the merging of efforts/resources both at the national level and within the fifth research framework program of the European Union.

The interest in the realization of a demonstration facility has lead to adopt a focalized "problem solving" approach of the engineering and technological issues raised by the ADS concepts. In this context, a great amount of R&D and engineering activities are being performed to demonstrate separate basic aspects of the ADS concept.

(1) Basic research is still very necessary, as it was reported in this AGM on the review of ADS national programmes, for instance:
   - Nuclear data below and above 20 MeV to validate nuclear model codes used to calculate microscopic cross sections both for incident neutrons and protons. This was clearly presented by the review of the extensive work carried out in the Russian Federation for the development of the MENDL-2 evaluation file (intermediate energy), and the CERN proposal to build a spallation driven neutron time of flight facility to measure neutron cross sections from 1 eV to 250 MeV (in collaboration with 43 universities and European national institutions);
   - Validation of codes and benchmarking are still required for quality assurance purposes (imposed by nuclear industry) and to confirm feasibility of a particular ADS concept;
   - As regards the conceptual designs, the approach adopted by the EIP to limit its effort by investigating two design options (i.e. the lead-bismuth cooled and gas cooled ADS), should not undermine the efforts carried out by other countries to develop other reference designs for purposes other than or in addition to transmutation of long lived highly radioactive nuclear waste, specifically molten salt systems and Th-U based systems for the production of electricity.

(2) Several confirmatory experiments are also required to confirm the soundness of the engineering and technological choices adopted. Experiments limited to the issues originated by the design studies of the demonstration facility should be encouraged, and participation should be extended to others (e.g. Republic of Korea, Japan, the United States, etc.…). For example:
   - MEGAPIE experiment/project (CEA-ENEA-PSI-CIEMAT-FZK);
   - SCK•CEN-PSI project in the context of the MYRRHA project;
   - CIRCE project.

(3) R&D activity in the accelerator field should be promoted; interconnection and collaboration with both the high intensity accelerator community (USA, Russian Federation) and the intense spallation source community (ESS, SNS, Russian Federation, Japan, Republic of Korea, etc.) should be encouraged.

(4) The EIP has established an excellent framework for the concentration of the efforts and R&D activities related to ADS and should be enlarged, or, alternatively, some of its findings made available to the community at large through IAEA or OECD/NEA (but one should be careful with proprietary rights).

4. RECOMMENDATIONS

The AGM's recommendations and proposals for new international activities in the ADS area can be summarized as follows:

- Establish the ADS Status Report as a 'living document' on the Internet, strongly connected to the ADS database (see next item);
- Finalize and maintain the ADS database as the core component of an IAEA ADS Webpage;

- Initiate a major conference on The Role of ADS in Nuclear Power for the 21st Century;
- Initiate benchmarks aiming at inter-comparisons of methodologies, data libraries and simulation codes; in particular for programmes with experimental backup, for example:
  - compact subcritical systems driven by neutron generators (e.g. MUSE, Minsk)
  - spallation and hybrid systems driven by accelerators (e.g. ISTC project 1371 and SAD project, both in Dubna)
  - measurement of spallation yields at intermediate energies (300–590 MeV);
- Initiate studies on compact ADS (based on reduced beam power accelerator designs);
- Initiate studies on ADS based on thorium fuels for fast and thermal neutron spectra;
- Initiate studies on the conceptual design of ADS based on the coupling of present-day reactors (fast and thermal) with accelerators;
- Initiate cross-field exchanges between the ADS community and the accelerator designers:
  - Symposium on project status of high energy accelerator development.
  - Workshop(s) on specific accelerator design topics and their relevance to ADS.

OPENING STATEMENT

Chang K. Park
Senior Vice President of KAERI

Welcome to Korea Atomic Energy Research Institute (KAERI).
It is my great pleasure to have the first advisory group meeting on accelerator driven system in KAERI. I would like to express my sincere thanks to IAEA and all other participating member countries for their support for this meeting.

We are at the edge of the centennial turning point from 20th to 21st. Nuclear energy is believed to be one of the greatest inventions man made in 20th century. It caused the revolutionary changes in the concept of energy. It seemed to provide new horizon in energy sector to human being. However, the safety, spent fuel, and proliferation problems have prevented the nuclear energy from playing the ambitious role we expected in the beginning. Those problems are still remained as the works to be solved.

Energy will be one of the most crucial problems in this coming 21st century. I do believe nuclear will be the major energy source we can lean on in the next century. Many R&D have been done to seek the solutions to make nuclear the best energy source. Some of them have been done on national basis and some are on international basis. In recent years, I can see that the international cooperations are getting more weights. I think it is very natural because the problems we, nuclear people are facing with can not be solved easily by one or two nations but they can be by the international collaborations. In terms of this kind of international cooperations, KAERI appreciates and supports strongly the activities of IAEA.

The accelerator driven system has in recent few years become a fast growing field and is getting more attentions in many countries. The reasons are, I believe that ADS can be expected to play a hopeful role to provide a breakthrough to make the nuclear energy a safe, economically efficient and acceptable, and environmental friendly energy source. However, enormous research efforts are supposed to be required to make ADS system working.

We all understand that it is not easy for one or two nations to invest such huge efforts for the development of ADS in terms of the required technical manpower and cost. That is the reason why we are getting together here today. We will discuss and exchange each nation's view on ADS for next 4 days. I am sure you can derive the most efficient ways to develop and make ADS system working throughout this meeting.

Autumn is the best season in Korea. You can experience almost spotlessly clean blue sky and fresh air during your stay in Korea. I do believe that your 4 day meeting will make a fruitful contribution to preserve this beautiful weather for our future generations.

OVERVIEW OF THE ONGOING ACTIVITIES IN EUROPE AND RECOMMENDATIONS OF THE TECHNICAL WORKING GROUP ON ACCELERATOR DRIVEN SUB-CRITICAL SYSTEMS*

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Abstract

ADS has become a major R&D topic in Europe because it promises new options for nuclear waste management in order to reduce the burden of geological storage. The resources presently allocated for ADS in Europe are significant and are related to a large number of activities spanning from accelerator to materials and fuel technology. The total effort in the last two years and foreseen for the year 2000 is estimated to be of the order of 400 man-year/year. Important ADS activities are also going on or planned outside Europe e.g., in Japan (JAERI and KEK Joint Project) and USA. This report provides an overview of the different ongoing activities on Accelerator Driven Systems (ADS) in various European countries.

1. EXECUTIVE SUMMARY AND RECOMMENDATIONS

Following the mandate of this Technical Working Group (TWG), this report provides an overview of the different ongoing activities on Accelerator Driven Systems (ADS) in various European countries along with an examination of the proposals in the final status for presentation to the 5th Framework Programme (FWP). From this review, it appears that the preliminary conclusions of the TWG Interim Report are still valid and applicable to the present state-of-the-art in terms of R&D priorities. A number of recommendations are formulated for the future development of these activities.

ADS has become a major R&D topic in Europe because it promises new options for nuclear waste management in order to reduce the burden of geological storage. The resources presently allocated for ADS in Europe are significant and are related to a large number of activities spanning from accelerator to materials and fuel technology. The total effort in the last two years and foreseen for the year 2000 is estimated to be of the order of 400 man-year/year (including European countries not represented in our TWG). The organisations performing the activities range from national R&D Bodies to Universities with a significant participation of several major nuclear Industries of the European countries.

Important ADS activities are also going on or planned outside Europe e.g., in Japan (JAERI and KEK Joint Project) and USA (ATW recently completed roadmap as requested by DOE). The TWG has identified important spin-offs of ADS activity outside the field of waste transmutation such as development of high intensity accelerator; spallation neutron sources and their applications, liquid metal technology, new structural materials, radioisotopes production, etc.

Besides the review of the national activities, the TWG has examined the proposals related to the item Partitioning and Transmutation (P&T) of the Nuclear Fission Key Action of the 5th

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* This paper was presented on behalf of the Technical Working Group (TWG) on Accelerator Driven Subcritical Systems by Stefano Monti, Scientific Secretary of the TWG. Membership in the TWG is presented in Appendix to this paper.

FWP, presently in the final stage of preparation. The intent was to identify overlaps of activities, missing points and the relative weights of the different technical areas. Although many activities are well covered (e.g. nuclear data, materials, etc.) the TWG has identified in the proposals some almost missed items (e.g. high intensity accelerator technology) or underrepresented activities (e.g. Minor Actinides (MA) based fuels). These are key issues and they need dedicated facilities, which should be available for demonstration of ADS technology.

Limited funds have been allocated by European Commission (EC) on P&T, and then to ADS. This seems not consistent with the importance acquired by the ADS technology as potential nuclear waste transmuter and with the urgent need of demonstrating the safe and efficient applicability of the concept.

The proposals of R&D within the 5th FWP cover the basic aspects and data needed for assessing the impact of this technology on nuclear waste transmutation and for the preliminary design of a reference "As Soon As Possible" (ASAP) demonstration facility, having the objectives and missions set in the TWG Interim Report.

Long term activities for the development of ADS technologies will require further important R&D programmes which should already start in a co-ordinated way within the present time frame.

Based on this review and the considerations of the prepared proposals the TWG strongly recommends an increased support and co-ordination of ADS related activities on a multinational level. Specifically, the following points are stressed:

- the need for a continuous and possibly enlarged support at national level;
- an increased support by European Commission (EC), at present only at the level of 10-20 % of the allocated resources in the national programs;
- a co-ordination at all levels; on this purpose the TWG suggests a multinational European co-ordination taking into account that wider resources are already allocated with respect to the 5th FWP and that priorities have been identified. A specific important point is the co-ordination needed between Research Bodies and European Industries. Information sharing could take benefit of the network already foreseen within the 5th FWP for ADS related proposals. The co-ordination should even include other countries and institutions not yet represented within the TWG. This could be set up within the frame of the TWG in order to avoid duplications and overlapping and enable synergies.

2. OVERVIEW OF ONGOING ACTIVITIES IN EUROPE

This chapter provides an overview of the ongoing activities in the European countries participating in the TWG. Other ongoing national programs in Europe, in particular in Sweden, UK and the Netherlands, are not covered due to lack of precise and complete information. However in all these three cases their contributions are at least partially included in the prepared proposals to the 5th FWP proposals (system studies, nuclear data, neutronics and fuel studies).

It is important to mention that at present the Paul Scherer Institut (PSI) in Switzerland, although not directly involved in the Partitioning and Transmutation program, has the highest technical potential in Europe as far as accelerator and spallation target experiments are concerned.

2.1. Ongoing activities in Austria

All the activities on ADS are carried out within the 5th FWP and are focused on scientific issues. In particular Austrian scientists will take part in the neutron time of flight (TOF) experiment proposed by CERN. Austria is planning to build "AUSTRON" - a spallation neutron source for scientific purposes - by 2006/2007. This provides interesting opportunities for specific scientific studies relevant for ADS. The internationalisation of the project and funding is under way but no final decision has been taken yet. In particular, Italy is interested in the design of the accelerator (Synchroton of Trieste).

2.2. Ongoing activities in Belgium

There is no interest in using ADS for energy production, but there is a will from the waste management body (ONDRAF/NIRAS) to consider the transmutation as potential way for waste management if this is agreed at international level. The SCK•CEN-Board has agreed on the proposal made by SCK•CEN Management to consider the transmutation as a research topic for the fuel cycle back end.

All the Belgian activity is mainly concerned with the development of the MYRRHA concept, a multi-purpose small-size ADS. The facility should replace the BR2 reactor for material and fuel research, radioisotopes production but could be useful even for transmutation research and medical applications.

Main parameters of the accelerator (H^+ cyclotron) are: E_p = 350 MeV, I_p = 2 mA. The spallation source is a liquid Pb-Bi windowless target. The sub-critical blanket is divided into two zones: an innermost fast spectrum zone and an outermost thermal spectrum one.

The ongoing activities are focused on identification of R&D needs. For the design of MYRRHA and a cost evaluation of the project based on a given design answering the scope of application given above.

First evaluation of the overall cost of the facility is around 100 MEuro of which less than a half is due to the accelerator. On the basis of a pre-design SCK•CEN will ask for founding to the Belgian Government. SCK•CEN is in order to cover 50 % of the project. Other international partners should cover the rest. SCK•CEN participates in the 5th FWP for what fits with the MYRRHA project, but no money will be asked to EC for the construction of the facility during the 5th FWP period. A more realistic cost estimate of the facility will be available at the end of 2000.

Existing collaborations are with CEA, PSI, ENEA, NRC Soreq (Israel), IBA (B) and UCL (B). MYRRHA is not considered as an ADS DEMO but is in the roadmap (first step) towards DEMO.

More specifically and besides the activities of SCK•CEN concerning the proposals for the 5th FWP, the activities related to ADS development presently conducted at SCK•CEN, UCL, IBA and BELGATOM are summarised hereafter:

- *SCK•CEN & UCL `*

SCK•CEN is willing to design and build an ADS system dedicated for R&D purposes. The applications foreseen in this facility will be on: material and fuel research, MA and LLFP transmutation demonstration, ADS technological demonstration, and radioisotope production for medical applications. As complementary applications we foresee neutron and eventually proton beam applications for applied physics purposes and also for medical application. The latest applications are conditioned by establishing collaboration contracts with the academic and medical communities. As support for the MYRRHA project, SCK•CEN is conducting or preparing R&D programmes in the following areas:

- Basic spallation data: An experiment is presently conducted at PSI (CH) in collaboration with PSI and NRC Soreq (Israel) consisting in bombarding a solid Pb-Bi target with protons of 300 MeV. The parameters to be assessed are:
  - the spallation neutron yield at 300 MeV (n/p),
  - the spallation neutron energy spectrum,
  - the spallation neutron angular distribution,
  - the spallation products induced by protons of 300 MeV.

- Windowless spallation target design: An experimental and theoretical programme is presently conducted by SCK•CEN in collaboration with the Department of thermalhydraulics of the catholic university of Louvain-la-Neuve, UCL, aiming at establishing a reliable design of a windowless spallation target made of liquid Pb-Bi. The programme foresees:
  - hydraulic experiments with hot water to simulate Pb-Bi,
  - numerical simulation using FLUENT CFD code,
  - ultimately Pb-Bi experiments will be necessary but are not yet planned

- Validation of the windowless concept in connection with the proton beam vacuum tube: A conceptual design of an experiment is actually undergoing at SCK•CEN. The experiment aims at demonstrating the feasibility of connecting a windowless target concept of MYRRHA with a vacuum environment compatible with the characteristics of the proton beam delivered by the cyclotron.

- Structural Material Research: A research programme is already going on at SCK•CEN based on irradiation in the BR2 MTR, to study the behaviour under high neutron dose irradiation of HT9 (martensitic steel) and EUROFER (Reduced Activation Ferro-Martensitic Steel, comparable to the F28H Japanese steel) as potential material candidates for MYRRHA structures.

- THOMOX Fuel Research: A R&D proposal in collaboration with TUI (Euratom) and Belgonucléaire (B) on a comparative study of fuel behaviour made of $(Th\text{-}Pu)O_2$ compared to classical MOX or advanced MOX (large grain) is presently under evaluation between the partners. The programmes foresees:
  - Fuel fabrication,
  - Irradiation in well characterised conditions (power level, burn-up, central temperature, neutron flux),
  - Irradiation in stationary and later on in transient conditions,
  - Post Irradiation Examinations: linear power, B-U, TOP-18 FP, profilometry, SEM, FG punction, gamma scanning, X-radiography.

- *IBA*

IBA is a partner of SCK•CEN aiming at the development of a reliable proton accelerator operating in CW mode and fulfilling the required specifications of beam availability needed for ADS systems. The present R&D work consists in developing a one or two stage accelerator of 350 MeV and currents of the order of 5 mA. The aim of the development being to ensure the requirements of the accelerator dedicated to MYRRHA machine (350 MeV, 2 mA).

- *BELGATOM*

BELGATOM has joined the European Industrial Partnership for developing the ADS Demo and have expressed its interest in investing some manpower on the ADS topic in the coming years. An implication of BELGATOM in MYRRHA project is expected in the near future.

2.3. Ongoing activities in Finland

The Finnish interest in transmutation is at this stage motivated predominantly by a general need to sustain competence in all fields of nuclear energy research. In order to credibly defend the "once through" cycle one must be aware of all new developments in nuclear waste disposal. It is recognised that transmutation, particularly accelerator-driven transmutation, shows some promise for a future "closed fuel cycle" although the need for a geological repository will remain even in this case. Finland has the capacity and intention to join as a partner or sub-contractor in some international ADS-related research.

2.4. Ongoing activities in France

In the context of the French Law of 1991 concerning the nuclear waste management, a twofold R&D programme is carried out, involving primarily CEA, CNRS and industrial partners: FRAMATOME and EDF.

The first part is dedicated to basic, generic R&D, in the framework of the "Groupement de Recherche GEDEON", including eight topics:

- High Intensity Accelerator (development, performances): France (CEA - CNRS) is working on the LINAC solution, in co-operation with Italy (INFN) and University of Frankfurt, around the IPHI project (High Intensity Proton Injector) and the development of superconducting cavities.
- «ADS class operation requirements» concerning reliability and stability are specified and taken into account.
- External source driven sub-critical reactor physics: MUSE programme (CEA - CNRS) involving the MASURCA facility and the GENEPI accelerator.
- System studies are performed, evaluating several core and fuel options with various coolants.
- Spallation target physics: theoretical and experimental studies are carried out by CEA and CNRS, mainly in the framework of international co-operation.
- Nuclear data: the effort in this field, and the development of powerful computer code systems lead to reduced uncertainties in the core and system studies and to unique simulation capabilities.
- Physico-chemistry of liquid metals and molten salts.

- Materials under irradiation: damage due to the proton beam, due to the spallation products, to the neutrons (spallation and fission).

The second part of R&D studies is dedicated to ADS design (industrial plant demonstrator) and namely to safety studies, in the framework of a document (to be issued at the end of 2000) related to the motivations for:

- ADS in the field of the fuel cycle back end (leading to specifications);
- a stepwise demonstration strategy (leading to specifications for a demonstrator).

The main topics are:

- Studies of scenarios comparing ADS to FR in the waste transmutation field. These studies are carried out consistently with the schemes used in the OECD/NEA expert group study.
- Assessment of the specific advantage of ADS for transmutation relating to sub-criticality including detailed studies of the reactivity balance in every situation of normal and accidental operation.
- Specific safety issues relating to the ADS concept and design: sub-criticality, containment barriers.
- Assessment of several ways leading stepwise to the «full size» demo:
- optimising the use of the main relevant (existing or foreseen) European facilities: IPHI, MASURCA, KALLA, SINQ (MEGAPIE project), CERN facilities, MYRRHA, etc.;
- taking into account the international (US, Japan) road mapping and development schedules.
- Advanced nuclear fuel development and qualification leading to the “ASAP” availability of significant amounts of fuel with high contents of MA at an early step of the demonstration (this part is developed in § 2.9). Conversely, the schedule of this development acts as a boundary condition on a cost/benefits analysis of the demonstration steps, through the (useful) power level choice of the fissile sub-critical core.
- Demonstrator: base case design studies (in France: LINAC, He as a core coolant, etc.).
- These studies are carried out by CEA - CNRS - FRAMATOME.

The next steps are:

- a report of technical and safety options for the demo (end of 2002);
- feasibility and definition studies carried out during the 6th FWP.

The French key milestone related to the 1991 Law is the year 2006.

Meanwhile, France is going on with participating in the 5th FWP and with bilateral co-operations in the ADS field:

- ENEA, INFN;
- CIEMAT;
- FZK;
- SCK•CEN (around basic R&D and MYRRHA concept);
- PSI (MEGAPIE);
- BNFL;
- NRG.
-

The R&D proceeds along two main lines:

- from the accelerator to the spallation source (with a significant power level in the target), through IPHI and MEGAPIE;
- from the sub-critical fissile core at «zero» power (MASURCA + GENEPI) to a fissile facility integrating a spallation module;

These lines have to converge towards the demonstrator. The specifications have to be determined.

The detailed definition of the demonstration steps and of the cost and benefits of each step (taking into account the availability of the technology and the licensing requirements) are currently investigated, in order to optimise the design of the first ASAP experimental facility operating at a significant power level (in the target and in the fissile core).
There is a good synchronisation between French context (91 Law) and the 5th FWP schedule, in particular for DEMO realisation. Decision for DEMO should be taken at the end of 2006, being the first steps:

- end of 2000: motivations (ADS and DEMO);
- end of 2002: main technical and safety options.

2.5. Ongoing activities in Germany

There is no possibility in Germany to build an ADS for power generation: there is only interest as transmuter.
The main research areas are:

- Design studies (neutronics, thermal-hydraulics, fuel behaviour, safety studies);
- Nuclear data (extension above 20 MeV, evaluation of Russian Measurements, MUSE exp. Exp. At FZK Van de Graaf);
- Materials/Corrosion Studies (Exp. at St. Petersburg and Obninsk, exp. at FZK);
- Pb/Pb-Bi Technology: KALLA facility (Technology loop for Oxygen control, Thermal-hydraulic loop, Corrosion loop).

In order to flatten the radial power distribution, a three targets concept has been proposed.
FZK is participating in the ISTC 559 project (FZK-proposal for an IPPE-target design; water model of ISTC 559 target) and will participate in the MEGAPIE exp.

Of great concern are:

- the development of codes for safety studies (SAS5A, SIMMER III)
- investigations on steels with modification by mean of Pulsed Electron Beam Facility (which shows no corrosion in Pb with Al alloy)
- the realisation of the KALLA laboratory, which includes three Pb-Bi loops (thermal-hydraulic and material investigations, Oxygen control). The smallest loop is under construction. An oxygen-control meter will be developed by FZK and compared to the one developed jointly with Russians.

Some other activities are also carried out by FZJ, which has competence in gas technology.

2.6. Ongoing activities in Italy

The Italian National programme on ADS is divided into two main parts:

- The TRASCO programme: a two years R&D programme promoted by ENEA and INFN and approved at governmental level (18 G£ ≈ 9 MEuro);

- The Industrial programme (an ANSALDO initiative in collaboration with CRS4, ENEA and INFN) devoted to the development an EA-like DEMO facility for nuclear waste transmutation.

The TRASCO programme addresses both the LINAC-accelerator (managed by INFN) and the sub-critical system (managed by ENEA). The programme also involves industrial companies (ANSALDO, CINEL, CISE, CRS4, FN, HITEC&SISTEC, SAES-GETTERS, ZANON), Italian Universities and research organisations (INFM). The different sub-programmes concern:

- The proton source
- The Low and medium energy accelerator section
- The High energy accelerator section
- The Neutron production for material characterisation
- The General design criteria and safety classification
- The Neutronics and transmutation efficiency
- The Thermal-Hydraulic analysis
- The Beam window technology
- The Material technology and compatibility with Lead and/or Lead-Bismuth alloy

Within the program the realisation of some accelerator components (proton source, first section of the RFQ, prototypes of SC cavities) and a Pb-Bi loop for material compatibility is foreseen.

As far as the industrial programme, ANSALDO, with the collaboration of ENEA, CRS4 and INFN, has issued a Reference Configuration of an EA-like ADS DEMO Facility for nuclear transmutation. The proposed EA_DF, already presented to the three-countries-TWG and included in the Interim Report as one of the two possible options for an ASAP_DEMO, is a Pb-Bi cooled 80 MWth sub-critical system driven by a 600 MeV – 3÷6 mA PSI-like cyclotron. The target is a with or windowless liquid Pb-Bi concept. Fuel pellets are as for the 2nd load of SPX1 reactor.

An extension of the TRASCO programme has been submitted to the Research Ministry at the end of 1998. The new proposal is being funded with further 9 G£ (about 4.5 MEuro). The extended programme foresees:

- realisation of a full-scale RFQ energising it at full power
- optimisation of the accelerator design;
- as recommended by the French-Italian-Spanish TWG, realisation of a test facility (CIRCE) to study the behaviour of the Pb-Bi eutectic as coolant of the sub-critical system and the spallation target.

Furthermore an extension of the industrial programme has been submitted to the Research Ministry at the end of 1998. This proposal, starting from the EA_DF conceptual configuration, deals with the preliminary design, inclusive of safety analysis, of the Pb-Bi cooled ADS demonstration facility. This in strict synergy with the parallel TRASCO programme extension.

2.7. Ongoing activities in Spain

There is a CIEMAT-ENRESA collaboration agreement on Research and Technological Development for the Partitioning and Transmutation of Long Lived Radioactive Isotopes. The agreement, signed on March 1999, is a consequence of the increasing Spanish and international interest on P&T. ENRESA is interested in P&T for evaluating the application of these technologies to the management of spent fuel and other radioactive wastes. CIEMAT has decided to launch an R&D programme on P&T technologies and basic research as a continuation of its previous activities. Both institutions would like to perform this research activities in the largest possible international collaboration and well integrated on the 5th. The approved and funded 5 years programme covers 5 main research areas:

- spallation process and neutronics for transmutation;
- participation on international experiments on spallation, sub-critical assemblies, neutronics and nuclear data;
- partitioning and separation of long lived radio-nuclides by advanced hydro-metallurgic processes;
- pyro-chemical reprocessing of fuels and irradiated targets;
- research and development of ADS components.

Projects of P&T research at CIEMAT are:

- FACET project: basic research on ADS + experimental studies (MUSE, FEAT, TARC, TOF experiment at CERN PS);
- Materials and technologies for use in presence of lead alloys;
- Irradiation effects on Martensitic Steels;
- Pyrometallurgic reprocessing;
- Hydrometallurgic reprocessing;
- ADS Safety.

Small loops for corrosion studies are already available including one with Pb-Bi forced circulation. Some of these projects include large participation of several Spanish Universities.

Other initiatives:

Empresarios Agrupados has joined the European Industrial Partnership proposal for the 5th FWP.

LAESA (E), University of Pavia (I), CRS4 (I), Technicatome (F), CRISA (E), Universidad Politécnica de Madrid (E), AIMA (F), Instituto Técnico Nuclear Fundacion F^2I^2 (E) and INTECSA (E) have been proposing, even to the E.U. 5th FWP, a new project concerning the "Conceptual and prototype designs of optimised transmutators based on cyclotron driven pebble bed reactors". This proposal is described at point 3.2.3 of this report.

2.8. The MEGAPIE project proposal (1 MEGAWATT Spallation Target Pilot Experiment)

The MEGAPIE proposal has been initiated by PSI, FZK and CEA. JAERI ha expressed its interest to join. The objective of the project is to build a 1 MW_{th} liquid Pb-Bi target (for which forced circulation is foreseen) to be put in the SINQ facility at PSI, replacing the present (and next) solid target. The horizon is beginning of 2004. The design of MEGAPIE will be supported by parallel R&D work on liquid metal thermal-hydraulics, to be performed mostly in the KALLA laboratory at FZK, and on materials. The experiment will be equipped with enough instrumentation to validate thermal and mechanical phenomena, spallation product build-up and their effects on LME and corrosion. The MEGAPIE target will benefit of the existing equipment at SINQ (handling machine, shielding, secondary cooling system).

The LISOR experiment (part of which is proposed in the frame of the proposal "Technologies, materials and thermal-hydraulics for Pb alloys" to the 5th FWP) is intended to be part of the MEGAPIE project and to provide essential answer to the possible fast embrittlement problems. The overall MEGAPIE project is estimated to about 10 MSFr (excluding manpower). Part of it can be proposed for the second "call for proposals" of the 5th FWP.

2.9. Fuels for transmutation

As far as the French program on fuels for transmutation (which is applicable to both critical and subcritical systems), the program guidelines are as follows:

- Incineration of MA in homogeneous mode (dilution < 10%) in standard fuel (LWR, FR) or CAPRA type (high plutonium content);
- Incineration of MA in heterogeneous mode (high concentration > 20%) in a dedicated fuel or a target;
- Transmutation of LLFP in targets.

2.9.1. Dilution of actinides (homogeneous mode)

The objective is to test the behaviour under irradiation, of fuel with MA addition, by comparison to standard fuel. The SUPERFACT experiment (oxide fuel) has performed and analysed different fuel target. This is also the case of TRABANT experiment in HFR (collaboration FZK-ITU-CEA). METAPHIX (UPuZr + MA,RE) experiment, under contract with TUI and with CRIEPI concerns metallic fuel and is foreseen in PHENIX.

2.9.2. High concentration actinides (heterogeneous mode)

A first option is MA compounds dispersed in a matrix.
The program addresses the choice of the inert matrix and the selection of the actinide compounds. The program also addresses the issue of micro vs. macro dispersion of the actinide in the matrix. A second possibility is a solid solution with MA. The program looks for the choice of elements leading to a solid solution with MA.

2.9.3. Matrix selection and behaviour under irradiation

Two objectives:

- to evaluate and increase the capacity of oxide matrices MgO and $MgAl_2O_4$ to withstand high fast fluence and FP damage (MATINA 1A, 2 experiment in PHENIX already performed);
- to enlarge the irradiation programme to other supports: oxide ceramics, nitrides, metals, (the MATINA 3 experiment to be performed in PHENIX);

2.9.4. Selection of an Am compound and target concept optimisation

Selection from out of pile R&D on materials: fabrication, characterisation and property measurements of the Am compounds, and successively of the CERCER.

Two sets of experiments are foreseen:

1st set (simple oxides + microdispersion):

T4, T4 bis experiments in HFR (in the frame of the EFTTRA collaboration)
ECRIX-H&B (locally moderated flux) in PHENIX

2nd set (stabilised oxide + macrodispersion):
T5 experiment in HFR (EFFTRA collaboration)
CAMIX and COCHIX in PHENIX

One more set of experiments to be performed in the BOR-60 reactor is under discussion (in the frame of a collaboration FZK-ITU-CEA). This experiment will concern targets fabricated by pyrometallurgy.

2.9.5. Solid solutions with high MA concentration

The objective is to select the more promising candidate materials by achieving as quickly as possible a high burn-up.

1st set of experiments: Pu to simulate Am, oxide or nitride solid solutions (PINPOM experiments at HFR, under discussion in the frame of a FZK-ITU-CEA collaboration).

2.9.6. Long lived fission products (LLFP)

a) ^{99}Tc target irradiation in PHENIX

ANTICORP1 (locally moderated flux) the objective is the incineration of ^{99}Tc, in metallic form. This is complementary to an irradiation already performed at HFR (EFTTRA collaboration).

^{129}I, ^{135}Cs target irradiation in PHENIX:

ANTICORP2 (locally moderated flux) still to be defined.

b) Some experiments on iodine compounds were already performed in HFR. Complementary out of pile R&D on compounds is needed before defining the irradiation.
Finally, some experiments on inactive caesium are foreseen.

2.9.7. Moderators

This has to be intended for the concept of transmutation in "moderated" subassemblies at the periphery of a FNR.

The objective is to define the utilisation conditions for solid state moderators (e.g.: hydrides) in a FNR.

First, basic R&D on materials is needed and a collaboration with MINATOM is underway on the properties and behaviour of Hydrides.

Then the test of behaviour under irradiation (fast flux) is planned. (MODIX experiment in PHENIX).

2.10. Accelerators for ADS

2.10.1. General remark concerning the conceptual design

The high-power proton accelerators needed to drive an ADS are a critical issue since the required performance is an extrapolation from existing machines. The TWG Interim Report has already analysed this matter in some detail.

The choice of the beam intensity and energy will result from an interdependent, and complex optimisation of the whole hybrid system, including financial constraints. Since the number of neutrons produced per incident proton is a function of both beam energy and the design of the spallation target, it is obviously an important ingredient for this optimisation. The neutron flux to be produced is essentially given by the choice of the effective criticality factor k_{eff} and the wanted thermal power P_{th}. Other considerations for the beam energy are the power loss in the target window, inversely proportional to the energy and the shielding of the accelerator, which sets in practice a high energy limit. As for the intensity, the accelerator should be able to provide variable beam power in order to account for the change of k_{eff} induced by the evolution of the core composition in time so that P_{th} may be kept constant.

The accelerator design has therefore to stress 3 major objectives:

- The beam intensity is a priori considerably higher than those achieved routinely today, correspondingly the beam losses have to be kept at an extremely low level. One may note, however, that the requirements for an "ASAP_DEMO" (see TWG interim report, page 9) could be somewhat less stringent, closer to what has already been achieved.
- The accelerator must achieve a level of reliability/availability, which has to be substantially above the usual request at laboratories devoted to fundamental research. In particular beam trips lasting in excess of some seconds should be limited to a few per year.
- The first two objectives, in particular when combined, ask for a rather conservative design philosophy. Being such an approach intrinsically expensive, it has to be counterbalanced by a particular effort for cost effectiveness.

Correspondingly, the specifications of the accelerator, to be defined in close contact with experts for spallation target and core design, should include:

- Maximum beam current and energy;
- Range of beam current and duty cycle for both start-up and operation;
- Accuracy of the beam current and energy control;
- Time constants for the current control;
- Performance of the beam system abort in case of an emergency shutdown;
- Reliability / availability and number of beam trips;
- Restart procedure after a beam trip;
- Monitoring and control of the beam profile on the target;
- Acceptable beam losses along the accelerator and resulting concrete shielding;
- Acceptable beam losses with respect to the access time to the machine area and remote control.

2.10.2. Present European efforts on conceptual design for a driving accelerator

The TWG Interim Report has considered two basic options for an accelerator: an "high-energy" facility based on a super-conducting LINAC and a "low-energy" facility based on coupled isochronous multi-sector cyclotrons (see Interim Report, pg. 10-12). There is considerable experience in Europe for both types of accelerators, and teams in several countries are performing important R&D efforts on accelerator performance.

PSI (Switzerland) operates a high intensity (room-temperature) cyclotron facility (1-2 mA, 600 MeV) and has studied together with LNS-Catania (Italy) (super-conducting heavy-ion cyclotron) an evolution of the PSI concept aimed at an intensity of 10 mA at 600-800 MeV. Furthermore, the operational experience from the MEGAPIE experiment (see section 2.8) will give precious insight in the topic of accelerator reliability. In Belgium, at the CRC of the Louvain-la-Neuve University, recently-built (warm) cyclotrons are used for nuclear astrophysics research where good efficiency is at prime. A historical synergy exists at the same location with the IBA company, a world-renowned major commercial cyclotron builder with clear competence for reliability and cost-efficiency aspects. GANIL (France) operates a warm coupled-cyclotron heavy-ion facility: a radioactive beam post-accelerating cyclotron, constructed together with the IPN Orsay (France) is presently under commissioning; all these accelerators have been vigorously optimised for high transmission. IPN (Orsay, France) and KVI (Groningen, the Netherlands) have built together a super-conducting heavy-ion cyclotron which operates since four years over a very broad particle and energy range. The JYFL (Finland) warm cyclotron facility is known for its radioactive beam experiments and the development of associated instrumentation.

Concerning linear structures, one has to underline the very remarkable development of super-conducting cavities and their associated radio-frequency driving systems. In Europe, outstanding motors have been TTF, the test-bed program initiated for a future 500 GeV electron linear collider at DESY (Germany), and the evolution of CERN's LEP into the LEP200. The very successful R&D effort for TTF has been assured by a collaboration of DESY with LAL/IPN-Orsay and CEA-Saclay (France) and teams from INFN (Italy). Its cavities are fully operational as well as those of the booster-accelerator ALPI developed by INFN for its heavy-ion laboratory at Legnaro. Concerning more specific development of high-intensity proton accelerator, the teams from CERN (Italy and France) quoted above, based on their experience with cavities for relativistic electrons, have started to develop those "low-β" cavities, which would accelerate above about 200 MeV. The low-energy part needs a warm linear injecting structure, and here the Italian and French teams have started a major design effort, the most difficult part being the RFQ injector. The Italian team is also studying, for the low energy part, a possible super-conducting solution based on independently phased resonators. A 100 mA injector is presently being built by a French Saclay-Orsay collaboration (IPHI Project).

Expertise in linear structures is also available at the German Heavy Ion laboratory in GSI and at the University of Frankfurt.

One should finally point out that the ADS projects in both the US and Japan have selected LINACs and strong expertise is existing for the various components in these countries (Los Alamos, CEBAF and JAERI-KEK, respectively).

3. PROPOSALS FOR THE 5TH FRAMEWORK PROGRAMME PREPARED BY THE RESEARCH AND INDUSTRIAL ORGANISATIONS

An informal co-ordination was started in June 1998 (Karlsruhe), initially among the participants to the 4th FWP in the field of Transmutation and ADS.
Two further meetings were held in Paris (September 1998 and January 1999). Minutes of these meetings are available. This initiative triggered a concerted proposal for projects in the

field of P&T and, in particular, ADS that will be summarised in what follows. In parallel, 2 proposals in the field of partitioning chemistry are being finalised, as well one proposal related to the new TOF facility at CERN.

Finally a proposal related to Thorium, up to now discussed in the frame of the "transmutation" package of the "Safety of Nuclear Fission" key item, will not be discussed here, since it will be part of a separate item of the 5th FWP.

3.1. The project proposals related to ADS and transmutation

The project proposals can be grouped under three major headings:

- Basic studies:
  - Nuclear Data (4 proposals)
  - Neutronics of the subcritical core (1 proposal)

- Materials and processes:
  - materials for target, window and coolant (2 proposals)
  - pyrochemical processes (1 proposal)[a]
  - fuel for transmutation (1 proposal)

- System and scenario studies:
  - scenarios for ADS utilisation in a power park (1 proposal)
  - system studies (3 proposals)[b]

For each proposal a summary content is given below along with information on their status, resources required, share among laboratories, when available.

3.2. The system and scenario studies

3.2.1. ADS design analysis (ADDA)

This proposal, driven by R&D organisations (BNFL, CEA, CIEMAT, CNRS, FZK, SCK•CEN, ENEA, INFN, FZJ, RIT-Stockholm, NRG, JRC-Ispra, Frankfurt Un.), is focused on the motivation for transmutation in ADS with Pu and MA-based fuel, both of small and large size, with Pb-Bi or gas cooling. The inter-comparison of small/large sizes is related to the reactivity coefficient issue and, since the issue is not the same with different coolants, both gas and Pb-Bi are considered. The inter-comparison with critical systems is a major deliverable, in order to assess effectiveness and peculiarities of the ADS. Among the tasks foreseen, the safety characteristics of an ADS, and the required specification (reliability/availability) for the high intensity proton accelerators will be the object of studies. A "MYRRHA" task is also foreseen. In this task the windowless concept is assessed and the safety issues related to licensing are considered. The implication for MYRRHA of the transmutation goal is also evaluated. The proposal ADDA includes the assessment of methods and calculation tools for use both in R&D and plant design. (98 man-years of which 17 man-years for accelerator task).

[a] Two separate proposals are prepared on separation chemistry as follow-up of the 4th Framework Program.

[b] One further proposal is under discussion, as follow-up to the « IABAT » project of the 4th Framework Program. The Thorium proposal is no more considered here.

3.2.2. Preliminary design of a demonstration facility

Following the recommendations of the Advisory Group and the TWG, an industrial platform (ANSALDO, BELGATOM, EMPRESARIOS AGRUPADOS, NNC, SIEMENS, FRAMATOME), referred to as EIP (European Industrial Partnership) in the rest of the text, has proposed a demo (~ 100 MWt) oriented project. The aim is to inter-compare three design options with respectively: Pb-Bi, Gas and Na cooling.

The goal of the proposal is to perform preliminary design studies to a level sufficient to: i) identify all the technical concerns of each design option, ii) to compare the different configurations (technical/technological aspects, feasibility, costs, basic and confirmatory R&D needs), iii) to make recommendations for detailed design studies to be performed within the 6^{th} FWP.

Two design options (Pb-Bi and Gas cooled reactors) will be developed up to the definition of a plant reference configuration including system and component design, nuclear island layout, safety and plant performance. A third (sodium cooled) option will be only partially developed mainly to give technical and economical elements to perform the comparison of the two forerunners concepts. Only the sub-critical system will be studied; the accelerator will be treated as functional and mechanical interface defining the main requirements in terms of performances and reliability/availability factors.

The proposed work is split in two main phases: the Conceptual Design Phase, lasting the first two years of the program, that is devoted to the general concept studies in order to examine and justify the main choices of each option and the Preliminary Design Studies Phase, with a duration of two years, dedicated to the preliminary design in order to consolidate the feasibility and cost evaluations, to confirm or assess R&D needs and finally to compare the design options to give essential elements for the selection of a reference configuration to be designed in detail and eventually realised. (125 man-years)

3.2.3. Conceptual and prototype design of pebble-bed transmutators

This project is aimed at the development of a Nuclear Transmutator based on a pebble-bed gas cooled sub-critical reactor activated by an intense neutron source driven by a cyclotron-accelerated proton beam.

Project objectives are twofold:
- to complete a conceptual design of an ADS system representing a family of nuclear waste transmutators;
- to complete the detailed design of a prototype of these systems with a nominal power around 10 MW.

The project is structured as a “combined project” with two parts:
- a Demonstration Project, devoted to the study of high current, high availability cyclotron to activate the neutron source;
- a Research Project for the conceptual and detailed designs of the nuclear system.

About the former, it is considered in the project that cyclotrons can offer a not-expensive solution for a broad range of accelerating voltage and current. Availability can be improved by using modern technologies for computerised control of the beam.

About the latter, emphasis will be put in analysing:

- transmutation performances of under-moderated pebble-bed sub-critical reactors;
- spallation source configuration for minimising proton and neutron fluxes (and energy deposition) in the target window in order to have low power density and temperature;
- safety features of the sub-critical reactor in relation to after-heat cooling and reactivity issues, looking for inherently safe mechanisms.

The project is based on a well-balanced collaboration between research institutions and industrial companies, co-ordinated by LAESA, a company specifically created for that purpose. (57 men-years)

3.2.4. P&T strategy studies

This proposal is the continuation of a project of the 4th FWP where P&T scenarios based on critical reactors were studied and inter-compared. The new proposal is related to innovative options (ADS, but also Thorium and the use of HTRs) and complementary options with respect to the items studied in the 4th FWP.

The inter-comparison will be made according to criteria such as the technical feasibility (including fuels fabrication, reprocessing and waste management issues) but also short term risks for workers and environmental impact.

Participants are: BNFL, BN, CEA, CNRS, ENRESA, FZK, FZJ, JRC-Ispra, NRG, SCK•CEN. (34 man-years).

3.3. Materials and processes

3.3.1. Irradiation effects in martensitic steels under neutron and proton mixed spectrum

The objectives of this proposals related to the investigation of irradiation damage in the target structural material (particularly the window target) of an ADS, are as follows:

- determine the properties of candidates as structural steels under spallation conditions;
- provide the basic mechanisms and modelling for the phenomena observed under spallation conditions;
- specify a reference material and give a path for development of an advanced window material.

Several experimental facilities (neutron irradiation and PIE) will be used in this context.

The participants are: CEA, CIEMAT, CNRS, NRG, ENEA, SCK•CEN, RIT-Stockholm, PSI.

(30 man-years + specific budget for some experiments).

3.3.2. Technologies, materials and thermalhydraulics for Pb alloys

The proposal main tasks are as follows:

- corrosion of materials in lead alloys;
- structure protection (e.g. oxide layers or specific coatings);
- mechanical behaviour of structural materials in contact with lead alloys (embrittlement mechanisms);
- effects of irradiation on lead alloys corrosion and material embrittlement (LISOR experiment at PSI);

- impurities control and removal;
- thermal-hydraulics experiments: small/medium scale analytical experiments and large scale experiments (global validation and gas enhanced circulation tests);
- development of oxygen control instrumentation.

Participants: CEA, CNRS, FZK, CIEMAT, RIT-Stockholm, SCK•CEN, ENEA, CNRS, JRC-Ispra.

(70 man-years)

3.3.3. Fuels and targets for americium transmutation

The objective of this project is the development of fuels and targets for transmutation of americium in ADS or critical reactors. To achieve this objective, fabrication methods will be developed and irradiation experiments will be prepared: the EFTTRA collaboration experiments T5 (irradiation in HFR) and F3 (irradiation in PHENIX).

In particular Am oxide targets in different inert matrices and with different particle dispersions will be studied (T5). New matrix candidates (ZrO_2, nitrides, thoria, CERMET, etc) will be screened, irradiating them without Am (F3).

Participants: BNFL, CEA, EDF, JRC-IAM, JRC-ITU, NRG, PSI

(22 man-years + specific budget for irradiation).

3.3.4. Pyrometallurgical reprocessing basic data acquisition

The main items of this proposal are as follows:
- separation by salt/metal extraction and by electrofining,
- conversion to halide form and salt decontamination,
- waste form modelling and system studies.

Participants: CEA, CIEMAT, BNFL, AEA-T, ENEA, NRI, ITU
(32 man-years).

3.4. Basic studies

3.4.1. The MUSE experiments for sub-critical neutronics validation

The MUSE experiments to be performed at MASURCA (CEA-Cadarache), will investigate several sub-critical configurations driven by an external neutron source provided by (d,d) or (d,t) reaction. The deuterons will be provided by a pulsed accelerator (GENEPI), to be installed in the MASURCA facility. The configurations foreseen will have uranium, plutonium, thorium fuels, with coolant Pb, Na, gas. Several levels of sub-criticality will be investigated (from $K = 0.95$ to 1). Measurements are foreseen of the sub-criticality (with static, dynamic and neutron noise methods) of the neutron spatial distributions, neutron spectra, effective delayed neutron fraction and neutron source importance parameter (φ^*). Extensive cross-comparison of codes is foreseen. Experimental reactivity control techniques will be developed, related to sub-critical systems operation. The participants aim to recommend validated calculational scheme able to treat sub-critical systems and to assess the associated uncertainty of the predicted parameters.

Participants: CEA, CNRS, BNFL, FZK, ENEA, CIEMAT, NRG, RIT-Stockholm, TU-Delft, JZJ, Polito-Turin, SCK•CEN

(45 man-years).

3.4.2. High and intermediate energy (20 - 200 MeV) experiments and data evaluation

This proposal is mainly related to microscopic experiments (Cr, Fe, Pb, Bi and Th), but has a component related to nuclear models. Data libraries evaluations are foreseen, for application in transport codes.

(10-20 man-years).

3.4.3. Neutron cross-section data for unstable fission products and actinides

This proposal concerns experiments to be performed at the FZK-Karlsruhe Van de Graaff and JRC-IRMM Linac. Both time-of-flight and activation measurements are foreseen. Three topics are defined:

- neutron capture cross-sections and related resonance parameters,
- neutron fission cross-sections,
- (n,p), (n,α) and (n,2n) reactions.

The selection of nuclei to be measured is underway (co-ordination with the TOF proposal of CERN).

Participants: FZK, JRC/IRMM, CEA.

(10-20 man-years).

3.4.4. Validation of neutron data for transmutation using Integral Experiments

The proposal comprises the following topics:

- Analysis of integral experiments in LWRs (MOX and high burn-ups);
- Analysis of integral experiments in FRs (PROFIL, SUPERPROFIL, PFR, KNK-II experiments);
- Analysis of experiments in critical assemblies (MINERVE facility in CADARACHE: OSMOSE experiments);
- Evaluation of basic nuclear data;
- Sensitivity Studies and Accuracy of Recycling Scenarios.

The proposal is meant to cover the validation of data for transmutation (mainly minor actinides and long lived fission products) in both thermal and fast neutron spectra.

(10 man-years).

3.4.5. TOF Proposal

This proposal includes:

- the spallation TOF facility (Design and Construction of the neutron beam, etc.);
- the measurements station (data acquisition system, front-end electronics, determination of beam characteristics, detectors for cross-section measurements);

- cross-section measurements in the field: data of relevance for Th-cycle; data for LLFP incineration; data relevant for actinide transmutation (targets: ^{237}Np, $^{238-242}$Pu, $^{241-243}$Am, $^{242-245}$Cm). The determination of required precision, the tests of the TOF performances and the calibration with mono-energetic neutrons, are also foreseen ;
- cross-section evaluation and dissemination.

Even when the actual effort required by TOF is estimated on 218 man-years, the TOF collaboration is requesting 4.52 MEuro (equivalent to 60 man-years) to the P&T call for proposals of the 5th FWP.

3.5. Perspectives

For most of the proposals, specific meetings have allowed to define objectives and the share of work. Some constraints on overall manpower/budget requirements have been given as guidelines, in order to be consistent as much as possible with the foreseen allocation of funds to the transmutation and partitioning item under the key action on nuclear fission (26 ± 3 MEuro).

The present informal co-ordination foresees a final meeting in Paris, Sept. 13-14. Moreover, a network is proposed (BNFL), to provide a continuity for a formal co-ordination during the 5th FWP. A final comment concern the possibility to reorient some of the proposals or to present new ones for the second series of "call for proposals", after the mid-term review. This could be the case of one-MWt liquid Pb-Bi target to be put in SINQ (the MEGAPIE project).

4. COMMENTS AND EVALUATION OF THE PROPOSED R&D RELATED TO ADS

The analysis has been focused on an evaluation of the prepared R&D proposals related to the P&T and ADS topics of the 5th FWP. The supplementary ADS related R&D activities on a national level need a further review to determine their contributions for the development of an ADS Demo facility.

4.1. System and scenario studies

The TWG recommends to stress the objectives of the ADDA (ADS Design Analysis) proposal in order to avoid misunderstanding with the EIP proposal "Preliminary design of a demonstration facility". The ADDA proposal should focus on the motivation of transmutation in ADS with Pu and MA-based fuel, both of small and large size, with Pb-Bi or gas cooling. The inter-comparison of small/large size is related to the reactivity coefficient issue and, since the results depend on the used coolant, both gas and Pb-Bi cooled systems are considered. The comparison include critical systems to allow a cost/benefit analysis of the different systems.

The MYRRHA contribution to the ADDA tasks should be clearly established. The TWG, based on the report of the MYRRHA representative, agreed on the relevancy of the assessment of the windowless concept and the safety issues related to the licensing aspect. The implication for the MYRRHA project design, taking into account the transmutation goal, should also be considered.
The ADDA proposal includes the assessment of methods and calculation tools (e.g. neutronics and Thermal-Hydraulics) to be used for R&D as well as for Plant Preliminary Design purposes.

Overlapping between both proposals (e.g. geometrical arrangements and outline design studies of small size plants and ADS Demos) should be avoided through a strict and efficient and effective co-ordination which can be achieved through the TWG and the ADS Network envisaged between the different partners involved in the ADS and P&T proposals. This co-ordination between R&D organisations and EIPs is recommended not only for System and Scenario Studies but also for Basic Studies and for Material and Processes.

The TWG supports the idea of using the expertise among the TWG and the ADS Network to co-ordinate and harmonise the activities on ADS.

The TWG also recommend to ensure that the expertise gained through the R&D Bodies get transferred to the European Industries and vice versa.

4.2. Materials and processes

In order to achieve a consistent solution, a best trade-off from the corrosion and the irradiation embrittlement point of view, a strong co-ordination is needed between the 3.3.1 and 3.3.2 proposals.

Being the ADS ASAP Demo the first step dedicated to the demonstration of the correct coupling of the accelerator and the sub-critical system, on the fuel aspects the TWG stresses to not underestimate the needed effort for developing, at least for the long term, a new fuel and even more a new fuel cycle. Indeed, in order to perform transmutation in a most effective way, fuel should be used with high contents of MA's. This could be on Th-based inert matrices or in fuel containing MA and Pu only. Other consideration could be the fuel form: metal fuel, nitride, or oxide. The R&D topics to be considered are:
- Fabrication, remote handling, new fabrication processes,
- Irradiation behaviour, He accumulation, swelling, thermal behaviour,
- Dry/wet reprocessing, remote handling, ^{238}Pu.

Even if some of those aspects are addressed in the present proposals and some activities will be carried out in the national programmes, the TWG stresses that the present effort is insufficient and should be increased already now even if the results are not strictly needed for a demo design. Finally the TWG stresses the fact that the availability of the appropriate facilities (hot laboratories, irradiation reactors with the appropriate neutron spectra) should be guaranteed and their performances upgraded to handle large amounts of MA.

4.3. Basic studies

As far as the TOF proposal, the TWG recognises its scientific merit and relevance in order to establish the credibility of the announced performances of transmutation in ADS. However, two points should be clarified, namely:
- the large requested financing of the installation and related instrumentation (approach quite different from the other experimental proposals) should be justified within the goals and scope of the Nuclear Fission key action of the 5th FWP;
- the experimental program of the proposal related to the transmutation should be dominant and monitored by experts coming from the ADS community (the TWG could provide such expertise).

These two points should be accounted for, in order to improve the technical and financial balance of the overall proposals.

5. OPEN POINTS

It has been investigated by the TWG whether there are important R&D needs for ADS, which are not covered by the envisaged 5th FWP proposals. In doing so, needs for the ASAP DEMO as well as long term needs were considered.

In accordance with the TWG Interim Report three options were evaluated with respect to the ASAP DEMO. The first option is a Demonstration Facility designed using existing technology (Na as primary coolant) for the sub-critical core. Second and third options would be facilities using either Pb-Bi or gas as coolant. Different power levels were discussed as well. It was the common understanding, in accordance with the conclusions of the TWG Interim Report, that the DEMO should primarily demonstrate the basic principle of coupling of the proton beam to the sub-critical blanket and should not use MA fuel (except possibly for irradiation experiments).

It is agreed that the envisaged proposals of R&D within the 5th FWP cover all the basic aspects and data needed to assess the impact of ADS on nuclear waste transmutation and for the preliminary design of a reference demonstration facility having the objectives and missions set in the TWG Interim Report.

Besides the ASAP DEMO objective, long term research for full “ADTT (Accelerator Driven Transmutation Technology) Demonstration” was also considered. A global long term demonstration programme would have to take into account:

- steps and goals for developing the programme;
- optimisation of the steps for an ASAP strategy: technology availability and acceptability (fuel, coolant, accelerator);
- cost-benefit analysis depending on the boundary conditions coming from technology;
- optimisation of the DEMO.

Topics for the long term programme that need additional research effort are the following ones:

5.1. Fuels and fuel cycle

The development of a new fuel up to industrial level may require an R&D effort of many years. This item has to address points related to:

- fabrication
- irradiation behaviour
- reprocessing, including new or specifically developed methods as pyrochemistry.

The R&D work must include alternatives on:

- chemical composition (nitrides, etc.);
- fuel configuration (particle, etc.);
- isotopic composition, with special attention to Minor Actinides fuels.

5.2. Accelerator

The two main lines were already considered in the TWG Interim Report:

- LINAC
- Cyclotron

In both cases it is stressed that there is the requirement of a very high reliability and availability of the accelerator driving a neutron source for an ADS system. This is fundamental for the operation and must be met by ADS accelerators, which must also reach a significant level of beam power (cf. 2.10).

5.3. In-service inspection and repair (ISI&R) for liquid metal systems

According to previous experience in liquid metal fast breeder reactor (LMFBR), this is a fundamental technological requirement for any ADS using liquid metal for any component; in particular the use of Pb-Bi as coolant will lead, due to corrosion issues and density value, to the need to develop upgraded ISI&R techniques and tools. However, these constraints are somewhat relaxed for the short-term ADS DEMO.

5.4. Spallation target

The isotopic composition and the chemistry of spallation products are an important issue, in particular for long-term performance optimisation.
The confinement and purification of radioactive products are major points in this context.
Material damage (radiation damage, contamination by evaporation products, corrosion, etc.) will also be a fundamental subject.

5.5. Shielding issues

Shielding aspects will have to be treated even in any DEMO design: their careful optimisation will be necessary to maximise radiation protection and to minimise activation problems, specifically for the very high energy neutrons.

5.6. Licensing

The ADS ASAP DEMO, and eventually bigger prototypes and commercial plants, will be based on several innovative technologies. Safety aspects will require a particular design effort to address all the critical issues in the early stage of the project. As suggested by the TWG Interim Report, the licensing process should start from the earliest stages of the design, in order to implement from the beginning the requirements of the relevant Safety Authorities.

Appendix

MEMBERSHIP IN THE TECHNICAL WORKING GROUP (TWG) ON ACCELERATOR DRIVEN SUBCRITICAL SYSTEMS

Chairman:	Carlo Rubbia, ENEA
Austria:	Helmut Leeb, Institut fur Kernphysik, TU Wien
Belgium:	Hamid Ait Abderrahim, SCK•CEN
Finland:	Mikael Björnberg, VTT Energy
France:	Bernard Carluec, NOVATOME/FRAMATOME
	Alex Mueller CNRS-IN2P3
	Massimo Salvatores[1], CEA
	Jean-Baptiste Thomas, CEA-DRN

[1] Invited guest.

Germany:	Gerhard Heusener, FZK
Italy:	Giuseppe Gherardi, ENEA-ADS Giuliano Locatelli, ANSALDO Marco Napolitano, INFN
Spain:	José Maria Martínez-Val, Scientific Adviser of the Spanish Office for Science and Technology Angel Pérez-Navarro, LAESA Enrique Gonzalez Romero, CIEMAT
Scientific Secretary:	Stefano Monti, ENEA

STATUS OF THE LOW ENERGY ACCELERATOR DEVELOPMENT FOR KOMAC

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Abstract

KAERI (Korea Atomic Energy Research Institute) has been performing the project named KOMAC (Korea Multi-purpose Accelerator Complex) within the frame work of national mid and long term nuclear research plan. The final objectives of KOMAC is to build a 20-MW (1 GeV and 20 mA) cw (100% duty factor) proton linear accelerator to study on basic researches and nuclear transmutation with spallation neutrons. As the first stage, the low energy accelerator up to 20 MeV is being developed in KTF (KOMAC Test Facility). The low energy accelerator consists of an injector, RFQ, CCDTL, and RF systems. The proton injector with duoplasmatron ion source has been developed, and the LEBT with solenoid lens is under installation. The RFQ linac that can accelerate a 20mA proton beam from 50keV to 3MeV has been designed and is being fabricated. The RF system for RFQ is being developed, and the CCDTL up to 20MeV is being designed. The status of the low energy accelerator will be presented.

1. INTRODUCTION

The KOMAC accelerator has been designed to accelerate a 20 mA cw proton/H^- with the final energy 1GeV cw by a super-conducting linac [1]. In the first stage of the project, we are a developing cw accelerating structure up to 20MeV, and operate the accelerator in pulse mode with 10% duty. And then, we will try to operate with cw mode. The 20MeV proton accelerator, KTF (KOMAC Test Facility), is constructing in the KTF (KOMAC Test Facility), and will be commissioned in 2003.

In KTF, we are developing the proton injector, LEBT, 3MeV RFQ, 20MeV CCDTL, and RF system. The proton injector was already developed, and the 3MeV RFQ will be constructed in this fiscal year. Also we have a plan to develop a basic super-conducting cavity technology for the second stage of KOMAC. Fig. 1 and Fig. 2 show the plan of KTF and the status of KTF respectively.

The status of the low energy accelerator developments in KTF will be introduced in this paper.

2. PROTON INJECTOR [2]

KOMAC requires the ion source with a proton beam current of 30 mA at the extraction voltage of 50 kV, for 20 mA proton beam at the final stage. Normalized rms emittance of less than 0.3 π mm⋅mrad is also required for good matching of ion beam into RFQ. The proton injector with a duoplasmatron ion source is shown in Fig. 2 (left side). The system is composed of a high voltage power supply, ion source power supplies in a high voltage deck, gas feeding system, and vacuum system.

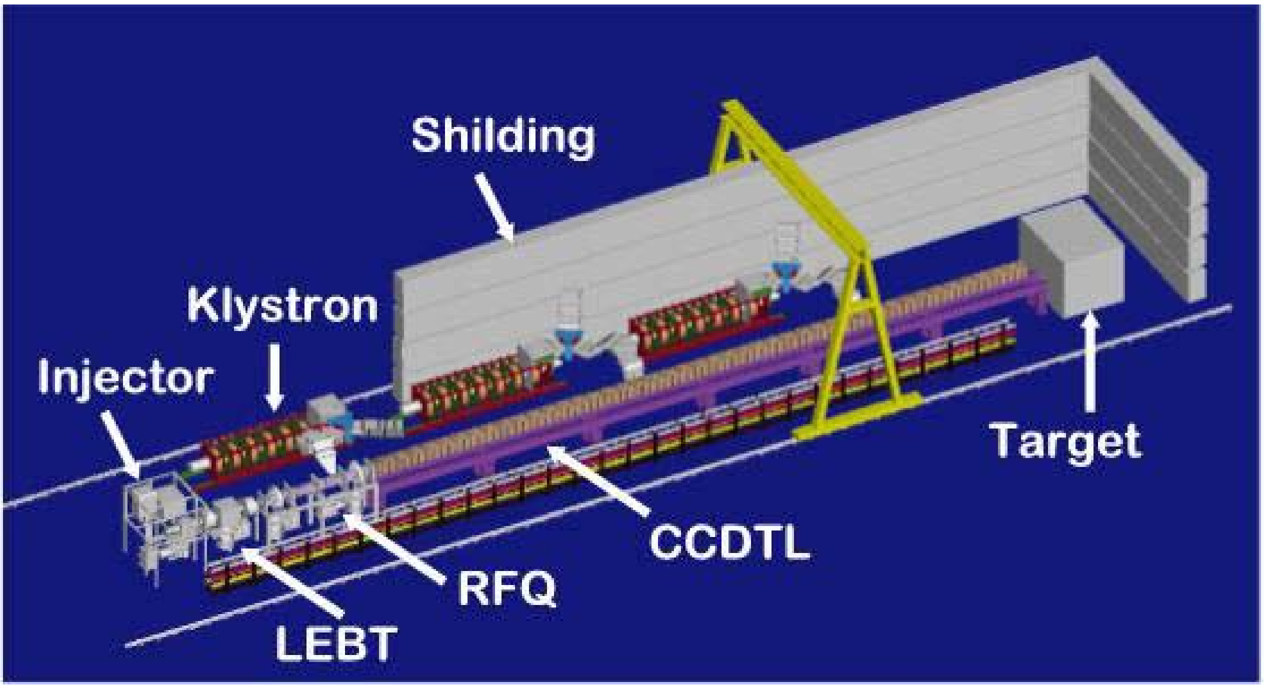

Figure 1. Plan of KTF 20MeV Accelerator.

Figure 2. Status of KTF 20MeV Accelerator.

The injector has reached beam currents of up to 50 mA at 50 kV extraction voltage with arc power of 150V 10A. The extracted beam has a normalized emittance of 0.2 π mm·mrad from 90 % beam current and proton fraction of over 80 %. The proton fraction is measured with deflection magnet and scanning wire.

The beam can be extracted without any fluctuation in beam current, with a high voltage arcing in 4 hours. The cathode lifetime is about 40hr. To increase the filament lifetime, it is necessary to lower the arc current.

3. LEBT

Low-energy beam transport (LEBT) consists of two solenoids, two steering magnets, diagnostic system, beam control system, and funnelling system to transports and matches the H^+ for 20mA and H^- for 3mA, beams from the ion source into the RFQ. The main goal of the LEBT design is to minimise beam losses. The calculation codes used are TRACE 3D and PARMTEQM. The PARMTEQM-simulated solenoid settings are B=2800G and B=3900G, the RFQ transmission rate is more than 90%. Two solenoid magnets constructed are 20.7cm-long, 16cm-i.d., are surrounded by a low carbon steel and provide dc fields ≤5000G on the axis. LEBT will be tested to obtain a proper matching condition with the RFQ.

4. RFQ [3]

The KTF RFQ bunches, focuses, and accelerates the 50keV H^+/H^- beams, and derives a 3.0MeV beam at its exit, bunched with a 350MHz. The RFQ is a 324cm-long, 4-vanes type, and consists of 56 tuners, 16 vacuum ports, 1 coupling plate, 4 rf drive couplers, 96 cooling passages, and 8 stabiliser rods. The RFQ is machined of OFH-Copper, integrate from separate four sections which are constructed by using vacuum furnace brazing. The RF system for the RFQ is operated with 350MHz at 100% duty-factor by one klystron of 1MW.

Its design was completed. In the RFQ design, a main issue is to accelerate the mixed H^+/H^- beam at the same time. The motion of the mixed H^+/H^- beam into the RFQ has been studied by using a time marching beam dynamics code QLASSI. The longitudinal beam loss increases with the concentration of negative ions by the bunching process, which is distributed by attractive forces when the ratio of H^- is more than 30%. Because of the space charge compensation in the low energy sections, the transverse beam loss decreases with the mixing ratio of H^-.

The average power of the RFQ cavity structure by rf thermal loads is 0.35 MW and the peak surface heat flux on the cavity wall is 0.13 MW/m^2 at the high energy end. In order to remove this heat, we consider 24 longitudinal coolant passages in each of the sections. In the design of the coolant passages, we have considered the thermal behaviour of the vane during CW operation and manufacturing costs. The thermal and structure analysis is studied with SUPERFISH and ANSYS codes. The temperature of the coolant passages on the cavity wall is varied to maintain the cavity on resonance frequency.

As a test bed for 3MeV RFQ, the design, construction, electrical test, and vacuum test of the 0.45MeV RFQ have been finished. Design of the RFQ was done by KAERI and POSTECH; a fabrication was done at Dae-Ung Engineering Company and VITZRO TECH Co., Ltd. One of the difficult processes in the fabrication of the RFQ was to braze. Because of the leak of the brazing surface and the strain of the RFQ structure by the furnace heat, it is important to determine an appropriate shape of the brazing area. To determine it, we had two brazing test. Fig. 3 shows a 96.4cm long 0.45MeV RFQ, which was brazed in a vacuum furnace. The RFQ was brazed in a vertical orientation with LUCAS BVag-8, AgCu alloy with a melting temperature of 780 ^{o}C. Testing of the brazed RFQ showed it to be leak-tight.

The coolant passages in the cavity wall and vane area were drilled with a deep hole. The entrances of deep holes at the vane end was brazed. The exact dimension of the undercut was determined empirically by cutting a vane of the hot model, which was fabricated of the OFHC. In the case of the RFQ with a modulated vane tip, the resonant frequency of the RFQ cavity linearly decreases with undercut depth. However, in the case of the RFQ cold model with a constant vane tip, the resonant frequency of the RFQ cavity non-linearly decreases with undercut depth.

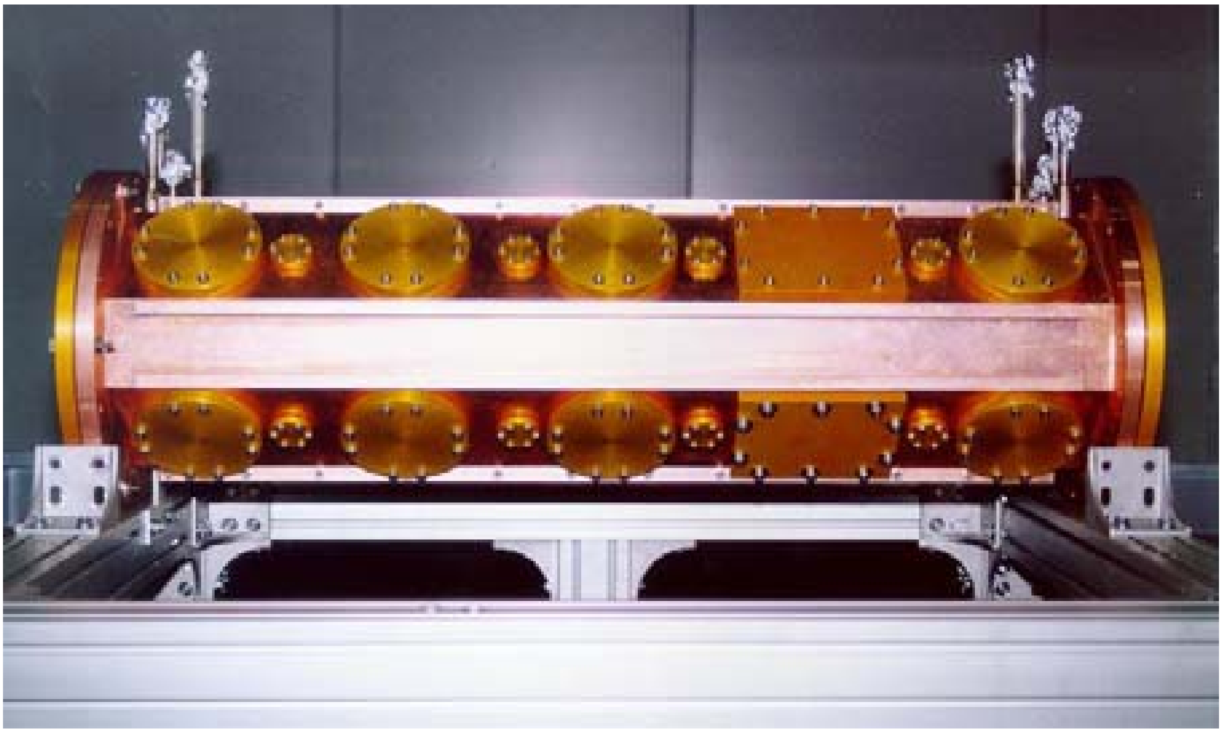

Figure 3. A brazed 0.45MeV RFQ.

The 3MeV RFQ cold model of aluminium was fabricated and tested. A low-level RF control system, which maintains proper amplitude (within ±1%) and phase (within ±1°), has been designed. A cold model of a tuner has been fabricated and is being tested. Assembly works of the 3MeV RFQ will be done in March 2001.

5. CCDTL [4]

CCDTL will accelerate the 3MeV 20mA proton beam to the energy of 20MeV. The structure design of CCDTL is based on the 100% duty factor.

The CCDTL cold models are fabricated to check the design, the tuning method, and the coupling coefficients and the fabrication method. The measured resonant frequency is 700.8 MHz without air and humidity compensation. The measured Q value of the cavity without brazing is 87% of the calculated Q by SUPERFISH without any surface cleaning. The super-drilled coolant path is well fabricated, and this type cooling method will be used for the CCDTL construction.

Table I: Major design parameters of CCDTL cavity

- Structure: 700MHz CCDTL
- Length: 25m
- Aperture Diameter: 10/15mm
- No of EMG: 130 (8 $\beta\lambda$ FODO)
- Total Structure Power: 1.15MW
- Structure Power per length: 50kW/m avg.
- Surface E: <0.9 Kilpatrick

The field profile is measured with a bead perturbation method. The field measurement system is shown in Fig. 4. A 2mm diameter and 2mm long alumina cylinder is used for the bead. The stepping motor drive system controls the position of the bead with an accuracy of 0.2mm. The frequency shift is measured with a network analyzer (HP4306A/85064A). Because the temperature controlled room is not available, the measurement was carried with the careful check of the unperturbed resonance frequency before and after the experiment.

Figure 5 shows a field profile measured in one cavity of the aluminium cold model. The measured field profile in a cavity agrees with the calculated profile. But, the field uniformity in the multi-cavity is not good. It is necessary to increase the field uniformity by the fine tuning of the cavity. This will be done with the brazed copper cold model that will be fabricated in this year. The copper model will be fabricated with the study. As a backup of the CCDTL, the design study for conventional DTL will be performed.

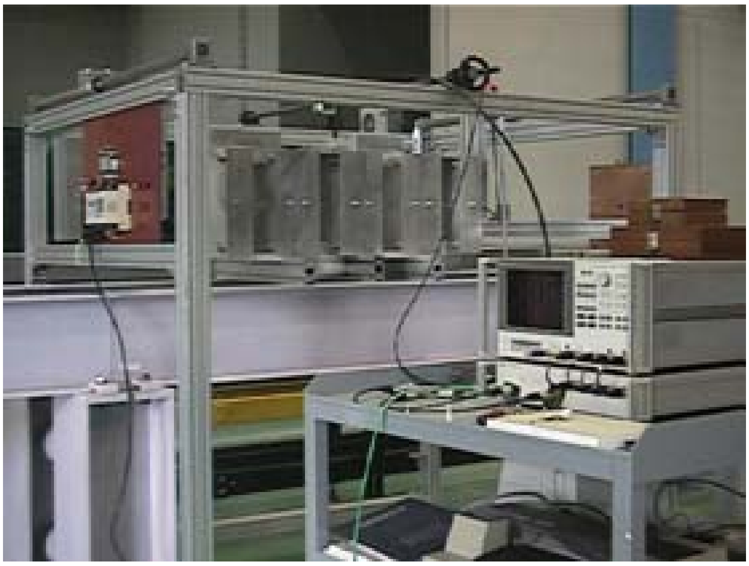

Figure 4. CCDTL Field Measurement.

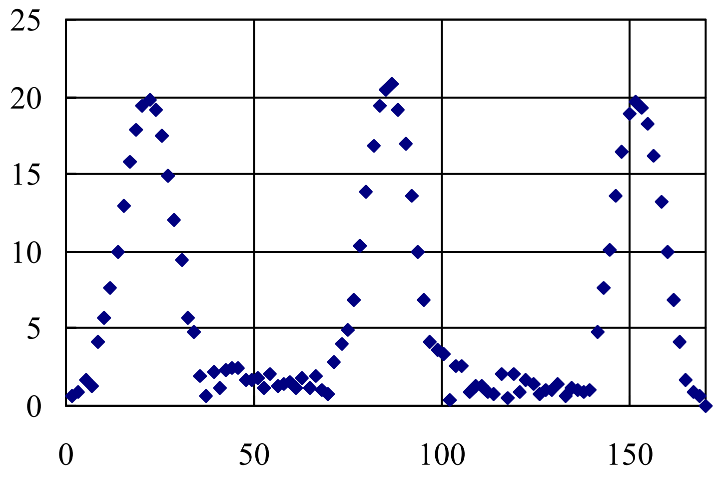

Figure 5. Measured field profile in one cavity (x: Position (mm), y: Field (Arb.)).

6. SUMMARY

The low energy proton accelerator for KTF is designed. The proton injector can provide the proton beam for RFQ. The RFQ is fabricating and will be tested with 1MW RF system. The CCDTL is studied with cold models, and the hot model will be fabricated.

ACKNOWLEDGEMENT

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REFERENCES

[1] CHOI, B. H., et. al., "Overview of the KOMAC Project", (Proc. of the 3rd International Conference ADTTA'99, 1999), (1999).

[2] CHO, Y. S., et. al., High-current ion source development for the Korea Multipurpose Accelerator Complex, Rev. Sci. Instr., **71**, (2000) 969.

[3] HAN, J. M., et. al., "A 3MeV KOMAC/KTF RFQ Linac", in this conference (2000).

[4] CHO, Y. S., et. al., "Cold Model Test of The KOMCA CCDTL Cavities", Proceeding of EPAC'2000, Vienna, (2000).

PLANNING OF ADS RELATED R&D IN THE RUSSIAN FEDERATION

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Abstract

ADS oriented R&D were initiated in Russia by ITEP in close cooperation with LANL in early nineties. In October 1998 the Scientific Council of the Ministry of Russian Federation for Atomic Power (MINATOM) decided to develop a Program of ADS related R&D coordinated and partly financed by MINATOM. Experimental investigations of ADS parameters include critical assembly in ITEP for precision measurements of neutron and kinetic parameters of subcritical multiplicating systems driven by pulsed neutron D-T generator; fast-thermal "neutron valved" blankets imitated at the BFS-1 facility (IPPE) coupled with microtron accelerator. BFS-1 and BFS-2 critical assemblies are used to investigate the transmutation of minor actinides (MA). Theoretical methods and related computer codes are developed for the calculations of physical and thermohydraulic parameters of ADS covering transition processes and emergency situations in thermal, fast and cascade (two-stage) multiplicating blankets; neutron yields and spectra for the targets of various geometries and structures irradiated by GeV protons and ions; heat release, beam moderation, gas production and radiation damages in the target and window beam materials.

1. INTRODUCTION

ADS oriented R&D were initiated in Russia by ITEP in close cooperation with LANL in early nineties. This activity was supported by the largest ISTC grant so far (Project 17, 3 million USD with more than 400 Russian scientists participating. In 1996-1997 a MINATOM Working Group prepared a conceptual ADS design for Russian-American Plutonium Committee.

In October 1998 the Scientific Council of the Ministry of Russian Federation for Atomic Power (MINATOM) decided to develop a Program of ADS related R&D coordinated and partly financed by MINATOM. The decision was approved and confirmed by Minister E. O. Adamov. A working group of the representatives of mainly MINATOM institutions was set up (Chairman O.V. Shvedov, ITEP, Moscow, Co-Chairman N. S. Rabotnov, IPPE, Obninsk) to prepare a Draft Program. The Draft was prepared and is now being considered by participating institutions and is to be presented shortly to the Ministry for approval.

Participating MINATOM institutions:

- SSC RF Institute of Theoretical and Experimental Physics (ITEP), Moscow.
- SSC RF A. I. Leipunsky Institute of Physics and Power Engineering (IPPE), Obninsk.
- Moscow Institute of Radiotechnology of the Russian Academy of Sciences (MRTI), Moscow.
- FNC All-Russian Institute of Experimental Physics (VNIIEF), Sarov.
- Experimental Design Bureau for Machine-Building (EDMB), Nizhny Novgorod.
- A. Bochvar All-Russian Research Institute for Inorganic Materials, (ARIIM) Moscow.
- "Hydropress" Pilot-Design Bureau (HYDROPRESS), Podolsk.
- Joint Institute of Nuclear Research (JNRI), Dubna

- Research and Design Institute for Power Engineering (RDIPE), Moscow.
- State Specialized Design Institute (SSDI), Moscow.
- "Khlopin Radium Institute" Research and Production Association (RI), St-Petersburg.
- All-Russian Planning and Design, Research and Technological Association (VNIPIET), Moscow.

2. STATUS AND SCOPE OF R&D RELATED RESEARCH IN RUSSIA

2.1. Experimental investigations of ADS parameters

- Critical assembly MAKET (ITEP) is used for precision measurements of neutron and kinetic parameters of subcritical multiplicating systems driven by pulsed neutron D-T generator.
- Fast-thermal "neutron valved" blankets are imitated of BFS-1 facility (IPPE) coupled with microtron accelerator. BFS-1 and BFS-2 critical assemblies are used to investigate the transmutation of minor actinides (MA).
- Experimental investigations of relevant neutron and proton induced nuclear reactions including transuranic fission for the energies up to 30 MeV are carried out on IPPE electrostatic accelerators.
- Nuclear processes in thick and thin targets of heavy materials irradiated by protons and ions with the energies from a few hundred to 9 GeV per nucléon are investigated on JTNR accelerators.
- ABV-F facility with two stage cascade fast-thermal blanket coupled to electron LU-50 accelerator generating neutrons by (gamma,n) reaction (VNIIEF) is used for the investigations of ADS neutronic and kinetics.
- Fission of transuranics is studied on the proton and neutron beams of PIYaF reactor and synchrocyclotron and on fast neutron time-of-flight beams of IBR-30 reactor (JCNR).
- Elemental and isotopic composition of PIYaF reactor fuel is investigated on the high resolution gamma-spectrometer.
- Fluctuations of the neutron multiplication coefficient of electron accelerator-driven IBR-30 reactor are studied in JINR
- Neutronic and thermohydraulics parameters of uranium-graphite assemblies and of tungsten water cooled targets irradiated by 500 MeV protons of linear accelerator are studied in INR RAS.

2.2. Theoretical studies and software development

Theoretical methods and related computer codes are developed for the calculations of:

- physical and thermohydraulic parameters of ADS covering transition processes and emergency situations in thermal, fast and cascade (two-stage) multiplicating blankets;
- neutron yields and spectra for the targets of various geometries and structures irradiated by GeV protons and ions;
- evolution of radionuclides inventory and resulting radioactivity;
- heat release, beam moderation, gas production and radiation damages in the target and window beam materials.

Data base MENDL-2 is developed of the evaluated cross sections of major nuclear reactions induced by neutrons with the energies up to 100 MeV and by protons up to 200 MeV on more than 500 stable and radioactive target nuclides. "Electronic handbook" of nuclear cross sections is under preparation containing the data for proton-, neutron- and pion-nuclear interactions in the energy range from 10 MeV to a few GeV.

2.3. Optimization of design and experimental imitation testing of ADS

Following topics are under development:

- key elements of pilot ADS-plants of low and medium power;
- demonstration liquid Pb-Bi cooled 1 MW target;
- thermohydraulics and physical chemistry of molten Pb-Bi and Pb, mass exchange problems on corrosion resistance of construction materials;
- thermohydraulics of Pb-Bi eutectics natural circulation;
- ADS control and safety systems;
- windowless beam path to the target;
- experiments on radiation endurance of the beam window.

2.4. The experiments on proton accelerator beams

Following problems are studied:

- heat release in the proton targets (IVFE, ITEP, JINR, PINP);
- the measurements of transuranic fission cross sections on time-of flight and moderation-in-lead spectrometers (INK RAS, IPPE, JINR, PINP;
- interactions of intermediate energy protons with the target and construction materials (ITEP, PINF);
- transmutation of transuranics and fission products by resonance neutrons adiabatically moderating in lead (INK RAS);
- cross sections measurements for target materials irradiated by intermediate energy neutrons produced by synchrotron proton beams (Uppsala 200 MeV; PINF 1 GeV, ITEP 3 GeV);
- radiation endurance of semiconductor materials, elements and devices for control and dosimetry systems of ADS.

2.5. Theoretical and experimental studies of advanced nuclear fuel cycles and technologies

• Development and economic assessment of MFC elements:
- high-level waste partitioning;
- reprocessing of transmutation products;
- management of short-lived radwaste;
- technological risks associated with radwaste storage.

• The efficiency of closed NFC:
- reduction of MA inventory in spent nuclear fuel;
- nonproliferation of fissioning materials and nuclear technologies;
- utilization of HEU and weapon grade Pu;
- transmutation of MA and reactor PU;
- extension of fuel resources;
- improved safety of nuclear power installations.

• U-Th fuel cycle:
- calculation of isotope kinetics and optimization of initial compositions of Pu-U-Th fuels;
- the analyses of blanket regimes ensuring constant reactivity;
- environmental and economic analyses of the problem of the accumulated U from Th;

• Development of the technologies of MOX fuel pins containing Th, U, Pu and MA;
• Conceptual development of subcritical cascade molten-salt reactor within CNFC.

The priorities formulated by the MINATOM Working Group for the development of low power experimental ADS

1. ITEP - Accelerator-driven neutron generator coupling heavy water TVR-reactor and ISTRA-36 proton linac.
2. JINR - reactor IREN coupled to 660 MeV proton synchrotron.
3. VNIIEF - ABV-F reactor coupled to high current electron linac LU-50.
4. INR RAS - a facility based on IN-06 neutron source of Moscow Meson Factory.
5. IHEP (Protvino), ITEP - multiplicating target on the RFQ accelerator.
6. MRTI, ITEP - cryogenic module of superconducting proton linac.
7. S-PbINP - GNEIS neutron source on 1 GeV synchrotron.

MINATOM funding limit for the Program's First Year (supposed to be 2000) is $3,000,000.

The top priority item (ITEP neutron generator) is getting money from diverse sources:
- MINATOM
- Ministry of Science and Technologies
- Russian Academy of Sciences
- Moscow City Government
- ISTC
- DOE Programs

MINATOM funds requested for the Program's First Year

	Topic		Funds, million rubles
1	assessment of the ADS potential role in nuclear power and nuclear technologies.		0.5
2	demonstration ADS facility		4.5
3	advanced nuclear fuel cycle		11.5
4	nuclear data and computer codes.		9.5
5	low power ADS.		47.5
	Total	Million rubles	73.5
		Million USD equivalent	3.0

3. SELECTED RESULTS

3.1. Accelerator development

Estimated parameters of three design options for ADS 1 GeV, 30 mA proton linac

Parameter	"Warm"	Supercond. 5 MeV/m	Supercond. 15 MeV/m
Proton energy, GeV	1	1	1
Beam power, MW	30	30	30
Length, m	1000	400	135
Resonator efficiency	0,4	1	1
Generator efficiency	0,65	0.65	0.65
HF power, MW	75	33	33
Grid power, MW	150	52	52
Accelerator efficiency	0,2	0,60	0,55
Cost of the accelerating system, $ mln	50	69	23
Cost of the generators, $ mln	125	49.5	49.5
Total cost of non-standard equipment, $ mln	275	120	72
Total cost of equipment, $ mln	313	179	109
Total cost of the accelerator	437	233	142

f = 600 MHz or 900 MHz

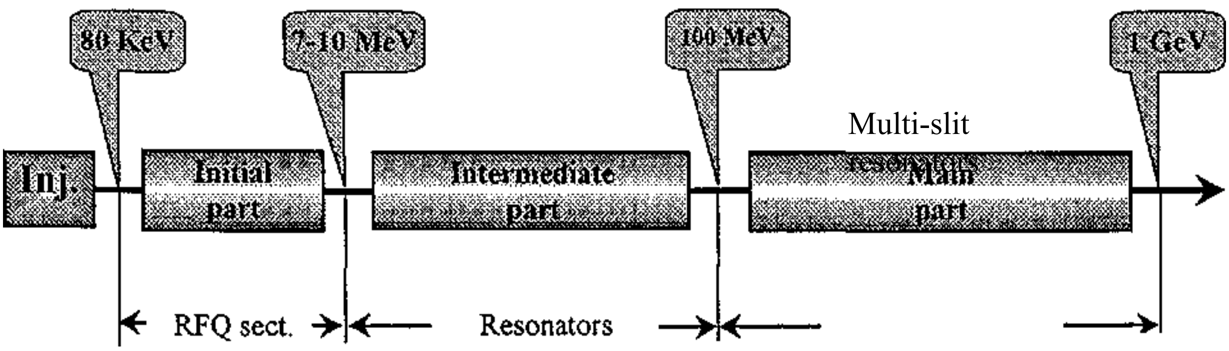


Figure 1. Accelerator's structure.

3.2. Proton target and beam window

Experiments and calculations carried out in IPPE resulted in determination of the coolant velocities distributions and temperature fields in the coolant and structural materials of 1 MW neutron generating target for operating, transitional and emergency regimes thus substantiating the target design in the framework of ISTC Project 559. Maximum temperature is reached in the most stressed element of the design - beam window. T_{max} = 450°C at the coolant velocity 1.5 m/s.

Calculations were also done of the window fatigue strength for the trips frequency 30/day and trips duration up to 1 min with a loss of plasticity under irradiation taken into account. Maximum stresses arise in the area of welding connections between different types of steel (ferrite- martensite steel of the window with the stainless steel of the target vessel).

Extensive calculations were done of the activation of the coolant by mixed proton and neutron irradiation. Following conclusions were made:

- The main contributors to the activity generated bot in Pb and Pb-Bi coolants are the radionuclides formed by spallation reactions, not Po isotopes.
- Total activities of Pb and Pb-Bi do not differ drastically (within a factor of two).
- Dominating contribution to the activity is due to spallation reactions induced by high energy part of neutron and proton spectra (>20 MeV), so most careful analysis of nuclear data in this energy range is needed.
- Fission products contribution to the activity is 10-15%.
- Tritium production rate is approximately equal for all the targets and its contribution to total activity is 50% for Pb target in the range of cooling times 3-30 years and about 25% for Pb-Bi target for the same cooling times.

Major topics to be investigated in 2000-2001:

- The development of the calculation methods and codes for the cross sections of spallation reactions induced by 20-2000 MeV protons: emission of clusters (d,t,alpha) responsible for gas production and radiation damage; high energy model of fission for the transition region from shell to drop model;
- The development of computer codes for the calculations of total and partial activities of the target and heat release by short-lived nuclides;
- Compilation of tested libraries of evaluated neutron and proton cross sections for the energies up to 2 GeV;
- Detailed experimental and theoretical studies of the beam window radiation and thermomechanical endurance aimed at realistic assessment of the window lifetime and at the ways of its prolongation;
- Precision experiments determining neutron fields on the surface of thick targets and nuclide yields for thin targets;
- Determination of thermohydraulic and mechanical parameters of the target including the deterioration of structural materials under continuous and interrupted irradiation.

3.3. ADS blankets

3.3.1. Conceptual design of fast-thermal solid fuel blanket

Inner fast neutron section is cooled by Pb-Bi at low pressure and is designed as MA burner. Outer part is heavy water moderated and cooled thermal neutron section to study ^{99}Tc and ^{129}I, transmutation and use Th-Pu MOX-fuel with americium component as burnable absorber. The whole system should work as "neutron valve" with low feed back and nearly factorized neutron multiplication allowing to obtain large total multiplication (k=0.99) at acceptable safety.

The parameters of fast-thermal partitioned blanket

Parameters	Units	Fast	Thermal
Thermal power	MW	300	2000
Coolant/moderator		Pb-Bi	Heavy water
Cooling method		Pumping	Pumping
Coolant flow	kg/s	7560	13900
Coolant pressure	MPa	-	11
Number of fuel subassemblies Number of control rods		18	684 24
Coolant velocity in the fuel	m/s	1.5	7.4
Core parameters: Height Diameter	m m	1.5 2.0	6.0 9.0
Power non-uniformity: Radial Axial		1.4 1.6	2.2 1.6
Maximum fuel temperature	°C	1610	494
Maximum outlet temperature	°C	630	mln
Moderator volume	m^3	-	160
Moderator pressure	MPa	-	0.1

3.3.2. Fast-thermal molten-salt blanket

Separate target and blanket create many interface problems so the idea to unite them looks promising and a unified molten-salt ADS system was proposed for development by a collaboration lead by Kurchatov Institute. It uses heavy mixture NaF (34 mol%)-PbF_2 (66 mol%) with melting point 498°C. It has following advantages:

- The salt serves as target material, base of the fuel composition and coolant which simplifies the design and servicing and improves safety;
- Eliminates a lot of structural materials from the core area;
- Simplifies chemical reprocessing;
- Neutron spectrum hard enough to burn MA in the central part.
- High neutron multiplication.

A 2GW(th) unit concept was developed. It has following potential features in various fuel cycles.

3.4. Advanced fuel cycles for ADS blankets

Plutonium cycle. 20 tons of Pu and MA are included in the cycle for as many years. 80% of Pu and 30% of Am are transmuted, with 500 kg of Cm and 1.5 kg of heavier actinides formed.

Neptunium transmutation cycle. Np itself is partially burned in that hard a spectrum and capture product ^{238}Pu is burned efficiently. 25 t of Np may be processed with transmutation coefficient of 0.985 for 40 years of the facility life-time. Such quantity of Np is produced by 40 PWR 1 GW(e) reactors during 40 years period.

Plutonium-thorium cycle. Following quantities of actinides may be processed during 40 years (transmutation coefficients in brackets): Th - 16 t (0.77); reactor Pu, Np and Am - 19 t total (0.7, 0.845, 0.23). An option may be realized in all those scenarios with Cm continuos extraction and placement into intermediate storage. Decay products $^{238, 240}Pu$ are returned to the cycle. Higher actinides (^{245}Cm and heavier) inventory is reduced a few times in this case. Two or three most promising fuel compositions should be selected for physical experiments on BFS (IPPE) and MAKET (ITEP) facilities.

3.5. Nuclear data. Experimental support

Intense development of intermediate energy physics is needed for the energy range 20-2000 MeV that is hundred times wider than reactor range. Experimental situation in this region is similar to reactor neutron data situation fifty years ago. Major figures are often known with the accuracy 10 times worse than needed. Typically results of different authors differ 2-3 times. Many reactions may not be studied experimentally at all so recent developments in theoretical models and computer codes are very important but the neutron sources of the next generation promise qualitative improvements in experimental situation.

3.5.1. Measurements on neutron and proton beams

ITEP. Main facility is U-10 proton synchrotron with two proton beams:

70-200 MeV, current 5×10^{10} protons/s.
800-2600 MeV, current 1.5×10^{9} protons/s.
The main technique is high-resolution gamma-rays spectroscopy. Thin targets: separated isotopes of W, Co, Cu, Tc, Th, natural uranium; thin targets: $^{27}A1$, ^{115}In, ^{197}Au, lead, lead bismuth eutectics, molten salt compositions.

Radium Institute. Experiments are prepared to measure fission cross sections of the separated isotopes of lead and neighbor elements of Bi and Tl in 20-200 MeV range. The results will be very important for the calculations of heavy metal coolant activation. Fission cross sections also will be measured in a wider energy range 30-3000 MeV for many nuclides from ^{181}Ta to ^{243}Am. Thin film counters and track detectors will be used.

PNPI. Main facility is synchrotron external beam with 1 mcA current ($6x10^{12}$ protons/s). A possibility exists of internal irradiations of targets by 3-5 mcA current in 10-1000 MeV energy range. A thick lead target on the external beam may be investigated by radiochemical methods. A technique was developed of the measurements with the energy varying due to beam moderation or by generation of secondary beams on thick targets. Priority tasks are the fission cross section measurements for natural lead, ^{209}Bi, ^{232}Th, $^{233,235,238}U$, ^{237}Np, ^{239}Pu from 200 to 1000 with 100 MeV steps and 5-10% accuracy. GNEIS time-of-flight neutron spectrometer will be used to measure the ratios of $^{233, 238}U$, ^{232}Th, ^{239}Pu, ^{237}Np, Pb and Bi cross sections to ^{235}U fission cross section in 1-200 MeV energy range.

JINR. A facility is prepared for the measurements on thick Pb and U targets irradiated by proton and ion beams with energies from a few hundred up to 10 GeV per nucleon. Low energy cross sections are measured on the neutron beams of IBR-30 reactor.

IPPE. Low mass actinide targets are used on quasimonochromatic neutron beams of electrostatic accelerators. Tandems present opportunity to enlarge the energy range up to 30 MeV. In the framework of ISTC Project 304 nuclear data for minor actinides were measured in 150 KeV-5 MeV energy range. Continuation of this work is planned for higher energies 5-30 MeV using new possibilities opened by EGP-15 tandem and heavier targets $^{241,242m,243}Am$ and $^{243,244,245,246,247,248}Cm$ made in VNIIEF by electromagnetic separation. Resonance neutron measurements will be done on Moscow Meson Factory neutron source (INR RAS).

3.5.2. Experiments on the accelerator driven subcritical assemblies

IPPE. Critical facility BFS-1 (2m diameter, 2.6 m height, 1200 tubes for materials pellets) is coupled a microtron (up to 30 MeV, 100 mA). Multiplied pulses may be studied on two flight paths (50 m and 250 m). The experiments are planned for two model blankets:
- Fast U-Pu nitride core cooled by lead;
- Fast test core surrounded by radial thermal or epithermal region.

The experiments will include: core mounting and measuring of the critical mass; measurements of the central ratios of fission and cross section ratios of various actinides to the fission cross section of ^{235}U; measurements of the central reactivity coefficients of actinides and structural materials; measurements of the reaction rates volume distributions and void and Doppler reactivity coefficients. ITEP.

The experiments will be done on the MAKET facility driven by D-T generator:

- Investigations of ^{235}U- ^{232}Th-heavy water uniform and non-uniform grids;
- ^{235}U-heavy water grids imitating the blanket of ITEP accelerator-driven neutron generator.

The objects of investigations: neutron spatial and energy distributions, reactivity coefficients, perturbations induced by heavy experimental equipment.

3.6. Low power experimental ADS

3.6.1. ITEP accelerator-driven neutron generator (ADNG)

ADNG is aimed at:

- development and testing of safe operation regimes for transmutation ADS;
- precision measurements of neutron functional and subsequent computer codes testing;
- fundamental research;
- testing of the elements of high current accelerators;
- production of the isotope emitters for electron-positron tomography;
- demonstration of the use of decommissioned experimental reactor. Major safety features:
- fixed level of subcriticality;
- low burn up (1% in 3-4 years) resulting in low inventory of fission fragments;
- blanket design excludes coolant's flow or excessive residual heat;
- low specific power prevents fuel pins damage.

3.6.1.1. The design and basic parameters

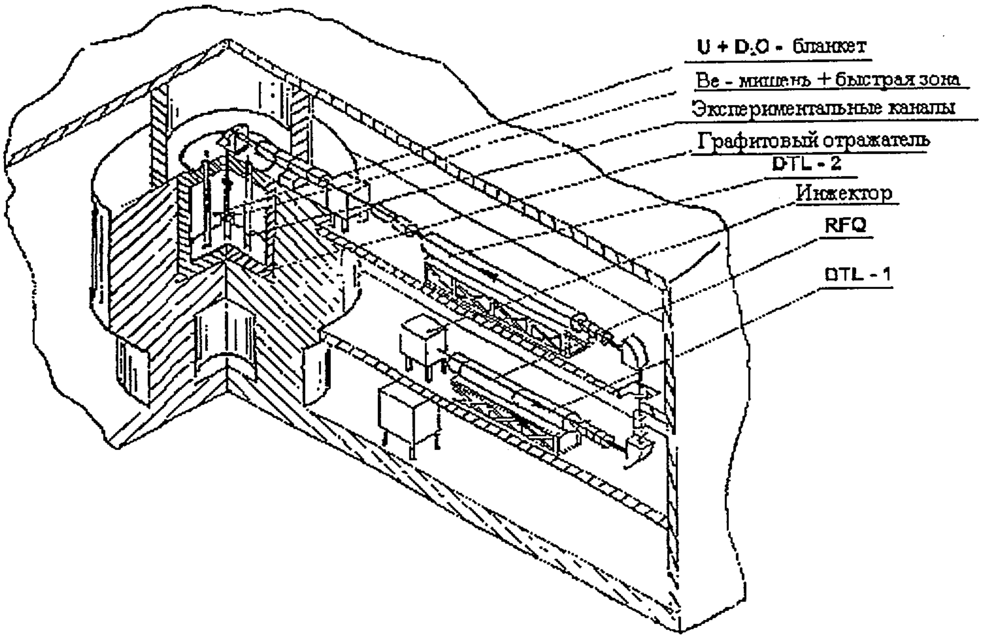


Figure 2. Outlay of ITEP accelerator-driven neutron generator.

- Proton energy	36 MeV;
- Proton pulse duration	150 mcs;
- Average proton current	0,5 mA;
- Fast neutrons intensity on the target	$1{,}5\times10^{14}$ n/s;
-K_{eff}	0,95 - 0.98;
- Power	100 KW
- Thermal neutron flux in the reflector	$1{,}5\times10^{12}$ /cm^{2}s;
- Fuel mass	^{235}U(20 %.) 1,5 кг;

2.6.3. INR RAS demonstration ADS transmutation facility

INR RAS neutron facility is based on an accelerator of protons and H ions, planned current 500 mcA and consists of a few neutron sources and beam catcher. It has biological radiation shielding, water cooling system and to neutron target boxes (height 4 m, lower diameter 1.6 m). Intermediate storage for irradiated components is located in the shielding.

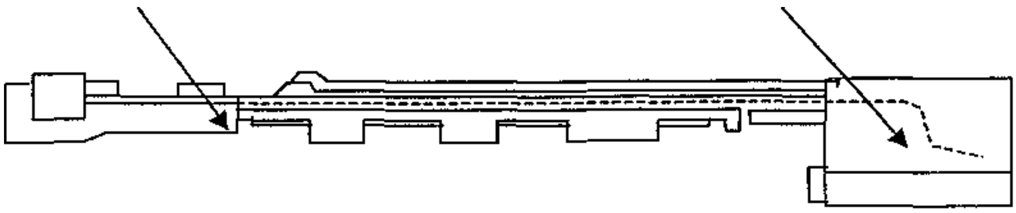
Intense neutron pulsed source in the first box will be used for the experiments on solid state physics and nuclear physics. Physical start up was in 1998.

Second box is planned to use for construction of ADS demonstration facility.

Beam catcher has a channel for irradiation of samples in mixed proton-neutron fluxes.

3.6.1.2. Construction schedule and funding (mln rubles)

		2000	2001	2002	Total
1.	Completion of design. Development of non-standard equipment.	12.0	1.0		13.0
2.	Purchase of standard equipment. Making of non-standard equipment and elements. Elements tests.	30.0	30.0	20.0	80.0
3.	Construction works	3.0	15.0		18.0
4.	Equipment assembly and adjusting		20.0	10.0	30.0
5.	Final adjustment. Physical start up.			10.0	10.0
Total		45.0	66.0	40.0	151.0



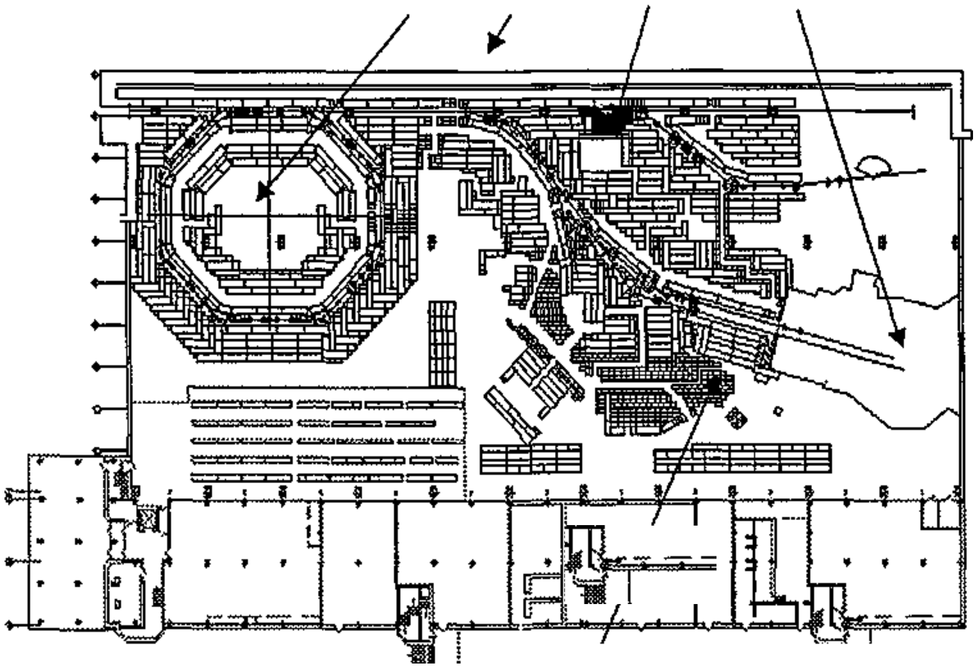


Fig.3. Proton linac and experimental building of Moscow Meson Factory. 1- accelerator, 2 - experimental building, 3 — storage ring, 4 — neutron sources, 5 — beam catcher, 6 — time-of-moderation-in-lead spectrometer.

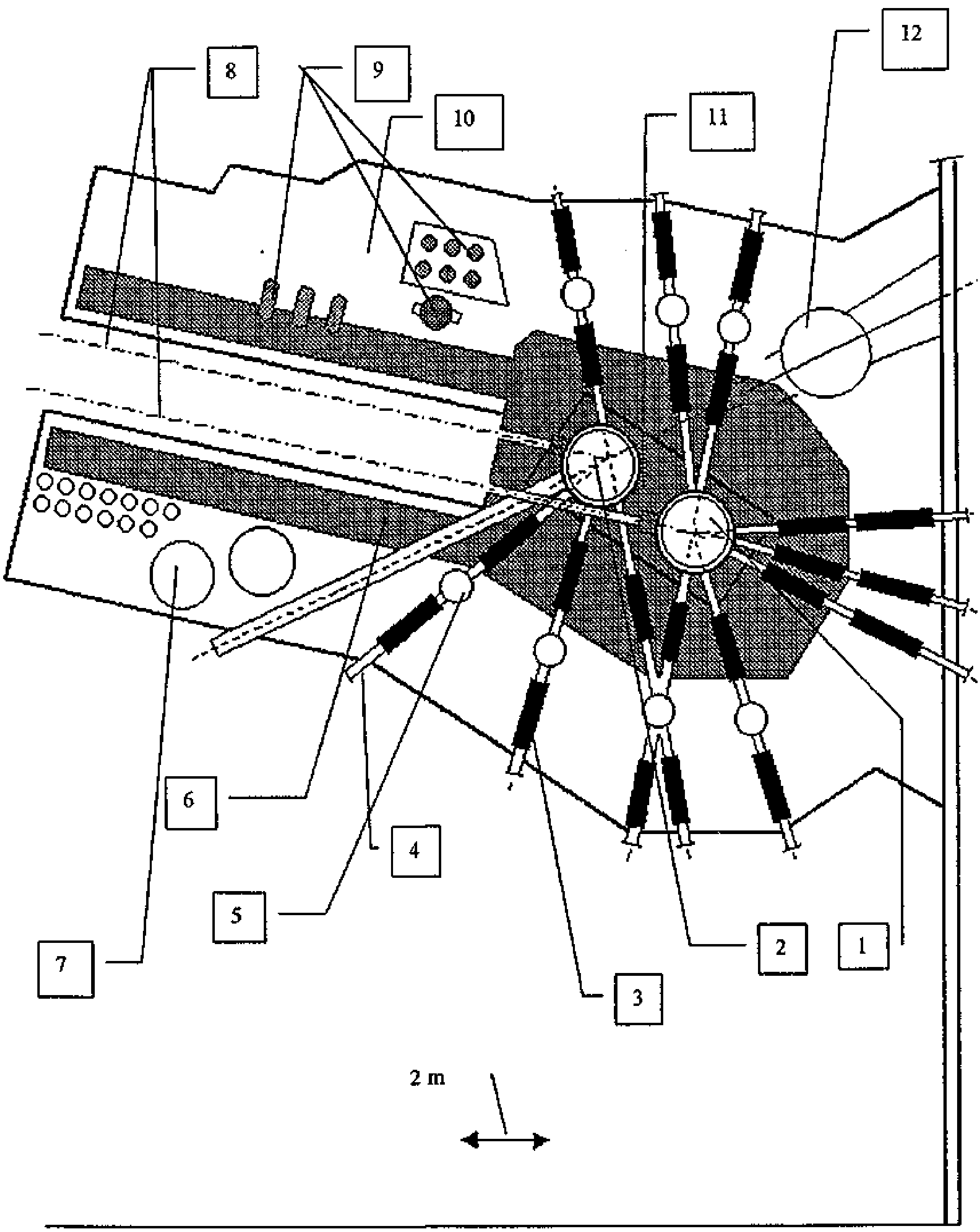


Figure 4. Moscow Meson Factory neutron sources. 1 - demonstration ADS box, 2 - pulse neutron source box, 3 - neutron beam windows, 4 - neutron channels, 5 - vertical channels for additional experimental equipment, 6 — steel shielding, 7 - storage of irradiated elements, 8 - proton beams, 9 - primary cooling circuit, 10 - concrete shielding, 11 - heat shielding, 12 - wide channel.

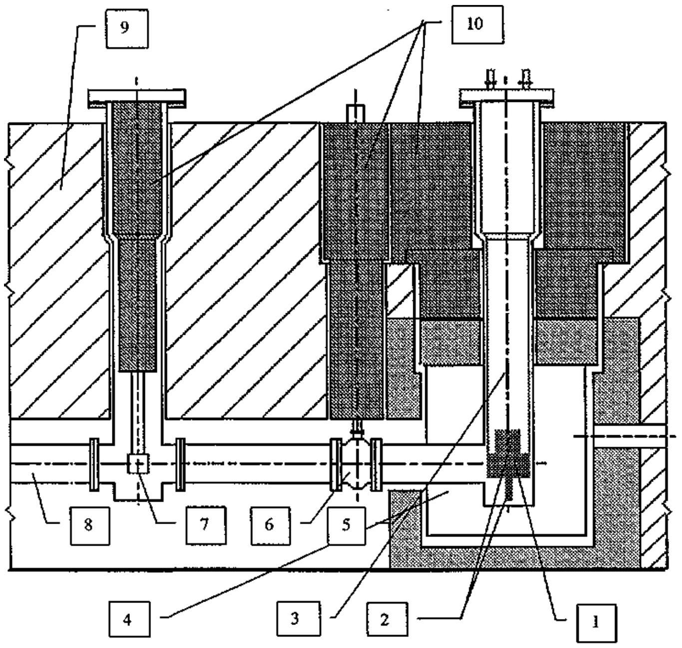


Figure 5. Outlay of the pulse neutron source: 1 - core 2 - moderator, 3 - core ampoule, 4 - gas tank, 5 - thermal shielding, 6 — remotely controlled sealing, 7 — beam position sensor, 8 — ion channel, 9 - steel shielding, 10 - steel plugs.

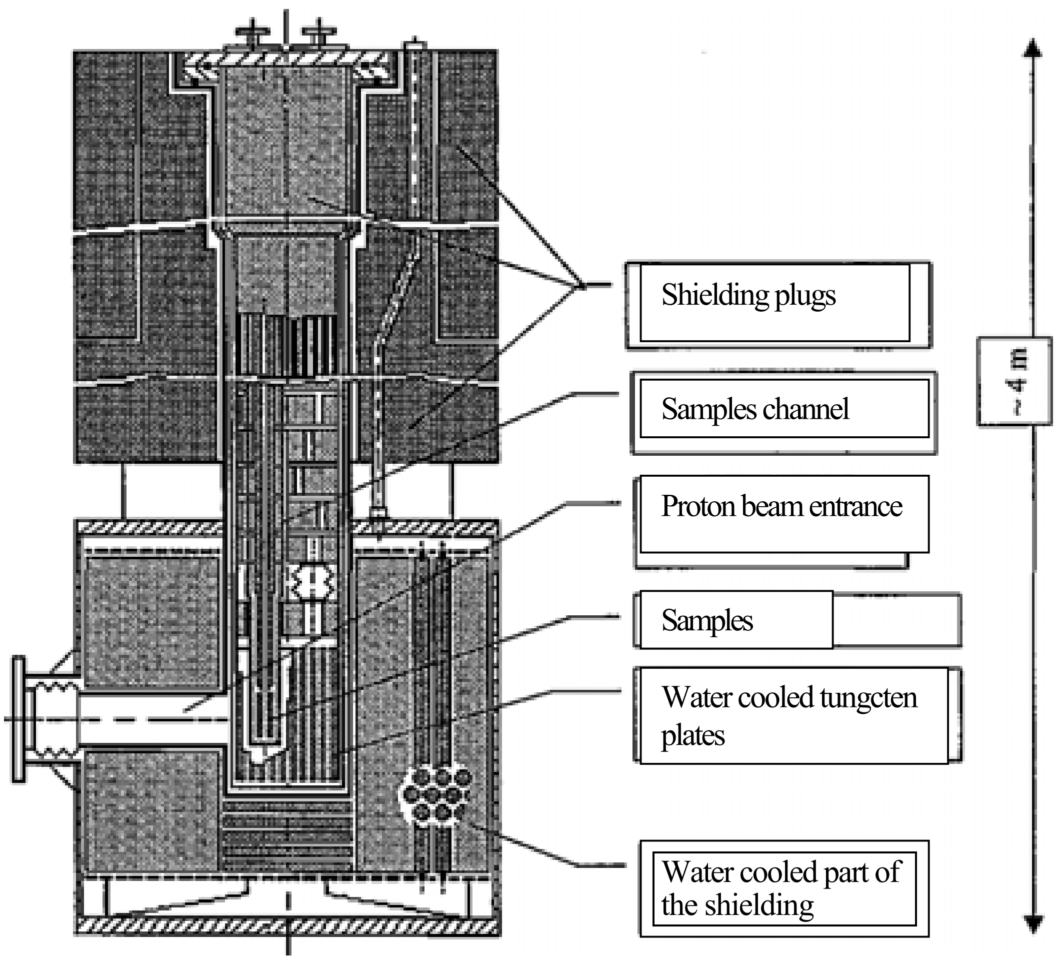


Figure 6. Beam catcher.

CONCLUSIONS

1. The priorities formulated by the MINATOM Working Group for the development of low power experimental ADS
2. ITEP Accelerator-driven neutron generator coupling heavy water TVR-reactor and ISTRA-36 proton linac.
3. JINR - reactor IREN coupled to 660 MeV proton synchrotron.
4. VNIIEF - ABV-F reactor coupled to high current electron linac LU-50.
5. INR RAS - a facility based on IN-06 neutron source of Moscow Meson Factory.
6. IHEP (Protvino), ITEP - multiplicating target on the RFQ accelerator.
7. MRTI, ITEP - cryogenic module of superconducting proton linac.
8. S-PbINP - GNEIS neutron source on 1 GeV synchrotron.
   MINATOM funding limit for the Program's First Year (supposed to be 2000) is equivalent to $3,000,000.

The top priority item (ITEP neutron generator) is getting money from diverse sources:

- MINATOM
- Ministry of Science and Technologies
- Russian Academy of Sciences
- Moscow City Government
- ISTC
- DOE Programs MINATOM funds requested for the Program's First Year.

	Topic		Mln rubles
1	Assessment of the ADS potential role in nuclear power and nuclear technologies.		0.5
2	Demonstration ADS facility		4.5
3	Advanced nuclear fuel cycle		11.5
4	Nuclear data and computer codes.		9.5
5	Low power ADS.		47.5
	Total	Million rubles	73.5
		Million USD equivalent	3.0

Supplement l
Data base on Experimental Facilities for ADS Research

Under a contract with IAEA an electronic version of the Database on Experimental Facilities for ADS Research was developed in IPPE and is now available on CD from the IAEA.

SUMMARY: "A ROADMAP FOR DEVELOPING ACCELERATOR TRANSMUTATION OF WASTE (ATW) TECHNOLOGY". A REPORT TO CONGRESS

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Abstract

The U.S. Congressional Conference Report accompanying the Fiscal Year 1999 Energy and Water Development Appropriation Act directed the U.S. Department of Energy, through its Office of Civilian Radioactive Waste Management, to conduct a study of accelerator transmutation of waste (ATW). It was transmitted to the U.S. Congress on November 1, 1999. The Report to Congress made it clear that the U.S. Administration, in transmitting the report, was not taking a position either way on those recommendations. If an ATW program were to be undertaken in the U.S., the pace and funding would have to be evaluated and planned in light of the currently unproven technologies involved, the potential benefits, and overall Government budget priorities.

1. INTRODUCTION

The U.S. Congressional Conference Report accompanying the Fiscal Year 1999 Energy and Water Development Appropriation Act directed the U.S. Department of Energy, through its Office of Civilian Radioactive Waste Management, to conduct a study of accelerator transmutation of waste (ATW). It was transmitted to the U.S. Congress on November 1, 1999, a few days before the IAEA AGM meeting in Taejon, Republic of Korea on national R&D programs on Accelerator Driven Systems (ADS). With the public release of the Roadmap Report, it was possible to summarize the results of the study and research program recommended by the study's Steering Committee. The Report to Congress made it clear that the U.S. Administration, in transmitting the report, was not taking a position either way on those recommendations. If an ATW program were to be undertaken in the U.S., the pace and funding would have to be evaluated and planned in light of the currently unproven technologies involved, the potential benefits, and overall Government budget priorities. In this regard, it should be noted that the Fiscal Year 2000 Energy and Water Development Appropriation Act provided the U.S. Department of Energy with $9 million to continue the program described in the Roadmap. What if any funding might be provided in the future remains to be seen.

2. THE ESTABLISHMENT AND INITIAL EFFORTS OF THE ATW ROAD MAP STEERING COMMITTEE

The preparation of the ATW Roadmap, including responses to the specific requests contained in the Congressional conference report, was assigned to a ten-person ATW Roadmap Steering Committee. The committee membership included representatives of four DOE program offices having specific responsibilities in technical areas associated with ATW; representatives of the directors of three national laboratories: Argonne National Laboratory (ANL), Brookhaven National Laboratory (BNL), and Los Alamos National Laboratory (LANL); and three individuals informally recommended by the National Academy of

Sciences. The following individuals were selected as members of the ATW Steering Committee:

Edward Arthur, Director's Representative, Los Alamos National Laboratory
William Bishop, Office of Defense Programs, DOE (since retired)
James Bresee, Office of Civilian Radioactive Waste Management, DOE, Chairman
Leslie Burris, Argonne National Laboratory (retired)
David Goodwin, Office of Science, DOE
John Herczeg, Office of Nuclear Energy, Science and Technology, DOE
Darleane Hoffman, Lawrence Berkeley National Laboratory (retired)
Michael Todosow, Director's Representative, Brookhaven National Laboratory
David Wade, Director's Representative, Argonne National Laboratory
Carl Walter, Lawrence Livermore National Laboratory

The Steering Committee first acknowledged the ongoing international R&D programs on accelerator-driven systems (ADS), requested advice from world experts and was able to obtain very useful recommendations from them for the direction of the Roadmap. The following foreign experts provided individual guidance either at the beginning of the Roadmap study or after reviewing an early draft of the report or both:
Dr. Vladimir Chitaykin, Institute of Power Engineering, Russia
Dr. Giuseppe Gherardi, ENEA, Italy
Dr. Waclaw Gudowski, Royal Institute of Technology, Sweden
Dr. Stefano Monti, ENEA, Italy
Dr. Takehiko Mukaiyama, JAERI, Japan
Dr. Guannadi Petrov, St. Petersburg Institute of Technology, Russia
Dr. Carlo Rubbia, ENEA, Italy
Dr. Massimo Salvatores, CEA, France

In addition, the following world experts from the United States also provided significant input to the planning of the program:

Dr. Charles Bowman, ADNA Corporation, Los Alamos, New Mexico
Dr. Gregory Choppin, Florida State University, Tallahassee, Florida
Dr. Mujid Kazimi, MIT, Cambridge, Massachusetts
Dr. George Lawrence, LANL, Los Alamos, New Mexico
Dr. Gregory Van Tuyle, LANL, Los Alamos, New Mexico

Although these world experts had considerable influence on the direction of the Roadmap program, the content of the ATW Roadmap, including its conclusions and recommendations, are the responsibility of the Steering Committee.

3. CONCLUSIONS AND RECOMMENDATIONS OF THE ATW STEERING COMMITTEE

After approximately six months of analysis and evaluation, the first conclusion reached by the Steering Committee was that ATW will not eliminate the need in the United States for a high-level nuclear waste repository. Such a repository will be needed for the disposal of defense waste such as the vitrified "logs" being prepared at DOE's Savannah River plant and also for the residual high level waste from ATW, if implemented. ATW would provide no basis for interruption of the current efforts to characterize the Yucca Mountain site in Nevada as a possible high level nuclear waste repository. Further, the design concept for the proposed Yucca Mountain repository would provide access to the waste for a hundred or more years after it is filled in case an advanced technology such as ATW were to become technically and economically feasible. Thus, the two programs are complementary.

The second conclusion was that ATW offers the potential to reduce the contribution, at least, of civilian spent nuclear fuels to the dose predicted by the total system performance assessment (TSPA) for the Yucca Mountain project. Applicability of ATW to other high level nuclear waste types in the United States will require further analysis.

The Total System Performance Assessment (TSPA) done for a 1998 Yucca Mountain Viability Assessment estimated the potential exposure to radioactivity of those who may be living along the natural flow path of an underground water reservoir deep beneath Yucca Mountain. The nearest potential exposures would be to farmers in Amargosa Valley, about 20 miles south of Yucca Mountain. The Roadmap shows that ATW has the potential to reduce such exposures. By how much and at what cost will only become clear after additional ATW R&D. Due to the removal of decontaminated uranium from the civilian spent fuel, the ultimate waste volumes may be somewhat lower than that of the initial waste.

The third conclusion was that operation of an ATW prototype or demonstration unit might take a long as 20 years to implement and cost as much as $11 billion. Before such investments, public acceptance must be addressed as well as other complex institutional issues.

The fourth conclusion was that full implementation of ATW in the United States for treating civilian spent nuclear fuel would require several decades and could cost hundreds of billions of dollars, but the scenarios used to estimate such future activities have great uncertainties. The potential energy recovery from the transmutation of actinides in the waste could be significant (up to 25% of the initial energy recovery) and power sales could potentially offset much of the transmutation costs.

The fifth conclusion was that ATW development in the U.S. could provide significant auxiliary benefits, among which would be a potential for greater influence in international efforts in the areas of nuclear non-proliferation and waste management. In a related sixth conclusion, the Committee felt that collaboration with current international efforts to develop accelerator driven systems could reduce the schedule and cost of the U.S. program compared with an independent effort. Such collaboration could have beneficial impacts on the foreign programs as well and offer opportunities for shared use of research facilities and, assuming success in R&D, even the possibility of joint demonstration facilities. The number of applicable research facilities throughout the world has been gradually reduced, and making optimum use of those that remain would be important to the international success of ATW/ADS.

The final conclusion of the Steering Committee was that, although there do not appear to be any technological "show stoppers" (known scientific or engineering barriers which would prevent the implementation of ATW) an initial six-year, science-based research program would be a prudent investment in the United States for the proper evaluation of the possible future role of accelerator driven systems. It would also offer advantages in related scientific and engineering areas such a materials science and high-powered accelerators.

The ATW Steering Committee made several recommendations for future actions. However, the Roadmap made clear that these recommendations did not reflect official U.S. policy with regard to accelerator transmutation of waste. The first and second recommendations expanded on the final conclusion. The initial two years of the six-year effort should be devoted to trade-off studies which would allow a more careful evaluation of ATW's potential role in nuclear non-proliferation and waste management as well as the potential costs and efficiencies associated with the various technical options for an ATW system configuration. Examples of technical options are cyclotrons vs. linear accelerators; aqueous vs. high temperature chemical processes; helium, sodium or lead-bismuth eutectic cooling; solid vs. molten spallation targets; and homogeneous vs. heterogeneous transmutation targets. Various neutron energies are possible in a transmutation system. Minor actinide fission is favored by fast neutron bombardment while fissile elements and long-lived fission products can be transmuted more efficiently in an epithermal or thermal neutron flux. Even though some experimental studies can be effectively carried out during the first two years, initial analytic studies are needed to focus the subsequent R&D program on the most productive avenues of research. Thus the trade-off studies and the experimental program should be mutually supportive, with experimental emphasis placed on the areas identified as the most feasible and potentially cost-effective technology options.

The third recommendation was that a strong international collaborative program should be established with countries having existing ADS activities in areas of mutual interest. The French and Japanese programs are the largest national programs in the world, and there are many areas of potential collaboration with each country. In addition, there is a multinational effort in Europe headed by Carlo Rubbia in which ten countries (Austria, Belgium, Finland, France, Germany, Italy, Spain, Sweden and the U.K.) are jointly investigating at "Energy Amplifier" concept proposed several years ago by Professor Rubbia. He chairs the Technical Working Group made up of scientific and technical representatives for the member countries. Their efforts have received ADS funding from the European Community through the EC Framework Program, which has been matched by equal funding from the supported countries. The involvement of European industry in the Rubbia effort adds some complications regarding intellectual property rights, but similar problems have been solved in the past in other technical areas.

Another opportunity for collaboration is through the International Science and Technology Center which sponsors R&D in the Russian Federation, some of which is directly applicable to a future U.S. ATW program. The best example is ISTC project #559, under which funding from the EC, France, Sweden and the U.S. is supporting the fabrication of a lead-bismuth eutectic (LBE) spallation target for experimental testing in the LANSCE proton accelerator at LANL. The use of LBE (with a melting point of 125 degrees Celsius and a boiling point above 1500 degrees Celsius) as a reactor coolant was pioneered by the USSR in nuclear submarines. Difficult steel corrosion problems were solved by careful control of the oxygen content of the circulating molten metal. LBE also is a potentially superior spallation target since heat removal by circulation provides advantages over solid spallation targets such as tungsten which require a separate cooling loop. Potential disadvantages of LBE include to formation of long-live radioactive lead and bismuth isotopes (lead-205 with a 15 million year half-life and bismuth-208 and 210 with 368,000 and 3 million year half-lives, respectively) as well as polonium-210, a strong alpha emitter with a 138 day half-life and a spleen-seeker if ingested.

The LBE loop is currently being fabricated at the Institute of Physics and Power Engineering, Obninsk, Russia and may be available for testing at LANL before a similar experiment is carried out at PSI in Switzerland.

Other Russian R&D efforts in ADS include investigations of molten salt as a transmutation target, capable of circulation for heat removal and potentially capable of continuous chemical treatment. The molten salt work in Russia, the Czech Republic and other countries is based in part on the successful operation of the Molten Salt Reactor (MSR) at ORNL in the 1960's. That effort was aimed at developing a thermal breeder based on the thorium-232/uranium-233 fuel cycle. It was not continued when the emphasis in the U.S. was focused on a sodium-cooled fast breeder reactor. However, the Energy Amplifier ADS concept of Carlos Rubbia is again considering the thorium/uranium cycle as part of a nuclear energy system to reduce the formation of the higher actinides.

It was the final recommendation of the ATW Steering Committee that a total funding of approximately $281 million over a six-year period be provided for a science-based R&D program. It should be divided roughly as follows: nine per cent for system studies, twenty-one per cent for accelerator development, twenty per cent for separations and waste form R&D, forty-four per cent for target and blanket studies and nine per cent for program management. At the end of the program, the following deliverables would be produced:

- A reference design of an ATW system at the preconceptual design level,

- A description of the status of ongoing science and technology efforts, highlighting major interim results and remaining technical needs,

- A development and demonstration plan for ATW, including plans for use of existing U.S. and/or international test facilities and the identification of demonstration facilities to be constructed,

- A preliminary cost estimate of the program costs through demonstration, and

- An institutional analysis identifying strategies for addressing and dealing constructively with issues such as regulation and public acceptance.

The final ATW Roadmap report has been made publicly available through the distribution of printed copies and through the Internet. It has also provided the initial basis for current ATW research efforts in the U.S.

HIGH POWER ACCELERATOR DEVELOPMENTS AT THE RUTHERFORD APPLETON LABORATORY

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Abstract

The Rutherford Appleton Laboratory currently operates a pulsed 200 μA, 800 MeV rapid cycling proton synchrotron accelerator. The laboratory is actively engaged in research and development of accelerator systems for high power pulsed and CW machines. In particular, the Laboratory is involved in the design of accelerators and compressor rings for the European Spallation Source (ESS). This is a pulsed (50 Hz) source of neutrons for material science research with an average power of 5 MW at an energy of 1.33 GeV. The R&D for this project includes the design and production of a prototype of the low energy acceleration section. This prototype consists of (sequentially) a high intensity negative ion source (H^-), a low energy beam transport system (LEBT), a 280 MHz radio frequency quadrupole accelerator (RFQ) and a drift tube linear accelerator (DTL). The prototype will produce a chopped beam of 60 mA at an energy of 20 MeV. With the completion of a second (identical) system, the technologies of funneling - where beams from the two systems can be combined in an RF structure - will be studied and prototyping undertaken. To accomplish this scheme, the technology of beam chopping is also being studied and will be part of the R&D prototype system. Although the project is at an early stage, a pre-prototype RFQ accelerator is being constricted, primarily as an upgrade to ISIS, but also to test design ideas appropriate to the design of the higher power RFQ, for which engineering and thermal studies are currently underway.

Considerable expertise has been gained in the study and tracking of particles in high intensity beams. In particular, the loss mechanisms, which are of great importance in high power machines are being studied.

Currently studies are beginning to take place for designs of the driver of the neutrino factory being considered for Europe.

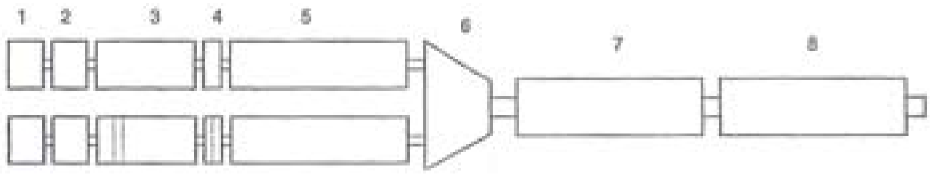
1. INTRODUCTION

The Rutherford Appleton Laboratory operates a pulsed 160 kW proton beam at 800 MeV for the production of spallation neutrons for condensed matter research (the ISIS facility [1]). High power accelerators are becoming important in a number of diverse fields. Of particular interest to ISIS is the planned European Spallation Source accelerator [2] to which the laboratory is an active contributor both in accelerator design, and to the R&D programme. The ESS accelerator, outlined in Fig. 1, is a pulsed machine with an average power of 5 MW, and consists of a pulsed linear accelerator system feeding a pair of accumulator rings. Machines like the ESS accelerator are likely to become the next practical step in high power accelerator construction. They are however, still of low power compared to the machines being considered today for transmutation and energy production, but the ESS and its like are important milestones along the path to these higher power machines. Table 1 illustrates the present state of proven technology.

There are particular features that must be addressed in all high power machines (including ISIS) connected with safety, reliability and low beam loss in absolute terms. These points are pertinent to the ESS and to ADS, and are discussed below.

TABLE 1. A REVIEW OF OPERATING HIGH POWER PROTON ACCELERATORS

Accelerator	Machine type	Beam energy [MeV]	Final (average) current [mA]	Beam power [kW]	Duty cycle
MMFL (Russia)	Linac	423	0.16	68	0.008 100 Hz
TRIUMF (Canada)	Cyclotron	520	0.2	104	CW
ISIS (UK)	Synchotron	800	0.2	160	2.25×10^{-5} 50 Hz
PSI (Switzerland)	Cyclotron	590	1.5	885	CW
LANSCE (USA)	Linac	800	1.7	1300	0.1 120 Hz
LEDA (USA)	Linac	7	100	700	CW



Legend:

1	High intensity H^- source	50 keV	75 mA
2	Low Energy Beam Transport (LBET) system		
3	280 MHz Radio Frequency Quadrupole Accelerator	2.5 MeV	60 mA
4	Fast Chopper		
5	280 MHz Drift Tube Linac (DTL)	20 MeV	60 mA
6	Funnel		20×60 mA
7	560 MHz Coupled Cavity DTL	100 MeV	
8	560 MHz Coupled Cavity Linac	1334 MeV	107 mA

FIG. 1. Schematic of the present ESS accelerator design.

2. R&D TOWARDS THE ESS ACCELERATOR

An R&D programme to tackle important design issues for the front end of the ESS (E<20 MeV) is currently underway at the laboratory. The goal of the R&D programme is the eventual construction of a low energy high intensity accelerator (elements 1-6 as shown in Fig. 1).

The accelerator design shown has evolved from the original reference design [2]. Detailed studies of the implementation of the original design showed problems in the funnel and with the chopping of the beams, indicating that not all design choices and aspects of high power machines are clear cut, since they are largely out of the experience of accelerator builders and only in the codes of the designers.

The programme involves the design of a negative hydrogen ion source with a sufficiently high current (~70 mA) beam quality and lifetime. A low energy beam transport system to couple the source to the first stage of acceleration is required. This latter is a critical piece since the neutralization of the beam in magnetic solenoid systems is not well studied at high power. The initial acceleration of the beam is made in a 280 MHz RFQ to accelerate the beam to 2.5 MeV. This device is designed for intense pulsed operation for the ESS, however, the development of this device to long pulse duty cycles is seen as advantageous to ADS systems. The pulsed nature of the machine and the injection of the beam into compressor rings (a characteristic of pulsed neutron sources) demands that no beam should be present while the system is not able to properly accumulate it. Such current must be removed as soon as possible and a beam chopper is introduced after the RFQ structure and before the drift tube linac. Finally, a drift tube linac structure could be built to provide a 20 MeV, 60 mA beam (1.2 MW) for the eventual testing of a beam funnel. The funnel is a complex radio-frequency device in itself. However, its presence is useful from the point of view of easing the initial stages of acceleration (space charge effects) and from. a redundancy point of view - since two ion sources are available.

The current R&D scheme for a pulsed system could be extended to longer duty cycles. Such a machine would represent a 140 MW accelerator if operated in a CW mode.

This programme is collaborative, involving collaborations in R&D and Beam Physics Research between many European Laboratories. The present planning — which is subject to available funds — expects to complete the R&D phase in three years. The construction of the ESS would be complete some 7 years later.

3. PULSED VS CONTINUOUS WAVE (CW) SYSTEMS AT HIGH POWER

Initial studies have been made of a comparison of mull-MW machines designed around various modes of operation:

- Room Temperature (RT) Continuous Wave (CW) operation;
- Room Temperature Pulsed operation;
- Superconducting Continuous wave operation;
- Superconducting Pulsed operation.

In high intensity machines the power extracted from the cavity to accelerate the beam is significant compared to the power in the cavity itself. For a room temperature cavity the power losses are essentially due to resistive effects and to the power removed by the beam. In a pulsed system where the accelerating cavities are only driven while the beam is being accelerated, the average resistive power losses are clearly duty cycle dependent. For a constant beam power, an increasing amount of power is removed from the cavity per pulse as the duty cycle of the beam is decreased. But since the resistive losses vary linearly with time the average power consumed is also decreased. Clearly there is a balance between the cavity power and the power extracted by the beam and the argument is strongest at lower power levels where the cavity losses dominate over the demands of the beam power. Superconducting accelerators provide a case where only the beam power is important, and in principle high fields can be achieved. (However, in principle, high fields could also be induced in a pulsed normal conducting cavity). In the case of superconducting accelerators at moderate powers one needs to compare the power required to the case of the pulsed RT scenario, and the demand to be reliable at the 99% level. Additionally, the technology used to drive the RF Cavities is also a point of discussion as new technology emerges in the for of inductive output tubes (IOT) which have efficiency advantages and lend themselves to be more readily pulsed compared to the conventional klystron.

4. SPACE CHARGE TRACKING AND LOSS MECHANISMS

The ISIS accelerator system accelerates particles to 800 MeV in a rapid cycling synchrotron. The process of injection - from a 70 MeV linac, capture and acceleration takes place in a period of some 20 ms. Losses during the injection and capture process account for almost all of the beam losses experienced during operation. Theoretical studies and modelling of this process — which is space charge dominated — has allowed the beam losses to be minimized to low levels in absolute terms. This is a critically important maintenance issue, since ISIS relies on "hands on" maintenance of the machine during shutdown periods. Such studies have also indicated the possibility of an intensity upgrade for ISIS that is now in progress by introducing second harmonic cavities to redistribute the space charge density of the beam allowing higher beam currents to be accelerated [5].

Analytical and numerical tracking tools have been developed for 2d transverse and ld-longitudinal space charge taking into effect the image charges on the beam tubes of the accelerator. These codes have been applied to the study of the beam dynamics of the ESS accelerator in the important topics of beam loss and halo formation.

A new RFQ code has been written to include more sophisticated geometries, realistic 3D space charge calculations and the effects of neighbouring bunches [6].

Development of 3d space charge tracking codes has also been progressed based on the ld-longitudinal and 2d-transverse codes. These have recently been used to track particles through the entire linac with promising results, indicating that the loss levels are extremely low. The eventual aim is to study the space charge dominated beam in the ESS accelerator with sufficient numbers of particles to generate a confidence in the low level predictions of beam loss and to allow a "hands on" approach to maintenance of the machine. The codes developed at Rutherford Laboratory have become benchmark codes. The importance of code comparison in this fields must *be* stressed.

5. CONCLUSION

Existing high power accelerators fall far short of the characteristics required for ADS solutions by at least one order of magnitude. The critical areas of future high power machines are being addressed in studies and in building demonstration machines. It is likely that pulsed accelerators designed for spallation sources will form the next generation of high power machines to be built.

The analytical tools for studying the physics of space charge forces in high power linear accelerators (and storage rings) are available and work towards testing the designs of high power accelerators against these codes is well underway. Beam losses under normal operation of existing machines have been studies in detail to achieve extremely low loss levels in absolute terms in the case of ISIS. The aim of current studies is to preserve the losses in high current machines at the same absolute level. This represents a computational resolution of a part in 10^7.

Beam Physics studies of the ESS accelerator for intense pulses are equivalent to studies of a 140 MW, 1.3 GeV CW machine.

Arguments have been presented which show that pulsed high power accelerators have power saving advantages over their CW counterparts, offering simpler and potentially more reliable technology than superconducting alternatives.

Hardware development at ISIS is aimed at pulsed machines in the 5 MW band but would be suitable for developments to higher powers by extending the operating duty cycle.

REFERENCES

[1] FINNEY, J.L., Europhys. News 20 (1989).

[2] ESS – A Next Generation Neutron Source for Europe - Volume III – The ESS Technical Study, ESS-96-53-M-3.

[3] SCHNEIDER, J.D., A Review of High Beam Current RFQ Accelerators and Funnels, Proc. 6th European Particle Accelerator Conference (EPAC98), 22–26 June 1998, Stockholm, Sweden, Institute of Physics Publishing, Philadelphia, PA, USA, ISBN 07503 0580 (CD-ROM)(1998).

[4] LAGNEIL, J.M., A Review of Linacs and Beam Transport Systems for Transmutation, Proc. 6th European Particle Accelerator Conference (EPAC98), 22–26 June 1998, Stockholm, Sweden, Institute of Physics Publishing, Philadelphia, PA, USA, ISBN 07503 0580 (CD-ROM)(1998).

[5] PRIOR, C.R., Simulation with Space Charge, Workshop on Space Charge Physics in High Intensity Hadron Rings, AIP Conf. Proc. No. 448, AIP, New York, (1998) 73.

[6] LETCHFORD, A.P., Code RFQSIM, Rutherford Appleton Laboratory, unpublished.

HIGH GAIN ENERGY AMPLIFIER EMPLOYING A FAST SPECTRUM BOOSTER COUPLED ONE WAY TO A THERMAL SPECTRUM SYSTEM

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Abstract

The Indian nuclear power programme has a special interest in Accelerator Driven Subcritical Systems (ADSS) from the point of view of utilization of .the large thorium deposits in the country. Our effort is directed towards devising an ADSS based on the ^{233}U-Th cycle, which produces a net energy output. In view of the constraints in accelerator technology for achieving high currents, we have also been investigating the possibility of having an ADSS with a reduced accelerator current. In an earlier work, these investigations were carried out using a simple model based on two groups diffusion theory. It was shown that it is possible to achieve a high neutron multiplication by designing a subcritical assembly made up of a booster and a main reactor region and ensuring that there is a more or less one way coupling between the two regions. In the present work, we have studied an ADSS, consisting of a fast spectrum booster coupled one way to a thermal spectrum reactor, using multigroup transport theory. The results of these calculations show that such a system has many advantages over the fast-fast system. It is shown that it is in principle possible to design such a system with a reasonable amount of ^{233}U inventory and which is driven by a 1 GeV proton beam having a current of 1-2 mA for a power of 750 MW(th). The power densities in the two regions are found to be in the range of typical fast and thermal reactors, respectively. Such a system could also be useful for waste incineration.

1. INTRODUCTION

Accelerator driven subcritical systems have evoked considerable interest in recent years. The Energy Amplifier, studied by the CERN group [1] is one such scheme which promises to burn plutonium and other actinide and fission product waste, and also produce power with a high degree of inherent safety and at a reasonable cost. An attractive feature of the energy amplifier over other schemes for transmutation is that it is based on a relatively 1ów current accelerator. Even so, the proposed current of 10 to 20 mA for a 1 GeV proton beam is one order of magnitude above the best achieved to date. For this reason, there have been suggestions [2] for reducing the current to levels that are presently achievable.

In two earlier publications [3, 4], we had proposed a one way coupled booster-reaátor system which could be operated at currents as low as 1-2 mA. The basic idea is to enhance the importance of the spallation neutrons by placing a booster region around the source locate4 at the centre with the main reactor region surrounding the booster. The booster contains a fuel having a relatively higher enrichment than the main reactor and is intended to enhance the spallation source neutrons. To obtain a large gain in the booster, it is essential to have its Keff as close to unity as possible. To ensure that the overall Keff of the booster reaçtor combination does not exceed unity, the arrangement has to be designed so as to have one way coupling between the two parts. Two practical ways of getting such a coupling were studyed. The first of these envisaged a fast booster and a fast (main) reactor separated by a large gap to get effective decoupling between the two. In the second concept, there was a fast booster and a thermal (main) reactor region separated by a large gap, and a thermal lining around the booster. In both these arrangements, neutrons from the booster enter the main reactor but very few neutrons from the latter return to the booster, ensuring a one way coupling. Very similar ideas have been independently proposed by some Russian groups [5, 7].

Our studies on these systems indicated that it may be possible to have a fairly large power of about 750 MW(th) with an accelerator current in the range 1-2 mA, while maintaining adequate sub criticality margin which ensures that the system does not become critical under various conditions. One difficulty with the scheme was that the power is very sharply peaked in the booster region making the fast-fast combination unviable. The fast-thermal combination is more benign in this respect but with the chosen moderator (light water), the power density in the (main) reactor region is too low if the power density in the inner booster is to be the same as that of existing fast reactors. These studies were carried out using two group diffusion theory with suitable boundary conditions at the material vacuum and for the absorber lining. Representative cross sections for typical fast and thermal systems were used.

In a recent paper [8] we presented studies of similar systems based on multigroup transport theory using the code DTF-IV with a 69 group WIMS library. The aim of the calculations was to confirm the earlier results obtained using a simple model and to examine other combinations of fast-thermal one way coupled systems, which would have power distributions, which are more acceptable. Specifically, we considered the case of a fast-thermal booster reactor combination in which the latter is a heavy water moderated reactor. The effect of having a heavy water reflector on the power distribution was also examined.

Our calculations suggested that the multigroup transport theory results give source multiplication values, which are generally comparable with those obtained earlier. The fast-fast combination requires very large fissile inventories due to very low specific powers in the outer region. However, the use of heavy water as a moderator in the outer main reactor region of the fast-thermal combination yields a power distribution such that the booster region would have power densities typical of fast reactors while the main reactor would have power density and specific power only slightly lower than existing heavy moderated reactors. The use of a heavy water reflector considerably flattens the power in the main reactor region because of the very large effective migration length in this system. In the present paper we explore the fast-thermal system further.

2. COMPUTATIONAL DETAILS

As stated earlier, the studies in [3, 4] were based on two group cross sections, which may be regarded as typical of fast and thermal systems but were not evaluated for any specific reactor system. The cross sections for the thermal region corresponded to light water reactors. However our studies showed that the power densities in this region were rather low and comparable to power densities of pressurized heavy water reactors (PHWRs). This indicated that the outer (thermal) region for the fast-thermal combination should be chosen to be a heavy water reactor. This means that the thermal region would contain the nuclides ^{233}U, Th, and D. Their concentrations and geometrical arrangement (homogeneous or heterogeneous) would of course be decided by various design optimization procedures and there would be other materials too. For our present study, we assume concentrations roughly corresponding to those of existing PHWRs.

As regards the inner system, we assume that it contains the materials, ^{233}U, Th and Pb. Their concentrations have been varied around what would be expected for a fast energy amplifier described in [1]. We have also added O to the mixtures as this has a softening effect on the fast spectrum, which would be the case if oxide fuels are used or involve other combinations, which include light elements. The concentration of O, however does not conform to any definite chemical formula.

The studies were carried out using the 69 group WIMS library and the one dimensional transport code DTF-IV [9]. While the library is mostly used for thermal systems, it has several fast and resonance groups and, may be expected to represent both the regions reasonably well. In any case we expect that the present studies would be sufficiently realistic to provide a confirmation of our earlier assertions based on simple calculations.

The fast and thermal mixtures could be treated as homogeneous or as heterogeneous mixtures. The WIMS formalism makes use of equivalence theorems for heterogeneous systems by adding an escape cross section to the potential scattering. Studies using different values of this parameter showed that for the fast systems, the results are not very sensitive probably because of the very hard neutron spectrum well above the resonance energy range. For the thermal system, heterogenity effects are very pronounced in the resonance region as is seen by the sensitivity of K_{eff} to this parameter. In the absence of a specific design, the choice of this parameter is somewhat arbitrary. In the thermal energy range, the effect of heterogeneity on the multiplication factor is small because of very low absorption of D_2O but there may be some effect on the thermal absorption and therefore the migration length.

The, calculations were done using the S¢ weights and directions in spherical geometry. Since the system has been seen to give good results with diffusion theory [2] it was felt that no purpose would be served by going to higher order quadratures. The spallation neutron spectrum is known to be much harder than the fission spectrum and also depends upon the thickness of the target. The number of neutrons produced by a proton also depends upon the thickness due to multiplication by (n,2n) and fast fissions and this is correlated with the spectrum. As an approximation, we have simulated the spectrum by introducing a source in the highest energy group (average energy 8 MeV). The number of neutrons per fission is assumed to be the same as in [2]. The latter quantity does not appear in the basic calculations but only in converting the neutron multiplication results into beam current requirements. The sensitivity of the neutron multiplication to the source spectrum was checked by comparing the above results with a pure fission spectrum source.

3. RESULTS AND DISCUSSION

Table I shows the concentrations of various materials used in our calculations. Table II shows the multiplying properties of the fast-fast assembly, for these material concentrations in the two zones. The main point to be noted is that the gap introduces one way coupling as observed in [3, 4] and the results are very similar. The large requirements of fissile fuel inventory for the highly decoupled system is also worth noting.

Table III shows our results for the fast thermal system. Note the effect of the gap and the thermal. lining ín providing the necessary one way coupling in this system. We see that with both these methods of decoupling we can achieve a subcritical K_{eff} of 0.98. A point to be noted is that in these more realistic calculations, the outer K_{eff} has to be chosen to be somewhat lower to get the same overall K_{eff} which means that our mechanisms for effecting the decoupling are less effective than was indicated by the earlier studies. However the neutron multiplication finally obtained is comparable.

In Table IV we compare the power distribution characteristics of a few selected cases. This table shows that the fast-thermal combination is likely to work because the specific powers in the outer region compare well with typically the values of existing thermal reactors while the inner region experiences power densities and specific powers that are lower or comparable to that of existing fast reactors.

Table I. Concentration (10^{24} atoms/cm^2) of various materials used in the calculations

Case	Region No.	O	D	Pb	Th	^{233}U	K
Fast-Fast 1	1.	2.8-2		1.94-2	8.48-3	8.93-4	1.200
	2.	2.8-2		1.94-2	8.75-3	6.29-4	0.975
Fast-Fast 2	3.	2.8-2		0.80-2	1.95-2	1.77-3	1.200
	4.	2.8-2		0.80-2	1.95-2	1.22-3	0.975
Fast-thermal	1.	2.8-2		1.40-2	1.23-2	1.226-3	1.200
	2	2-2	6.4-2	-	2.52-3	3.80-5	0.975

Table II. Variation of KFe, neutron multiplication (M) and accelerator current requirement (for 750 MW(th) power) with the degree of decoupling (gap) for the Fast-Fast (2) system

S.No.	R_1 (cm)	R_2 (cm)	R_3 (cm)	^{233}U-mass (tons)	K_1	K_2	K_{12}	M	Acc. curr. (mA)
a	59.2	59.2	64.0	0.7	0.953	0.120	0.980	100	3.4
b	59.2	160	198	7.8	0.953	0.813	0.980	175	1.9
c	59.2	210	284	27.6	0.953	0.920	0.980	276	1.2
d	59.2	245	370	71.6	0.953	0.952	0.980	371	p.9
e	52.1	52.1	66.4	0.7	0.900	0.414	0.980	97	3.5
f	52.1	75	112	2.3	0.900	0.762	0.980	131	2.6
g	52.1	100	172	8.5	0.900	0.908	0.980	194	1.7
h	52.1	123	296	48	0.900	0.959	0.980	279	1.2

Table III. Effect of gap, absorber lining and reflector on the K_{eff} of the Fast-Thermal combination

R_1 (cm)	R_2 (cm)	R_3 (cm)	^{233}U mass (tons)	Absorber	Reflector	K_1	K_2	K_{12}	M	Acc. Cur. (mA)
71.5	71.5	195.0	-	No	No	0,954	0,948-	1.067		-
71.5	71.5	195.0	-	Yes	No	0.954	0.948	1.025		-
71.5	210.0	310.0	-	No	No	0.954	0.948	0.999		-
71.5	210.0	310.0	2.0	Yes	No	0.954	0.948	0.980	335	1.0
68.5	210.0	310.0	1.9	Yes	Yes	0.937	0.948	0.980	348	1.0

Table IV. Peak and total power produced in the two regions for some selected cases

Case No.	Peak-Power in region 1 (KW/lit)	Peak Power in region 2 (KW/lit)	Total Power in region 1 (MW)	Total Power in region 2 (MW)	Specific Power (W/gm) in peg. 1	Specific Power (W/gm) in reg. 2
F-F (2c)	680	26	208	542	29.1	1.2
F-T	326	15.6	172	577	21.5	6.8
F-T-R	237-	12:37	109	641"	15.5 '	7.5

In Fig. 1 we show the power distribution for the fast-fast combination. The dashed line indicating linear power density refers to the power per unit radial interval and is shown to give an idea of the relative power produced in the inner and the outer regions. This is remarkably similar to the one obtained by simple two group theory. Fig. 2 show the power distribution in a fast thermal system while Fig. 3 shows the effect of adding a heavy water reflector to the outer core. Two points are worth noting here. The first is that there is a spike in the power density in the inner region close to the gap. This was not observed in the two group approach and may be attributed to the epithermal neutrons from the outer region which are not adequately shielded by the Cd liner. The other point is that the reflected system shows a much flatter power distribution as compared to the bare case.

We have also studied the effect of lowering the overall K_{eff} to 0.96 by reducing the radius of the inner sphere, for the unreflected system. This has the effect of reducing the multiplication by more than half to 150 with a current requirement of 2.3 mA. We have also studied the effect of accidental filling up of the void with D_2O. The reactivity effect is about +3mK for the reflected system and +14mK for the unreflected case. While both systems stay subcritical, a smaller positive or a negative effect is clearly preferable.

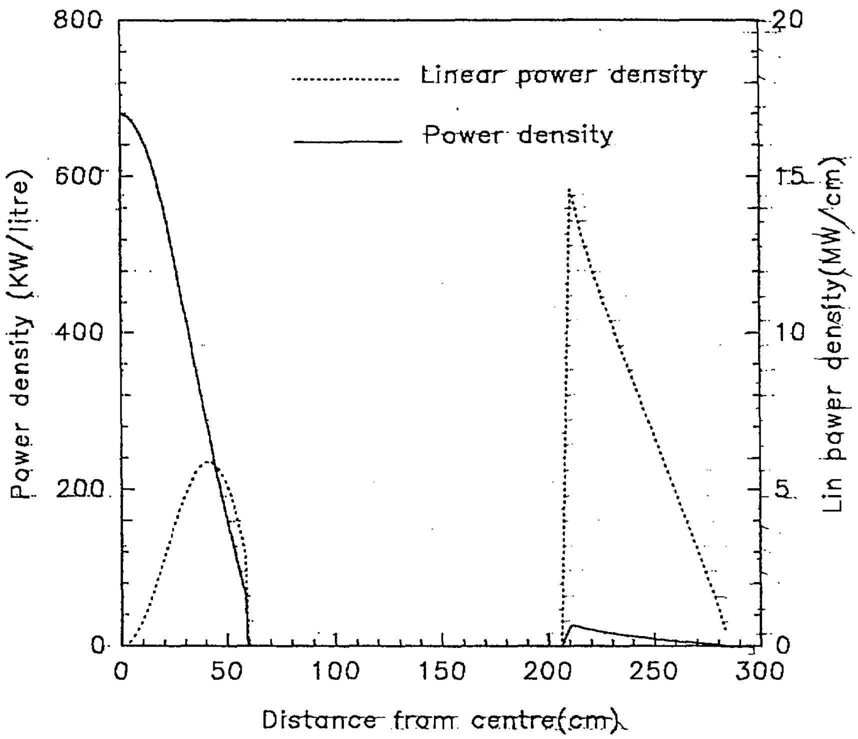

Figure 1. Power distribution in the fast-fast system.

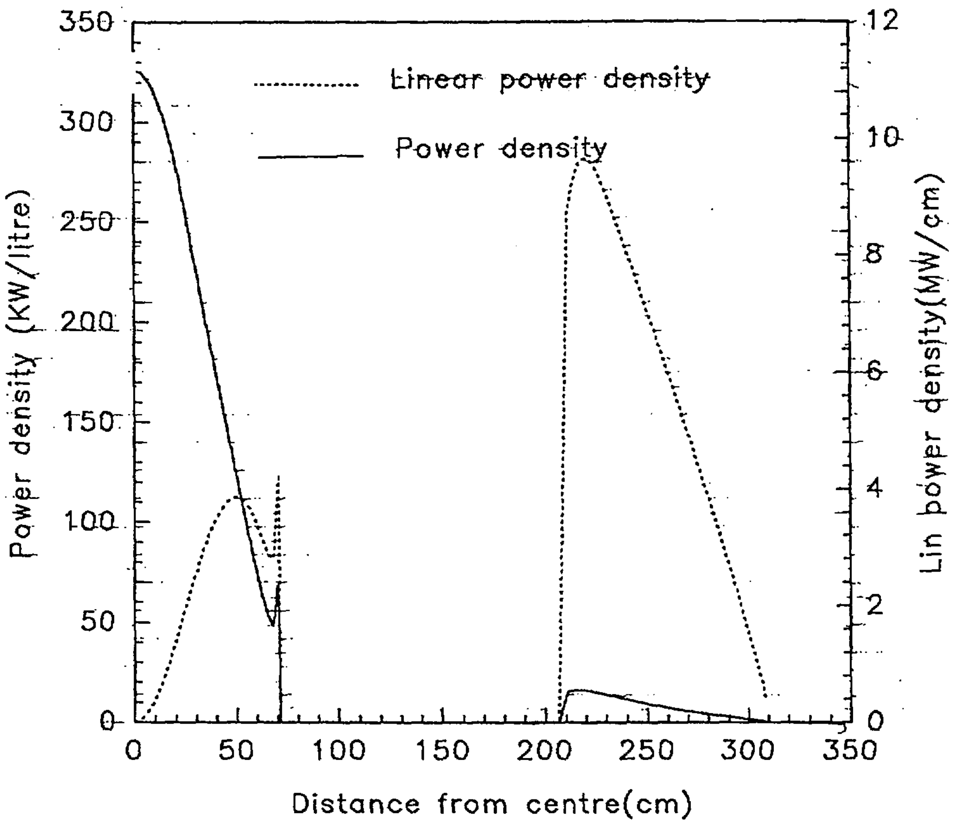

Figure 2. Power distribution in the fast -thermal system.

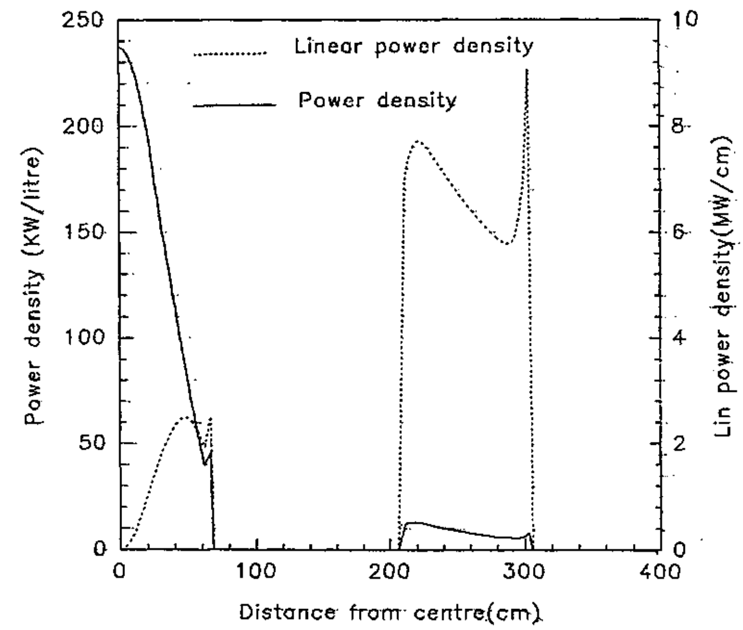

Figure 3. Power distribution in the fast-thermal system with a reflector.

4. CONCLUSION

We have presented multigroup transport theory results of calculations of our concept of a one way coupled booster-reactor for enhanced neutron multiplication. The results by and large confirm the validity of the concept studied earlier using simple models. Some adjustment in the size or parameters such as the K_{eff} are indicated. A fast-fast combination does not appear to be economical because of the large fissile inventory required and rather low specific powers in the outer regions. However, a fast-thermal combination with a heavy water reactor moderated core for the outer zone together with a heavy water reflector gives a much more acceptable power distribution. More detailed studies including burnup effects and substituting our simple one dimensional studies with two dimensional calculations for a realistic reactor geometry would be required to assess the practical feasibility of our concept.

REFERENCES

[1] RUBIA, C., et al, Report CERN/ÁT/95-44 (ET) (1995).

[2] DANIEL, I. and PETROV Yu.V., Nucl. Instr. Meth. Phys. Res. A 373, 131.

[3] DEGWEKER S.B., LAWANDE S. V. and KAPOOR S. S., Paper presented at the IAEA TCM on Feasibility and Motivation for Hybrid Concepts for Nuclear Energy Generation and Tansmutation, Madrid, (1997).

[4] DEGWEKER S.B., LAWANDE S. V. and KAPOOR S. S., Ann Nucl. Energy 26, (1998) 123.

[5] KOCHUROV B.P. AND SELIVERSTOV V.V., Int. Meeting on ADS, Moscow, (1999).

[6] KOCHUROV B.P. et al, ICENES'96, Obninsk, Russia, (1.996).

[7] ALEKSEEV P.N., et al, IAEA TECDOC-6110/3 (1998).

[8] DEGWEKER S.B., SAHNI D.C. and KAPOOR S.S., Int. Meeting on ADS, Moscow, (1999).

[9] LATHROP K.D., Report LA-3373 (1965).

RESEARCH ON ACCELERATOR DRIVEN SYSTEMS IN THE NATIONAL ACADEMY OF SCIENCES OF BELARUS

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Abstract

By now a lot of theoretical papers was published where basic aspects of ADS concept were discussed: production of energy, transmutation of radioactive waste, tritium production and incineration of weapon plutonium. The experimental research in this field is rather scare because the experiments on available high energy accelerators are difficult and expensive, and in some case even unfeasible. In this regard experimental research of various aspects of ADS on the basis of low energy ion accelerators are of great importance. The possibility of using low energy accelerators to investigate physical characteristics of subcritical target/blanket systems follows from the mechanism of nuclear reactions in high (0.8 - 2 GeV) and low (15-20 MeV) energy ranges as well as from features of nucleon-meson cascade development. It was the foundation for creation of an experimental facility consisting of a uranium-polyethylene subcritical assembly (maximal multiplication factor $\approx$ 0,975) driven by high intensity ($\approx 1.5\times10^{12}$ n/s) neutron generator. The core is a rectangular prism having base 400 $\times$ 400 mm and height 570 mm. It is assembled from polyethylene blocks with the channels to place the fuel pins disposed in a rectangular lattice with pitch equal to 20 mm. The fuel pins are made of UO_2 with enrichment equal to 10% in ^{235}U. The subcritical assembly contains 265 fuel pins. Central part of the blanket is a neutron producing lead target 10 cm in diameter and 60 cm in length. There are four channels for location of detectors of neutron flux level control. For irradiation of investigated samples the experimental channels with diameter of 25 mm are located at subcritical assembly radii 5, 10, 16 cm. The core is placed in a well of a stack of graphite bricks serving as lateral reflector. Calculations have shown that energy spectrum of neutrons in the thermal assembly is unique one and differs from energy spectra in thermal reactors. Namely in the energy range from 1 to 10^4 eV dependence of neutron flux on energy is very weak what allows to get data on neutron cross sections for a resonance region. Measurements of energy spectra at different points inside experimental channels will be performed by means of activation technique having different advantages comparing to other techniques. At present experiments on measurement of transmutation rates of long-lived fission products (^{129}I, ^{135}Cs, ^{126}Sn,....) and minor actinides (^{237}Np) are being planned.

1. INTRODUCTION

The investigations in the field of nuclear physics, development of calculation methods of nuclear reactors, neutron physics etc. are being carried out at the Scientific and Technical Center "Sosny" of National Academy of Sciences of Belarus (ASTC "Sosny" of NASB) since the 60s after putting into operation the research reactor and the critical assemblies. During 30 years the wide range of different configurations of the critical assemblies (geometry, composition) has been constructed to study physics of the special reactors. Some later the researches in the field of relativistic nuclear physics, the development of calculation methods of interaction of high energy particles and nuclei with fissile media and evaluation of nuclear data of fissile isotopes Th-U and U-Pu fuel cycles were begun. These researches have been performed in accordance with the program of joint investigations with such well-known scientific centers as Physics and Power Engineering Institute in Obninsk, I. V. Kurchatov' Institute of Atomic Power in Moscow and Joint Institute of Nuclear Research in Dubna.

At present the researches on the physics of the subcritical systems driven by high energy accelerators for energy production and transmutation of long-lived fission products and minor actinides are being carried out at ASTC "Sosny" in following main directions:

- evaluation of nuclear data for nuclides Th-U and U-Pu fuel cycle;
- development of the calculation methods for Accelerator Driven Systems characteristics;
- experimental investigation of neutronics of subcritical systems driven by high energy accelerators.

2. EVALUATION OF ACTINIDE NUCLEAR DATA

A large number of the evaluated data files are available for isotopes Th-U and U-Pu fuel cycles, i.e. BROND, ENDFB-6, JEF-2.2 and JENDL-3 libraries. Unfortunately, the current state of minor actinides evaluated data files is characterized by severe discrepancies. In case of using weapon Pu for fuel fabrication it is necessary to improve data for these nuclei. The main reason of discrepancies is the scarcity of experimental data. The most obvious way of data files improvements is usage of different nuclear models: coupled-channel optical model, evaporation statistical model with pre-equilibrium emission of first neutron and statistical model with systematics of parameters of level density. The latter approach helps to analyze irregularities in the experimental data and subsequently predict the same kind of irregularities in cases of poorly studied nuclei. For example, the one of step-like energy dependence of (n,2n) reaction cross section for ^{239}Pu in a near-threshold region, which we should take into account in other cases of Z-even, N-odd target nuclei. The mentioned above methods are generally used for neutron data evaluation of major actinides like ^{235}U, ^{239}Pu, ^{233}U etc. in BROND-2, ENDFB-6 and JENDL-3. There are data on (n,f), (n,2n) and (n,3n) reaction cross sections for 235-U and 238-U. A consistent analysis of these data provides an insight into the competition of fission and neutron emission above (n,nf) reaction threshold. The same approach coupled with statistical model allows to predict fission, capture, inelastic scattering and multi-particle emission cross sections. The neutron cross-section files for ^{233}U, ^{242}Cm, ^{244}Cm, ^{235}U, ^{240}Pu, ^{239}Pu, ^{236}U, ^{241}Pu were created. The file for ^{233}U was included into the BROND-2 library (CJD, Obninsk, Russia) in 1993. The files of evaluated neutron data for ^{243}Cm, ^{245}Cm, ^{246}Cm, ^{241}Am, ^{243}Am, ^{242m}Am, ^{242g}Am, ^{238}Pu, ^{242}Pu, ^{238}Np were created in the frame of the ISTC project #B-03 "Actinide Nuclear Data Evaluation". The files created in NAS of Belarus for ^{243}Cm, ^{245}Cm, ^{246}Cm will be included in ENDFB-6 library. In the frame of the project which have been submitted to the ISTC the neutron data evaluation laboratory is planning to create the files for ^{232}Th, ^{232}U, ^{233}U, ^{234}U, ^{238}U, ^{231}Pa, ^{232}Pa, ^{233}Pa in the energy region up to 150 MeV.

These evaluations are of interest for safeguard application as well, in particular for the control of nuclear materials of Th and U fuel cycles. New photon and neutron spectrometry methods used for the control of NM are under development now. They require more complete and precise data on energies and quantum yields of X rays and gamma rays, which can be measured using available at the Center modern spectrometers with high purity germanium detectors in wide range of energies. Then ND can be evaluated as well.

3. THE RESEARCH ON ACCELERATOR DRIVEN SYSTEM

3.1. The code SONET to calculate accelerator driven system

Application of high energy accelerators for production of energy and radioactive nuclides as well as transmutation of nuclear waste is based on large scale using of spallation reactions for neutron generation in thick targets and following multiplication of the neutrons in subcritical blankets. Correct description of such a complicated process as high energy particle interaction with matter can be performed by means of Monte Carlo method. It is the most suitable tool to take into account all details of nuclear reaction mechanism and particle transport both in high and low energy range. At present computer codes LAHET, HETC and HERMES [1, 2] are widely used for such purposes. For description of inelastic reactions they use cascade-evaporation models, which do not describe correctly energy distribution of neutrons generated both in spallation reactions and in thick heavy-metal targets bombarded by high energy protons [3-5]. The code SONET [3-5] being developed for many years in NAS Belarus is dedicated for Monte Carlo simulation of intra- and inter-nuclear cascades in thick targets of arbitrary geometric configuration and material composition. One can calculate transport of nuclei ($A<16$), nucleons and pions in the energy region up to 10 GeV. Electro-magnetic showers are simulated by means of the well-known EGS4 computer code [6], which is included in the package also. Ionization energy losses, range straggling, pion decay, multiple Coulomb scattering, inelastic interaction of particles (n, p, $\pi^{\pm}$, γ) and nuclei with nuclei as well as high energy fission have been taken into account. Calculation of particle transport in high energy range is performed by Monte Carlo method also. The main problem by calculation of a high energy beam interaction with target/blanket system is description of inelastic nuclear reactions in wide energy range. The code SONET describes nuclear collision products by means of a cascade-exciton model [7]. The model treats a nuclear reaction as a three-stage process (cascade, pre-equilibrium and equilibrium stages) unlike well-known two-stage model. The first stage is an intra-nuclear cascade in which primary particles can be re-scattered several times prior to absorption by the nucleus or escape from it. The excited residual nucleus remaining after cascade particle emission determines a particle-hole configuration which is starting point for the second, i.e. preequilibrium stage of the reaction. Subsequent relaxation of nuclear excitation is treated in terms of exciton pre-equilibrium decay model [4,7], which includes equilibrium particle evaporation as the third stage of the reaction. The equilibrium (evaporation) stage includes evaporation-to-fission competition for heavy nuclei and Fermi break-up model [8] for light nuclei. To describe nuclear level density is used by Ignatyuk et al. parameterization [9, 10] in the frame of Fermi gas model. Fission barriers for different nuclei are calculated in the frame of the liquid drop model. The fission process for heavy nuclei ($A>200$) is described in the frame of Fong [11] or Alsmiller [12] model. Nucleus-nucleus inelastic interactions are described on the basis of cascade mechanism of nuclear reactions too. It is assumed that neutral pions decay π^0-2 γ at the point of the inelastic interaction. Charged pions lose their energy for ionization and excitation of atoms as well as can decay $\pi \rightarrow \mu + \nu_{\mu}$ or initiate a nuclear reaction $\pi^{\pm} + A \rightarrow xn + \gamma\, p + A'$. It is assumed that positive pions coming to rest decay immediately. Negative pions having lost their energy can form π^- - atoms. It is assumed that pions can be absorbed in the nuclear diffusive layer by a pair of intranuclear nucleons. Decay of stopped π^- - mesons are also taken into account. Hadron interactions with a hydrogen nucleus (h+p) are the last but not least nuclear reaction channels taken into account.

After completion of spallation and fission reactions in high energy range low energy neutrons ($E < 10$-20 MeV) represent main part of a nucleon-meson cascade. The code SONET provides complete description (in energy as well as angular and spatial coordinates) of the low energy neutrons produced in spallation reactions. Further low energy neutron transport is described by means of methods developed for radiation shielding and nuclear reactor calculations. The first alternative is represented by the code SYNTES-Q [13]. It allows to calculate neutron transport for subcritical systems in multigroup diffusional approximation in 2D geometry and is based on iterative synthesis method for solving the neutron transport equation with an external source. The second alternative is represented by the well-known code MCNP 4A [14]. One of the versions of the code SONET allows to calculate the evolution of a nucleon-meson cascade in time, which is of interest when considering external pulsed sources.

The code SONET allows to calculate the following quantities:
• neutron energy-angle leakage spectra as well as neutron and nuclide yield;
• neutron fluence as well as specified reaction rates for specified geometric cells;
• time evolution of fuel nuclide composition;
• thermal power due to (n,f) and (n, γ) reactions;
• energy production in different geometric cells;
• distribution of fission and capture rates for different nuclides.

3.2. Comparison with experimental data

It is obvious that for ADS design study it is necessary to use computer codes, which can predict the most important quantities such as yields and energy spectra of neutrons escaping from spallation targets because these quantities are main factors determining ADS performance. In order to validate accuracy of nucleon-meson transport calculations with the code SONET some comparisons of calculation results with recent experimental data have been carried out. A comparison of experimentally measured and calculated neutron yields from thick lead and tungsten targets bombarded by high energy protons [15, 16] is presented in Fig. 1 and 2.

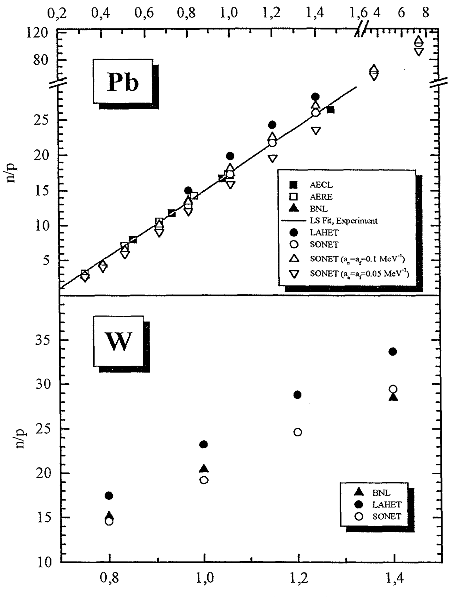

Figure 1. Measured and calculated spallation neutron yields (n/p) for a thick lead target (10.2-cm diameter x 61-cm length) and a thick tungsten target (10.2-cm diameter x 40-cm length) [15].

It can be seen that the code SONET predicts the number of neutrons escaping the thick lead and tungsten targets with high accuracy in wide energy range. The computer code LAHET overestimate relevant neutron yields for proton energies near 1 GeV [15].

One of the most important characteristics of ADS is energy spectrum of neutrons generated due to, spallation reactions in heavy metal targets. Therefore a comparison has been made of experimental and calculated energy spectra for neutrons escaped a thick lead target at different angles [16]. It can be seen from the latter paper that the package NMTC/JAERI + MCNP 4.2 underestimates significantly the spectra in the energy range from 20 to 100 MeV.

The package SONET + MCNP 4A reproduces the experimental energy-angle spectra successfully (Fig. 3). Thus one can state that taking account of pre-equilibrium emission of nucleons as well as dependence of level density parameter on shell corrections and excitation energy allows to describe experimental results on the energy-angle spectra with high accuracy in the whole energy range covered.

3.3. Transmutation of ^{129}I and ^{237}Np with relativistic protons

First experiments on transmutation of ^{129}I and 23'Np using 1.5, 3.7 and 7.4 GeV, protons performed by "Marburg-Dubna" collaboration at Joint Institute for Nuclear Research (Dubna, Russia) since 1996 are of great importance [17, 18]. Sectional views of the experimental target-moderator system are shown in Fig. 4.

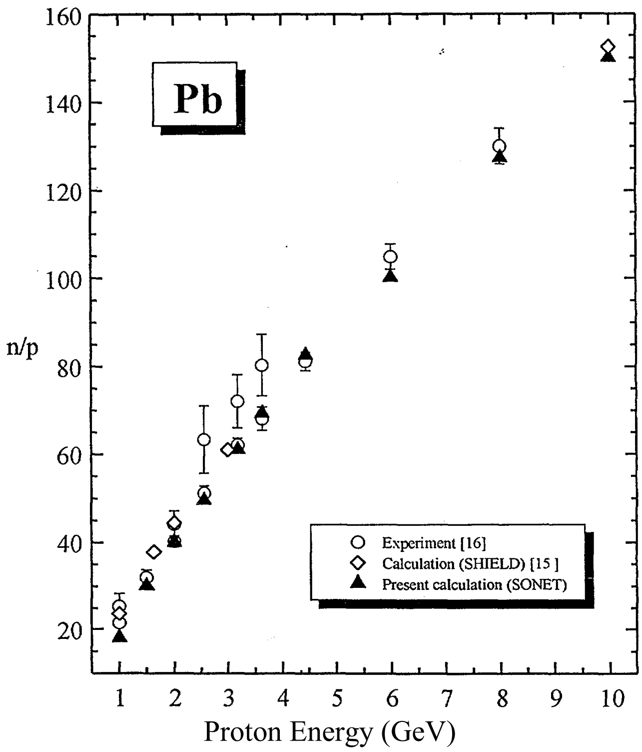

Figure 2. Measured and calculated spallation neutron yields (n/p) for a thick lead target (20-cm diameter x 60-cm length) [16].

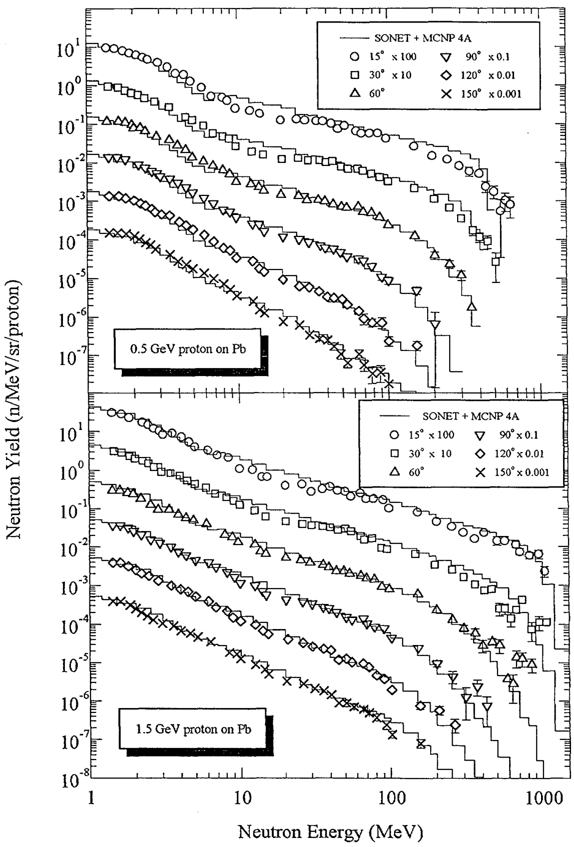

Figure 3. Experimental and calculated leakege neutron energy spectra from a thick lead target of 15 by 15 cm in witdth and 20 cm in length bombarded with energy protons [16].

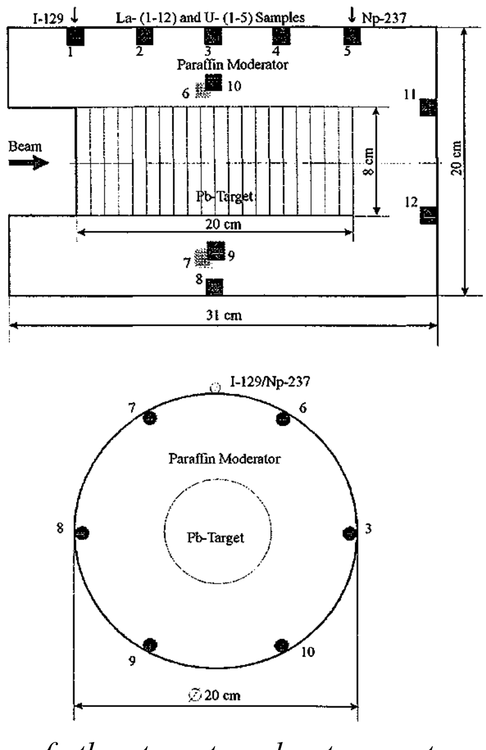

Figure 4. Sectional views of the target-moderator system used in experiments on transmutation of ^{129}I and ^{237}Np with relativistic protons.[17,18].

The calculations have been performed by means of the high energy transport code SONET. The sectional views in Fig. 4 represent one of two experimental systems used [17, 18]. In the second one there is a natural uranium core 3.6 cm in diameter surrounded by a lead shell 2.2 cm thick instead of the lead target 8 cm in diameter, other things being equal.

All details of the experimental target-moderator system including geometry configuration of NaI- and NpO_2-samples as well as Al-capsules were taken into account. The sealed aluminum capsules contained ^{129}I (0.425 g) and ^{127}I (0.075 g) in form of NaI as well as ^{237}Np (0.742 g) in form of NpO_2. In Tables I and II below calculated reaction rates are presented in comparison with experimental data for the two target-moderator systems used.

Table I. Measured and calculated reaction rates B defined as number of produced particles per single incident ion and per 1 gram sample for the Pb target

T_p GeV	**B (^{140}La), 10^{-4}**		**B (^{239}Np), 10^{-4}**		**B (^{130}I), 10^{-4}**		**B (^{238}Np), 10^{-4}**	
	Experiment	SONET[1]	Experiment	SONET[1]	Experiment	SONET[1]	Experiment	SONET[1]
1.5	1.7±0.4	2.5±0.2	0.75±0.15	0.71±0.09	0.9±0.2	1.9±0.3	8.1±1.6	7.8±1.5
3.7	6.0 ± 0.9	4.7 ± 0.1	2.9 ± 0.4	1.8 ± 0.3	3.1 ± 0.5	3.6 ± 0.5	44 ± 7	14.6 ± 2.5
7.4	7.3 ± 1.5	6.7 ± 0.2	3.3 ± 0.7	2.5 ± 0.4	4.0 ± 0.8	4.7 ± 0.6	4 1 ± 9	21 ± 3.6

[1]Given uncertainty is statistical one (2σ).

Table II. The same as in Table I but for the U(Pb) target

T_p GeV	**B (^{140}La), 10^{-4}**		**B (^{239}Np), 10^{-4}**		**B (^{130}I), 10^{-4}**		**B (^{238}Np), 10^{-4}**	
	Experiment	SONET[1]	Experiment	SONET[1]	Experiment	SONET[1]	Experiment	SONET[1]
1.5	3.1 ± 0.7	5.2 ± 0.2	1.4 ± 0.3	2.0 ± 0.4	2.3 ± 0.5	4.6 ± 0.5	9.0 t 1.8	13.8 ± 2.2
3.7	10.5 ± 1.6	10.6 t 0.5	4.5 ± 0.7	3.6 ± 0.5		9.5 ± 0.9		27.0 ± 3.2
7.4	15.4 ± 3.1	15.9 ± 0.3	6.0 ± 1.2	5.3 ± 0.5	14.7 ± 3.0	14.0 ± 1.4	50 ± 10	44.5 ± 5.0

[1]Given uncertainty is statistical one (2σ).

The reaction rates are the following: (i) $^{139}La(n,\gamma)^{140}La$; (ii) $^{238U}(n,\gamma)^{239}U \rightarrow {}^{239}Np$; (iii) $^{129}I(n,\gamma)^{130}I$; (iv) $^{237}Np(n, \gamma)^{238}Np$. When calculating the reaction rates $\sigma_{n,\gamma}$ cross sections from ENDFB-VI were used.

Monte Carlo calculations performed to validate measured capture reaction rates of ^{129}I and ^{237}Np reveal reasonable agreement with experimental data. The most significant discrepancy between the measurements and calculations (within a factor of 3) is observed for the NpO_2 sample and Pb target at 3.7 GeV proton energy. The discrepancy is not properly understood. When considering the calculations one can state that spallation neutron yields from such a lead target for 1.5 GeV proton energy are predicted by the code SONET with high accuracy (see Figs. 1, 2). Most of these neutrons escaping from the target have energies below 20 MeV and their transport through the paraffin layer is simulated by Monte Carlo method. Therefore one may expect high accuracy in the calculated reaction rates. One possible reason for the discrepancy between the measurements and calculations is inadequate description of sample positions in radial direction. High neutron fluence gradient in radius near the outer surface of the paraffin layer was observed experimentally [18].

To get insight into detailed properties of the target-moderator systems neutron spectra were calculated for all the samples. In Fig 5 such spectra are presented for the sample No. 3.

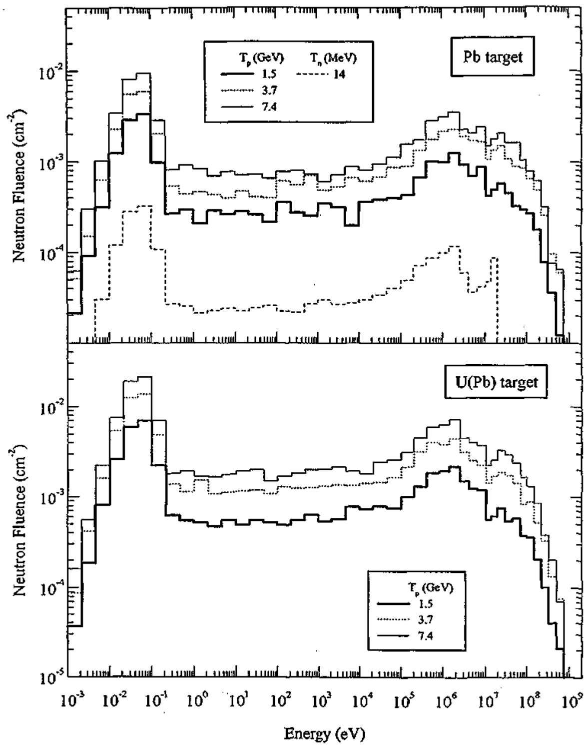

Figure 5. Calculated neutron energy spectra averaged over the La-sample (No. 3) volume for the Pb and U(Pb) target-moderator systems (see Fig. 3) at different projectile energies for monodirectional protons and neutrons as projectiles. Normalization was performed per one incident particle.

It can be seen that all of these spectra are similar in shape. Thus the only physical difference between the six energy-target combinations is difference in neutron multiplicity. At the same time similar spectra can be formed when using low energy neutrons as projectiles. Relevant spectrum for a monodirectional 14 MeV neutron beam is also presented in Fig. 4. Therefore when considering (n,γ) reactions such investigations can be performed using, for example, a low energy neutron generator.

3.4. Research of the peculiarities of transmutation of LLFP and MA on the base of the subcritical assembly driven by neutron generator

At present experimental research of various aspects of ADS using high energy accelerators are difficult, expensive, and in some cases even unfeasible. In this regard research of ADS with low energy ion accelerators are of great importance. The possibility of using of low energy accelerators to research the physical characteristics of the subcritical target/blanket systems is based on the mechanism of nuclear reactions in high (1 GeV) as well as low energy range (10 - 20 MeV) and nucleon-meson cascade. It was shown that spallation source can be simulated by 14 MeV neutrons irradiating the heavy element targets [20]. It was the foundation for creation of the experimental facility consisting of uranium blanket (subcritical assembly) where the central part is neutron producing lead target and high intensity (10^{12} n/s) neutron generator. Performed calculations have shown that in the experimental channels of the subcritical system is possible to form different neutron spectra: fast, resonant and thermal.

The experimental set-up "Yalina" (Fig. 6) consists of a subcritical assembly driven by a neutron generator and corresponding engineering systems: rabbit system, measuring and dosimetric devices, vital provision system, system of physical protection etc.

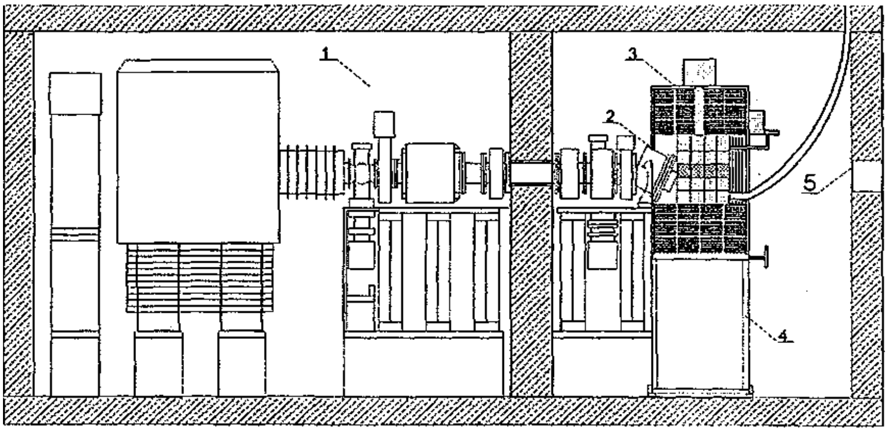

Figure 6. The subcritical facility "Yalina": 1 -the neutron generator; 2 - Ti-^{3}H target system; 3 - the subcritical assembly; 4 - movable platform; 5 - collimator.

The facility will be put into operation by the end of this year. The set-up allows to carry out experimental research using different subcritical assemblies with arbitrary composition, geometry and neutron spectrum. The neutron generator NG- 12-1 consists of a deuteron ion accelerator with magnetic separation of accelerated beam and rotating Ti-^{3}H target:

- Maximal yield of neutrons - (1.5-2.0) 10^{12} n/s
- Accelerating voltage - 250 keV
- Current of accelerated ions - (10-12) mA
- Diameter of the beam ~ 20mm

The set-up allows to carry out experimental research using different subcritical assemblies with arbitrary composition, geometry and neutron spectrum.

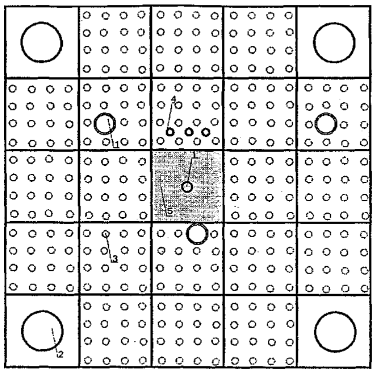

Figure 7. The core of the sutrcritical assembly "Yalina": 1 - experimental channels; 2 - channels for neutron flux level monitoring system; 3 - fuel channels; 4 - channels for compensation rods; 5 - neutron producing target.

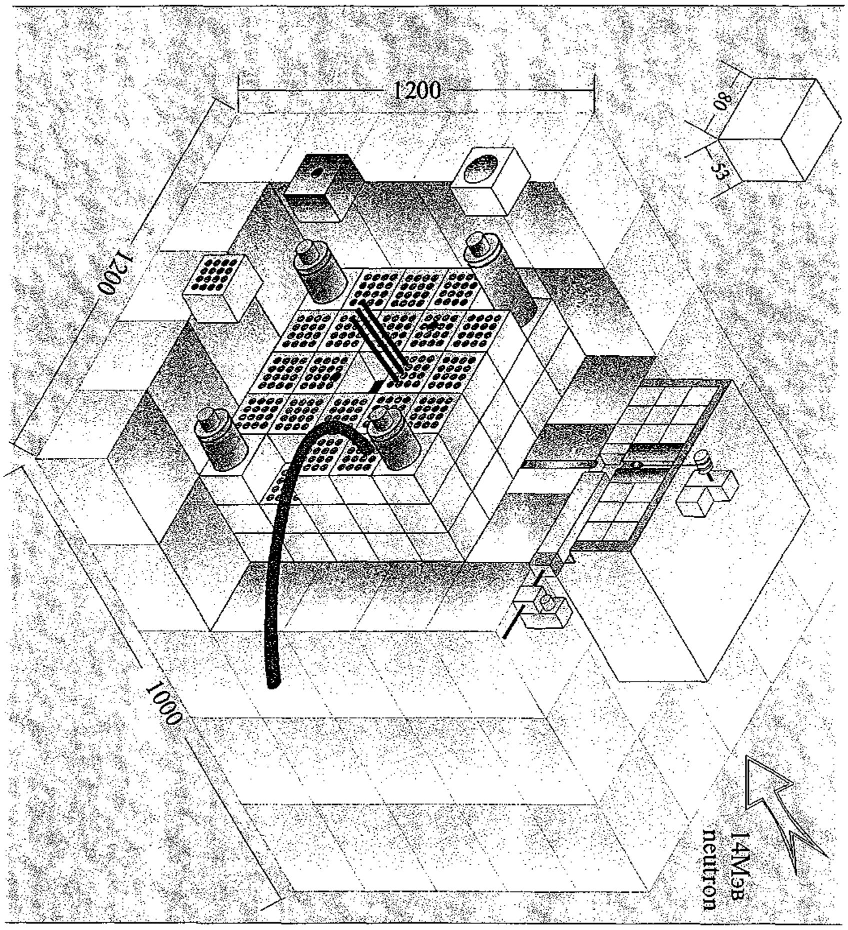

Figure 8. General view of the sutrcritical assembly "Yalina".

At the first stage the uranium-polyethylene subcritical assembly (maximal multiplication factor ~ 0,975) will be put into operation (Figs. 7, 8). The fuel pins are made of UO_2 with enrichment equal to 10% in ^{235}U. The subcritical assembly contains 265 fuel pins and has the reactivity compensation system. The device consists of remotely operating rods containing boron carbide. For irradiation of investigated samples the experimental channels with diameter of 25 mm are provided for radii 5, 10 and 16 cm. The subcritical assembly is mounted on the movable platform, which allows to shift the assembly in two directions with respect to the ion beam axis. Special attention was paid to aspects of nuclear safety of the subcritical assembly. The proposed design and material composition of the subcritical assembly allows to maintain pre-determined value of multiplication factor $k_{src} \leq 0.98$ for all possible variations of external factors (temperature, pressure etc.) as well as in hypothetical case of flooding the room where the facility are located. When elaborating the technical project of the assembly the following main characteristics were determined:

- The number of fuel pins ensuring $k_{max} \leq 0.98$ for the subcritical assembly with chosen material composition and geometry as well as with lead target in central part of the assembly equals to 265 fuel pins. There is a regular set of detectors for neutron flux level monitoring system.
- Reactivity worth of a fuel pin located at minimal distance from the core center - **0.34 $**
- Reactivity worth of a peripheral fuel pin - **0.22 $.**

- Reactivity worth of a complete fuel subassembly containing 16 fuel pins and situated near the lead target - **5.42 $.**
- Reactivity worth of a complete peripheral fuel subassembly - **3.25 $.**
- Reactivity swing due to insertion of the lead target into central part of the subcritical assembly - **1.49 $.**
- Reactivity effect due to filling the subcritical assembly with water - **0.54 $.**
- Integral energy deposition rate for the assembly when operating with the Ti - ^{3}H target (it is supposed that neutron intensity equals to 10^{12} n/s) - **100 Wt.**
- Temperature coefficient of reactivity by accidental increase of temperature up to 100°C **-0.2 $.**
- Effective fraction of delayed neutrons β_{eff} - **0.00738**.

Calculations have also shown that energy spectrum for the thermal assembly differs from any reactor spectra. Namely in the energy range from 1 to 10^4 eV dependence of neutron flux on energy is very weak which allows to get data on neutron cross sections for resonance region (Fig. 9).

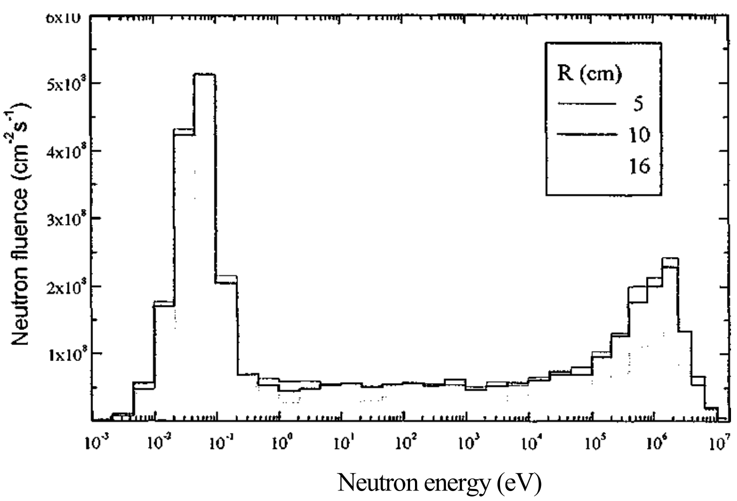

Figure 9. Neutron spectra averaged over sample volumes in the experimental channels of the subcritical assembly with thermal spectrum driven by the neutron generator. It is supposed that neutron source intensity equals to 10^{12} n/s.

easurements of energy spectra at different points inside the experimental channels will be performed by means of activation technique. In the energy region 30 keV - 15 MeV the measurements will be performed by means of solid-state nuclear track detectors and thin-film break-down counters. For spectrum unfolding in the energy region 30 keV-15 MeV fission reactions will be used for the following nuclei: ^{232}Th, $^{235,\,236,\,238}U$, ^{237}Np, $^{239,\,240,\,241}Pu$. The reactions (n,α) on nuclei ^{10}B and ^{6}Li will be used also. In the energy region $E_n < 30$ keV foils made of indium, gold, tungsten, and manganese (E_γ = 1.457; 4.906; 18.8; and 337 eV respectively) as well as some others will be used. Data on possible samples: cross sections and their mass are presented in the Table III. Relevant calculations have been performed taking into account exact position of each sample in the experimental channels of thermal assembly.

In the nearest future modernization of neutron generator will be carried out. As a result pulse mode of operation will be created. Pulse duration will be 5÷100 μsec. That will permit us to investigate dynamic characteristics of subcritical assemblies.

Table III. Some data on the samples which will be irradiated in the experimental channels of the thermal subcritical assembly driven by the neutron generator. It is supposed that neutron intensity when operating with tritium target equals to 10^{12} n/s, irradiation time equals to 10^4 s, activity of each sample equals to 10^8 Bk.

№	Sample	σ, barn (therm)	m, g	ε, registration efficiency of the detector	S_{imp} number of registered impulses
1	Sr-90	0,9+0,5 0,014+0,0024	0,00002	0,017	2,6 10^1
2	Tc-99	20+1 22,9±1,3	0,16	0,032	1,7 10^6
3	Sn-126	0,297	0,0952	0,016	9,5 10^5
4	I-129 #	27±2,2	1,53	0,026	5,8 10^9
5	Cs-135	8,7+0,5	2,3	0,021	1,7 10^7
6	Cs-137	0,11±0,033 0,25+0,02	0,000031	0,015	9,5 10^2
7	Np-237 #	σ_c=169+3 σ_f=0,0019±0,003	0,382	0,018	2,9 10^9 & 1,5 10^{10}
8	Am-241	σ_c =832+20 σ_f=3,15	0,00079	0,018	2,5 10^7 & 9,8 10^7
9	Am-243	σ_c =79,3±1,8 σ_f =0,2+0,11	0,0136	0,023	1,4 10^8 & 4,8 10^8

ε - efficiency of the 3 cm HP Ge detector in γ-spectrometer;
- at activity 10^7 Bk; & - number of fission events.

4. CONCLUSION

The experiments on measurements of transmutation rates for some long-lived fission products and minor actinides in the subcritical assembly driven by the neutron generator will be performed. They will allow to make conclusions about trends of subsequent investigations, estimate discrepancies in evaluated nuclear data files for fission products and minor actinides as well as compare results obtained by means of computer codes with experimental data. It will be possible to carry out some experiments for research of peculiarities of dynamics of target/blanket systems driven by high energy accelerators.

REFERENCES

[1] CLOTHET, P., et. al., Report JUL2230, Forschungszentrum Juelich, Germany (1988).
[2] VAN TUYLE, G. J., et al., Report BNL-4666365, Brookhaven National Laboratory, Upton, N.Y., USA (1991).
[3] BARASHENKOV, V., CHIGRINOV, S. E., TONEEV, V. D., Atomnaya Energiya, 37, 475 (1974) (In Russian).

[4] CHIGRINOV, S. E., et al., Hyperfine Interactions, **77**, 144 (1993).
[5] CHIGRINOV, S. E., et al., (Proc. Int. Conf. on Emerging Nucl. Energy Systems (ICENES'93), 1993, Makuhari, Chiba, Japan), (1993) 434.
[6] NELSON, W. R., et al., Report SLAC-265, Stanford Linear Accelerator Center, Stanford, California, USA (1985).
[7] GUDIMA, K. K., MASHNIK, S. G., TONEEV, V. D., Nuclear Physics, A401, (1983) 329.
[8] FERMI, E., Progr. Theoret. Phys., 5, (1950) 1570.
[9] IGNATYUK, A. V., et al., Sov. J. Nuc. Phys., **21**, (1975) 485.
[10] IGNATYUK, A. V., et al., Ibid., **29**, (1979) 875.
[11] FONG, P., Phys. Rev., 135B, (1964) 1338.
[12] ALSMILLER, F. S., et al., Nucl. Sci. Eng., **79**, (1981) 147.
[13] SELEZNEV, E. F., Voprosy Atomnoy Nauki i Tekhniki, Ser. Fizika i Tekhnika Yadernih Reactorov, **6**, (1984) 56 (in Russian).
[14] BRIESMEISTER, J. F., (editor), Report LA-7396-M, Los-Alamos National Laboratory, New Mexico, USA (1986).
[15] ZUCKER, M. S., et al., Nucl. Sci. Eng., **129**, (1998) 180.
DEMENTYEV, A. V., et al., Preprint INR-0880/95, Inst. for Nuclear Research, Russian Academy of Sciences, Moscow, Russia (1995).
[16] TAKADA, H., et al., IAEA-TECDOC-985, (1997) 117.
[17] BRANDT, R., et al., JINR Preprint E1-97-349, Dubna, Russia (1997).
[18] OCHS, M., et al., JINR Preprint E1-99-1, Dubna, Russia (1999).
[19] CHIGRINOV, S. E., RAKHNO, I. L., KIEVITSKAIA, A. I., (Proc. Int. Conf. on the Physics of Nucl. Science and Technology, Long Island, N.Y., USA, 1998), V.2, (1998) 1455.
[20] CHIGRINOV, S. E., et al., (Proc. ADTT'96, Kalmar, Sweden, 1996), V.2, (1996) 737.

DESIGN CHARACTERISTICS OF HYPER SYSTEM

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Abstract

Korea Atomic Energy Research Institute (KAERI) is developing an accelerator driven system called HYPER (Hybrid Power Extraction Reactor) for the transmutation of nuclear waste. HYPER program is within the framework of the national mid and long-term nuclear research plan. KAERI is aiming to develop the system concept and a type of roadmap by the year of 2000. Some major design features of HYPER system have been developed. On-power fueling concepts are employed to account for the rapid drop of core reactivity. In order to increase the proliferation resistance, whole TRU without any actinide separation will be transmuted in the HYPER system. The long-lived fission products such as ^{99}Tc and ^{129}I will be destroyed using the localized thermal neutrons separately in the HYPER. A hollow cylinder-type metal fuel (TRU-Zr) has been chosen because of its high compatibility with pyro-chemical process. Pb-Bi is adopted as a coolant and spallation target material. The heat removal system is designed based on 3 loop concept. 1 GeV 6mA proton beam is to be provided for HYPER. HYPER is to transmute about 380 kg of TRU a year and produce 1000MWth power. The support ratio of HYPER is believed to be 5 ~ 6.

1. INTRODUCTION

Korea Atomic Energy Research (KAERI) is developing the HYPER system for the transmutation of nuclear waste. One of the major features of HYPER system is the subcritical operation that needs an accelerator for the production of seed neutrons.

The HYPER system is being developed within the framework of the national long-term nuclear research plan. The whole development schedule is divided into three phases. The basic concept of the system and the key technical issues are derived in Phase I (1997 - 2000). A couple of experiments will be performed to confirm the key technical issues in Phase II (2001-3). A thermal hydraulic test for the Pb-Bi, an irradiation test for the fuel and a spallation target test are the major experiments that KAERI is considering. In Phase III (2004-2006), a conceptual design for HYPER system will be finished by completing the development of design tools based on the experiments.

2. CORE DESIGN

The HYPER core adopts hexagonal type fuel array to render the core compact and to achieve hard neutron energy spectrum by minimizing neutron moderation [1]. The reference core consists of 186 flow channel, 54 reflector assemblies, 60 shield assemblies, 4 fission product burning locations, and 2 emergency safety units. Liquid Pb-Bi is employed as the coolant material and it is used as the spallation target material as well. The core configuration is shown in Figure 1.

The core is designed to produce 1,000 MW(th) with an average temperature rise of 170°C. The active core height is 1.75 m and the effective core diameter is 3.8 m. Each fuel channel contains 4 unit assemblies of which the length is 43.7 cm. The reflector assemblies filled with liquid Pb-Bi are located at the core perimeter. HT9 shield assemblies are located at the outermost core perimeter to prevent excessive irradiation damage to reactor structures and components surrounding the core. Four fission product assemblies are provided to transmute ^{99}Tc and ^{129}I. In order to achieve the effective neutron spectrum to transmute these fission products, the assemblies have calcium hydride (CaH_2) block in the central region.

Basically, the HYPER core is uranium free. The absence of uranium material (especially ^{238}U) raises two disadvantages in terms of core neutronic behaviors. The first is little Doppler coefficient that contributes to make the fuel temperature coefficient negative. The second is a relatively large reactivity swing in the core. As the fuel burns up, the reactivity inside the core

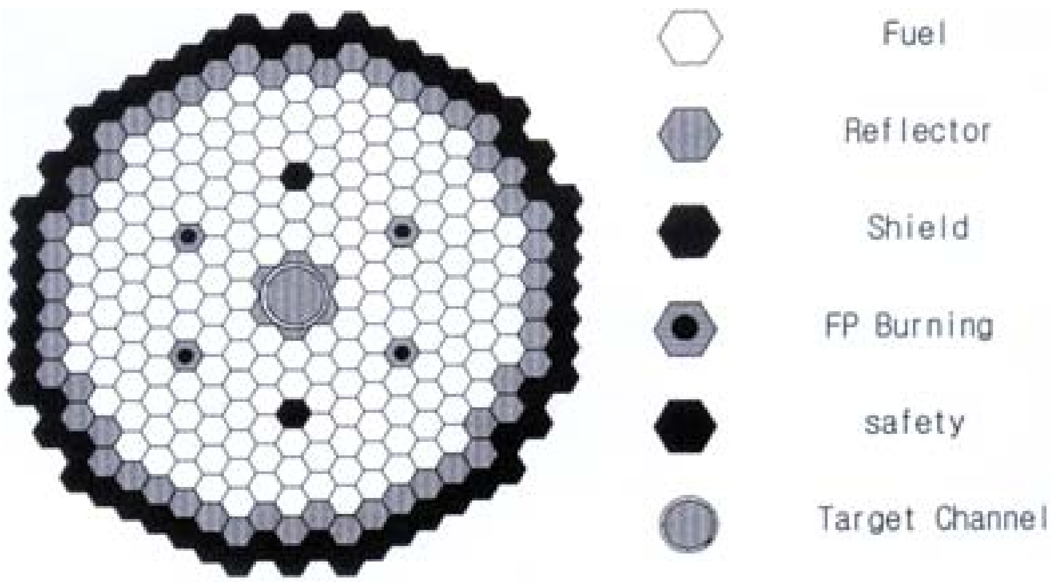

Figure 1. HYPER core configuration.

is reduced and more accelerator power is needed to maintain constant power. However, the reactivity runs down so quickly that the system cannot be operated effectively with a batch-type fuel loading scheme. The simulation shows that k_{eff} drops down from 0.97 to 0.90 in less than 10 months, which means the system needs more than 300% of the designed accelerator power. To avoid such a large amount of reactivity change and minimize the fluctuation of the required accelerator beam power due to the fuel burnup, the HYPER system is to be operated with on-line refueling scheme similar to that of the CANDU reactor system. Detailed engineering design of the refueling system is not set up at this point and an extensive research is now in progress to develop an effective refueling system. In addition, a burnable poison concept is also being considered for the reactivity compensation. To operate the HYPER system on 1,000MWth with k_{eff} of 0.97, ~6mA of 1GeV proton beam is required. HYPER burns (or transmutes) about 380 kg of TRU a year.

The HYPER core is designed to incinerate TRU based on the fuel cycle shown in Figure 2. The average duration time of the fuel assembly is about 1 year. The core is believed to reach an (pseudo) equilibrium condition in terms of neurtonic characteristics after 5~6 year operation. ^{240}Pu becomes the most dominant nuclide at the equilibrium core. Figure 3 and 4 show the K-inf and TRU inventory variations, respectively, as the cycle goes. The main design parameters of the core are described in Table I. The core-averaged neutron flux is about 6x10^{15} neutrons/s-cm^2. The HYPER core has inherently higher power peaking compared to other typical LMRs because of its sub-criticality. In order to reduce the peaking power, the core is divided into three different zones. Figure 5 shows the relative assembly power distribution.

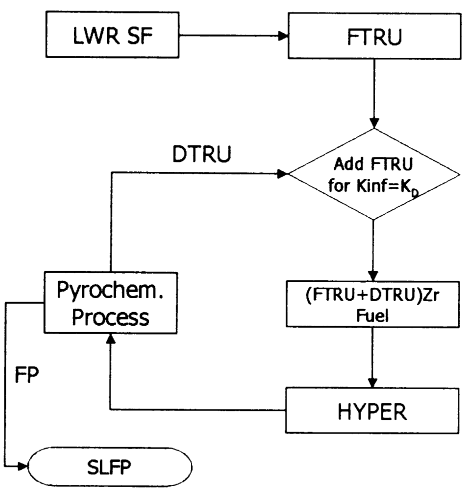

Figure 2. Fuel cycle concept in HYPER system.

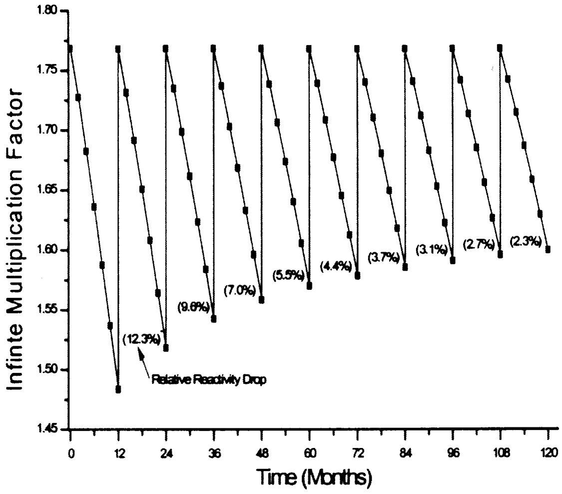

Figure 3. Assembly K-inf variation as cycle goes.

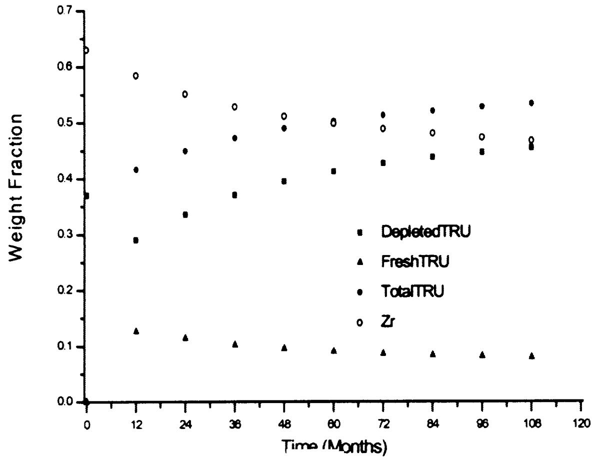

Figure 4. TRU inventory variation.

Table I. Design Parameters of the Core

Parameter [unit]	Values
- Core Thermal Power [MW]	1,000
- Active Core Height [m]	1.75
- Effective Core Diameter [m]	3.8
- Total Fuel Mass [TRU-kg]	2,961
- System Multiplication Factor	0.97
- Accelerator Beam Power [MW]	6
- Ave. Discharge Burnup [%at]	18
- Transmutation Capability [kg/a]	380
- Ave. Number of Fuel Assemblies to be loaded [#/day]	2
- Ave. Linear Power Density [KW/m]	13.5
- Ave. Neutron Energy [keV]	600
- Ave. Neutron Flux [n's/s-cm^2]	6x10^{15}

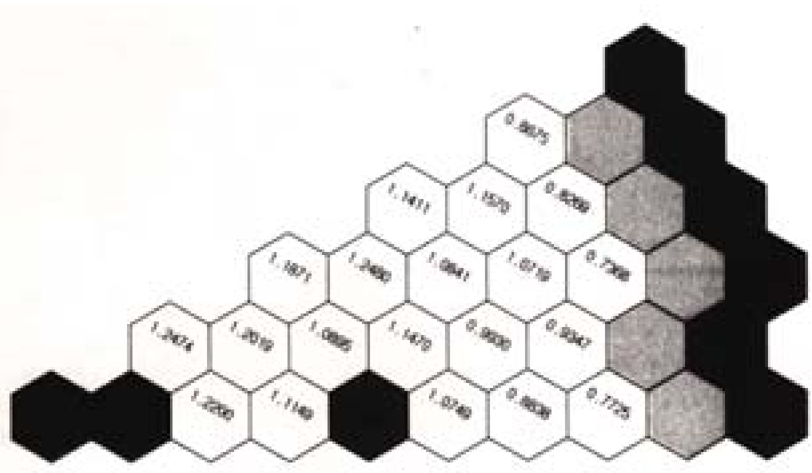

Figure 5. Relative assembly power distribution based on 3 zone concept.

3. FUEL/FP TARGET DESIGN

TRU and Fission Products (FP) are to be loaded into the HYPER system for incineration. They have different loading types and will be loaded into different sites because of their different neutronic characteristics. TRU is loaded as a fuel to drive the system. On the other hand, FP is just a target to be incinerated.

3.1. Fuel design

TRU-Zr alloy is used as the blanket fuel of the HYPER system. Because of the fuel cycle concept, the weight fraction of TRU varies from cycle to cycle. However, the fraction of TRU will be about 50% when the HYPER core reaches an equilibrium state (Figure 4). Table II and Figure 6 describe the fuel design parameters and the fuel assembly configurations, respectively. When using a metallic fuel, a fairly large plenum is usually required at top or bottom side of fuel rod to accommodate the fuel swelling and the fission gas release. However, the plenum at top or bottom side of fuel rod would disturb the core axial power shape very much. In addition, top or bottom side plenum would increase core height. As results, a hollow cylinder type fuel is selected for the HYPER system. The ratio of the central hole volume to the fuel volume is 1.0, which is believed to be reasonable for the system burn-up of 18 at%.

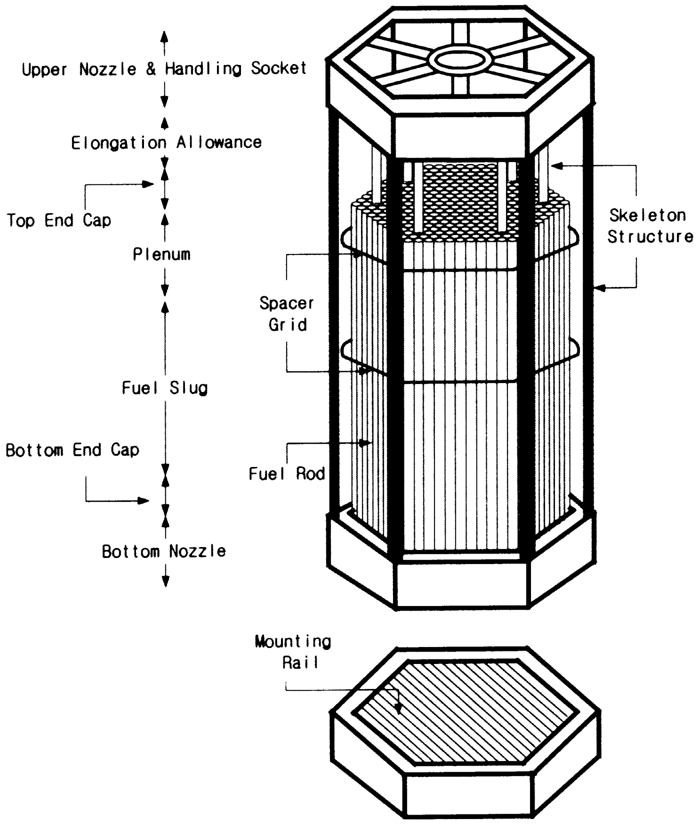

Figure 6. Schematic structure of HYPER fuel assembly.

There are no experiences on TRU-Zr type metal fuel. Most of experimental data and experiences are on U-Zr or U-Pu-Zr types that have Zr fraction no more than 10%. In general, the physical characteristics and irradiation behavior of fuel strongly depend on its fabrication methods. Two types of fabrication technologies (cast alloy and dispersion) are being considered currently. A couple of fuel performance analysis have been done based on the cast alloy type fuel using the MACSIS-H code developed by KAERI.

Figure 7 shows the measured conductivity for U-11.4 wt%Zr, U-40 wt%Zr alloys, and the calculated thermal conductivity of TRU-50 wt%Zr. High Zr fraction degrades the thermal conductivity. In addition, the analysis showed that the burnup reduces the fuel thermal conductivity down to 70% level of fresh fuel's. The temperature difference between the fuel rod centerline and the fuel surface is expected to be about 270K at 3.4 at% burnup.

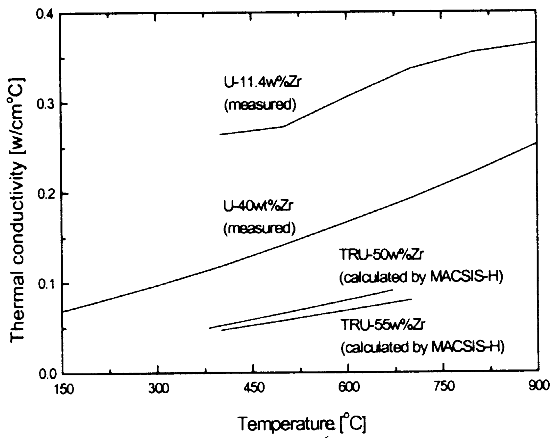

Figure 7. Thermal conductivity variation.

Figure 8 shows the fraction of the fission gas released as a function of fuel burnup for 4 different Zr fractions. More than 90% of fission gases are released at 5 at% burnup. The simulations predict that the fission gas release rate is almost independent of Zr fractions when the Zr weight fraction varies from 45% to 55%. In addition, the calculational results showed that the fuel rod internal pressure caused by fission gas release is linearly proportional to the size of gas plenum.

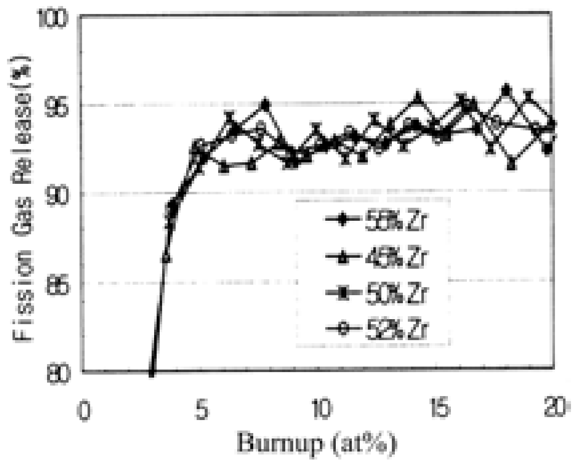

Figure 8. Fission gas release rate.

Figure 9 is the results of the cladding strain analysis. The cladding strains start to increase abruptly after 7at% burnup and exceed the design limit (2% deformation) at 13.5% burnup. The increase of gas plenum size does not make any changes on the cladding strains. It seems that the cladding strains are originated from FCMI (fuel cladding mechanical interaction).

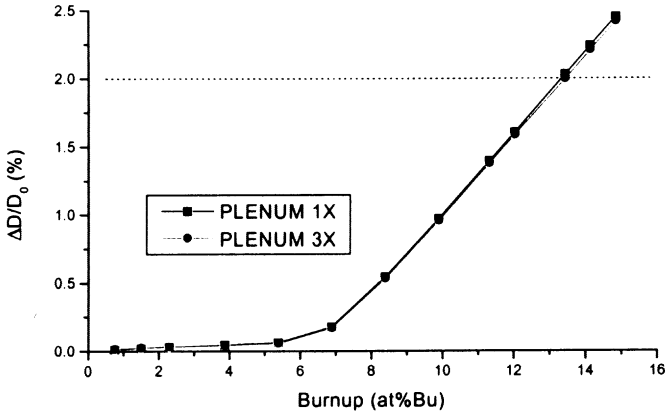

Figure 9. Rod deformation rate as fuel burns up.

Table II. Design Parameters of Fuel

Parameter	Values
Assembly	
- Ass. Pitch (cm)	19.96
- Flow Tube Outer Surface Flat-to-Flat Distance (cm)	19.52
- Tube Thickness (cm)	0.3556
- Tube Material	HT-9
- Rods per Assembly	169
Fuel Rod	
- Composition (TRU-Zr)	0.45-0.55
- Active Height (cm)	30.0
- Outer Diameter (cm)	1.034
- Pitch/Diameter	1.5
- Cladding Material	HT-9
- Cladding Thickness (cm)	0.055
- Fuel Meat Outer Dia. (cm)	0.8
- Central Plenum Dia. (cm)	0.5

3.2 FP Target Design

^{99}Tc and ^{129}I are to be transmuted among long lived fission products. They will be irradiated in a special region of the reactor core (thermal neutron region). However, the target forms are different for Tc and Iodine, respectively. Many researches have been done on the selection of suitable target forms for Tc and Iodine [2]. The preliminary results of these studies have shown that pure metallic form is the most desirable one for the incineration of ^{99}Tc: a fabrication route for casting the Technetium metal has been developed and irradiation experiments did not show any evidence for the swelling or disintegration of the metal. On the other hand, an elemental form was found not to be acceptable for Iodine because of its volatility and chemical reactivity. Thus, metal iodides are being considered. Sodium iodide (NaI) is expected to be the best candidate so far [3].

Figure 10 shows the cross-sectional view of the FP target assembly concept. A calcium hydride block is installed in the central region of the target assembly. The fission products having relatively high absorption cross section in thermal energy range are almost transparent to the incoming fast neutrons but strong absorbers to the outgoing thermal neutrons. This kind of configuration is devised to reduce the power peaking increment in the fuel assemblies surrounding the fission product target assembly. In addition, Tc and Iodine targets are designed to have different configurations for the maximization of transmutation amount.

4. COOLING SYSTEM DESIGN

Thermal efficiency of power cycle strongly depends on the temperature at which the heat is supplied by the primary to secondary coolant. It is obvious that coolant temperature should be set as high as possible for high efficiency. However, mechanical and corrosion characteristics of structural materials set the upper limit. According to the Russian results [4], the maximum allowable temperature of Pb-Bi coolant is approximately 650 ^{o}C. The lower limit of the coolant temperature can be started from the Pb-Bi melting point, 125 ^{o}C. For safe operation, Pb-Bi temperature must be sufficiently above 125 ^{o}C. Therefore 125 and 650 ^{o}C can be the basic temperature limits of Pb-Bi coolant. The core inlet and outlet temperature of Pb-Bi coolant are determined to be 340 and 510^{o}C, respectively based on the Russian design experience [5]. This temperature range marginally satisfies the basic temperature limits. Resulting core flow rate in order to cool 1000MW thermal power is 46569.0 kg/s.

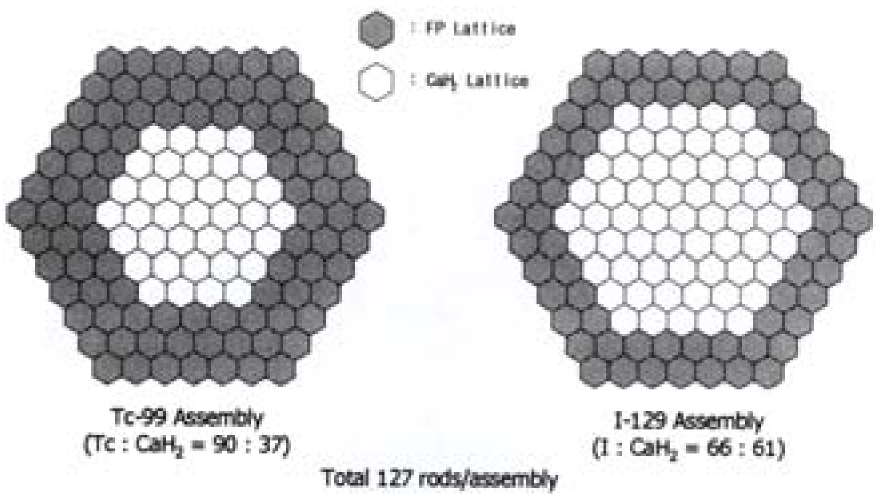

Figure 10. Fission product target assembly.

Coolant velocity of primary cooling system can also cause a design constraint. Coolant velocity affects the integrity of structural materials and the pumping load. The primary cooling system of HYPER should be designed with low coolant velocity as long as it can satisfy another design requirements. Since Pb-Bi does not significantly absorb or moderate neutrons, it allows the use of a loose lattice, which favors lower coolant velocity. The P/D (Pitch-to-Diameter) of HYPER's core is chosen to be 1.5 and the corresponding Pb-Bi velocity is 1.1 m/s, which is a relatively lower coolant velocity compared to the value of typical power reactors. Instead of wire spacer commonly used for tight lattice, grid spacers are suitable to ensure proper separation of the fuel rod. Although the Pb-Bi velocity within the fuel channel is sufficiently reduced, roughly estimated pressure loss across the reactor vessel is similar to that of typical power reactors. This is mainly from the heavy characteristics of Pb-Bi density. Complex geometry of inlet and outlet components of the core also contributes to the large amount of pressure loss. Therefore, it is expected that natural circulation does not play an important role to cool the HYPER's core under normal operating conditions.

Loop type configuration is selected for the preliminary design of HYPER system and three loops system is chosen as an optimal system for HYPER. The number of loop is determined by considering the coolant velocity and pressure drop across the loop. The mass flow rate flowing into a heat exchanger is 15523.0 kg/s. It is possible to eliminate intermediate heat transport system with Pb-Bi coolant. Steam cycle is adopted for HYPER due to its long and successful experience. Figure 11. shows the overall view of the cooling system of HYPER.

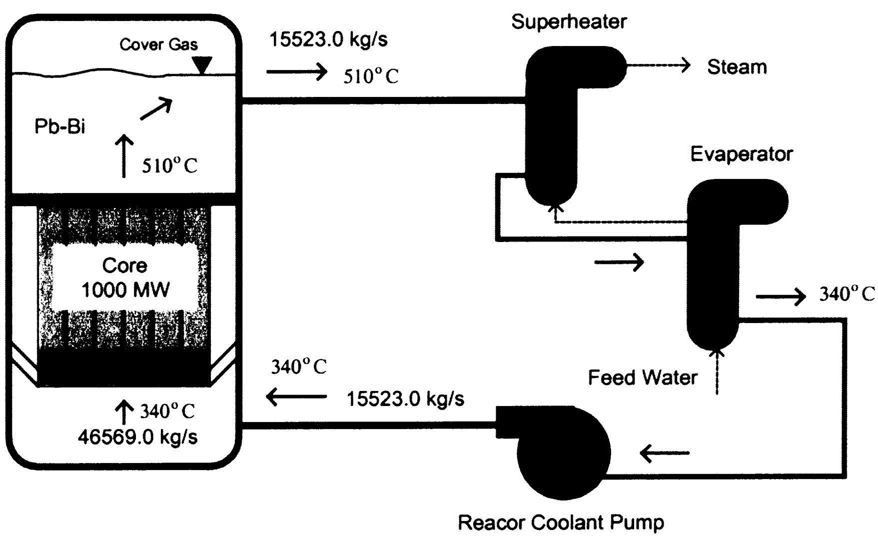

Figure 11. Overview of cooling system.

5. BEAM TARGET

The core coolant is also used as the spallation target. Figure 12 shows the structure of target channel and beam window system. Pb-Bi comes from the bottom of the reactor and encounters beam window before going out of the top of the reactor. The double beam windows are adopted to enhance the window cooling by forcing Pb-Bi to flow between two windows.

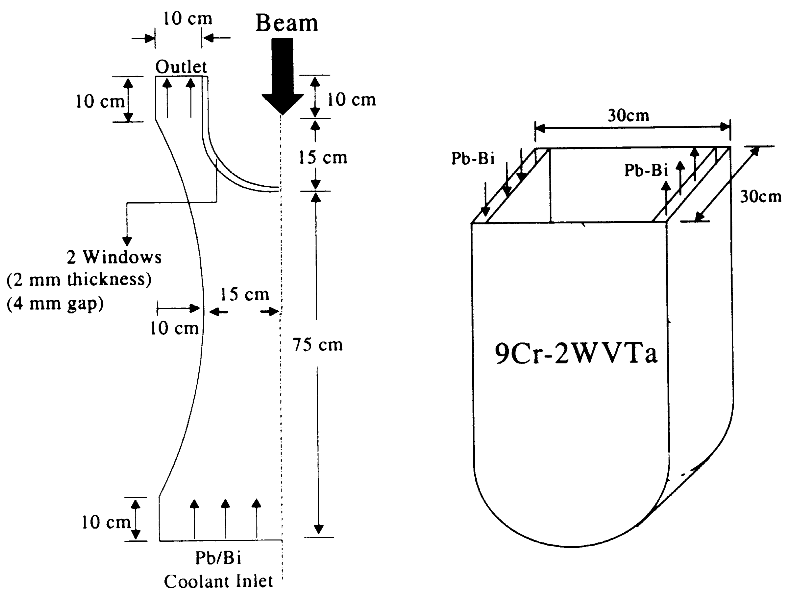

Figure 12. Target and beam window.

There are some design goals for the stable and safe operation of the target and reasonable lifetime of the beam window. We set the maximum allowable temperature and stress of beam window to be 700°C and 200MPa respectively. The temperature of Pb-Bi is set to be less than 600°C and lifetime of the beam window is set to be 1 year. We performed calculations to obtain appropriate design parameters and operating conditions for achieving the design goals. We used FLUENT code for temperature calculations and ANSYS code for stress calculations. We fixed beam to have parabolic density distribution with circular shape of 10 cm diameter. The inlet temperature and velocity of Pb-Bi coming from the bottom of the reactor are given 340°C and 2m/s respectively. The temperature of inlet Pb-Bi flowing between two windows is also given 340°C. Then we varied the inlet velocity of Pb-Bi between two windows and beam current.

The temperature and stress calculations are still under progress. Our goal is to estimate the maximum current, which does not exceed our design limits. We will change the beam and cooling conditions to achieve as high current as possible. The factors affecting lifetime of the beam window are corrosion due to Pb-Bi and radiation damage. We expect the effect of radiation damage is dominant in deciding the window lifetime. To predict the lifetime, first we should know the amount of radiation damage and then how the properties of the window material change as the radiation damage increases. Therefore we calculated dpa (displacements per atom) and He production rates by using LCS (LAHET Code System). When beam is 20 mA, parabolic distribution and circular shape with the diameter of 10 cm, dpa is 380 per year at the window center. About 20% of total dpa are caused by neutrons and the other by protons. Most of He production is caused by protons and calculated to be 60000appm per year at the window center. In order to check if the window can be used for 1 year without exchange, we are studying how the window properties change for 1 year based on available experimental data related to radiation damage.

6. SUMMARY

The design goal of the HYPER system is to transmute the nuclear waste. Many considerations have been given to maximize the incineration capability as well as the power production efficiency. HYPER core does not have any fertile materials. On-line refueling concepts are being considered to compensate for the sharp reactivity drop. In addition, a type of burnable poison is also being investigated. No experimental data are available for TRU-Zr fuel. In addition, the development of Pb-Bi coolant/target technologies is also very challengeable work. HYPER system concept is to be finalized and Key technical issues related with TRU-Zr fuel, Pb-Bi coolant/target are to be derived by the end of 2000. The experiments to solve the key technical issues will be performed from 2001 to 2003 and the conceptual design will be completed by 2006.

ACKNOWLEDGEMENT

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REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Accelerator Driven Systems: Energy Generation and Transmutation of Nuclear Waste, Status report, IAEA-TECDOC-985, Vienna (1997).

[2] WOOTAN, D. W., et al., "Irradiation test of Tc-99 and I-129 transmutation in the Fast Flux Test Facility", ANS Transaction Vol. 64, (1991) 125.

[3] OECD/NEA, "Report for the Partitioning and Transmutation", to be published.

[4] YACHMENYOV, G. S., et al., "Problems of corrosion of constructional materials in lead-bismuth coolant", HLMC (Heavy Liquid Metal Coolants) '98 Conference, 1998.

[5] GROMOV, B. F., et al., "Heavy coolant fast neutron reactor BRUS-150 for minor actinides burning and U-233 build-up," International Conf. On Reactor Physics and Reactor Computations, 1994.

THE CONCEPT OF NUCLEAR INCINERATION OF PWR SPENT FUEL IN A TRANSMUTER WITH LIQUID FUEL AS APPLIED IN THE CZECH REPUBLIC PROGRAMME

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Abstract

Introductory studies made in the Czech Republic during the years 1994 and 1995 and close contacts with other laboratories, the national project was started in 1996. The four major nuclear research institutions of the country (Nuclear Physics Institute of Czech Academy of Sciences, Nuclear Research Institute Rez plc, R&D Basis of the SKODA Nuclear Machinery Ltd. and Faculty of Nuclear Sciences and Physical Engineering of the Czech Technical University) formed a consortium focused on an adoption of the world - wide experience and a development of a national project of a transmutation technology (experimental transmuter LA-0) or an efficient participation in the international effort in that field. A brief description of the results obtained so far is given in this paper.

1. INTRODUCTION

After introductory studies made in the Czech Republic during the years 1994 and 1995 (including study stay in LANL) and close contacts with other laboratories and centres, the more realistic national project started to be developed in 1996. The four major nuclear research institutions of the country

- Nuclear Physics Institute of Czech Academy of Sciences,
- Nuclear Research Institute Rez plc,
- R&D Basis of the SKODA Nuclear Machinery Ltd. and
- Faculty of Nuclear Sciences and Physical Engineering of the Czech Technical University

formed a consortium focused on an adoption of the world - wide experience and a development of a national project of a transmutation technology (experimental transmuter LA-0) or an efficient participation in the international effort in that field. A brief description of the results obtained so far is given in the following chapters.

Because of the LA-0 transmuter concept of subcritical reactor with liquid fuel based on molten fluorides driven by an external neutron source has been adopted; the contributions will be focused on those three regions.

The first is devoted to the problem of a suitable neutron source, the second to a pre-conceptual design of a blanket being convenient for burning of actinides contained in spent fuel from PWRs and simultaneously for nuclear incineration of long - lived fission products as well as other radionuclides from radioactive waste produced during the operation of NPP with PWRs. The third chapter will be describing the experience from a specific field of dry (fluorine) processing of spent fuel and a preparation of liquid fuel in the form of molten fluorides for the transmuter LA-0.

2. NEUTRON SOURCES FOR ADTT BASED ON MEDIUM-ENERGY ACCELERATORS

2.1. Motivation

In so far presented concepts of strongly subcritical AD transmuters the high-energy (GeV)-high-power (100 MW) future proton linacs and spallation reactions are mostly considered as an effective base for required strength (10^{19} n/s) of an external neutron source. Nevertheless, the length (1000 m) and cost (1B US$) of these facilities are usually used to be assigned incommensurably large [1] with respect to expected cost and dimensions of a reactor-transmuter itself, especially if local transmuters (i.e. transmuters, located on the territory of the nuclear-power plant) are considered. Therefore, in some studies a distinct effort is devoted to conceptual development of a transmuter system with lower subcriticality. In a Russian project SCANUR [2] the subcritical molten salt reactor with the cascade multiplication scheme is proposed. For this assembly a substantially lower strength of $3\text{x}10^{16}$n/s is required for the external neutron source. In this "burner"-reactor scheme, the external neutrons from bremstrahlung reaction are provided by electron linacs (100 MeV, 5MW of the beam power, five simultaneously operated facilities, molten salt target) with overall beam power of 25 MW for a 1250 MW transmuter.

At present, the most powerful electron accelerator-based neutron source ORELA (180 MeV/60 kW maximum) located in Oak Ridge National Laboratory provides 10^{14} n/s, so only future generation of electron linacs will capable to reach source power, required in the SCANUR concept. Note, that neutron intensities, needed for SCANUR, are presently available only at two facilities (800 MeV/ MW proton linac, Los Alamos and 590 MeV/0.9 MW cyclotron facility at PSI, Villigen), both being presently the most powerful spallation neutron sources in the world.

As an alternative, the break-up of light nuclei at medium incident energies could also be considered for the ADTT external neutron source, taking into account, (i) that the energy cost of produced neutron lies inside the range, calculated for bremstrahlung and spallation reactions. Further, (ii) fast neutrons emitted from these reactions are strongly forward-directed, which - in principle - could enable to separate neutron-producing target from the blanket, avoiding thus the problem with the beam-power dissipation inside the core and finally, (iii) for some type of targets the beam-window problem could be avoided (liquid jet target).

Of course, none of fast neutron sources based on break-up reactions has yet demonstrated the source strength beyond the value of 10^{14}n/s (extracted beam power in 10 kW range), in spite of a wide class of medium-energy cyclotrons for medical employment (neutron therapy) has been developed. Although the cryogenic technology, (approved as effective technical base of compact medical cyclotrons) offers to reach further beam-power improvements for cyclotrons, the Alvarez linear accelerators with the RFQ (radio-frequency quadruple) injector are accepted to be more promising for the significant increase of beam intensity. Within this technology, the reported conceptual design [3] of the International Fusion Material Test Facility (IFMIF) presents most powerful fast-neutron source (up to 10^{17} n/s), which is based on two linear deuteron accelerators (35-40 MeV/5 MW, 40 m length, evaluated cost of 0.12B US$ each) and liquid lithium jet target. Being highly developed, the IFMIF neutron source is expected to be ready before 2010 [4].

Therefore, it seems appropriate to enlarge the conceptual ADTT studies considering the low-power local transmuters, based on medium-energy accelerators. The processing (i.e. the database and computer codes) of neutron generation and the target-blanket neutronics in energy range from 20 to 100 MeV is of the primary interest not only within the task of low-parameter transmuters. Such request is strongly stressed for ADTT conceptual studies altogether [5].

2.2. Neutronic studies for low-parameter transmuters at NPI

The neutron facility NGl (see Fig. 1) on external beam line of the NPI isochronous cyclotron U-120M (variable energy up to 35 and 17 MeV of protons and deuterons, respectively) was based on the d (17MeV)+Be (thick target) reaction and originally developed for military oriented biophysical research. The design provides a well-collimated, highly homogenous neutron beam with the averaged energy of 6 MeV and the fluency rate up to 10^8 n/cm^2 s at 70 cm distance from the target. The same version of NG1 was used to the study of the transmission of fast neutrons through tungsten target (see chapter 2 above).

The upgrading of NG1 facility was then dedicated to the extension of source characteristics requested by the ADTT-database program. With this aim, the fast neutron spectrometer consisting of the organic scintillators (NE 213, stilbene), two-dimensional n-gamma discrimination technique and the PC data-acquisition system was developed. In the research program, the spectral yields of fast neutrons from proton and deuteron induced break-up reactions on various light-nuclei targets (H_2, D_2, D_2O, He and Be) were collected [5]. Employing different technique (gaseous, liquid or solid) of thin targets and the variable energy of incident beam, the monoenergetic neutrons with energies approximately from 4 to 30 MeV and the flux up to 10^8 n/sr.s can now be provided by NG1 facility.

New facility with improved neutron source strength (NG2) is now projected at NPI to take advantage of higher-power mode of external beams (foil-stripping technique for extraction of accelerated negative ions), recently implemented on the U-120 M cyclotron. Owing to this technique of beam extraction, the proton- and deuteron-beams of power up to kW are routinely extracted on targets for radionuclide production. Preliminary analysis showed, that the neutron flux up to $3x10^{12}$ n/sr.s and the averaged neutron energy about 15 MeV could be

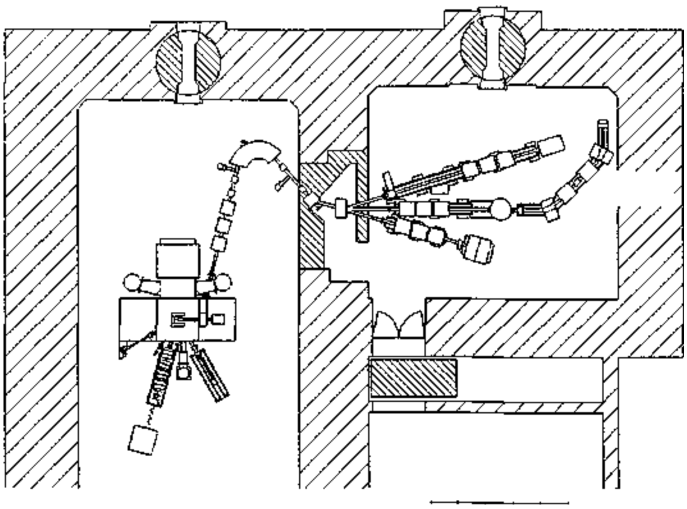


Figure 1. The neutron facility NG1 on external beam line of the isochronous cyclotron U-120M.

achieved for the NG2 facility from deuterium thick target (heavy water or gas), installed on beam-line of foil-extractor and irradiated by 35 MeV protons.

In the conceptual form, the program of neutronic studies is now formulated for the target/blanket system, which consists of the high-power NG2 external neutron source and the elementary module LA-0 of proposed molten-salt transmuter (see following chapter).

3. THE CONCEPT OF A TRANSMUTER WITH LIQUID FUEL

3.1. Introduction

There are principle drawbacks of any kind of solid nuclear fuel listed and analysed in the first part of this chapter. One of the primary results of the analyses performed shows that the solid fuel concept, which was to certain degree advantageous in the first periods of a nuclear reactor development and operation, has guided this branch of a utilization of atomic nucleus energy to a death end (not having been able to solve principle problems of the corresponding fuel cycle in an acceptable way). On the basis of this, the liquid fuel concept and its benefits are introduced and briefly described in the following part of the chapter.

As one of the first realistic attempts to utilize the advantages of liquid fuel, the reactor/blanket system with molten fluoride salts in the role of fuel and coolant simultaneously, as incorporated in the accelerator-driven transmutation technology (ADTT) being proposed in [6], will be studied both theoretically and experimentally. There is a preliminary design concept of an experimental assembly LA-0 briefly introduced in the following paragraph, which is under preparation in the Czech Republic for such a project.

Finally, there will be another very promising concept [9, 10] of a small low power ADTT system introduced, which is characterized by a high level of safety and economical efficiency. This subcritical system with liquid fuel driven by a linear electron accelerator represents an additional element -nuclear incinerator- to the nuclear power complex (based upon thermal and fast critical power reactors) making the whole complex acceptable and simultaneously giving an alternative also very highly acceptable nuclear source of energy and even other products (e.g. radionuclides, etc.). In the conclusion, the overall survey of principal benefits, which may be expected by introducing liquid nuclear fuel in nuclear power and research reactor systems, is given and critically analyzed. The other comparably important principles (e.g. the general subcriticality of reactor systems principle) are mentioned which being applied in the nearest future may form a basis for an absolutely new nuclear reactor concept and a new nuclear power era at all.

In spite of the fact that all what is following is well known it seems to be worth to remind it in the new circumstances of nuclear power at the end of the 20^{th} century while starting to search new nuclear energy systems and fuel cycle options for the 21^{st} century. Since the discovery of the reaction of atomic nucleus fission, the main goal of all efforts was to utilize it for an energy generation. As one of the most important conditions for an efficient achievement of this goal self-sustaining of fission chain reaction was demanded in an assembly containing fissionable nuclei of nuclear fuel without an external source of neutrons. If this was reached, the assembly was defined as being critical. Let us note that it was by definition (theoretically) critical on prompt neutrons released, immediately, from fission reactions only. Very early, it was observed experimentally that the assembly reaching criticality is in fact very slightly subcritical on prompt neutrons and that there is a not very strong natural source of delayed neutrons originated from radioactive decay of some of the fission products always added (which, fortunately, allowed easier control of the system).

At the early stages, reaching criticality was one of the most difficult tasks and all the effort and ideas had been devoted to this aim. The reason was that there were only small amounts of fissionable materials available in those times in the form of the low (0.7%) content of ^{235}U in natural uranium. Therefore, solid phase metallic uranium with highest as possible density was used and in the form of blocks with a specifically defined size arranged in a heterogeneous lattice filled in by a solid (graphite) or liquid (heavy water) moderator with a certain pitch determined by optimal neutronic conditions. This arrangement has remained nearly exclusive one being used even in latter systems with fuels enriched by ^{235}U content up to much higher levels than the content of natural composition of uranium. The reasons had been of different nature, however, the designs have mostly started from what became already an approved conventional principle - solid fuel blocks in a heterogeneous lattice - which has been kept even in the case of pure or high enriched fuel in a fast neutron system without moderator.

One of the next consequences of the adoption of the solid fuel concept has been a type of control system which has been mostly applied for a short term control of nuclear reactors - the concept of solid absorbers - and what is more the concept of a negative neutron source (neutron poison) at all. This, and a number of other consequences, can be traced to start all from the initial tension in neutron economy when the principle of a self-sustaining fission chain reaction and consequently the concept of a critical reactor have been adopted. They all begin to form a magic circle of convention in which the short term and finally even long term operational behaviour of nuclear, namely power, reactors is being imprisoned and limited in its ability to give a positive and broadly acceptable development. Let us explain this thesis in some following more see-through examples.

The adoption of the solid fuel concept leads to the principal necessity to keep the fuel blocks at a certain position in the reactor core for a shorter or longer period of time. This in-core residential time is especially long in power systems where at least a quasi-continuous exchange of fuel would be very complicated and expensive. Therefore, the following very inconvenient consequence arises: the whole time, the block of solid fuel remains at a certain position in the reactor core, there are fission fragments and by neutron capture induced radionuclides (let us call them altogether products) being accumulated in the volume of the fuel block. There are several secondary consequences caused by this fact which contribute to the above mentioned magic circle forming:

(1) reactivity margin for a short term as well as long term negative influence of the increasingly accumulated products has to be applied which has to be compensated by another artificial negative source of neutrons. It has in principle a consequence in greater amount of fuel being present in the core than really necessary for a demanded power and then the more products including actinides is generated;
(2) the original fuel is finally so heavily poisoned by the products that it cannot keep the selfsustained fission chain reaction any more and a further operation of the reactor under original conditions is impossible. There is a principle change in the operation and structure of the reactor unavoidable, which means an outage, and exchange of at least a part of the original fuel charge;
(3) the most controversial problem what to do with spent solid fuel arises and a vicious circle has been closed or a solid fuel concept "trap" snapped.

The above briefly described solid fuel concept shows its most important and sensitive drawbacks:

(1) continuous accumulation of products during the whole residential time of fuel blocks in the core,

(2) following necessity to stop the operation, discharge spent fuel and store it for a necessary period of time (in order of magnitude of years until it reaches a desirably low level of radioactivity) in a specific storage,

(3) the last and the most difficult drawback is the need of an optimal decision of the following destiny of spent fuel.

Up to now, the only two possible solutions were developed either to reprocess (chemically) it and to prepare next generation of solid fuel (it means with basically the same class of drawbacks) or to dispose it in a depository of a corresponding quality (which sometimes is called repository because a possible reuse of the disposed product is supposed). In the former case mostly chemical methods and processes are applied. In the latter, a lot of branches are involved; however, nearly all of them are of a classical (non-nuclear) nature. The only nuclear process, which is employed, is the natural radioactive decay.

This fact contains one very controversial principle or better say a violence of a basic principle which can be described as follows: The energy generation in nuclear reactors utilizes enforced nuclear process which are simultaneously producing products or nuclear waste (including secondary raw materials e.g. actinides). The treatment of the products needs to apply an adequate technology in an adequate scale. This principle has not been applied and fulfilled in those so far developed and designed systems for spent solid fuel management. There is an adequate technology which only one can utilize nuclear processes and which can transfer the high level and long-lived radionuclides towards short-lived or even stable nuclides-transmutation technology performed in a suitable nuclear reactor device and combined with a continuous separation of certain components of its core or reprocessing of the reactor fuel as to avoid the consequent induction of radioactivity by neutron irradiation of stable and short-lived nuclides. One of the principle concepts allowing to reach such a technology in an industrial scale is the concept of liquid nuclear fuel.

3.2. Liquid fuel concept for neutron source-driven transmutation technology (NSDTT)

3.2.1. Molten fluoride salt fuel for neutron source-driven transmutation technology

The concept of a neutron source-driven subcritical blanket for a nuclear incineration of nuclear waste is well known for a several recent years [6]. Let us recall at least very briefly the main features of the last developed version of this concept and let us show a part of a proposed research program to approve its ability for an efficient realization in the industrial scale.

The fuel material is in the form of the fluoride salt AcF4 dissolved in a molten salt carrier whose composition is a mixture of ^{7}LiF and $^{9}BeF_2$. The carrier's melting point and operating temperature are about 500°C and 650°C, respectively. The molten salt flows over either the outside of a close-packed set of cylindrical high-purity graphite blocks or inside cylindrical channels coaxially situated in e.g. hexagonal graphite blocks.

There has been an experimental research system designed by the authors preconceptually in [7, 8] which should be developed and realised in the Nuclear Research Institute Rez plc. in the Czech Republic. The final purpose of the system would be an experimental testing of a given type of transmuter reactor/blanket core neutronics and possibly also other physical and technological characteristics and properties including time behaviour. For the very first stage, the following scheme can be applied which will allow to reach the first results very cheaply and relatively soon. There can be an elementary, however, a sufficiently representative sample of the investigated reactor blanket lattice inserted into an existing experimental reactor core serving like a driver and the basic set of its characteristics can be experimentally measured and verified. The suitable experimental reactor can be e.g. the experimental reactors LR-0 (full-scale core modelling in Nuclear Research Institute Rez) or VR-1 (training reactor at Czech Technical University Praha - Fig. 2.) which have been successfully operated for core analyses of thermal reactors since 1982 and 1990, respectively.

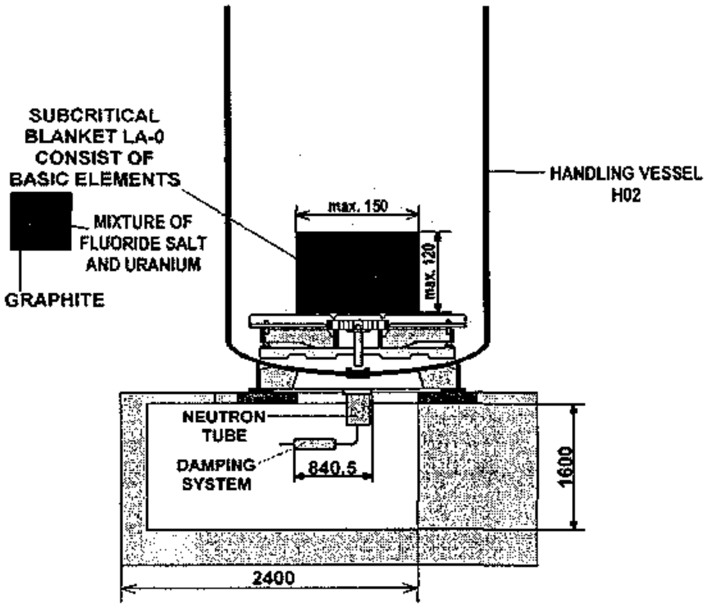


Figure 2. Subcritical blanket LA-0 with a neutron generator in the VR-1 experimental reactor.

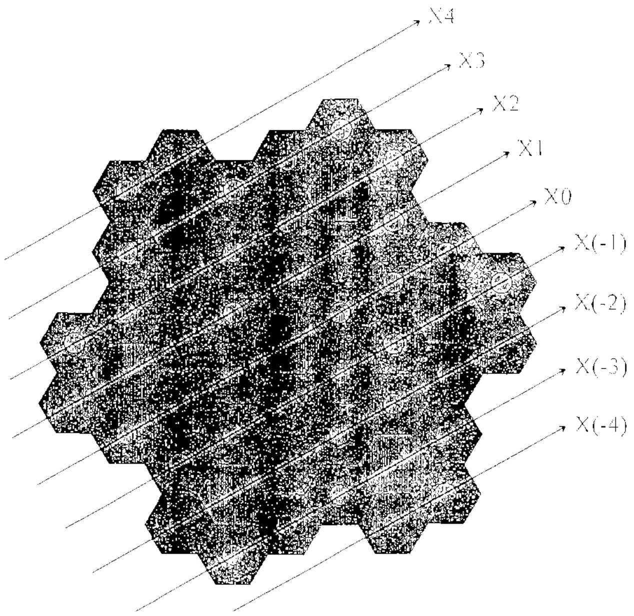
3.2.2. Low power ADTT system

The molten salt reactors (MSRs) with the continuous control of nuclide composition almost do not require an initial reactivity margin. In such reactors, subcriticality may be reduced up to the minimum value ß where ß is the effective delayed neutron fraction. However, with such a small subcriticality and in view of available uncertainties in nuclear data and nuclide concentrations, the difference between subcritical and critical MSR in a great extent disappears: in both cases the nuclear safety is ensured by the large negative temperature reactivity effect. The deeper subcriticality is of course substantiated by the fact that under such conditions we exclude the necessity to control a reactor-burner in a dynamic mode, that is a bit difficult and poorly known.

In this case, the e.g. accelerator - driven positive source performs only one of the usual functions - the function of a reactor control system without inertia, an alternative to, up to now usually used as reactor control organs, negative sources like e.g. absorbers or decreasing of the dimensions of the system, etc. The high level parameter proton accelerator with its all disadvantages (like e.g. the length - 1 km, the investments - U$ 1 billion, etc.) having been applied e.g. in the Los Alamos concept is not necessary more in the system and a low level parameter accelerator can be employed.

3.2.3. A blanket concept of a transmutes for PWR spent fuel incineration

There has been a convenient blanket concept for an efficient nuclear incineration of PWR spent fuel developed as a combination of those two ideas described in two paragraphs above. The Fig. 3 illustrates the concept where two zones are indicated, one under-moderated and thus better equipped for actinides burning and the second well-moderated and thus more convenient for fission products incineration. There might be an analogous axial profile of the blanket core adopted, e.g. the above mentioned structure is applied for the upper half while the lower half is formed by a homogeneous non-moderated zone with even harder neutron spectrum convenient for an efficient burning of actinides. These systems and modifications has been still a subject of intensive, namely computational, studies and an optimal version will be employed in an experimental transmuter LA-0 system finally.



Graphite

Fuel (LiF 70% +BeF_2 20% +UF_4 10%)

Figure 3. Two zone blanket concept for PWR spent fue1 efficient nuclear incineration.

4. CHEMICAL PROBLEMS OF SPENT NUCLEAR FUEL TRANSMUTATION

4.1. Introduction

In the beginning of the 90s, there has been a transmutation process [11 14] proposed in the Los Alamos National Laboratory which enables burning of actinides contained in spent nuclear fuel and their decomposition to fission products in a subcritical reactor blanket driven by an external neutron source (to be sufficiently strong the spallation reaction initiated by a beam of highly accelerated charged particles e.g. protons by a high power linear accelerator being operated in LANL was suggested as the external source of neutrons). Thus, it would make possible to transform spent fuel from commercial reactors as well as from military production (including plutonium warheads) to short-lived fission products.

The principle of the chemical treatment applied in those projects has been based on the experience obtained in the sixties during the operation of nuclear reactors with molten fluorides in the ORNL [15-20]. Two types of reactors have been operated there: Molten Salt Reactor (MSR) and Molten Salt Breeder Reactor (MSBR). Both reactors have originally been designed for the thorium fuel cycle, where fissionable ^{233}U is formed from ^{232}Th. However, during the experimental verification ^{235}U and ^{239}Pu have also been used. There was a mixture of BeF_2, LiF and ThF_4 used as a carrier for both the fissionable and breeding components. A pump has circulated molten salt from the graphite-moderated reactor via a heat exchanger back to the reactor. An original process for continuous removal of the melt was worked-out. The melt was processed in order to obtain ^{233}U produced and to separate fission products. The purified molten salt together with ^{233}U obtained was returned to the reactor.

The chain fission reaction was initiated with ^{235}U, the use of ^{239}Pu has also been experimentally verified. Circulating a melt composed of LiF, BeF_2, ThF_4, UF_4 or PuF_3 has served as a basic load of the reactor where the fission reaction proceeded. At the same time, it served also as a heat-exchanging medium. The temperature of the salt in the reactor was 500 - 700°C. In the heat exchanger, the molten salt was cooled down from 700°C to 550 - 500°C giving heat to the secondary circuit. A mixture of molten $NaBF_4$ and NaF was selected as a coolant circulating in the cooling circuit where the coolant is not exposed to radiation and neutron flux and, therefore, cheaper material of a lower melting point could be used. Metallic parts of the reactor and of the equipment for the salt treatment being in contact with molten salts were produced from a hastelloy N type material. Its main component is nickel containing approximately 16% Mo, 7% Cr, 4% Fe and 0,05% C. This material proved fully satisfactory and did not show corrosion or radiation damage during three years of operation.

4.2. Chemical processes taking place during the isolation of U, Pa and FPs from the MSR and MSBR type reactors [21 24]

The operation of a reactor with the fluoride molten salt needs a continuous removal of the melt. The aim is to reprocess the melt, i.e., to obtain uranium and protactinium and to separate fission products. For the primary separation of uranium elementary fluorine is used passing through the molten salt and escaping in the form of UF_6 together with some volatile fluorides of fission products, such as MoF_6 or TcF_6.

Metallic bismuth having a low melting point of 271°C is used for extraction and reducing extraction. It is immiscible with basic components of the molten halide mixtures containing fluorides, chlorides and bromides and has a negligible vapour tension in the range of the temperatures used. Further, bismuth dissolves some metals such as lithium, thorium, uranium, protactinium and rare earth elements. Dissolved lithium is able (under the given temperatures) to reduce fluorides to a metal according to the general equation

$$MF + nLi(Bi) = M(Bi) + LiF.$$

In addition, it has been found that after the reduction of rare earths it is possible to extract them selectively from bismuth to LiCl or LiBr.
Molten salt composed of LiF 72%, BeF_2 16%, and ThF_4 12% contains (at a continuous removal of 0.3 mole % of UF_4) approximately 0.0035 mole % of PaF_4. About 99% of uranium

are separated from the salt by fluorination. Remaining uranium and protactinium in contact with liquid bismuth and dissolved lithium (when the salt is passing through the counter-current extractor) are going to the metal. Metals dissolved in bismuth are converted to non-volatile fluorides by passing-through hydrogen fluoride. The fluorides formed are mechanically segregated. Technological scheme has been verified at a laboratory scale and recommended as a part of the 1000 MW(e) reactor.

4.3. Chemical processes connected with reprocessing of spent fuel in ADS

General scheme of the whole process supposed for the basic types of the processed material is given in Figure 4.

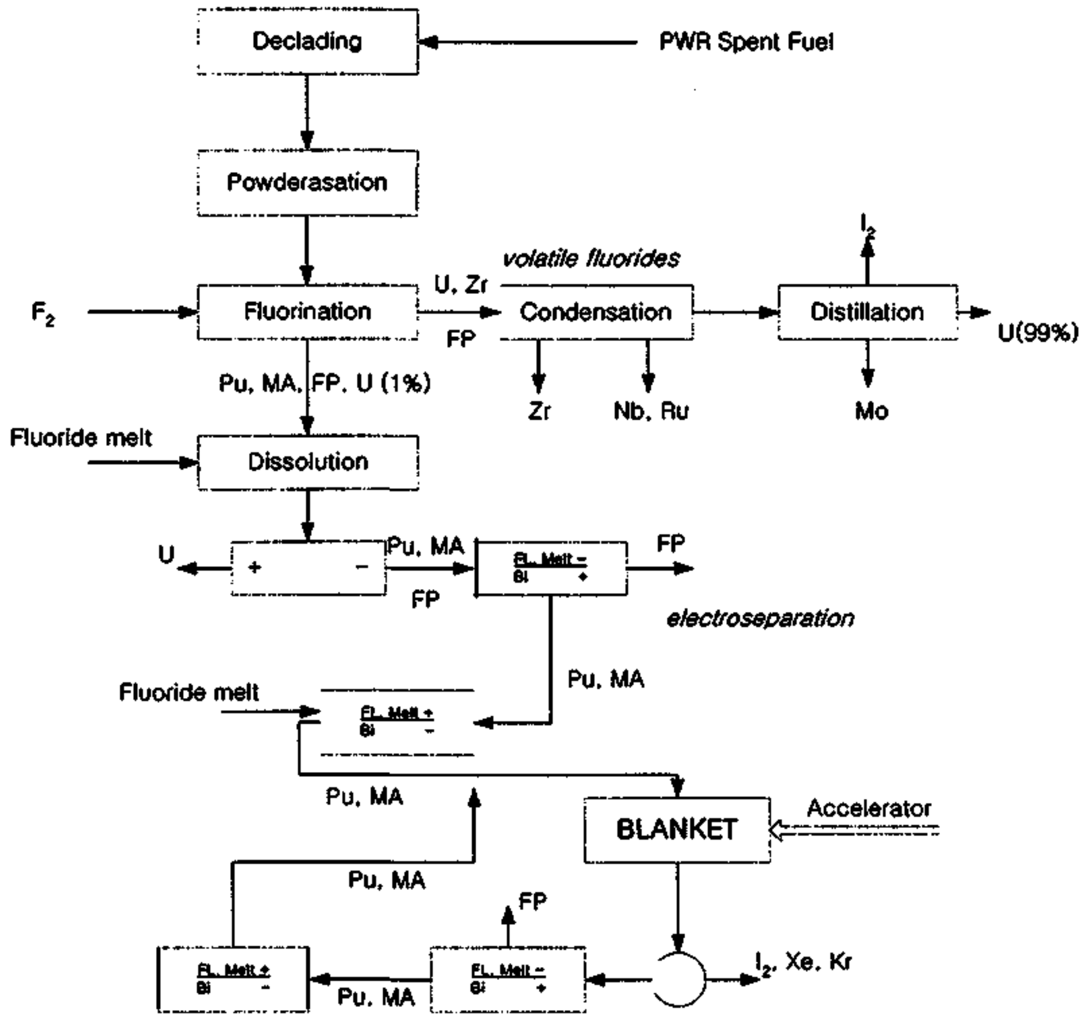


Figure 4. Simplified scheme of fuel processing in an ADS based on molten fluoride.

The fuel is adjusted before the processing, i.e., all metallic parts of fuel elements and packings of plutonium warheads are separated. The scheme of fuel adjustment before the processing by neutron irradiation is as follows. Fuel elements with Zr and Nb coating are dissolved in BeF_2+LiF melt under a continuous bubbling-through of hydrogen fluoride. Hydrogen is released during the process getting off with the volatile components of the fuel, predominantly with Xe and Kr. Before entering the reactor, the melt containing dissolved fuel element undergoes an electrolysis in order to separate some metallic components such as zirconium, uranium and some fission products. The melt is then pumped into the reactor where it surrounds the neutron source inducing the nuclear reaction desired.

After a certain reaction time, the melt is continuously taken off in order to separate fission products and remaining actinides by the method of reduction extraction with liquid bismuth and probably also by the centrifugation method. The basic fluoride melt (carrier medium) is purified, its composition is adjusted and then returned to the process.

The Department of Fluorine chemistry of the Nuclear Research Institute Rez has been engaged in the field of inorganic fluorides for more than 30 years. Experience thus obtained was used in nuclear chemistry, especially to the separation of a series of fluorides of uranium, plutonium, fission products and some transplutonium elements [30-35].

4.4. Experience in fluoride volatilization and chemical problems associated with molten salt technology application in ADTT

A technological process has been worked-out for the separation of uranium and plutonium from the spent fuel by the so-called fluoride method. The whole process was upgraded to a pilot plant scale with a capacity of 1-3kg of processed fuel/hour. There was a part of the technological equipment built and verified at the inactive scale at the Nuclear Research Institute, Rez. The whole technological process was then realised in the Institute of Atomic Reactors at Dimitrovgrad, Russia.

All equipment including fittings, measuring instruments and accessories have been built in the former Czechoslovakia, the plutonium part of the pilot plant has been built in the former USSR. A certain experience has also been obtained on the uranium isotope separation by ultra-centrifugation. Solutions of UF_6 in perfluoroorganic compounds have been treated. The separation effect was determined by mass-spectrometry.

The experience gained in the course of the research is going to be applied in developing fluoride chemistry based separation processes for the use in the Accelerator Driven Transmutation Technology (ADTT).

The process consists in fluorination of the fuel, separation of plutonium, uranium, and fission product fluorides by partial thermal decomposition and uranium hexafluoride rectification in a distillation column. Fluorination of the powdered oxides was performed in a flame fluorinator in a combination with a fluidised bed fluorinator (for the secondary fluorination of plutonium oxides).

The fluorides formed passed through two types of condensers and an apparatus for thermal decomposition of plutonium hexafluoride, apparatus for uranium hexafluoride purification by rectification and columns packed with sodium fluoride and aluminium oxide pellets. The whole line called Fregat was installed in hot cells of the Institute of Atomic Reactors in Dimitrovgrad.

4.5. Involvement in the ADTT

The transmutation process enables the decomposition of transuranium elements into fission products in a subcritical reactor. Thus, it is possible to transform the spent fuel from nuclear reactors as well as the military Pu and Am waste to the shorter-lived fission product nuclides. Spent fuel from the LWR reactor contains about 95% of uranium (UO_2), about 5% of cladding metal (Zr, Nb), fission products and transuranium elements. Chemical processing in the molten salt fluorides is suitable for simple separation of uranium by fluorination. According to the experience of ORNL with the molten-salt breeder reactor, MSBR, it would be possible to obtain 99% of uranium by fluorination from fluoride salt melt. The program was proposed and is being carried out at present on the development of spent LWR reactor fuel reprocessing before its transmutation in the ADTT process. The main task of the reprocessing is the separation of uranium; transuranium elements and some fission products and their fixation in a fluoride glass. Technological scheme of the process was developed and some of the operations were verified experimentally at a laboratory scale. The scheme supposes the dissolution of fuel elements in molten alkaline fluorides in the presence of hydrogen fluoride, separation of uranium by reacting with elementary fluorine to form volatile uranium hexafluoride, separation of transuranium elements and some fission products by the use of physical or chemical procedures. The separated fission products are fixed in the form of fluoride glass by melting with calcium difluoride.

On the base of literature information and our own experience a technological flow sheet of the reactor fuel treatment before and after the transmutation was proposed. It comprises the following operations:

1. Dissolution of spent reactor fuel elements (pelletised UO_2) and cladding material (Zr) in fused fluoride salts in the presence of hydrogen fluoride. This reaction was studied at a laboratory scale last year. We have measured the kinetics of the dissolution of U, Al, Zr and UO_2. The molten fluoride salts composed of LiF, KF and NaF have a m. p. of 450°C.
2. Contacting dissolved uranium, zirconium, fission products and transuranium elements with elementary fluorine in fused fluoride salts carries out fluorination. Uranium is released from the reactor in the form of gaseous UF_6 together with the volatile fission products, for example MoF_6, TcF_6, IFS, etc. Only non-volatile fluorides, transuranium and fission product elements remain in the molten salt.
3. Separation of other components from molten salt. There is no experience on this subject in our laboratory, but according to information from Los Alamos, electrowinning or other physical methods are recommended for separation of Zr and some fission product elements.

4.6. Conclusions

In order to determine optimum flowsheets of chemical processes both for spent fuel processing before the introduction in the reactor and for the treatment of the fluoride molten salts taken-off the reactor, the following range of problems needs to be considered by research organisations:

1. Dissolution of spent fuel elements and materials containing plutonium and other transuranium elements in a molten salt medium by the reaction with anhydrous hydrogen fluoride.
2. Separation of certain metals before the irradiation fluorination (by electrolysis, extraction with metals).

3. To elaborate a simple technological flowsheet by utilising the pieces of knowledge acquired in the work on the separation of fission products by molten salt processing (within the framework of MSBR) and the information on new processes based on physical methods of separation of elements.
4. To elaborate a process for the regeneration of the valuable LiF-BeF_2 mixture.
5. To verify experimentally the electrolytic precipitation of the individual elements or of whole groups of elements.
6. To verify experimentally the efficiency of fluorination with F_2 in the molten salt medium.

REFERENCES

[1] ALEKSEEV, P. N., et al., Proceedings of GLOBAL' 95, (1995) 1500.
[2] ALEKSEEV, P. N., et al., IAEK Rep. 5857/2. (in Russian).
[3] MARTONE, M. (ED.) FRASCATI, IFMIF/CDA - Final Report, Rep.,RT/ERG/FUS/96, http://fabios.frascati.enea.it/cda/FinalReport/Default.htm.
[4] IFMIF/CDA - Cost Report, compiled by M. J. Rennich, ORNL, M-4908, (1996) http://fabios.frascati.enea.it/cda/CostReport/Default.htm.
[5] BDM, P., et al., Nucl. Instr: and Meth. **A 425** (1999) 522.
[6] BOWMAN, C.D., et al., Nuclear energy generation and waste transmutation using an accelerator - driven intense neutron source, Nucl. Instr. and Meth. In Phys., Res. **A320** (1992) 336.
[7] HRON, M., et al., The LA-0 Transmuter: Development of New Technologies of Nuclear Incineration of VVER Spent Fuel... NRI R&D Project (1998).
[8] HRON, M., A Preliminary Design Concept of the Experimental Assembly LA-0, LA-UR 95-3761 Los Alamos National Laboratory (1995).
[9] ALEKSEEV, P. N., et al., "Subcritical enhanced molten salt reactor concept", GLOBAL'95 Versailles, France, (1995).
[10] ALEKSEEV, P. N., et al., Molten Salt Reactor Concept with a Higher Safety, Report of the Russian Scientific Center Kurchatov Institute IAE 5857/2 (1995).
[11] NOVY, P., PEKA, I., CHOCHLOVSKY, I., Technological Regulations for the Uranium Branch of the Fluorine Reprocessing Method for Fast Reactor Spent Fuel, Report OJV-9081 CH, T. Nuclear Research Institute, Rez, (1989) (in Czech).
[12] NOVY, P., BARKOV, V.V., et. al., Testing of the Uranium Branch of the Line FREGAT-2 with Non-Irradiated Fuel, Report ÚN-9060 CH, T. Nuclear Research Institute, Rez, (1989) (in Czech).
[13] GRUBBLY, T., Accelerator-Driven Transmutation of Waste (ADTT), Project Office MS 854. Los Alamos National Laboratory, Los Alamos, USA. (1995).
[14] GAT, U., ENGEL, J. R., DODDS, H.L., Molten-salt reactors for burning dismantled weapons fuel, Nucl. Tech. **100** (1992) 390.
[15] GAT, U., et al., "Dismantled weapons fuel burning in molten-salt reactors", Global'93, Seattle, USA. 1993.
[16] Williamson, M.A., Molten-salt electrochemistry of accelerator driven transmutation systems. Private communication. Los Alamos National Laboratory, USA. 1995.
[17] ROSENTHAL, M. W., HAUBENREICH, P.N., MCNEESE, L.E., Recent progress in molten-salt reactor development, Atomic Energy Review, **9** (1991) 601.
[18] KASTEN, P. R., BETTIS, E. S., ROBERTSON, R. C., Design Studies of 1000 MW(e) Molten-Salt Breeder Reactors, Report ORNL3996. Oak Ridge National Laboratory, USA. (1996).

[19] ROBERTSON, R. C., et. al., Two-Fluid Molten-Salt Breeder Reactor Study, Report ORNL-4528. Oak Ridge National Laboratory, USA (1970).
[20] GRIMES, W. R., BAMBERGER, C.E., Energie Nucl., **14** (1970) 137.
[21] BETTIS, E. S., BOWMANN, H. F., Power Engineering **42** (1970).
[22] HAUBENREICH, P. N., Nucl. Eng. Int. **14** (1969) 325.
[23] McNEESE, L. E., Rare-Earth Removal Using The Metal Transfer Process, Report ORNL-TM-3140, Oak Ridge National Laboratory, USA (1971).
[24] McNEESE, L. E., Report ORNL - (TM) – 3137, Oak Ridge National Laboratory, USA (1971).
[25] YOUNGBLOOD, E. L., McNEESE, L. E., Metal Transfer Process Development, Report ORNL-TM-3259, Oak Ridge National Laboratory, USA (1972).
[26] HANNOFORD, B. A., KEE, C. W., McNEESE, L. E., Semi-Continuous Reductive Extraction Experiments In Mild Steel Facility, Report ORNL-TM-3141, Oak Ridge National Laboratory, USA, (1971).
[27] FURUKAWA, K., et al., “Opening of rational safe thorium utilization way - THORIMS-NES - by plutonium burning and U-233 production”, Conference EEWW2, 1995, Sinta, Portugal (1995).
[28] FURUKAWA, K., et al., Opening of rational safe thorium utilization, Ukr. Zhurnal Chim. (1994) 456.
[29] FURUKAWA, K., et al., “Thorium fuel-cycle development”, International Atomic Energy Agency, Tech. Comm. Meeting, Vienna, Austria, 1995.
[30] FURUKAWA, K., et al., “Rational Pu disposition for U-233 production”, Conference on Thorium Molten-Salt Nuclear Energy Synergetics, Obninsk, Russia, 1994.
[31] LECOCQ, A., FURUKAWA, K., “Chemistry of dry processes in the THORIMS-NES concept”, Conf. on Molten-Salt in Nuclear Technology, Dimitrovgrad, Russia, 1995.
[32] NOVY, P., et al., Experiments On The Technological Line Fregat-2 (Uranium Branch) With Irradiated Uranium Fuel, Report ÚN-9060 CH. Nuclear Research Institute, Rez. Czech Republic, (1989) (in Russian).
[33] NOVY, P., Verification Of The Fluoride Reprocessing Technology, Final report RÚ 159-113, Report UN-9062 CH. Nuclear Research Institute, Rez, Czech Republic. (1989) (in Czech).
[34] NOVY, P., PEKA, I., CHOCHLOVSKY, I., Technological Project Of The Fluoride Volatility Process For Reprocessing Of Spent Fuel From FBR, Report UJV-9081 CH,T. Nuclear Research Institute, Rez, Czech Republic (1989) (in Czech).
[35] PEKA, I., NOVY, P., et al., Use Of Fluoride Technologies For The Application In Industry, Report ÚN9333 CH. Nuclear Research Institute, Rez, Czech Republic (1989), (in Czech).

MOLTEN SALT FUELS FOR NUCLEAR WASTE TRANSMUTATION IN ACCELERATOR DRIVEN SYSTEMS

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Abstract

In order to discuss various criteria for achieving actinide reduction goals, the goals for long lived radwaste management strategy must be defined themselves. Some possible goals and the reasons behind these will be presented. An attempt is made to arrange the possible systems on order of performance with regard to their potential to limit actinide inventory in the fuel cycle, reduce actinides losses to waste, and the same time, to keep minimal uranium-235 support and minimal neutron captures outside actinides. The objective of this paper is to discuss the feasibility of molten salt reactor technology for treatment of plutonium minor actinides and fission products in accelerator driven systems. Fluid nature of the fuel gives extra flexibility to get: simpler back end of the fuel cycle due to the easy of the fuel preparation and treatment in the operation, increased burnup time in the system and as a result decreased TRU mass flow rate in the fuel cycle. This contribution aims: to review the results of the works performed, particularly in Russia, on molten salt reactor technology development and to evaluate the importance of remaining uncertainties for molten salt burner concept implementation. Fuel properties, container materials, fuel radiation stability, pump experience and cleanup of fuels with emphasis on experiments will be of priority. Recommendations are made regarding the types of experimental studies needed on a way to implement molten salt technology to back end of the fuel cycle. To solve some of the essential issues for molten salt burner concept, Russian Institutes [RFNC-All-Russian Institute of Technical Physics (Snezinsk), RRC-Kurchatov Institute (Moscow), Institute of Chemical Technology, Moscow) and Institute of High Temperature Electrochemistry (Ekaterinburg)] have submitted to the ISTC the Project # 1606 *"Experimental study of molten salt technology for safe and low waste treatment of plutonium and minor actinides in accelerator driven and critical systems":* The major developments that we believe should be pursed in the work of the ISTC # 1606 will be briefly discussed.

1. INTRODUCTION

Though, molten salt nuclear fuel concept has been proven by successful operation experience of MSRE experimental reactor at ORNL [1-4] this approach has not been implemented in industry. The fuel chosen for operation of MSRE and for subsequent reactors of this type was a mixture of $^7LiF\text{-}BeF_2\text{-}Zr\ F_4(ThF_4)\text{-}UF_4$.

The interest to explore molten salts, as a future option in nuclear power both to back end of the fuel cycle or for its overall simplification,, currently is revisited. It is naturally to expect that the molten salt reactor (MSR) technology could find a role in symbiosis with standard reactors in the management of plutonium, minor actinides and fission products. Introduction of the new molten salt burner (MSB) concept in the future nuclear power system should provide:

• Low plutonium and minor actinides total inventory in the nuclear fuel cycle;
• Reduced actinides losses to waste;
• Minimal uranium-235 support;
• Minimal Neutron Captures Outside Actinides (Coolant & Structural Material Activation Products).

New MSB concept requires a reconsideration of prior concept of MSRs, including optimization of the neutron spectra in the core and R&D which led to a selection of the salt composition and approaches to its treatment.

The results of recent studies have demonstrated that a broad range of MSBs with PuF_3 and minor actinides as the startup fuel is conceptually feasible. The basic reactor flowsheet for the MSB is essentially the same as that for the reference ORNL designs [1,2]. The only differences are in the core/blanket configuration, details of the fuel salt composition and the fission product cleanup system. uranium-free fuel matrix, as well as, addition of ThF_4 and UF, in homogeneous solution, are conceptually feasible for MSB.

At one end of this range is a high power density, well thermalized graphite moderator reactor in which fluid fuel consists of a molten mixture of ^{7}Li,Be/F and Na,Be/F, containing appropriate quantities plutonium and minor actinides as trifluorides. At the other end is a MSB operating with fast spectrum without graphite moderator, in which solvent system prepared from NaF-^{7}LiF (and/or other possible constituents, like CaF_2, ZrF_4 , PbF_2). This MSB concept will need a concentration of PuF; much higher than that of previous one. Phase behavior of some such mixtures appears suitable to permit use of a high concentration of PuF_3 in melts whose freezing point will acceptable for single fluid MSBs.

Chloride salts have been also proposed as an alternative fluid fuel to obtain a fast neutron spectrum due to high solubility of plutonium and transuranics (TRU) in the melt, but severe problems related to structural materials corrosion, chemical stability of such systems and poor separation ability between some representatives of actinides and lanthanide's groups have been pointed out [5,6]. Also, during irradiation ^{35}Cl transmuted to ^{36}Cl with $T_{1/2}$=300000a.

Ternary and quaternary systems of fluorides fuels still remain an interesting way out. Desirable characteristics of these innovative fuels are:

(1) stability and negligible pressurization at high temperatures;
(2) good hydrodynamic and heat transfer properties;
(3) stability in high radiation field;
(4) low neutron absorption cross sections and
(5) low solubility's of gaseous fission products, which makes possible a several fold increase in burnup time in the system.

The advantages of the MSR as a burner reactor follow not only from a possibility for its effective combination with the dry technique of fuel processing, which has a prospect to have low cost and produce a small volumes of wastes, but also from its capability to use fuel of different nuclide compositions. The MSRs have the flexibility to utilize different fissile fuel in continuous operation with no special modification of the core as it was demonstrated during MSRE operation for ^{233}U,^{235}U and plutonium. The MSRs further can tolerate denaturing and dilution of the fuel, as well as contamination by lanthanides.

In MSR the molten salt is pumped from the core to heat exchanges where heat generated by fission is transferred to a molten secondary salt. The fuel processing plant is conceived to operate continuously on a small side stream of molten fuel or in a batch mode, to remove fission products for discard as waste. The ability to treat the fuel rapidly for fission gases and rare earth removal is one of the most important characteristics of MSRs.

In Russia the MSR program was started in the second half of 70's. Russian Research Center-Kurchatov Institute (RRCKI) was a basic organization under which supervision a collaboration of specialized institutions was formed and functioned.

A reduction of activity appeared after 1986 due to Chernobyl accident and general stagnation of nuclear power and nuclear industry. Then at the end of 80’s there was an increase of conceptual studies as a result of the interest to the inherent safe reactors of new generation.

The MSR study was organized around the following issues [6]:

• exploration of possible use and niches for MSR concepts,
• reactor physics and reactor safety,
• container materials for fuel and coolant salts,
• physical and chemical properties of molten salt mixtures,
• heat transfer and hydraulics of fuel and coolant,
• handling and circulation of fuel and coolant salts,
• process and radiochemical bench tests of model installations,
• radiation chemistry of molten fluoride fuel salt.

The first two issues constituted theoretical studies, the rest - both theoretical and experimental ones. An extensive review of MSR technology development in Russia through 1989 is given in the book [6].

In 1994-1996 a feasibility study of molten salt fuel as applied to Pu burning and long-lived radwaste transmutation in accelerator driven system (ADS) had been supported by the International Science and Technology Center (ISTC) under the project #17 [7]. Currently some study on feasibility of new concept of MSB fueled by plutonium and minor actinides is being supported by MINATOM [8].

In order to evaluate the importance of remaining uncertainties and to identify the additional work needed for MSB concept implementation the next sections discuss experimental studies based on this publication and those appeared afterwards [6-20].

2. THE FUEL SALT FOR MSB CONCEPT

Many chemical compounds can be prepared from several "major constituents" mentioned above. Most of these, however, can be eliminated after elementary consideration of the fuel requirements [1].

Consideration of nuclear properties alone leads one to prefer as diluents the fluorides of Be, Bi, Li, Pb, Zr, Na, and Ca, in that order. Simple consideration of the stability of these fluorides toward reduction by structural metals, however, eliminates the bismuth fluorides from consideration (see Table I). Note, that ZrF_4, as a part of basic solvent, was found to distill from melt and condense on cooler surfaces in the containment system [1]. Control of the ZrF_4 mass transport was considered to difficult to ensure, so the $2LiF\text{-}BeF_2$ solvent system was chosen at ORNL. Also, in order to minimize problems associated with chemical treatment of the fuel salt and associated reduction of the basic components, the priorities should be given to the system with lowest possible ZrF_4 and PbF_2 content.

Table I. Thermodynamic Properties of Fluorides

Compound /solid state/	$-\Delta G_{f,1000,}$ ccal/Mol	$-\Delta G_{f,298,}$ ccal/Mol	E^0_{298}, V (Me/F_2)
LiF	125	140	6,06
BeF_2	208	231	5,0
ZrF_4	376	432	4,69
PbF_2	124	148	3,20
BiF_3	159	200	2,85
NiF_2	123	147	3.18
UF_4 *(UF_3)*	381(*301*)	428 (*328*)	*4,74*
PuF_3(PuF_4)	320	357 (*400*)	5,16
AmF_3	325	365	5,27
CeF_3	345	386	5,58
LaF_3	348	389	5,63

Trivalent plutonium and minor actinides are only stable species in the various molten fluoride salts [2]. Tetravalent plutonium could transiently exist if the salt redox potential was high enough. But for practical purposes (stability of potential container material) salt redox potential should be low enough and corresponds to the stability area of Pu(III).

The trifluoride species, PuF_3, AmF_3, CmF_3, NpF_3 and the rare earth's, have a relatively limited solubility in fluoride salt mixtures and tend to coprecipitate when their solubility is exceeded [2,10].

PuF_3 solubility is maximum in pure LiF or NaF and decreases with addition of BeF_2 and ThF_4. Decrease is more for BeF_2 addition, because the PUF_3 is not soluble in pure BeF_2.

The solubility of PuF_3 in LiF-BeF_2 and NaF-BeF_2 solvents is temperature and composition dependent and PuF_3 solubility seems to be minimal in the "neutral" melts. It reach about 0,5 mol.% at the liquidus temperature 600°C and increases to about 2,0 mole % at 800°C. These values are adequate for the use of plutonium as a fuel and would allow the accumulation of rare earth fission products for as long as 10 years without processing if this were thought desirable [2]. As expected substitution of a small quantity of $TRUF_3$ scarcely changes the phase behavior of the solvent system.

As expected single fluid fast spectrum MSB will need a concentration of PuF_3 much higher than that for the 2LiF-BeF_2 system with graphite moderator. LiF-PuF_3 system has eutectic at 743°C and 20 mol.% PuF_3. NaF-PuF_3 system has eutectic at 24 mol.% PuF_3, melting at 727°C [3]. Inspection of the diagram for Li,Na,Pu/F system reveals that a considerable range of compositions with about 10 mol. % PuF_3 will be completely molten at 600°C. It is possible that some addition of e.g. CaF_2 would provide liquidus for quaternary system at lower temperatures. As expected from general similarity of PuF_3 and minor actinides trifluorides substitution of a relatively small quantity of AmF_3, CMF_3, NpF_3 for PuF_3 scarcely changers the phase behavior. Accordingly, the phase behavior of the fuel will be dictated by that of Li, Na, Pu/F system. Significant quantities of minor actinides in the mixture will complicate phase behavior of the fuel.

The molten fluoride chemistry (solubility, redox chemistry, chemical activity etc) for the $2LiF-BeF_2$ system is well established and can be applied with great confidence, if PuF_3 fuels are to be used in the $2LiF-BeF_2$ solvent. But the chemistry of other solvent systems are different and less understood (for example, in the more acidic Li,Na/F system) and requires a comprehensive studies.

For MSB's needs next more important is consideration of PuF_3 chemical behavior in these solvent systems: PuF_3 solubility in Li,Be,/F and Na,Be,/F solvents; phase transition behavior of the ternary Na,Li,Pu/F and quaternary Na,Li,Ca,Pu/F fuels, oxide tolerances of such mixtures and redox effects of the fission products.

3. CONTAINER MATERIALS STUDIES

An important part of Russian MSR program dealt with the investigation of the container materials [6,11] The development of domestic structural material for MSR was substantiated by available experience accumulated in ORNL MSR program on nickel -base alloys for UF_4 -containing salts [1,2]. In addition, compatibility tests were conducted to re-examine to possibility of using iron based alloys as container materials for MSR's. These alloys are more resistant to tellurium penetration and generate less helium under irradiation than Ni - based alloys. However, their use would limit the redox potential of the salt and operating temperature in MSR.

Corrosion resistance of materials was studied by two methods. The first is the method of capsule static isothermal test of reference specimens in various molten salt mixtures. Also, flibe, flinak and sodium fluoroborate eutectic salts have been circulated for thousands of hours in natural and forced convection loops constructed of iron and nickel based alloys to obtain data on corrosion, mass transfer, and material compatibility. Not only normal, but also high oxidation conditions were present in the loops.

The alloy HN80MT was chosen as a base. Its composition (in wt.%) is Ni(base), Cr(6.9), C(0.02); Ti(1.6), Mo(12.2), Nb(2.6). The development and optimization of HN80MT alloy was envisaged to be performed in two directions:

• improvement of the alloy resistance to a selective chromium corrosion,
• increase of the alloy resistance to tellurium intergranular corrosion and cracking.

About 70 differently alloyed specimens of the HN80MT were tested. Among alloying elements there were W, Nb, Re, V, Al and Cu. The main finding is that alloying by aluminum at a decrease of titanium down to 0.5% revealed the significant improvement of both the corrosion and mechanical properties of the alloy. The chromium corrosion and intergranular corrosion have reached the minimum value at AI content in the alloy -2.5%. Irradiation effect on a corrosion activity of fuels was also studied. It was shown that at least up to the power density 10 W/cm' in fuel composition $LiF-BeF_2—ThF_4-UF_4$ there is no radiation induced corrosion.

Then the radiation study of 13 alloy modifications were carried out. Specimens (in nitrogen atmosphere) were exposed to the reactor neutron field up to the fluency of $3 \cdot 10^{20}$ n/cm^2. Experimental results of alloy mechanical properties at temperatures of 20, 400 and 650°C for nonirradiated and irradiated specimens permits to rest only four modifications. These alloys modified by Ti, Al and V have shown the best postradiation properties.

At last, corrosion under the stressed condition was studied. It is known that tensile strain promotes an opening of intergrain boundaries and thus boosts intergranular corrosion and create prerequisites for an intergranular , cracking. The studies did not reveal any dependence of intergranular corrosion on the stress up to the value 240 MPa, that is 0.8 of a tensile yield of the material and 5 times higher than typical stresses in MSR designs.

The results of combined investigation of mechanical, corrosion and radiation properties various alloys of HN80MT permitted to suggest the Ti and Al-modified alloy as an optimum container material for the MSR. This alloy named HN80MTY (or EK-50) has the following composition (in wt.%): Ni(base), Fe(1.5), Al(0.8-1.2), Ti(0.5-1), Mo(11-12), Cr(5-7), P(0.015), Mn(0.5), Si(<0.15), C(<0.04). The comparison of our corrosion data with those obtained at ORNL for Hastelloy-N indicates that corrosion resistance of HN80MTY is higher and it's maximum working temperature could be up to 750°C. Still the weldability of the alloy deserves an improvement.

The high temperature, salt redox potential, radiation fluence and energy spectrum poses a serious challenge for any structural alloy in a MSB system. Data for UF_4 -containing salts provides a roadmap for establishing the corrosion properties for PuF_3- containing salts, but additional corrosion testing under MSB conditions will be required to quantify corrosion rates of candidate container materials.

NaF, LiF, BeF_2, CaF_2, ThF_4, PuF_3, AmF_3, CmF_3 and NpF_3 can not be oxidized in system considered and can be reduced only to the metals, and then only by reducing agents very much stronger than the constituents of Hastelloy-N. Mixtures of these materials would not be expected to be corrosive. Recent capsule experiments [12] have demonstrated that PuF_3 addition in LiF-BeF_2 solvent system did not make the corrosion situation worse on both nickel- and iron-base alloys. The chemistry of PuF_3 needs further testing in corrosion loops studies for redox control.

Also, included in further evaluation should be an assessment of lower salt redox potentials from the stand point of allowing the use of stainless steels as structural materials and establishing the potentials that must be maintained to avoid intergranular cracking for nickel-based alloys. Techniques developed under other reactor programs to improve the resistance of stainless steels to helium embrittlement should be extended to include nickel-base alloys.

4. RADIATION STABILITY OF FLUORIDE SALTS

At RRC-KI in-reactor loop tests of radiation stability of fluoride molten-salt fuels were carried out [6]. The radiolysis products (first of all, F_2) were detected in the gas phase over the melts. It was stated, that:

- the radiolytic evolution of fluorine from molten fluoride mixtures when irradiating in a nuclear reactor is insignificant;
- the measured values of radiation chemical yield $G(F_2)$ (the number of F_2 molecules, evolving per 100 eV of absorbed energy) are in the range of 10^{-5}-10^{-6}, therefore the fluoride fluid fuel may be attributed to the category of radiation resistant materials in the temperature range up to 1200°C;
- the frozen fuel salt at 50°C has the $G(F_2)$ value about 10^{-2}, that is much better, as compared to water, which is used as the primary coolant in LWRs.

The high radiation resistance of molten fluorides allow them to be recommended as the fuel for high temperature reactors.

5. THE HEAT AND MASS TRANSPORT PROPERTIES

The design of MSR requires detailed information on heat and mass transport properties of the proposed fuel and coolant fluoride melts.

Many of the physical properties dealing with heat and mass transport processes of molten fluoride salt have been obtained during the development of the MSR program in ORNL [4]. The specific physical properties which were either measured in Russia include density, heat capacity, heat fusion, viscosity, thermal conductivity and electrical conductivity [13,14]. Particular emphasis has been placed for U/Th fueled reactor cores.

Physical property data on 100 fluoride salt mixtures (binary coolants, ternary coolants, binary fuels, ternary fuels and quaternary fuels) have been evaluated. The salt composition used in the U/Th - fueled MSWs are not the same as presently proposed for the MSB concepts. It may also be noted that only incomplete property data are available for some of candidate solvents, however, the tabulations available are such as to permit interpolation of reasonable accuracy for preliminary design purposes if melt contains small (< 1 mol %) addition of $TRUF_3$. When the melt is complicated by the addition of large quantities of $TRUF_3$ the situation becomes considerably less favorable and need additional measurements of required physical properties.

At RRC-KI heat transfer studies with different fuel and coolant salts (Prandtl modulus range from 2 to 35) flowing in forced and natural convection loops were made in a wide range of parameters typical for MSR designs [15-17].

Forced convection runs have covered a Reynolds modulus range from 5000 to 20000 and a heat flux q_s range up to 10^2 kW/m^2. The criterion dependencies generalizing the heat transfer data were obtained for natural convection heat exchangers at separate or joint actions of surface and volume heat sources in the fluid up to q_s=30 kW/m^2 and q_v=7 MW/m^3.

No evidence of influence of corrosion and irradiation products on heat transfer was found (at temperatures up to 750°C, neutron fluences up to 2. 10^{20} cm^{-2} and concentration of Cr, Fe, Ni metal impurities less than 10^{-1}% mole fraction by mass.).

The conclusion is that the use of accurate physical property data with correlation's for normal fluids (Pr>1) is adequate for heat transfer for design with fluoride salts respect to forced and natural convection.

6. PUMP EXPERIENCE

At RRC-KI we know how to make reliable short-shaft centrifugal pumps for molten salt test loops, having built and operated with capacities to 2000 rpm [6,16].
Although it may take some years to produce the larger pumps for demonstration of MSB prototype, the problems are well understood.

We haven't experience with the long-shaft pump configuration for molten salt systems. However, in Russia, it has been used in lead-bismuth and sodium systems developed for fast reactors program.

We expect that results of the long shaft development program for liquid metal pumps will have direct application to salt pumps. Same time we understand that the requirements for lead-bismuth (or sodium) and salt pumps are sufficiently different that in other important respects they are unique.

7. FUEL TREATMENT

The MSB would manage the noble gas removal by sparging with helium in the same manner as others MSR concepts [1,2]. The problem here is to prevent the xenon from entering the porous graphite moderator. Development of sealing techniques should be continuing, both with the "pulse-impregnation" technique and with isotropic pyrolitic coatings put on at a somewhat higher temperatures.

After the xenon poisoning has been limited to low value more than 85% of the neutron losses to the remaining fission products are due to the rare earth.

Since plutonium and minor 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 TRUs from the fuel salt and their reintroduction to the purified solvent. This plutonium reintroduction circuit has the advantage of also returning americium, curium and californium to the reactor fuel and permitting only very small losses of any transuranium elements to the waste stream. Since the higher actinides would always accompany the plutonium, this operation would never produce a "clean" material would be attractive for diversion [2].

Thermodynamic considerations of the fluorides systems have suggested a possible recovery schemes of rare earth, based on reductive extraction, electrorefining, precipitation and their combinations.

Important parameters for chemical treatment (reductive extraction, electrorefining, etc) are electrochemical potential/free energy of formation $\Delta G^{\circ}_f(T)$ of substances and separation factors of elements between molten salt and liquid metal Θ [18]. These separation factors for different elements with the same valency (for example actinides and lanthanides) could be given by:

$$\ln\Theta = \{\Delta G^{\circ}_f(Ln, T) - \Delta G^{\circ}_f(An,T)\}/(RT) \qquad (1)$$

where T is the temperature (K), R is the molar gas constant ($J\text{-}mol^{-1}\cdot K^{-1}$).

Thermodynamic consideration predicts to need more efforts for separating Am and Cm from Ce and La in chloride system in comparison with fluoride one (Figure 1). The calculated separation factors of various elements in fluoride and chloride systems suggest for fluorides a better separation ability between actinides and lanthanides groups but poorly within groups.

The distribution of the An and Ln between a molten salt and a liquid metal or a liquid alloy has been investigated experimentally in ORNL. The Bi-Li alloy is used for reductive extraction from different melts according to the scheme: MF_n (melt) + nLi^0 (Bi) = M^0(Bi) + nLiF (melt).

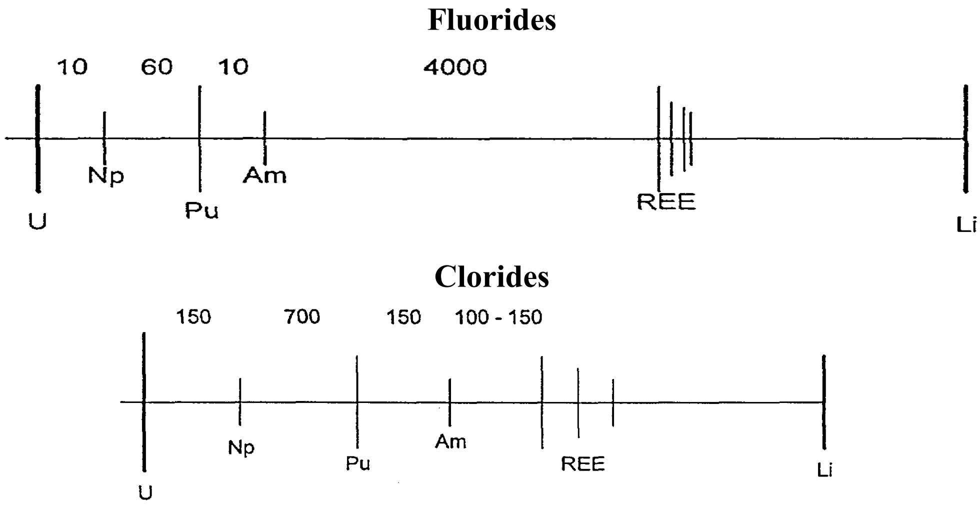

Figure 1. Separation factors between actinides and lanthanides for chloride and fluoride systems.

The separation factors Θ for Li,Be/F and Li,Na,K/F solvent systems are approximately equal to 10^{-3} ($\Theta = (X_{An}/X_{AnFn})(X_{LnFn}/X_{Ln})$, where X_{An}, X_{Ln} - mole fraction of An and Ln in alloy, X_{AnFn}, X_{LnFn} -mole fraction of AnF_n and LnF_n in a molten salt). These values are very convenient for the lanthanide's separation. However, when the melt is complicated by the addition of large quantities of ThF_4 the situation becomes considerably less favorable ($\Theta_{Ln\text{-}Th} \sim 1$).

Rare earth removal unit based on Bi-Li reductive extraction flowsheet developed in ORNL for $LiF\text{-}BeF_2$ solvent system could provide negligible losses of TRU ($\sim 10^{-4}$) by use of several counter current stages. In this process the rare earths are extracted from a fuel salt stream into lithiated bismuth after all the, plutonium and minor actinides have been previously removed.

For fluoride based systems alongside with reductive extraction other methods of lanthanide's removal could be considered. Particularly, this is electrochemical deposition of elements as metals on liquid or solid electrodes in electrochemical cells, filled with fuel salt and bismuth or lead. The applicability of the given method for fluoride based systems is defined by that the fuel salt at temperature 500-800°C is good electrolyte, the conductivity of which is comparable to conductivity of liquid electrolytes. In LiF-NaF salt solution the standard potentials for salt components are higher than that for actinide and lanthanide trifluorides.

As on account (see Table I) the difference of potentials of deposition Am and Ce in fluoride systems reaches 0.2V, it is possible to assert, that at electrolysis of fuel salt there will be the selective separation of these elements. The degree of separation will depends on value of an overvoltage on electrodes. As shows experience of industrial production for a number of alkaline and alkaline-earth elements by electrolysis, at difference of deposition potentials of elements more than 0.1 V, a degree of their separation is more than 100 (at nominal currents about $1A/cm^2$). Thus it is possible to expect that a separation factors between actinides and lanthanides will be more than 100, and at several stages of electrochemical clean up the contents of actinides in lanthanides can be reduced till to 10^4 times.

In experiments [19] a successful attempt was made to precipitate mixed uranium, plutonium, minor actinides and rare earths from LiF-NaF molten salt solution at temperatures 700-900°C by CaO (Al_2O_3) oxidation. The rare earths concentration in the molten salt solution was about 5-10 mol.%. It was found the following order of precipitation in the system U-TRU-Al-Ln-Ca. Essentially all U and TRU were recovered from the molten salt till to rest concentration $5\text{-}10^{-4}$ %, when rare earths still concentrated in solution. If treatment of a MSB fuel on a cycle-time of 100 days or more is practicable, such an oxide precipitation might be used as a batch operation.

With this and other possibilities for chemical processing relatively unexplored there seems to be much room left for interesting studies of electrochemical data and kinetics of reactions in the units when TRU are the fuel.

8. SUMMARY

Though, the molten salt nuclear fuel concept has the solid background, its specific application for the treatment of plutonium and minor actinides required additional studies [20].

To solve some of the mentioned in previous sections essential issues, Russian Institutes (RFNC-All-Russian Institute of Technical Physics, (Chelyabinsk-70), RRC-Kurchatov Institute (Moscow), Institute of Chemical Technology (Moscow) and Institute of High Temperature Electrochemistry (Ekaterinburg)) have submitted to the ISTC the Project# 1606 *"Experimental study of molten salt technology for safe and low waste treatment of plutonium and minor actinides in accelerator driven and critical systems"*.

The major developments that we believe should be pursed in the frame work of the project are the following:
- Conceptual designing and development of MSB.
- Experimental study of behavior and fundamental properties of prospective molten salt compositions.
- Experimental verification of candidate structural materials for fuel/coolant circuits and fuel correction unit. .

Last two objectives are considered crucial to the MSB concept further development. The experimental data would be fed into the conceptual design efforts. The objectives of the conceptual design and development program are to first identify candidate flowsheets (reactor + fuel correction unit) for MSB concept, that will be technologically feasible. Then basing on experimental data received in the Project, the choice of optimal flowsheet, including fuel composition and design parameters, will be done.

One of the aims of this contribution is to attract an attention of potential foreign partners to the ISTC Project # 1606 and to invite them to take part in collaboration in terms of the project.

ACKNOWLEDGEMENT

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REFERENCES

[1] WEINBERG, A. M. et. al., A review of molten salt reactor technology, Nucl. Appl.&Tech., **8**, 2, (1970).

[2] ENGEL, J. R., RHOADES, W. A., GRIMES, W. R., DEARING, J. F., Molten salt reactors for efficient nuclear fuel utilization without plutonium separation, Nucl. Tech., **46**, Nov., (1979) 30 43.

[3] BARTON, C. J., REDMAN, J. D., STRELOW, R. A., J. Inorg. Chem., **20**, 1 2, (1961) 45.

[4] CANTOR, S., Physical Properties of Molten-Salt Reactor Fuel, Coolant, and Flush Salts, ORNL-TM-2316, (1963).

[5] MIGAI, L. L., TARITSINA, T. A., Corrosion Resistance of Structural Materials in Halides and Their Mixtures, Metallurgy, Moscow, (1988), (in Russian).

[6] NOVIKOV, V. M., IGNATIEV, V. V., FEDULOV, V. I., CHEREDNIKOV, V. N., Molten Salt Nuclear Energy Systems - Perspectives and Problems, Energoatomizdat, Moscow, (1990), (in Russian).

[7] GREBENKINE, K.F., "Review of activities on accelerator-based transmutation at Chelyabinsk-70: results of the ISTC project and prospects", (Proc. of the Second International Conference on Accelerator Driven Transmutation Technologies and Applications, Kalmar, Sweden, 1996) (1996) 221 227.

[8] IGNATIEV, V. V., GREBENKIN, K. F., ZAKIROV, R Y., "Molten salt reactor technology for partitioning & transmutation and harmonization of the future nuclear fuel cycle", (Proc. of the Fifth Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, Mol, Belgium, 1998), (1998).

[9] GREBENKINE, K. F., et. al., Plutonium trifluoride as a potential fuel for power reactors and its synthesis from a weapon plutonium, Atomnaya Energya, **83**, (1997), 140 147 (in Russian).

[10] NAUMOV, V. S., et. al., "Interaction of actinide and rare-earth element fluorides with molten fluoride salts and possibility of their separation for ADTT fuel reprocessing", (Proc. of the Second International Conference on Accelerator Driven Transmutation Technologies and Applications, Kalmar, Sweden, 1996) (1996) 1144 1155.

[11] IGNATIEV, V. V., et. al., The State of the Problem on Materials as Applied to Molten-Salt Reactor: Problems and Ways of Solution, Preprint IAE-5678/11, Moscow, (1993).

[12] PANOV, A. V., GREBENKINE, K. F., SUBBOTIN, V.G., "Investigation of structural materials corrosion resistance to fluoride metal environment", (Proc. of the Second International Conference on Accelerator Driven Transmutation Technologies and Applications, Kalmar, Sweden, 1996), (1996) 943 946.

[13] Desjantnik, V. N., et. al., Experimental Study on Heat and Mass Properties of Li, Na, Be, U, Th Fluoride Melts, Rep. of the Ural Politechnical University, Ekaterinburg, (1979), (in Russian).

[14] KHOKHLOV, V. A., SMIRNOV, M. V., FILATOV, E. S., "Thermophysical properties of molten alkali fluorides", (Proc. of First International Symposium on Molten Salt Chemistry and Technology, Kyoto, Japan, 1983), (1983) 391 394.

EA-MC NEUTRONIC CALCULATIONS OF THE EAP80

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Abstract

A detailed neutronic analysis has been performed of the 80 MW(th) Energy Amplifier Demonstration Facility, fuelled with uranium and plutonium mixed oxides and cooled by lead-bismuth eutectic. The neutronic calculations presented in this paper are a result of a state-of-the-art computer code package developed by the EET group at CERN. Both high-energy particle interactions and low-energy neutron transport are treated with a sophisticated method based on a full Monte Carlo simulation, together with modern nuclear data libraries. The code is designed to run both on parallel and scalar computers. A series of experiments carried out at the CERN-PS (i) confirmed that the spallation process is correctly predicted and (ii) validated the reliability of the predictions of the integral neutronic parameters of the Energy Amplifier Demonstration Facility.

1. INTRODUCTION

This paper summarizes the neutronic characteristics of ANSALDO's Energy Amplifier Demonstration Facility [1] based on molten lead-bismuth eutectic cooling, classical MOX-fuel technology and operating at 80 MW(th).

The sequence of phenomena from high-energy protons induced cascade in a heavy-Z material (lead or lead-bismuth eutectic), producing neutrons that subsequently interact until they are finally absorbed or escape, is rather complex. For a correct simulation of the Energy Amplifier [2], it is essential to understand the fine details of the physics related to all these phenomena. In order to do so, an innovative simulation was developed using Monte Carlo techniques. This code allows the description of complex geometrical models. Special attention was devoted to the development of techniques (for kinematic calculation, cross-section evaluation, etc.) to minimize the computer time in order to provide sufficient statistics [3].

The design of accelerator-driven fission devices (also called hybrid systems) requires powerful simulation tools for:

- The modeling of the high-energy cascade and of the neutron production;
- The transport of the low energy ($E \leq 20$MeV) neutrons;
- The description of the fuel evolution as a result of neutron interactions and nuclear decays;
- The nuclear transmutation and activation of the structural materials and coolant due to the presence of a neutron fluence and of the proton beam induced high energy cascade.

The simulation tools developed for nuclear reactors cannot be applied immediately to externally driven sub-critical systems and to the Energy Amplifier in particular [2]. Indeed, the spatial distribution of the neutron flux is expected to be radically different in the two cases. While in a critical reactor the flux distribution inside the volume is determined essentially by the boundary conditions, in an Energy Amplifier the effect of the initial high-energy cascade is dominant. In fact, in a sub-critical arrangement the neutron flux along any radial direction starting from the center must fall-off in an approximately exponential manner.

The corrections to the exponential behavior depend primarily on the shape of the source and they are important when close to it.

Such a complex simulation had to be validated. It was one of the main goals of the FEAT and TARC Experiments [4,5] carried out at the CERN-PS accelerator complex. Detailed comparisons of the neutron multiplication factor and energy and space distributions of the neutron fluence give an excellent agreement with experimental data. These sets of experiments confirm in particular that the spallation process is correctly predicted and validates the reliability of the predictions of the integral neutronic parameters of the Energy Amplifier Demonstration Facility.

2. THE EA-MC CODE PACKAGE

EA-MC is a general geometry, "point-energy", Monte Carlo code which stochastically calculates the distribution of neutrons in three-dimensional space as a function of energy and time. The neutron data are derived from the latest nuclear data libraries [6]: ENDF/B-VI 6 (USA), JENDL-3.2 (Japan), JEF-2.2 (Europe), EAF-4.2 (Europe), CENDL-2.1 (China), EFF-2.4 (Europe) and BROND-2 (Russia).

2.1. Structure of the program

The general architecture of the EA-MC code is shown in Figure 1. The geometrical description is first automatically translated into FLUKA's combinatorial geometry, and the FLUKA simulation is carried out [7]. Neutrons are transported down to 20 MeV and then handed over to EA-MC which continues the transport.

The EA-MC code is designed to run both on parallel and scalar computer hardware. Having used standard language elements, the code can be implemented on different systems. A common initialization section is followed by a parallel phase where every CPU runs an independent simulation with the same initialization data. A parallel analysis program collects the results and calculates the standard deviation among the different CPUs. This gives an estimate of the statistical fluctuations [3].

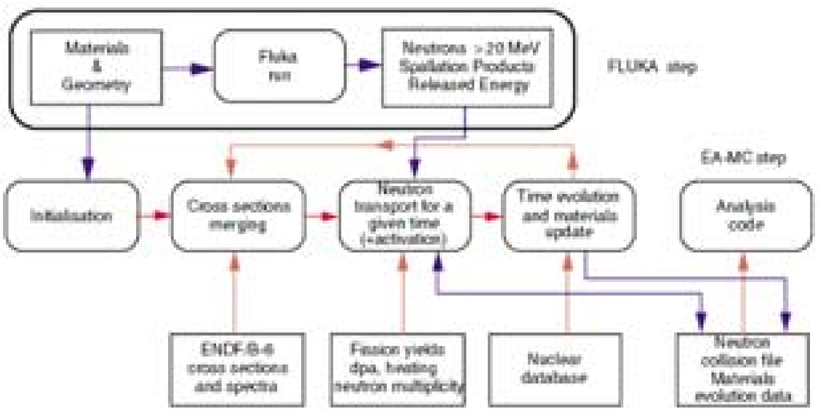

Figure 1. General architecture of the EA-MC simulation of neutron transport and element evolution.

2.2. Neutron transport and time evolution

The EA-MC code has taken the most challenging approach to the problem of neutron transport and burn-up simulation. This simulation package integrates neutron transport and evolution of the material composition in the same code. To take into account the time evolution, every proton simulated by the high-energy transport code, FLUKA [7], is scaled-up to a large number of "real" protons. Once this scale factor is fixed, the value chosen for the accelerator current determines the "real" elapsed time corresponding to one batch of incident protons.

The material composition is updated according to the neutron interactions simulated in every zone. Material composition evolution, due to nuclear decay, is performed between transport steps. High-energy spallation products generated by FLUKA are added to the material composition on a proton by proton basis. The contribution of each fission, spallation, decay and activation product is considered during neutron transport, no "lumped" fission products are used in our code. Every material can have a different temperature that is inherited by all the nuclear species produced there. New temperatures can be generated during the run either by Doppler broadening or by linear interpolation of existing evaluations. A typical reactor fuel material can contain a few hundred elements with cross sections and up to 2000 different nuclear species. Even in this case, the time to transport a neutron is approximately 8 μs/processor on our Convex SPP2000 machine.

At the end of each transport phase, all materials are evolved in a time step that varies adaptively during a complete burn-up simulation. To avoid all approximations in the concentration evolution, a complete solution of the Bateman equations involving thousands of elements is computed. A combination of storage techniques and decay chain algorithms allows evolving all materials in few seconds. No time stepping is used, the solution is analytically correct within the assumptions made. Different smoothing techniques have been developed in case of very short-lived elements or strong concentration gradients. The algorithm has been checked for stability against different time durations. An example of element evolution is shown in Figure 2.

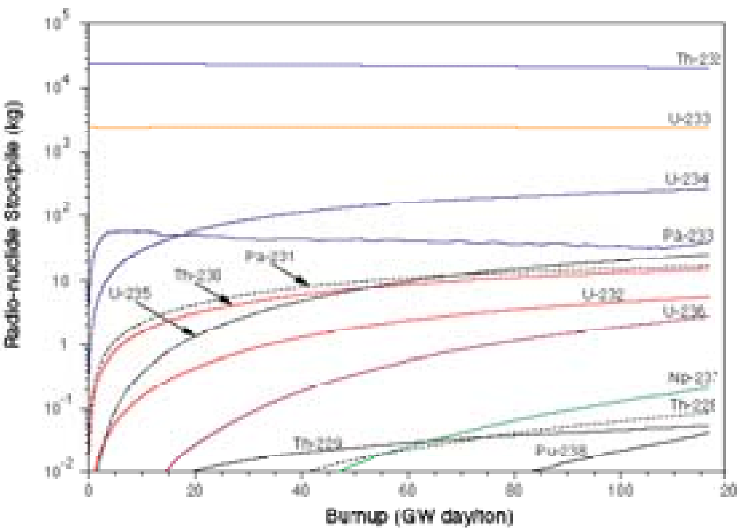

Figure 2. Evolution of the actinide concentrations during burn-up for a typical energy amplifier configuration.

The composition of all the materials is then updated and transport continues for a further cycle. This does not require several separate runs of the simulation/evolution chain, as all is handled by the same code. Moreover, this is done with the full precision inherent to the Monte Carlo transport, and all the effects due to the geometry of the model and to the variation of the neutron energy spectrum are taken into account. This constitutes the innovative aspect of our simulation code.

2.3 Nuclear data

For each nuclide we have selected one evaluation out of those available [6] on the basis of a systematic comparison [8]. In practice, the selection was done isotope per isotope according to the evaluation of the resonances and the number of reaction cross sections. When both the resonance region and the number of cross sections evaluated are similar, then the most recently evaluated cross section was selected. This resulted in a database of 800 nuclides with reaction cross sections out of which 400 have also the elastic cross section. For all nuclides the corresponding information on isomeric state exists, and isomer dependent reactions are treated correctly whenever they are available. Activation/Transmutation data for unstable nuclides are taken from the EAF-4.2 compilation that we have converted to ENDF format [9], including files 8, 9 and 10. All cross section files produced by PREPRO-94 [10] have been subsequently processed with a home-grown code package PROCESS [11] to create a direct access library containing: neutron cross sections, cumulative secondary neutron energy distributions and cumulative neutron angular distributions. Auxiliary routines [12] have been developed to sample several quantities of interest during transport: the products of a reaction, including the isomeric state of the residual nuclei; the mass-correlated fission fragments in case of fission, the energy-dependent number of neutrons emitted in a fission and the angular and energy distributions of secondary neutrons emitted in non-fission reactions.

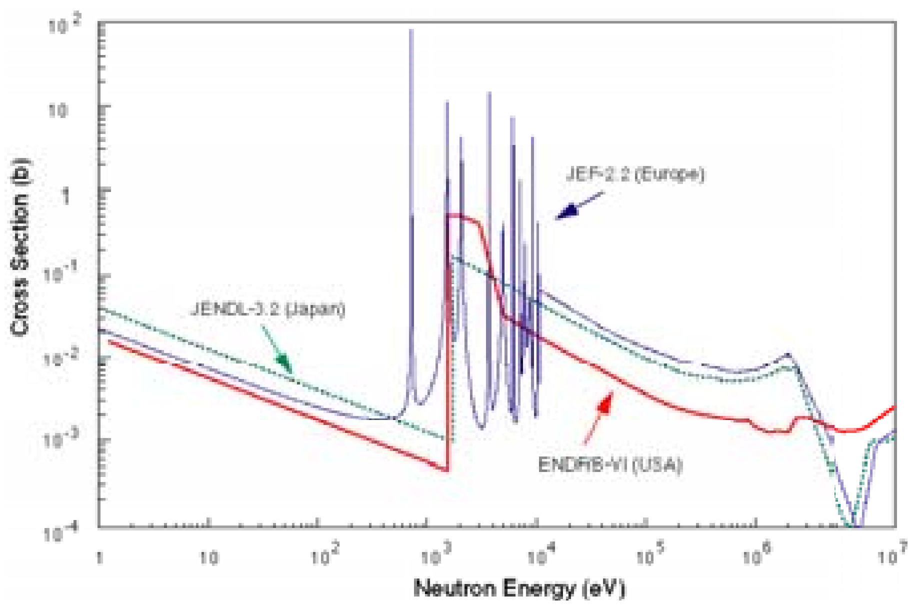

Figure 3. Comparison of the ^{137}Cs capture cross section in the different databases available. The one chosen in this case for the EA-MC corresponds to the European Database (JEF-2.2).

For nuclear transmutation studies, accurate mass and decay tables are needed. The EA-MC Nuclear Database [13] contains 3200 isomers and has been assembled via a careful comparison of several sources. In particular, we have used the Brookhaven nuclear database [14], NUBASE by Audi et al. [15], the National Radiological Protection Board (NRPB) database [16], the ICRP database [17] and nuclear data information from ENDF files. For each isomer we store the atomic number, the chemical symbol, the mass number, the isomeric level, the isospin and parity, the mass excess, the half-life, the decay modes, branching ratios and Q values where applicable, the natural isotopic abundance and the inhalation and ingestion radiotoxicities [18].

This database represents a major improvement over all existing ones for use in a Monte Carlo. All available information about isomeric state decays and production has been included. Nuclear data have been extensively checked for inconsistencies. In particular all decay paths are closed (i.e. they terminate in a nuclide present in the database) and all branching ratios are consistent (i.e. sum = 100%). We also make sure that the information between neutron cross sections and nuclear data is consistent. All nuclei produced by nuclear reactions for which the cross section data exist, must be present in the nuclear database. All nuclear masses are consistent between cross sections, isomer yield files and nuclear databases. This database is probably the most extensive and effective summary of nuclear data available at this time (Figure 4).

2.4 Geometry modeling

The material regions in the system are described using simple geometry techniques which construct the material zones by the combination of simple geometric bodies. The geometry allows for the separate specification of individual components of the geometry model (known as parts). The parts can be included as a single item within other parts to form new parts. Thus, the user can build an EA-MC geometry model as it is done in reality by assembling individual components to form larger components, which eventually form the completed system. This allows a very accurate model of the system to be created.

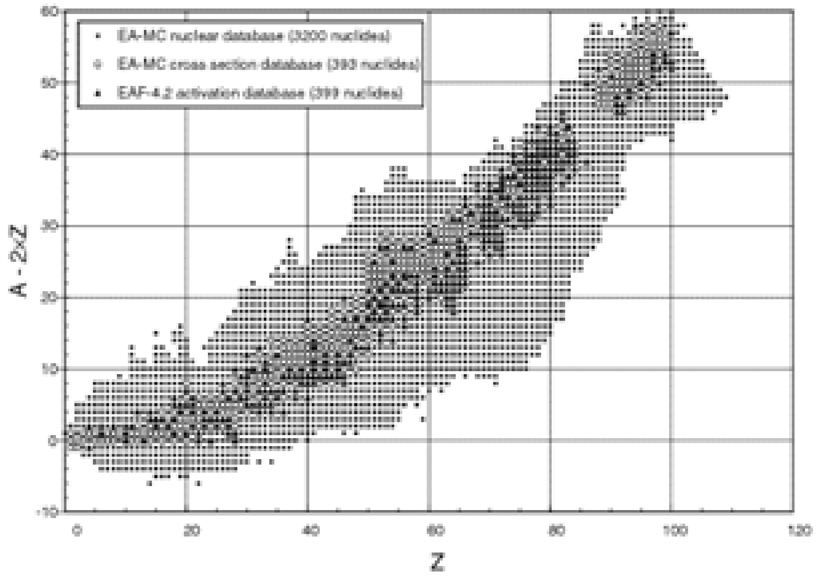

Figure 4. Summary of the element content in the EA-MC nuclear database, illustrated by a plot of A - 2Z vs. Z, showing the larger extent of the EA-MC database.

Particular attention has been paid to efficiency: typical execution times on Convex SPP2000 machines have been kept at around a few μs per call. Specially optimized algorithms have been developed for hexagonal geometries, square lattices typical of pressurized water reactors, and other particular geometrical models of the Energy Amplifier. Automatic translation tools have been developed to and from the FLUKA combinatorial geometry [19].

A suite of user-friendly visualisation packages has been developed for geometry display and error diagnosis. The GeoView program [20] outputs a rendering of the three-dimensional geometry model using simulated optical ray-tracing techniques.

3. VALIDATION OF THE EA-MC CODE PACKAGE

The validation base for EA-MC is extensive and covers all the areas in which it has been used. The most fundamental form of validation is the analysis of highly specified experimental benchmarks, often performed at CERN [4,5]. The validation base also includes the comparison of EA-MC calculations against other particle transport codes [21].

3.1. The FEAT experiment

The concept of the Energy Amplifier [2] was originally developed on the basis of a computer simulation of the rather complex interplay between the production of neutrons by spallation and their subsequent fission-based multiplication in a sub-critical assembly. Even though there existed indirect evidence that these simulations were valid, there was an obvious need to submit these predictions to the test of a dedicated but simple experiment [4].

The core of the concept (giving in fact its name to the “Energy Amplifier”) is the production of substantial amounts of energy, over and above the kinetic energy brought in by the accelerator beam. From that stems the concept of an "energy gain" G. In conditions of practical interest, the gain is predicted to be G = 30 to 60, which, taking into account the relevant efficiencies, was easily shown to be much more than is needed to power the accelerator. The main purpose assigned to the test is therefore to ascertain that there is such a gain and that its magnitude is in agreement with the value predicted by the EA-MC simulation.

We obviously wanted to use an existing accelerator to perform this test. Indeed with the further requirement of minimizing the amount of radioactivity produced, the beam intensity that we used (of the order of 10^8 protons/second) was much smaller (by five orders of magnitude) than the one normally delivered by the CERN Proton Synchrotron (PS). All in all, the power produced during the test was 1 Watt, i.e. nine orders of magnitude less than that of a fully fledged 1000 MW EA unit necessitating a dedicated high intensity accelerator (typically, a few mA of proton beam at 1 GeV). Under those conditions, we had to resort to very sensitive methods to measure the energy produced (hence the energy gain). In so doing, we gained precious information on the spatial distribution of the neutrons within the system [4], which differs profoundly from that of a critical reactor [22]. Also, because the protons came in very short pulses, it was possible to measure the time-dependence of the number of neutrons during the multiplication process. Finally, the energy of the accelerator could be varied and one could therefore measure the gain as a function of the energy of the incoming protons. Most important was the continuous validation of the Monte Carlo simulation that came from the agreement with the experiment.

The test was performed with an existing sub-critical assembly of natural uranium and water. It consists of small cylindrical rods of natural uranium metal, with aluminum claddings, immersed in ordinary water which has the function of moderator. We note that with such choices of moderator and target, the device can never become critical. Its "infinite multiplication factor" is $k_\infty = 0.97$.

The geometrical dimensions of the assembly are schematically shown in Figure 5. We highlight only the fact that the rods are arranged in a hexagonal lattice.

The total energy release in the volume of the assembly is calculated by taking the heat release measured at the different points by the thermometers[1] and the variation with distance of the energy release to perform the integration over the volume. The integration can either be done analytically using the dimensions r = 44 cm, h = 107 cm, and an artificial homogeneous density of uranium. The integration is somewhat tedious and one can also use the almost equivalent analytical procedure of finding a sphere of equivalent radius. However the most exact procedure is to perform a numerical integration using a Monte Carlo calculation that takes into account the detailed inhomogeneous structure of the fuel bars.

Having determined the energy release for different incident beam energies, we can plot the result which is shown in Figure 6. The gain is approximately constant above 1 GeV and drops significantly below 800 MeV.

The neutronic behavior of the assembly has been calibrated with the help of a 58 GBq neutron source (Am-Be) inserted in the centre of the device. The neutron flux measured with a boron loaded counter is shown in Figure 7, and confirms the expected exponential behavior as a function of the distance from the source. We find a multiplication factor for a point-like centered source of $k = 0.915 \pm 0.010$. This is in good agreement with EA-MC calculations which give $k = 0.920 \pm 0.005$.

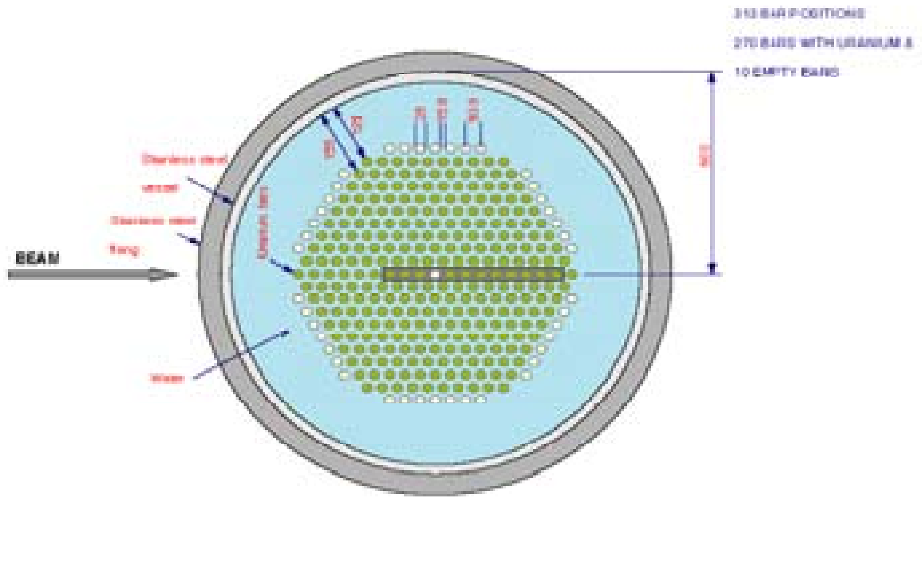

Figure 5. Top view of the sub-critical assembly (dimensions are in millimeters).

[1] The thermometers register the complete energy release not only from fission fragments but also from gammas following neutron capture and from radioactive decays.

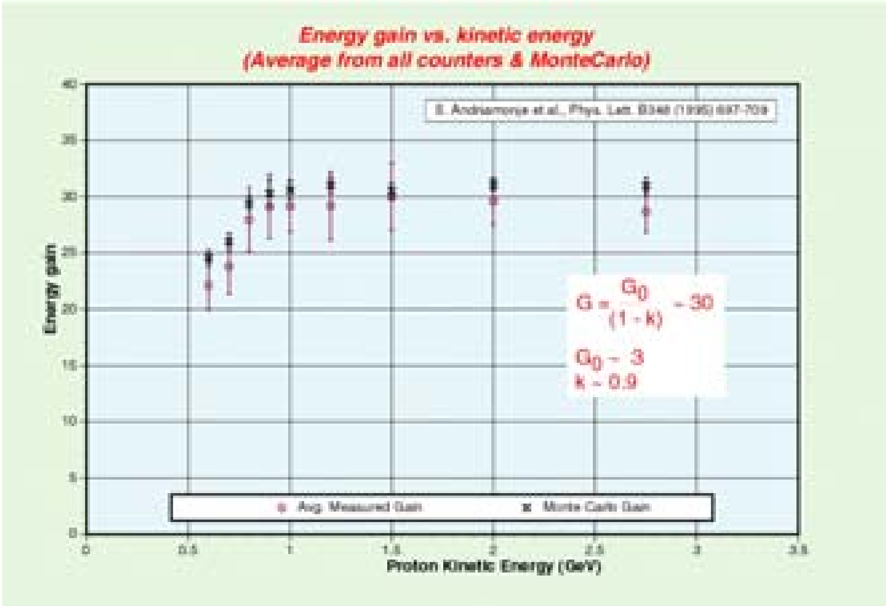

Figure 6. Average energy gain as a function of the proton beam kinetic energy.

The comparison indicates that one cannot use the "critical reactor" formalism to describe a sub-critical system, since all the ortho-normal modes of the "buckling" equation representing the neutron flux distribution must be evaluated, and not only the fundamental mode. The use of the fundamental mode alone, results in an underestimation of the neutron multiplication factor, *k*, since the escape probability is enhanced by the "cosine-like" distribution of the fundamental mode with respect to the real distribution which is exponential. In particular, the data in Figure 7 show that Monte Carlo calculations carried out in the "reactor mode" with MCNP-4B [23] give large disagreements with measurements (e.g., $k = 0.868 \pm 0.002$).

3.2 The TARC Experiment

TARC (Transmutation by Adiabatic Resonance Crossing) is part of a broader experimental program designed to directly test some of the basic physics concepts applicable to the field of radioactive waste elimination. In a system such as the Energy Amplifier [2], where the transuranium elements are destroyed by fission, the long-term (≥ 500 years) radiotoxicity of the waste is dominated by long-lived fission fragments which can, in practice, only be destroyed by nuclear decay following neutron capture.

The main goal of TARC is to test a new idea relying on the properties of spallation neutrons diffusing in lead, the use of adiabatic resonance crossing, to efficiently destroy long-lived fission fragments. More generally, TARC is a systematic study of the phenomenology of spallation neutrons in pure lead.

In fact, neutrons produced by spallation at relatively high energy (a few MeV), after having been quickly moderated by (n,xn) and (n,n') reactions down to energies of a few hundred keV, will slow-down quasi adiabatically with small iso-lethargic steps and reach the capture resonance energy of an element to be transmuted, where it will have a high probability of being captured.

This is the case of ^{99}Tc, which has a strong neutron capture resonance at 5.6 eV (4000 barn), covering four average lethargy steps. The ^{99}Tc resonance integral is 310 barn while the cross

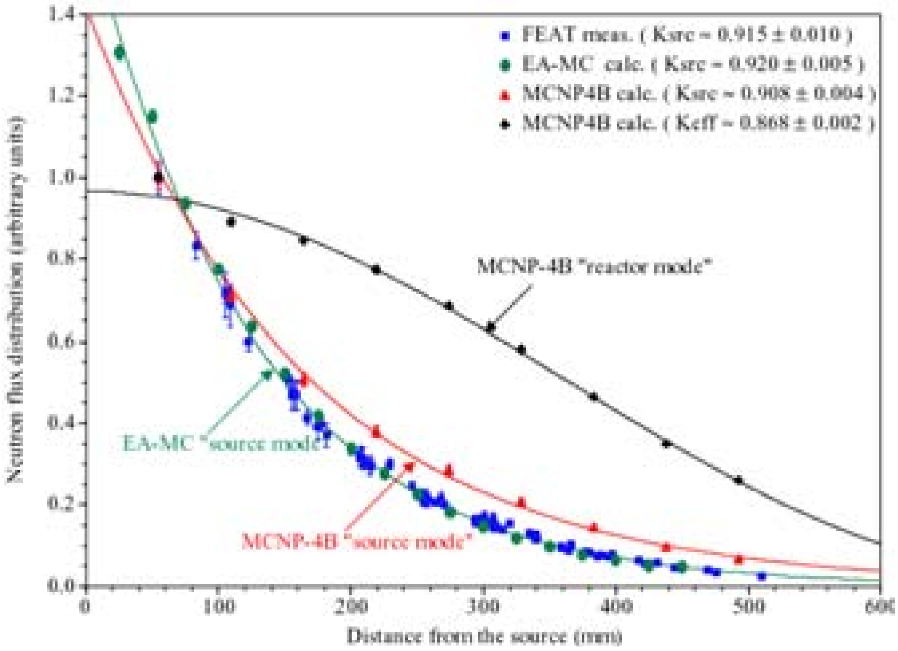

Figure 7. Comparison of the neutron flux distribution between measured data and Monte Carlo calculations carried out (i) in the "source mode" with EA-MC and MCNP-4B, and (ii) in the "reactor mode" with MCNP-4B.

section at thermal neutron energies is only of the order of 20 barn. Neutron capture on ^{99}Tc ($t_{1/2}$ = 211,100·a.) produces ^{100}Tc ($t_{1/2}$ = 15.8 s) which then decays to ^{100}Ru, a stable element. Thus, the radiotoxicity can be eliminated in a single neutron capture, and, since ^{100}Ru has a small neutron capture cross section and both ^{101}Ru and ^{102}Ru are stable, essentially, no new radioactive elements are produced.

The experiment known as PS211 [24] was set up at the CERN Proton Synchrotron (Figure 8). The experiment was designed in such a way that several basic processes involved in accelerator-driven systems could be studied in detail:

- Neutron production by GeV protons hitting a large lead volume;
- Neutron transport properties;
- Efficiency of transmutation of long-lived fission fragments in the neutron flux produced by spallation neutrons.

All of this program was achieved by performing (i) neutron flux measurements over a broad energy range, from thermal up to a few MeV (Figure 9), and (ii) neutron capture rate measurements on ^{99}Tc (both differential and integral) and on ^{127}I and ^{129}I (Figure 10).

Both the energy and space distributions and the absolute magnitude of the neutron fluence, throughout the entire lead volume, are well described by the EA-MC simulation. For instance, neutron fluence data from thermal energies up to 2 MeV, over eight orders of magnitude, are very well reproduced at both proton beam energies (Figure 9).

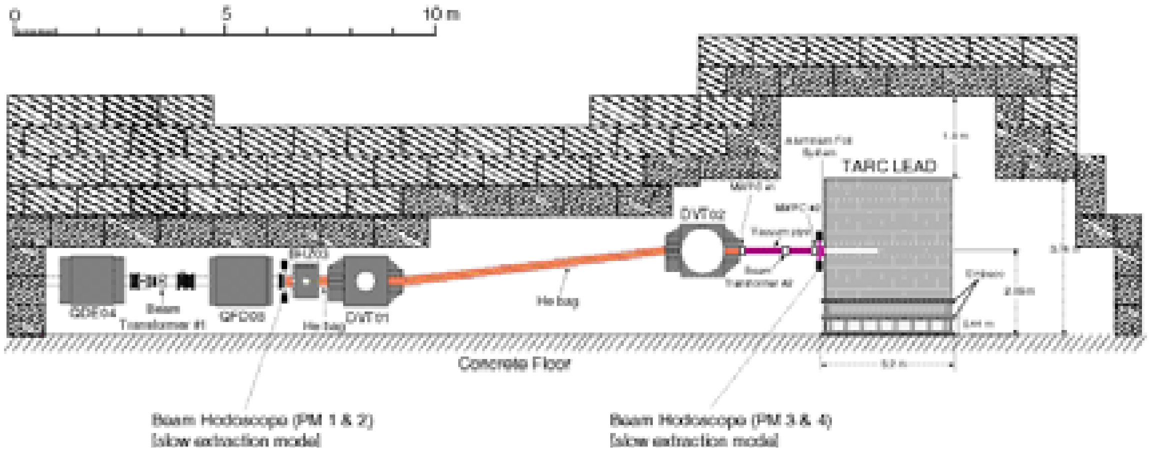

Figure 8. Side view of the TARC experimental area showing the details of the beam line and the lead spallation target.

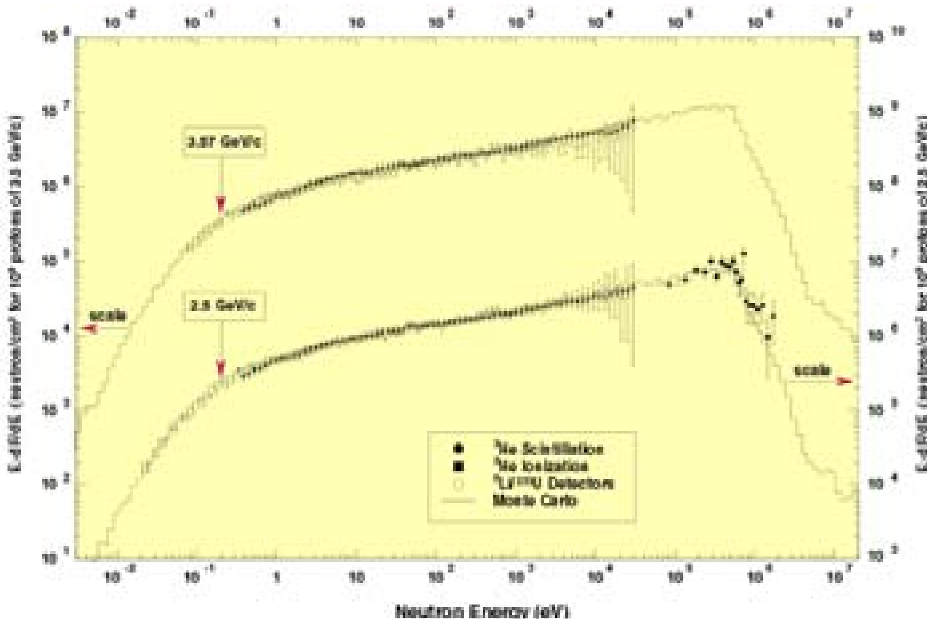

Figure 9. Example of TARC measurements of the neutron fluence. E×dF/E is shown as a function of neutron energy for 2.5 and 3.57 GeV/c protons. The Monte Carlo predictions are shown as histograms. The error bars include both statistical and systematic uncertainties added in quadrature.

We find that about 70% of the spallation neutrons survive the lead capture resonances. The ratio of fluences (between 0.1 eV and 10 keV) measured at two proton momenta (3.5 and 2.5 GeV/c) is found to be 1.52 ± 0.10 consistent with the expectation from the simulation: 1.57 ± 0.01 (statistics) which is essentially the ratio of kinetic energies of the beams.

The global uncertainty in the Monte Carlo prediction comes mainly from the uncertainties in the spallation process (10%) and in the neutron transport in lead (mainly the uncertainty on the lead cross section, ~ 10%) and amounts to a total of about 15%. The impurity content is small enough, with well measured concentrations, so that the corresponding uncertainty on the good agreement with the entire TARC data set validates in detail our Monte Carlo simulation. It confirms in particular that the spallation process is correctly predicted within the errors mentioned above. Typically, we expect 100 neutrons per 3.5 GeV/c proton (neutrons transported down to 19.6 MeV). The dependence of the neutron fluence on energy and space validates the neutron transport in the EA-MC code and tests the reliability of the lead cross sections. The correct prediction of integral and differential transmutation rates for ^{99}Tc, ^{127}I, and ^{129}I validates the efficiency of the TARC method for the elimination of long-lived fission fragments (Figure 10).

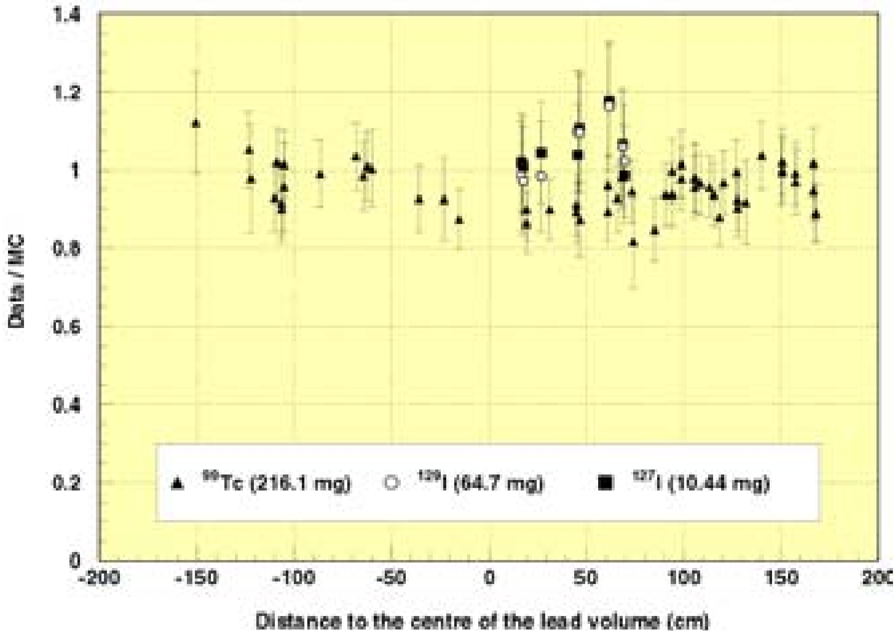

Figure 10. Ratio between measured data and EA-MC predictions of the transmutation rates of ^{99}Tc (216.11 mg), ^{129}I (64.7 mg) and ^{127}I (10.44 mg) samples as a function of distance from the centre of the lead volume.

4. NEUTRONIC CALCULATIONS OF THE EAP80

The key objective of the Energy Amplifier Demonstration Facility (EADF), in a first approximation, aims at demonstrating the technical feasibility of a fast neutron operated accelerator-driven system cooled by molten lead-bismuth eutectic (instead of pure molten lead) and in a second phase that of incinerating TRUs and long-lived fission fragments while producing energy. Therefore, it was adopted to use SuperPhénix type U-Pu MOX fuel (instead of thorium-based fuels) which had already been certified and which utilization in this particular context could stand close scrutiny.

4.1. Reference configuration

As in the case of the Energy Amplifier's conceptual design [2], the EADF core consists of an annular structure immersed in molten lead-bismuth eutectic which serves as primary coolant and spallation target (Figure 11). The central annulus contains the spallation target unit which couples the proton accelerator to the sub-critical core. The core is arranged in a honeycomb-like array forming an annulus with four coaxial hexagonal rings of fuel sub-assemblies. The fuel core is itself surrounded by an annular honeycomb-like array of four rings of dummy sub-assemblies, which are essentially empty ducts. The detailed description of the EADF reference configuration can be found in ref. [1].

The present version of the EA Monte Carlo code package enables a rather complete and detailed modelisation of the EADF reference configuration at the level of individual fuel pins or heat exchanger tubes (presently arranged in square lattices). All the major core components have been taken into consideration, such as:

- the fuel sub-assemblies including hexagonal wrapper tubes, spacer grids, central supporting tubes and individual fuel pins;
- the off-core structures such as the radial/axial reflectors and neutron protection structures;
- the primary cooling circuit including the heat exchangers, primary coolant purification units, gas-riser units and primary flow guides;
- the spallation target unit including its dedicated heat removal system;
- the reactor and safety containment vessels including RVACS system.

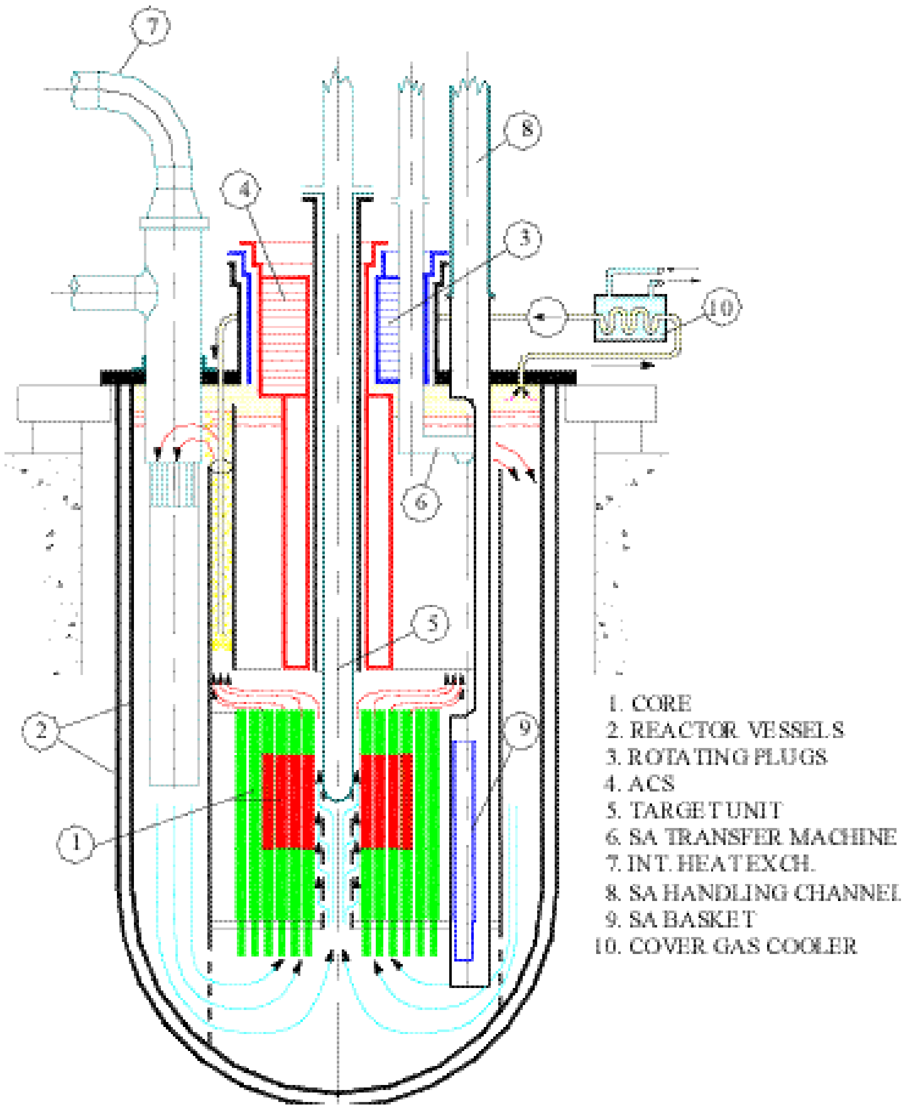

Figure 11. Schematic view of the reactor system assembly (Source: Ansaldo Nucleare [1]).

Contrary to the preliminary neutronics analysis carried out by ENEA [25], the fuel sub-assemblies are not homogenized. In fact, the fuel pin is sub-divided into 4 radial zones consisting of the steel clad, intermediate gap region, the fuel pellet and its central void region; and 21 axial zones taking into consideration the lower and upper plenum together with both depleted-uranium axial blankets, the supporting and spacer grids, and a series of 13 fuel segments.

4.2 Global neutronic parameters at beginning-of-life

The results reported in the paper refer to simulations carried out with 10^4 "source" protons, in which case the statistical error associated with the estimation of the neutron multiplication coefficient is of the order of 0.1% and 2% for the proton beam current. Figure 12 illustrates the variation of the neutron multiplication factor and proton current estimations as a function of the number of primary particles transported.

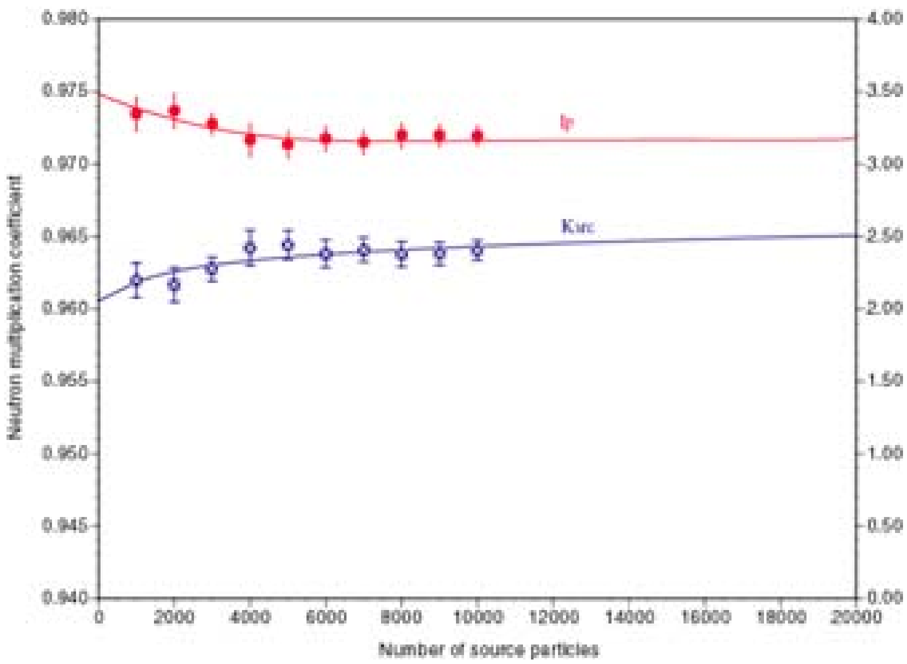

Figure 12. Variation of the k_{src} multiplication coefficient and of the proton beam current as a function of the number of simulated primary protons.

From the results of the convergence run, we can assume that the fission source spatial distribution, i.e. k_{src}, has achieved equilibrium after some 7×10^3 primary protons have been transported. The very poor initial guess for the spacial distribution of fissions[2] causes the first cycles estimates of k_{src} to be extremely low. This situation occurs because only a fraction of the spallation neutron source enters cells that contain fissionable material.

The main global results for the Beginning-Of-Life performance of the EADF reference configuration are summarized in Table I. Table I reports a source multiplication coefficient (k_{src}, depending on the external spallation source which is different, having in general a slightly higher numerical value, from the k_{eff} ~0.962, an intrinsic core feature) of 0.964 at

[2] At k_{src} ~ 0.965 neutron source density is dominated by fissions.

Beginning-Of-Life. This value has been chosen in such a way that criticality conditions are prevented, with adequate margins, under all normal conditions as well as following transient and accident conditions. Despite the large sub-criticality margin (see Table I), the radial and axial power profiles have a moderate peaking factor, mostly as a result of the diffusivity of lead-bismuth eutectic coolant.

The power distribution in a sub-critical system is sensitive to the core intrinsic multiplication factor. As the latter increases towards unity (criticality conditions), the neutron flux distribution flattens on the core fundamental harmonic. Figure 13 shows the axial flux distribution averaged over the fuel elements of a given hexagonal round, thus making apparent the almost axially symmetric nature of the power distribution.

Table I. Main parameters of the EADF reference configuration

Global Parameters	Symbol	EADF	Units
Initial fuel mixture	MOX	$(U\text{-}Pu)O_2$	
Initial fuel mass	m_{fuel}	3.793	ton
Initial Pu concentration	m_{Pu}/m_{fuel}	18.1	wt.%
Initial Fissile enrichment	$Pu^{39,41}/U^{38}$	18.6	wt.%
Thermal Power Output	P_{th}	80	MWatt
Proton Beam Energy	E_p	600	MeV
Spallation Neutron Yield	$N_{(n/p)}$	14.51 ± 0.10	n/p
Net neutron multiplication	M	27.80 ± 0.56	
Multiplication Coefficient	k=(M-1)/M	0.9640 ± 0.0007	
Energetic Gain	G	42.73 ± 0.88	
Gain coefficient	G_0	1.54	
Accelerator Current	I_p	3.20 ± 0.07	mA
Core Power Distributions			
Av. fuel power density	P_{th}/V_{fuel}	255	W/cm^3
Av. core power density	P_{th}/V_{core}	55	W/cm^3
Radial peaking factor	P_{max}/P_{ave}	1.25	
Axial peaking factor	P_{max}/P_{ave}	1.18	

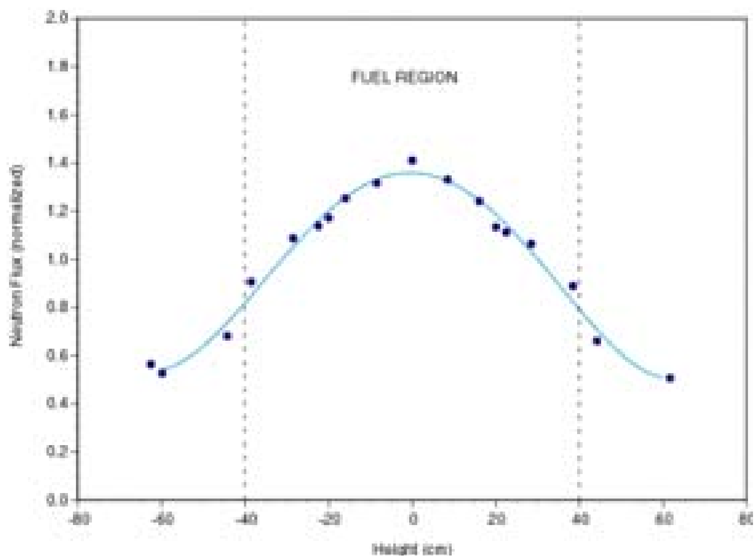

Figure 13. Axial distribution of the neutron flux in the core of the EADF.

The EA-MC code allows for the straightforward production of neutron flux spectra at selected locations. Figure 14 shows some spectra, which allow further insight into the neutronic characteristics of the device. Note in particular the flat neutron spectrum in the primary coolant, where the dominating process is elastic scattering with very small lethargy variation.

The neutron flux spectra generated by the EA-MC code can be used to estimate the heating and damage to structural materials by neutrons with energy below 20 MeV. Indeed, in the EADF one can consider separately the high energy portion of the spectrum, due to the primary proton shower, with its intensity proportional to the beam current, and a lower energy region associated with the fission multiplying medium, proportional to the reactor power. Table II reports results of damage and heating calculations concerning only this second effect. An accurate calculation is made by using the detailed geometric model of the device and by computing the neutron induced damage self-consistently in each geometric element (made by a given material) for each interacting nuclide [26]. The heating and damage-energy cross sections are extracted from the latest nuclear data libraries [6].

The lead-bismuth eutectic pool, in spite of the low absorbing characteristics of the eutectic, allows an important attenuation of the neutron flux and energy, so that the reactor fixed internal structures can be located at a distance of about one meter from the active region of the core, with acceptable dose damage of a few dpa's over the reactor lifetime.

In the EADF heat exchangers, the secondary coolant is a synthetic diathermic oil (with composition close to CH). Neutrons damage the oil by breaking molecular bonds, in an amount roughly proportional to the energy deposited by the neutrons in the oil itself. In order to evaluate such an effect, the energy delivered by the neutrons to the oil has been calculated self-consistently by the EA-MC code. Of course, most of this energy is delivered to elastically scattered hydrogen nuclei: the modification of the neutron spectrum is apparent from Figure 14, where the spectrum in the secondary coolant is plotted.

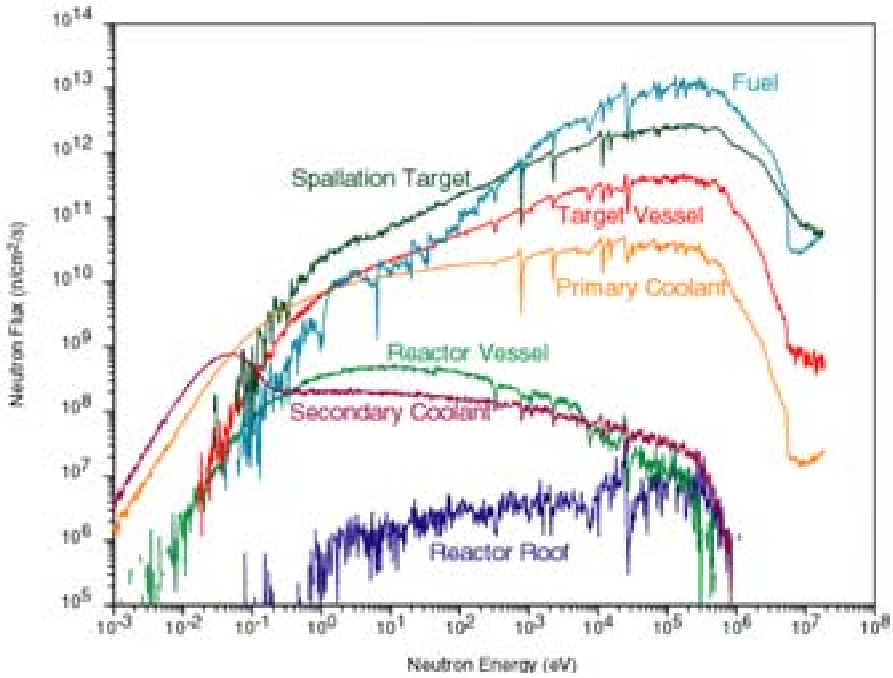

Figure 14. Neutron flux spectra at selected locations of the EADF.

A detailed inventory of the neutron induced reactions taking place in several locations of the device is given in Tables III and IV.

Table II. Integrated flux, heating and damage in the reactor internal structures due to neutrons with energy below 20 MeV computed by EA-MC

Region	Flux (n/cm^2·s)	Heat (W/cm^3)	DPA/a
Reactor roof	$7.0x10^{8}$	$3.7x10^{-6}$	$6.7x10^{-7}$
Reactor vessel	$1.2x10^{11}$	$2.9x10^{-3}$	$8.3x10^{-6}$
Safety vessel	$3.6x10^{10}$	$7.7x10^{-4}$	$2.5x10^{-6}$
Spallation target	$5.9x10^{14}$	0.515	1.566
Target vessel	$8.2x10^{13}$	0.263	0.230
Heat exchangers	$5.8x10^{11}$	$1.6x10^{-2}$	$1.5x10^{-4}$
HX secondary coolant	$7.2x10^{11}$	$2.3x10^{-4}$	$1.1x10^{-4}$
Core neutronic protection	$1.5x10^{13}$	0.184	$5.7x10^{-3}$
Av. fuel	$1.2x10^{14}$	260	0.772
Av. fuel cladding	$3.2x10^{13}$	0.098	0.141
Core radial reflector	$7.1x10^{13}$	0.327	0.146

Table III. Neutron reaction inventory at several locations of the EADF

Neutron Absorption Inventory		Neutron Absorption Inventory	
Reactor vessel	0.33 %	Core upper reflector	5.49 %
Spallation target	1.96 %	Core radial reflector	2.04 %
Flow guides	0.14 %	Core lower reflector	6.91 %
Heat exchangers	0.80 %	Fuel core	72.63 %
Purification units	0.03 %	Primary coolant	6.83 %
Gas injection units	0.16 %	Escapes	0.16 %
Neutron shield	2.52 %	Total	100 %
Main Nuclear Reactions		**Main Nuclear Reactions**	
Capture	66.28 %	Others	0.43 %
Fission	30.95 %	Escapes	0.16 %
n,Xn	2.18 %	Total	100 %

Table IV. Neutron reaction inventory in the fuel core of the EADF

Neutron Absorption Inventory		Neutron Absorption Inventory	
MOX-Fuel	89.60 %	Sub-assembly wrapper	2.01 %
Cladding	3.83 %	Coolant	4.56 %
Main Nuclear Reactions		**Main Nuclear Reactions**	
Capture	54.44 %	n,Xn	2.40 %
Fission	42.61 %	Others	0.55 %

The SuperPhénix high-enrichment UPu MOX fuel, which contains a fissile concentration of about 20%, enables a very high fission-to-capture ratio, close to 0.95, to be achieved. Parasitic captures in the fuel core as well as in the entire device have been reduced to a minimum, i.e. 10 and 25% respectively. Half of the absorption reactions occuring in the lead-bismuth eutectic coolant are due to (n,Xn) reactions, which further increase the overall neutron multiplication in the system. Non-fission multiplicative processes account for more than 2% of the total non-elastic scattering reactions taking place in the EADF.

4.3 Evolution of the main parameters with burnup

The evolution of the EADF fuel core composition in terms of the initial loading of fissile materials, minor actinides bread, and accumulated fission products and poisons is simulated with the EA-MC code. It gives a measure of the transmutation/incineration capability of the EADF for plutonium and minor actinides which may be both relevant for the depletion/denaturation studies for plutonium elimination and for the assessment of the total radiotoxicity eventually left in the environment.

The time dependence of the generic parameters and neutronic characteristics of the EADF after one burnup cycle is summarized in Table V.

Table V. Main parameters of the EADF reference configuration after one burnup cycle

Global Parameters	BOC	EOC	Units
Fuel mixture	$(U\text{-}Pu)O_2$	$(U\text{-}Pu)O_2$	
Fuel mass	3.793	3.723	ton
Pu concentration	18.1	16.7	wt.%
Fissile enrichment	18.6	18.1	wt.%
Fuel burnup	-	20	GW·d/t
Cycle length	-	900	EFPD
Thermal Power Output	80	80	MWatt
Proton Beam Energy	600	600	MeV
Spallation Neutron Yield	14.51 ± 0.10	14.51 ± 0.10	n/p
Net neutron multiplication	27.80 ± 0.56	15.29 ± 0.65	
Multiplication Coefficient	0.9640 ± 0.0007	0.9346 ± 0.0009	
Energetic Gain	42.73 ± 0.88	23.55 ± 1.00	
Gain coefficient	1.54	1.54	
Accelerator Current	3.20 ± 0.07	6.00 ± 0.09	mA

Table V reports the variation in the neutron multiplication coefficient after a fuel burnup of 20 GW·d/t, that is 900 days of operation at 80 MW(th). During this period of operation the reactivity of the EADF drops by 2.94% in Δk, which is compensated by a factor two increase in the accelerator current to 6.0 mA in order to maintain a constant power output .

In Figure 15 the variation of the UPu MOX fuel composition is plotted as a function of time.

The rate of plutonium incineration rapidly settles to approximately 7 kg/TW·h, at the expense of long-lived minor actinides, namely ^{237}Np, ^{241,243}Am and ^{244}Cm, which are produced at a constant rate of 0.23, 0.17 and 0.01 kg/TW·h respectively. This is compared with the production rates of 11 kg/TW·h for plutonium, 0.6 kg/TW·h for ^{237}Np and ^{241,243}Am, and 0.05 kg/TW·h for ^{244}Cm, in a standard PWR.

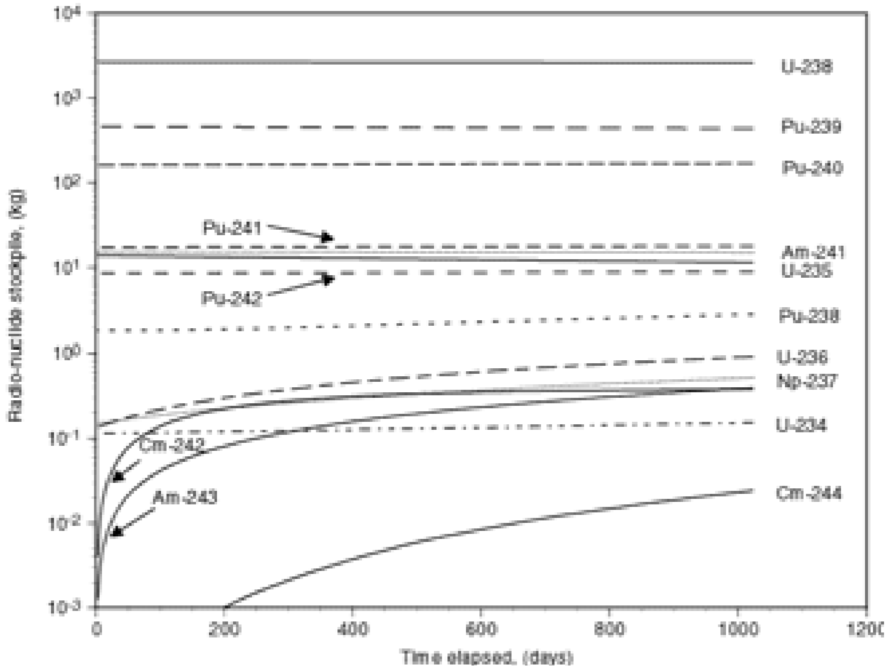

Figure 15. Evolution of the actinide concentrations as a function of time for the UPu MOX fuel in the EADF reference configuration.

5. CONCLUSIONS

By means of 3-D Monte Carlo simulations we have analyzed a number of neutronic features of the proposed Energy Amplifier Demonstration Facility reference configuration. Realistic models of some in-vessel structures which may affect the overall neutronics have been included in the simulations.

The results show that the desired subcriticality level and power distributions can be achieved, and that neutron-induced damage of the structures can be kept at tolerable values. Detailed information on power distribution, neutron spectra and material damage has been generated, which may be useful for the thermal-hydraulic and mechanical designs of the device.

The described core is the initial EADF core, constituted of fuel assemblies loaded with fertile uranium and fissile plutonium MOX fuel at moderately high plutonium concentrations (~20%), which is a well-proven fuel used in SuperPhénix. Different fuel assemblies may be tested subsequently, spanning on a vast range of fuel types and compositions depending on future strategies which will be pursued in the management of the nuclear fuel cycle:

- $^{232}Th/^{233}U$ based fuel to drastically reduce the production of actinides;
- Inert matrix fuels to maximize the actinides transmutation rates;
- Metal, carbide or nitride fuels to exploit different technological advantages such as better thermal conductivity, easier reprocessing, etc…

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REFERENCES

[1] Energy Amplifier Demonstration Facility Reference Configuration: Summary Report, ANSALDO Nucleare, EA-B0.00-1-200 – Rev. 0, (1999).

[2] RUBBIA, C., et al., Conceptual Design of a Fast Neutron Operated High Power Energy Amplifier, CERN report CERN/AT/95-44 (EET), Geneva, (1995).

[3] ATZENI, S., KADI, Y., RUBBIA, C., Statistical Fluctuations in Monte Carlo Simulations of the Energy Amplifier, CERN report CERN/LHC/97-12 (EET), Geneva, (1997).

[4] ANDRIAMONJE, S., et al., Physics Letters B 348 (1995) 697–709 and CALERO, J., et al., Nuclear Instruments and Methods A 376 (1996) 89–103.

[5] ARNOULD, H., et al., Physics Letters B 458 (1999) 167–180.

[6] OECD/NEA Data Bank, Issy-Les-Moulineaux, Paris, France, 1994.

[7] FASSÓ, A., et al., "Intermediate Energy Nuclear Data: Models and Codes", (Proceedings of a Specialists' Meeting, Issy les Moulineaux (France) 1994), published by OECD, (1994) 271, and references therein.
FASSÓ, A., et al., Nuclear Instruments and Methods A, 332, 459 (1993), also, CERN report CERN/TIS-RP/93-2/PP (1993). See also FERRARI, A., et al., "FLUKA-99 Manuel", INFN, Milan, (1999).

[8] COBO, M., et al., Notes on the Study of the Most Reliable Neutron Cross Section Data, CERN internal report, CERN/AT/ET 95-035, Geneva, (1995).

[9] CARMINATI, F., Conversion of EAF 4.2 File for Use by EET Group, CERN internal report, CERN/LHC/EET 96-022, Geneva, (1996).

[10] CULLEN, D. E., "The 1994 ENDF pre-processing codes (PRE-PRO 94)" IAEA Nuclear Data Section, Vienna, IAEA-NDS-39 Rev. 8, (1994).

[11] CARMINATI, F., "PROCESS: A Cross Section Processing Utility", CERN internal report, CERN/LHC/EET 96-005, Geneva, (1996).

[12] CARMINATI, F.,, "EET Code Collection User's Guide", (1996).

[13] CARMINATI, F., I. GOULAS and J.P. REVOL, A Nuclear Data Base for the EA Monte Carlo, CERN internal report, CERN/AT/EET 95-008, Geneva, (1995).

[14] TULI, J. K., National Nuclear Data Center, Brookhaven National Laboratory, Upton, New York 11973, USA, (1995).

[15] AUDI, G., and WAPSTRA, A. H., The 1995 update to the atomic mass evaluation, Nuclear Physics A595 (1995) 409–480.

[16] Final tables of dose coefficients as sent to IAEA for inclusion in the BRSS. National Radiological Protection Board, Chilton Didcot, Oxon OX11 0RQ, U.K., 1995.

[17] International Commission on Radiological Protection, ICRP, S-171 16 Stockholm, Sweden, Edition 91, 919950.

[18] CARMINATI, F., GOULAS, I., and REVOL, J. P., Data for the Simulation of Heat Deposition from Nuclear Decays and Update of the Toxicity Values, CERN internal report, CERN/AT/EET 95-009, Geneva, (1995).

[19] DAVID, S., et al., Link of the Geometry and Material Routines Between the EA-MC Code and FLUKA-99, CERN internal rep. CERN/SL/EET 99-XXX EET, Geneva, (1999).

[20] PILI, P., et al., GeoView: An Interactive 2D/3D viewer for the EA-MC code, (1999).

[21] CARMINATI, F., and Y. KADI, "ADS - Neutronic benchmark (stage 1): a new approach to the design of Accelerator Driven Systems", (Proc. of the IAEA Technical Committee Meeting on the Feasibility and Motivation for Hybrid Concepts for Nuclear Energy Generation and Transmutation, Madrid, Spain, 1997) (1999).

[22] RUBBIA, C., An Analytic Approach to the Energy Amplifier, CERN internal report, CERN/AT/EET 94-036, Geneva, (1994). See also C. RUBBIA, Why So Many Exponentials?, CERN internal report, CERN/AT/EET 94-032, Geneva, (1994).
[23] BREISMEISTER, J. F., MCNPTM - A general Monte Carlo N-particle Transport Code, Los Alamos National Laboratory report, LA-12625-M, (1993).
[24] ARNOULD, H., et al., Neutron-driven nuclear transmutation by adiabatic resonance crossing TARC, European Commission, Nuclear Science and Technology, Euratom, Project Report EUR 19117 EN, (1999).
[25] ATZENI, S., Preliminary Neutronics Analysis of the Energy Amplifier Prototype, ENEA internal report, EA D2.02 4 407, Frascati, (1998).
[26] CARMINATI, F., and Y. KADI, Implementation of a New Routine for Damage Calculations for the EA Monte Carlo, CERN internal report, CERN/LHC/EET 98-004, Geneva, (1998).

Appendix

PAPERS DISTRIBUTED BUT NOT PRESENTED

OMEGA PROGRAMME IN JAPAN AND ADS DEVELOPMENT AT JAERI

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Abstract

Since mid 70’s, JAERI has been developing technologies for a dedicated partitioning process and transmutation system based on the double strata fuel cycle concept. The activities cover the development of a wet partitioning process, design study of an actinide burner reactor (ABR) and an accelerator-driven system (ADS), the development of nitride fuel cycle technologies, and basic research such as nuclear data, fuel property data measurements. Development of a high-intensity proton linac has been carried out under the Neutron Science Project of JAERl which aims at construction of a superconducting proton linac of 8-MW for a 5-MW spallation neutron source of neutron scattering facility and for ADS experimental facility. Since the autumn of 1998, JAERI and the High Energy Accelerator Research Organization (KEK) have been discussing a new proposal to pursue frontier science in particle physics, nuclear physics, materials science, life science, and ADS technology, using a high beam power new proton accelerator. Previously, KEK proposed the Japan Hadron Facility (JHF) that covers a broad range of science from high-energy physics to materials science, by using the primary and various secondary beams from the high power proton accelerator. The present new joint plan, temporarily called the “Joint Project”, is based on these two past proposals. It is also proposed that accelerator facility complex of this Joint Project be constructed at the JAERI site. The Joint Project is preparing the budget proposal for facility construction starting from FY2001.

1. OVERVIEW OF OMEGA PROGRAM

Japan Atomic Energy Research Institute (JAERI) started the development of partitioning process for high-level liquid waste and the design study of transmutation system in mid 1970s. In October 1988, the Atomic Energy Commission (AEC) started a long-term program for research and development on nuclide partitioning and transmutation technology, called "OMEGA". The R&D programs have been jointly stimulated by the collaborative efforts of JAERI and Japan Nuclear Cycle Development Institute (JNC, former PNC). In the public sector, Central Research Institute of Electric Power Industry (CRIEPI) also has been carrying out R&D on this subject.

The JAERI has been developing technologies for a dedicated wet partitioning process for high-level liquid waste of the PUREX reprocessing and the dedicated transmutation system with fast neutron spectrum system with nitride fuel. These developments aim at the establishment of the double strata fuel cycle, in which the partitioning and transmutation will be carried out in a fuel cycle independent from the electric power generation. The JNC has been devoting its efforts for developing an advanced fuel recycling system with TRUEX process for U, Pu and minor actinides (MA) co-extraction and with MOX-FBR for transmutation. The CRIEPI has been developing advanced recycling technologies, based on pyroprocess, for metallic-fueled FBR.

The program is conceived as a research effort to pursue benefits for future generations through the long-term basic R&D, but not to seek a short-term alternative to established or planned fuel cycle back-end policies. The program is to be proceeded in two steps: the phase-I was originally intended to cover a period up to about 1996, and the phase-II to about 2000. The first check and review of the phase-I of the program by the AEC was started in February 1999.

2. R&D ACTIVITIES FOR OMEGA PROGRAM AT JAERI

Since the mid 1970s, JAERI has been developing a partitioning process of high-level liquid waste (HLLW) and a concept of a dedicated transmutation system based on the double strata fuel cycle (Fig. 1). The goal of technological development is the 99.5% reduction in the inventory of MA, ^{99}Tc and ^{129}I which are source terms for long-lasting radiological potential hazard (Fig. 2). JAERI's activities cover the following areas: (a) development of the four-group

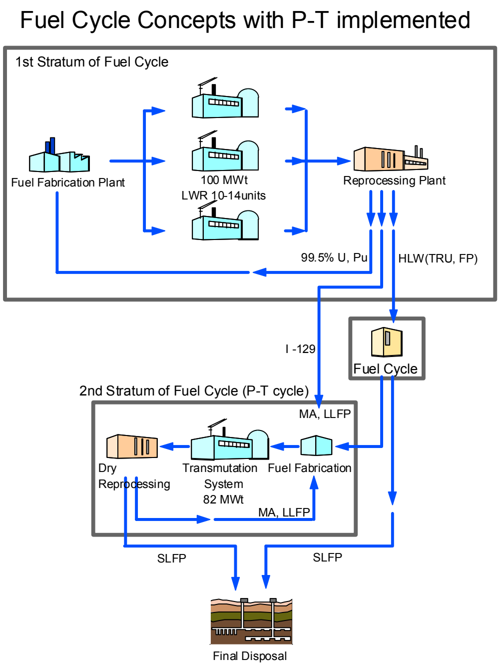

Figure 1. Double-strate Fuel cycle concept

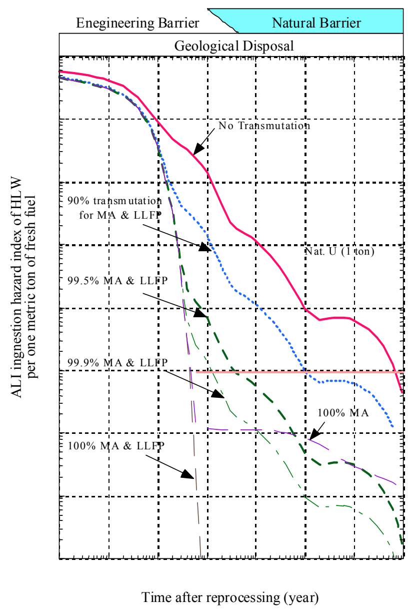

Figure 2. Reduction of Potential Radiological Toxicity of Long-lived Nuclides by Transmutation
LLFP : FPs with $T_{1/2}$ □ 30 years

partitioning process, (b) design study of the actinide burner reactor and the accelerator-driven system, (c) development of an intense proton accelerator, (d) development of nitride fuel cycle technologies, and (e) basic research for supporting the development of transmutation systems. The R&D activities at JAERI are illustrated in Fig.3.

2.1. Double strata fuel cycle [1]

JAERI developed the concept of the double strata fuel cycle, which consists of a power reactor fuel cycle and a P&T cycle. A remarkable point is that the latter cycle is independent from the former cycle, i.e., Partitioning and Transmutation (P&T) is dedicated for HLW management. Since P&T cycle handles only the elements contained in HLLW from commercial reprocessing, heavy metal throughput in the cycle is about 1/30 of that of the commercial fuel cycle, which will result in compact and economical P&T cycle facilities and in effective partitioning and transmutation. Introduction of P&T may be easier due to the independence of P&T from the commercial fuel cycle. Troublesome MAs, from the viewpoint of handling of heavy neutron emitter, can be confined in one small P&T cycle.

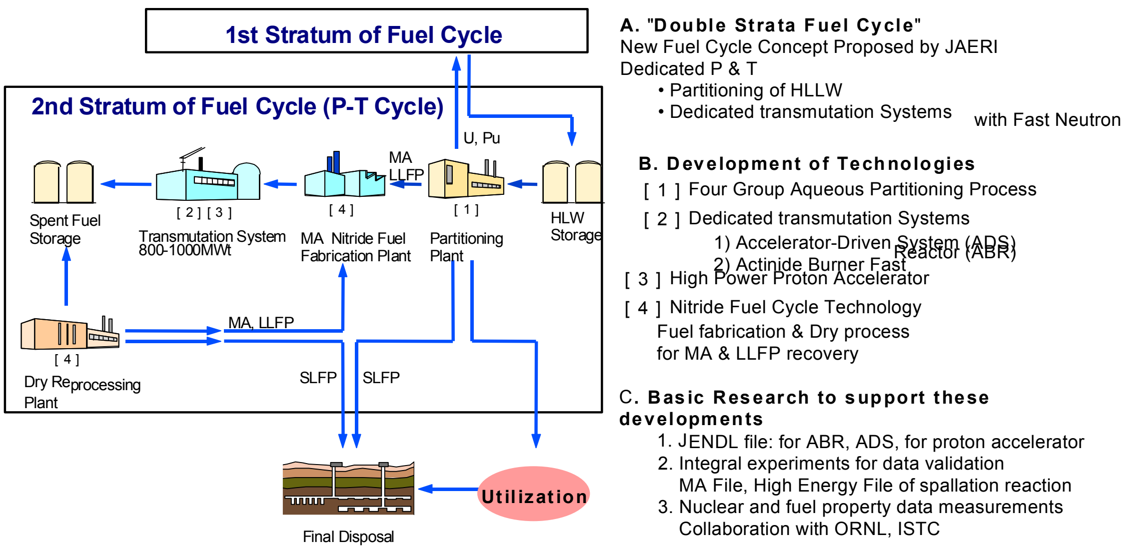

Figure 3. JAERI's activities for OMEGA program.

2.2. Design study of the dedicated transmutation systems

The reactor physics and ex-core fuel handling characteristics were compared between MA transmutation in LWR, FBR and in a dedicated transmutation system with a very hard neutron spectrum. The production of much heavier MA such as Bk, Cf in LWR and FBR is significantly higher than those in the dedicated system. The increase of heavier MA needs the reinforced neutron radiation shielding for the fresh fuel handling (manufacturing and transportation) in power reactor fuel cycle. This may cause the increase of electricity generation cost. In the case of a dedicated system, the shielding and the decay heat removal are much severer problem but the cost of construction and operation of the dedicated system is not significant because of significantly smaller throughput in the P&T cycle of the double strata fuel cycle [2]. From these viewpoints, the design study at JAERI has been devoted to of the transmutation systems has been devoted to dedicated systems with very hard neutron spectra. In the design study, nitride fuel was selected because of its good thermal property and compatibility with pyrochemical reprocessing. Two types of dedicated transmutation systems were designed, one is a MA burner fast reactor and the other is an accelerator-driven subcritical system.

a) MA burner reactors (ABR: actinide burner reactor) [3, 4]

Two types of ABRs were designed: a lead-cooled pin fuel ABR (L-ABR) and a helium-cooled particle fuel ABR (P-ABR). The fuel material in these ABRs is nitride mixture of 65% of MA and 35% of highly enriched (90%) uranium. The ABRs have very hard neutron spectra with core-averaged energy around 720 keV. The amount of MA fissioned in an ABR is about 250 kg/a. In a dedicated MA burner reactor, the content of ^{238}U in fuel should be minimal in order to avoid further production of MA because ^{238}U is a mother material of MA. Due to the very hard neutron spectrum and no content of ^{238}U in ABR fuel, the effective delayed neutron fraction in an ABR becomes very small. To increase the fraction, highly enriched uranium is mixed in fuel.

The design parameters of the ABRs are summarized in Table I.

Table I Design parameters of Actinide Burner Reactors

	L-ABR [1)]	P-ABR [2)]
Fuel concept	pin-bundle	coated particle
material	$(64NpAmCm\text{-}36U^{3)})_{1.0}N^{4)}{}_{1.0}$	$(65NpAmCm\text{-}35U^{3)})_{1.0}N^{4)}{}_{1.0}$
MA initial loading, kg	918	2870
MA/U	588/330	1865/1005
Reactor power, MW(th)	180	1200
Coolant material	Lead	Helium
Neutron flux, 10^{15}n/cm^2·s	3.1	6.6
Core averaged mean neutron energy, keV	720	720
Reactivity (% Δk/k)		
Coolant-void reactivity/core	-1.3	——
Doppler reactivity/core (Δt=300°C)	-0.01	-0.01
Kinetic parameters		
β_{eff}	2.6×10^{-3}	2.6×10^{-3}
L_p, sec	1.3×10^{-7}	1.5×10^{-7}
Cycle length, full-power days	550	300
MA burnup, %/cycle	11	13

(1) L-ABR: MA nitride fuel with lead cooling burner reactor
(2) P-ABR: MA particle fuel with He cooling burner reactor
(3) 90% enriched uranium
(4) 15N enriched

b) Accelerator-driven transmutation system (ADS) [5, 6]

Two types of ADS were designed: (a) ADS with Na-coolant and tungsten target, and (b) lead-bismuth target/coolant ADS. In these ADS with subcritical core of k_{eff} of about 0.95 and a 1.5 GeV - 45 mA proton beam, MA undergoes fission at a rate of approximately 250 kg/a (10% of inventory) and generates 820 MW fission energy. The design parameters of Na-cooled and lead-bismuth cooled ADS are shown in Table II.

Table II Design parameters of 820 MW(th)-ADS
(Proton Beam 1.5 GeV - 45 mA, ~30 Spallation neutrons/Proton)

Coolant	Na	Pb-Bi
Target	Solid Tungsten	Pb-Bi
Initial Inventory of MA	1950 kg	2500 kg
k-effective (Initial/Max./Min.)	0.93 / 0.94 / 0.90	0.95 / 0.95 / 0.94
Coolant void reactivity	+ 4.5 % dk/k	-4.8 % dk/k
Doppler coefficient	$- 2.2\times10^{-4}$ T dk/dT	$- 3.7\times10^{-4}$ T dk/dT
Power density (Max./Av.)	550 / 380 MW/m^3	310 / 180 MW/m^3
Coolant Temperature(Inlet/Outlet)	330 / 430 °C	
Coolant velocity (Max.)	8 m/s	2 m/s

2.3. Comparison of ABR and ADS

As it was discussed in the section 2.2 of this paper, the dedicated system is preferable for MA transmutation. In a critical burner reactor, ABR, the highly enriched uranium is needed to increase □eff. Under the present Nuclear Non-proliferation Treaty regime, the use of highly enriched uranium in a reactor is eventually impossible. The advantages of ADS are the design flexibility for subcritical core and the large margin for criticality-related accidents. From these reasons, we can conclude that the ADS is the best option for MA transmutation.

2.4 Development of a high-intensity proton accelerator, nitride fuel technology

(1) A high-intensity proton linear accelerator [7].

A high-intensity proton linear accelerator with a beam power up to ~8 MW has been proposed for basic science and ADS development at JAERI. Development of an accelerator is carried out as the part of Neutron Science Project of JAERI (the details are discussed in the section 3 of this paper). The layout of the proposed accelerator system is illustrated in Fig. 4.

(2) Development of nitride fuel cycle [8].

Feasibility of fabrication and pyrochemical separation of nitride fuel has been studied. The favorable thermal properties of the nitride fuel make full utilization of a cold-fuel possible. In a cold fuel concept, fuel temperature is to be kept lower than one third of its melting point to reduce mass transport. Since reduced mass transport results in smaller swelling and gas release, thickness of coating layer can be minimized to give large heavy-metal density in order to obtain hard neutron spectrum in a core.

A droplet of the actinide nitrate solution with a carbon suspension turns into a solid mixture in a form of (oxide + carbon) microspheres by a microwave gelation apparatus, then the microspheres are converted to the nitride by a carbothermic synthesis. In the particle-fuel concept, the TiN coating consists of both high-density and low-density layers.

Irradiated nitride fuels are electrorefined in a LiCl-KCl eutectic melt. The recovered metals have to be converted to nitrides. The recovery of U metal has been demonstrated in laboratory runs of the fused salt electrolysis of UN. Conversion of the metal to the nitride has been readily made in liquid Cd with N_2 cover gas.

3. NEUTRON SCIENCE PROJECT OF JAERI [9]

JAERI has been proposing the Neutron Science Project which aims at the construction of the world's most powerful spallation neutron source and related research facility complex to enhance basic sciences and ADS development. The major facilities would be a superconducting proton linac, a 5-MW target station allowing neutron pulses for neutron scattering research, and research facilities for ADS experiments, neutron physics, materials irradiation, and spallation radioactive ion (RI) beam production for exotic nuclei science.

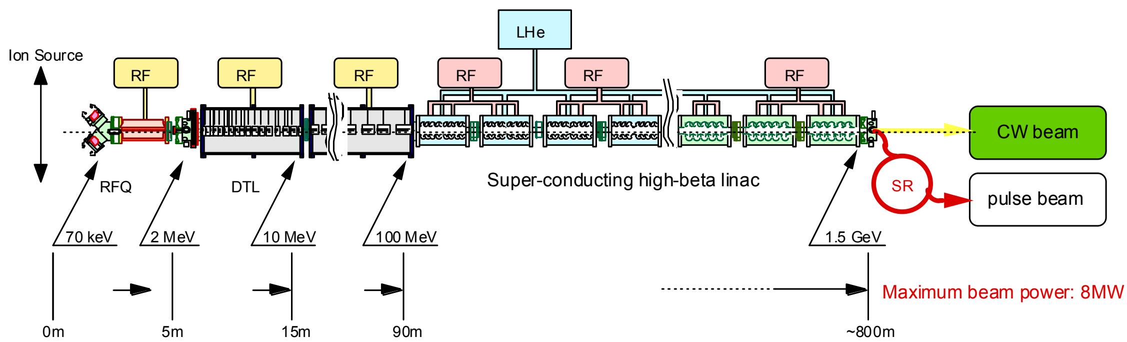

Figure 4. Conceptual layout of intense proton accelerator.

The ADS experiments under the project would cover the two steps of ADS development. The first step will be a feasibility study of the system concept by using subcritical system of UO_2 fuel at a very low power level. These experiments will check the system operability and MA transmutation capability. The second step will be demonstration tests with a 30 - 60 MW experimental reactor. Technical feasibility of spallation target and beam window will be also tested in the second-step.

With an initial power level of 1.5 MW (1.5 GeV, 1mA) on target with the pulse mode, the superconducting linear accelerator complex would be upgraded to 8 MW (5 MW for a spallation neutron source) with cw mode (pulse mode for a spallation neutron source).

Since one of the major technical challenges of a high-current accelerator resides in the low-energy section, JAERI has been carrying out R&D for ion source, RFQ, and DTL for a high power linac since 1990. In a recent beam test with a 150 mA (peak) ion source and a 2-MeV RFQ, the peak current of 70 mA with a duty factor of 8% was achieved. A hot test model of a DTL for mock-up of the low-energy portion (2 MeV to 100 MeV) was fabricated and tested for high-power and high-duty (50%) operation.

Another challenge is the development of a super-conducting linac for accelerating protons. A superconducting cavity for β =0.5 structure was manufactured and the world record of the maximum surface field of 47MV/m at 2.1K was obtained [7]. The layout of the proposed facilities is illustrated in Fig. 5.

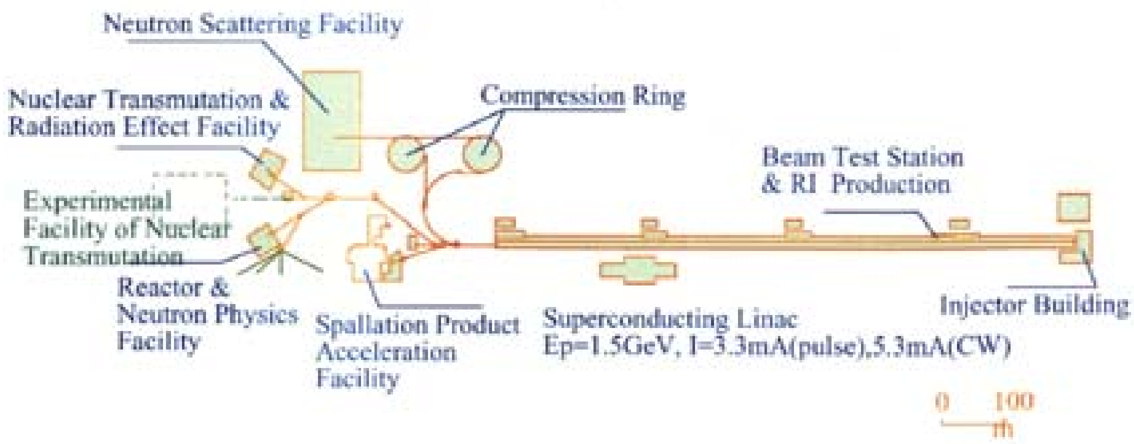

Figure 5. Neutron Science Project (NSP) at JAERI.

4. THE JOINT PROJECT FOR HIGH-INTENSITY PROTON ACCELERATOR AND RESEARCH FACILITY CONSTRUCTION BETWEEN JAERI AND KEK

The Joint Project is a new proposal to pursue frontier science in particle physics, nuclear physics, materials science, life science, and ADS technology, using a new proton accelerator complex at the highest beam power in the world. The plan has been discussed and it is proposed jointly by the High Energy Accelerator Research Organization (KEK) and JAERI. Previously, these institutions proposed the Japan Hadron Facility (JHF) at KEK and the Neutron Science Project at JAERI, respectively. The discussion was stimulated since the two projects proposed independently the construction of the common facilities, namely, high-intensity proton accelerators and high-power spallation neutron sources, The present new joint plan, temporarily called the "Joint Project", is based on these two past proposals. It is also proposed that accelerators and other facilities of this Joint Project be constructed at the JAERI site [10].

The JHF Project at KEK aimed to provide the highest beam intensity in the world among accelerators of these energies to promote hadronic sciences. Its accelerator complex would consist of a 200-MeV proton linac, a 3-GeV booster proton ring and a 50-GeV main proton synchrotron ring. Science programs at the JHF project cover a broad range from high-energy physics to materials science, by using the primary and various secondary beams from the accelerator complex [11]. The facility layout of JHF is illustrated in Fig. 6.

The Joint Project has two phases. The "Phase 1" accelerator complex consists of

- 400-MeV normal-conducting Linac,
- 600-MeV Linac (superconducting) to increase the energy from 400 to 600 MeV,
- 3-GeV synchrotron ring, which provides proton beams at 330 μA (1 MW), and
- 50-GeV synchrotron ring, which provides proton beams at 15 μA (0.75 MW).

In addition, an upgrade towards 5-MW proton beam power at the few GeV energy region is proposed as a "Phase 2" project of the present proposal. Fig. 7 shows the basic configuration of the proposed "Phase 1" accelerator complex.

At the initial stage, the normal conducting 400 MeV Linac will be used as an injector to the 3-GeV ring. At the stage when the superconducting 600 MeV Linac becomes stable, however, this 600-MeV Linac will be switched as the injector to the 3 GeV ring.

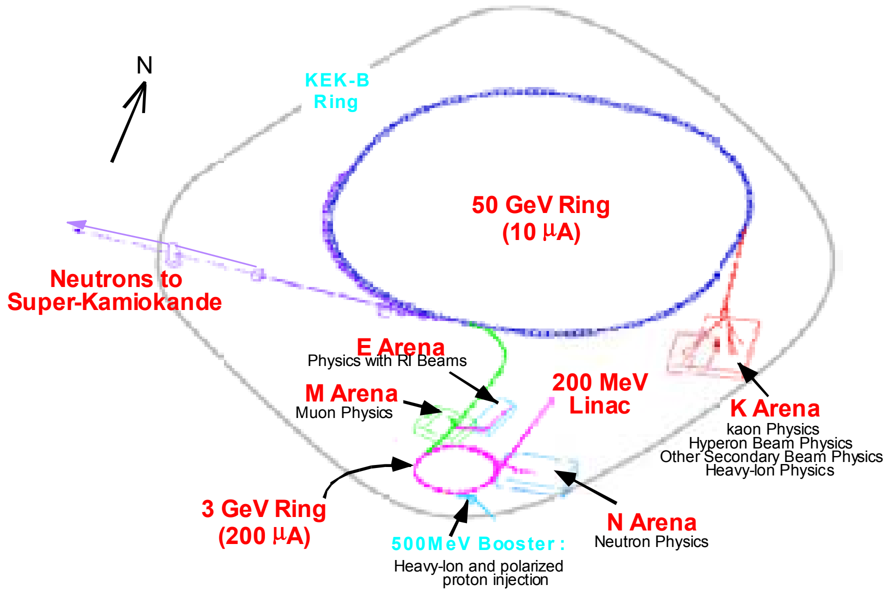

Figure 6. Japan Hadron Facility (JHF) Project at KEK.

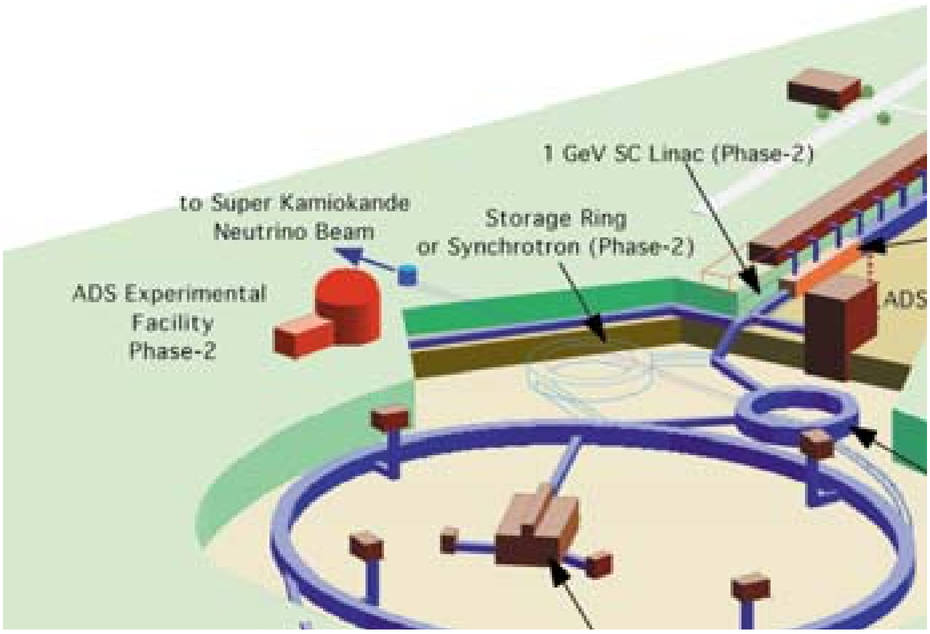

Figure 7. JAERI-KEK joint project.

At the 50-GeV Proton Synchrotron nuclear/particle physics experiments using kaon beams, antiproton beams, hyperon beams and primary proton beams are planned. Experiments on kaon rare decays, an experiment on neutrino oscillation from ν_μ to ν_τ using the Super-Kamiokande as a detector, etc. will also be carried out.

The 3-GeV ring will be used as a booster synchrotron for the 50-GeV main ring. In addition, it is designed to provide beam power of 1 MW. Extensive physics programs which cover nuclear/particle physics, condensed matter physics, materials sciences and structural biology will be carried out there. Among them the major highlights are materials sciences and structural biology using neutrons produced in spallation reactions.

In addition to neutrons, muon beams are also important in which μSR (muon spin rotation/relaxation), muon catalyzed fusion, and other materials sciences can be conducted. Also, the high-current 600-MeV Linac will be used for experiments for development of ADS.

The beam power is sufficiently high for all the above sciences. However, a higher beam power will enrich, in particular, neutron sciences. Thus, we plan in Phase 2 to upgrade the accelerator to 5 MW for spallation neutrons. The construction of the Phase 1 will be started in 2001 and will be completed in 2005.

5. ADS DEVELOPMENT UNDER THE JOINT PROJECT[12]

The major technical issues for developing ADS are; liquid metal (lead-bismuth) coolant, beam window under a high differential pressure load as well as thermal stress, material problem under severe radiation of both energetic proton and neutron, stable operation of an accelerator-driven subcritical system, a high intensity proton accelerator with high reliability and very low beam trip frequency. The scenario for the development of the accelerator-driven transmutation system is illustrated in Fig. 8. As shown in the scenario, the experimental program will proceed in a stepwise manner, depending on the available coolant/target technology, the accelerator beam power and the funding.

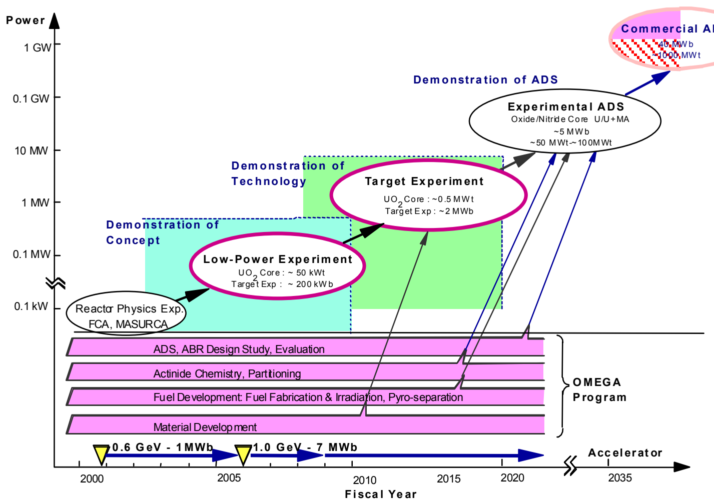

Figure 8. Development path for accelerator-driven transmutation system.

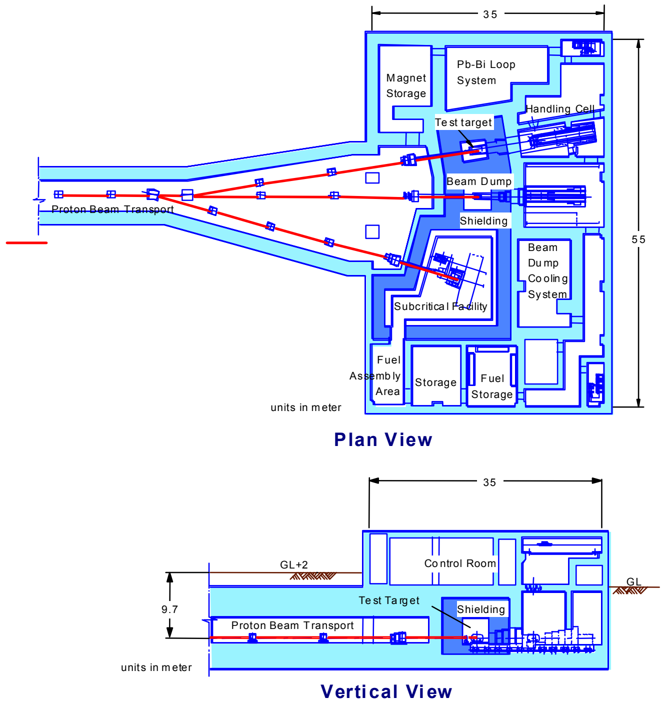

Figure 9. Layout of ADS experimental facility.

The initial zero phase experiments will be carried out using the existing fast critical facility FCA of JAERI with a Cf neutron source installed for reactor physics experiments for a subcritical system of various k_{eff}.

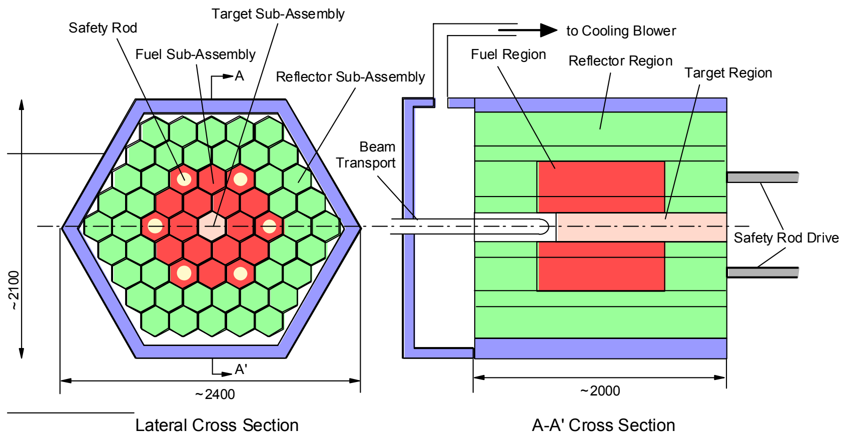

Fig. 10. Low power subcritical experimental facility.

The objectives of the first phase experiments under the joint project are the demonstration of the basic concept of an ADS at the very low fission power level (50kW) and the demonstration of key technologies, namely, beam windows and lead-bismuth coolant/target at the proton beam power of up to 200kW. The reactor-physics experiments will be also carried out. A proton beam is introduced horizontally to the ADS experimental facility and then the beam will be splitted into two, one beam into a subcritical core facility, the other beam into an engineering test facility for a beam window and a lead-bismuth target with a loop (Fig. 9). At the subcritical core facility, the stable operation will be tested for a hybrid system composed with a solid spallation target and a subcritical core driven by a proton beam at a low power level ~kWt. The subcritical core is a fast neutron system loaded with 20%-enriched uranium-oxide fuel, and cooled by forced flow of air (Fig. 10).

The second phase will be engineering tests of the ADS. The subcritical core facility will be upgraded up to a power level of around 500 kW. The subcritical core is loaded with 20%-enriched uranium oxide fuel and cooled by natural convection lead-bismuth. Thermal conduction and natural convection of lead-bismuth are sufficient to remove the heat of 500 kW. Beam window and lead-bismuth target/coolant test will be conducted with beam power of 2MW.

The third phase is to demonstrate the ADS technologies, the safety of the system, and the transmutation performance. The demonstration will be made on an integrated target/core assembly at a moderate power level of around 50 MW with a continuous-wave (CW) proton beam. The fuel is 20%-enriched uranium oxide fuel in an initial core and in the second core uranium nitride fuel. The core is cooled by forced convection of lead-bismuth. The proton beam required will be 1 GeV, 1 - 5 mA in average current, operated in a CW mode. The third phase is presently beyond the scope of the Joint Project. It will be undertaken according to check and review results after the second phase.

ACKNOWLEDGEMENT

The Joint Proposal was developed under the collaborative efforts of the JHF Project Office directed by Dr. S. Nagamiya of KEK and the Center for Neutron Science of JAERI.

REFERENCES

[1] MURATA, H., and KUROI, H., "A proposed concept for actinide-waste transmutation", in Nuclear Technologies in Sustainable Energy System (selected Papers from an IIASA Workshop) ed. G.S. Bauer and A. McDonald (Springer-Verlag, 1983).
MURATA, H., and MUKAIYAMA, T., Fission reactor studies in view of reactor waste programmes, Atomkernenergi-Kerntech., 45 (1983).

[2] MUKAIYAMA, T., et al., "Importance of double strata fuel cycle for minor actinide transmutation", (Proc. OECD/NEA 3rd Information Exchange Meeting on P&T), Cadarache, France, (1994).

[3] MUKAIYAMA, T., et al., ”Conceptual study of actinide burner reactors", (Proc. International Reactor Physics Conf., Jackson Hole, USA, 1988), Vol. IV, (1988) 369.

[4] MUKAIYAMA, T., OGAWA, T., and GUNJI, Y., "Minor actinide burner reactor and influence of transmutation on fuel cycle facilities", (Proc. Technical Committee Meeting on Safety and Environmental Aspects of Partitioning and Transmutation, 1993), IAEA-TECDOC-783, (1995) 105.

[5] TAKIZUKA, T., NISHIDA, T., AND SASA, T., "Conceptual design study of accelerator-driven systems for nuclear waste transmutation", (Proc. 2nd Int. Conf. on Accelerator-Driven Transmutation Technologies and Applications, Kalmar, Sweden, 1996) Vol. l, (1996) 179

[6] TSUJIMOTO, K., et al., “Conceptual study of the lead-bismuth cooled accelerator driven transmutation system”, to be presented in 2nd ANS Topical Meeting on Nuclear Applications of Accelerator Technology (AccApp’98), 1998, Gattlinburg, USA.

[7] MIZUMOTO, M., et al., “Proton linac development for neutron science”, this conference Section D (ADTT’99, Praha, CZ, 1999).

[8] OGAWA, T., et al., "Nitride fuel cycles on pyrochemistry", (Proc. Int. Conf. on Future Nuclear Systems, GLOBAL ’97, Yokohama, 1997) Vol.2, (1997) 812.

[9] MUKAIYAMA, T., et al., "Neutron science project at JAERI", (Proc. Topical Meeting on Nuclear Applications of Accelerator Technology, Albuqueraue, USA, 1997) (1997).

[10] Joint Project Team of JAERI and KEK, The Joint Project for High-intensity Proton Accelerators, JAERI-Tech 99-056 (KEK Report 99-4, JHF-99-3), (JAERI, 1999).

[11] JHF Project Office, Proposal for Japan Hadron Facility, KEK Report 97-3, JHF-97-1 (High Energy Accelerator Research Organization, 1997).

[12] Transmutation Group of JAERI, Section 2.5 Transmutation in the report The Joint Project for High-intensity Proton Accelerators, to be published 1999. (in Japanese)

ADS ACTIVITIES IN THE NETHERLANDS

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Abstract

The problem of long- lived nuclear waste was under consideration in the Netherlands during a number of years. Within this framework different reactor concepts have been considered. One of the most promising concepts appears to be the Accelerator Driven System (ADS) or Fast Energy Amplifier (FEA), as it has been proposed by C. Rubbia of GERN. To unite the effort on accelerator driven reactor concepts in the Netherlands, KEMA, NIKHEF, ECN, KVI, and IRI have joined their forces in a co-operative program since 1996. Cupertino covers the technical fields of nuclear data, reactor physics, and innovative fuels.

1. INTRODUCTION

In recent years, a lot of attention has been given in the Netherlands to the problem of long-lived nuclear waste, which is one of the main objections against nuclear energy. Within this framework different reactor concepts have been considered. One of the most promising concepts appears to be the Accelerator Driven System (ADS) or Fast Energy Amplifier (FEA), as it has been proposed by C. Rubbia of GERN. Apart from the obvious advantages in terms of low waste production and transmutation capacity, this concept also has advantages with regard to nuclear safety and efficient long-term use of energy resources.

To unite the Dutch effort on accelerator driven reactor concepts, KEMA, NIKHEF, ECN, KVI, and IRI have joined their forces in a co-operative program CLEAN (Col-Laboration on Energy Amplification in the Netherlands) since 1996. In order to develop the CLEAN project in coming years, collaboration has been sought with other teams working on ADS.

In 1998 a bilateral collaboration agreement between CEA and ECN (now NRG) has been signed in the field of "Innovative Systems for the Back-end of the Fuel Cycle". Cooperation covers the technical fields of nuclear data, reactor physics, and innovative fuels.

Each of these Dutch institutions is introduced below together with a summary of the activities related to the ADS concept. Contributions to international ADS projects are also presented. They directly arise from current activities, and available competence.

As the nuclear departments of KEMA and ECN has merged into a new company NRG (Nuclear Research and Consultancy Group) in 1998, the activities of the former nuclear departments of KEMA and ECN are presented as the activities of NRG.

2. NRG

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Who are we?

NRG, an ECN-KEMA company, was established in 1998 through the merger of ECN's and KEMA's business activities in the nuclear fields. NRG provides expertise and services in support of the safe, ecologically sound and efficient use of nuclear installations and develops and applies spin-off technology for the non-nuclear markets. In addition NRG participates in technology development programs for advanced nuclear power reactor designs.

NRG is divided in six product groups:

- Materials, Monitoring & Inspection;
- Fuels, Actinides and Isotopes;
- Risk Management & Decision Analysis;
- Radiation & Environment;
- Irradiation Services;
- Plant Performance and Technology.

The product groups offer services for energy utilities, government organisations and various branches of industry.

The activities described in this paper are being performed in the product groups Fuels, Actinides (FAI), Plant Performance and Technology (PPT), Materials, Monitoring & Inspection (MMI), and Isotopes and Irradiation Services (IS). NRG has a wide experience in the irradiation and Post Irradiation Examination (PIE) of fuels, and related nuclear and thermal calculations for such experiments. Further nuclear research is performed on several levels, ranging from basic nuclear data and transport modelling to full-core optimisation and core calculations using 3-D steady-state and transient analysis codes. The product group IS operates the High-Flux-Reactor HFR-Petten and the Hot Cell Laboratories, where respectively the irradiation and PIE are performed.

A major programme is the RAS programme to investigate partitioning and transmutation of actinides and long-lived fission products. NRG has also played a key role in the previous Thorium project and also in EFTTRA experiments.

What are we doing on the accelerator option?

NRG is mainly interested in Accelerator-based reactor systems from the point of view of waste minimization and reduction.

NRG has main R&D programmes on
a) - Recycling of Actinides and Fission Products (RAS).
This activity of NRG mainly is focussed on the recycling of Actinides and fission products under the project acronym RAS. The ADS is being studied as waste management option rather than as electric power supplier.
b) - The development of nuclear data files for activation and intermediate energy neutron and proton-induced cross sections. There is close international collaboration with NEA/IAEA and institutes working on nuclear data for intermediate energies. NRG co-operates with KVI of Groningen University (AGOR facility) for measurements of proton-induced reactions.

NRG was involved in two EU studies, one on Thorium and one on Accelerators, in the NEA exchange programme on Partitioning and Transmutation, and in the CRP of IAEA on Accelerator-based systems.

Present and forthcoming activities related to ADS

- Evaluation of P&T (RAS), including ADS as option for waste transmutation;
- Fuel and targets irradiation experiments, including:
  - Tc, I, Am irradiations;
  - Th fuel irradiations;
- Nuclear models and nuclear data files in the energy range up to 200 MeV;
- Reactorphysics studies, including:
  - code development;
  - Monte Carlo calculations with burnup;
- System studies and thermal-hydraulics analysis;
- Scoping study on HFR driven by cyclotron;
- Material Science:
  - Window steel irradiations.

Fuel and Targets

Transmutation of I, Tc, and Americium

In order to be able to transmute americium, plutonium and fission products (e.g. I and Tc) in an efficient manner, detailed knowledge on the impact of irradiation on the behaviour of these elements or inert matrices containing these elements is required. Table 1 gives an overview of the irradiation experiments in the HFR studying this topic. Part of the irradiation programme of NRG is being performed in close cooperation with the EFTTRA group. EFTTRA, which is the acronym for Experimental feasibility of Targets for TRAnsmutation, is a network of research organisations in France, Germany and the Netherlands. The OTTO project is performed in cooperation with JAERI and PSI.

The fission product iodine has been irradiated in the form of CeI_3, NaI and PbI_2 contained in stainless steel tubes. The irradiation of CeI_3 and NaI was very successful and detailed PIE has been performed. The irradiation of PbI_2 was less successful due to strong chemical interaction between the stainless steel container and the PbI_2. Metallic technetium rods have been irradiated very successfully to a rod average transmutation of 18% and a local peak value of 35%. No change in the microstructure and an insignificant swelling of the rod were observed.

In order to be able to transmute plutonium and americium efficiently, inert matrices are required in order to dilute the actinide and serve as a host material. The inert matrices should fulfil a variety of constraints, such as: high melting point, high thermal conductivity, low swelling, low neutron cross section, no formation of long-lived radiotoxic isotopes, etc. Various materials are presently under investigation, such as $MgAl_2O_4$, $(Zr,Y)O_2$ and $Y_3Al_5O_{12}$. ThO_2, which has a good behaviour under irradiation, is also suggested as an inert matrix, although some long-lived actinides are formed during irradiation.

From the research that has been performed up till now it can be concluded that:
1. The fission products I and Tc can probably be transmuted without to large material problems, but their neutron absorption cross sections are rather low.
2. The actinides Am and Pu can be transmuted, but several topics are still under investigation. These topics are:
i) which inert matrix should be used,
ii) how should the actinides be dispersed in the inert matrix,
iii) which fuel-rod design should be used.
These topics are subject of the new EFTTRA project in the 5th Framework Programme of the EU.

Table I. ADS related transmutation irradiation experiments in the HFR.

Name	*Description*	*Status*
T1	technetium and iodine	completed
T2	technetium	completed
	neutron damage in inert matrices	completed
T2bis	neutron damage in inert matrices	completed
T3	neutron damage in inert matrices	PIE ongoing
	inert matrix fuel using enriched UO_2	PIE ongoing
T4	americium in spinel	completed
T4bis	americium in spinel	irradiation ongoing
T4ter	central fuel temperature of spinel/UO_2	irradiation ongoing
OTTO	inert matrices containing plutonium	design phase
T5	americium in inert matrices	proposal
Thorium	ThO_2 based fuel	proposal

Th fuel irradiations

The use of thorium offers challinging options for nuclear waste reduction, both at the back end and at the front end of the fuel cycle. In the past 4th Framework Programme of the EU the Thorium fuel cycle has been studying aiming at waste reduction. In the 5th Framework Programme a new project is being started to supply key data for application of the Th-cycle in PWRs, FRs and ADS, related to Pu/TRU burning and reduction of the lifetime of nuclear waste. To achieve this, the irradiation behaviour of Th/Pu fuel at high burnup and for relevant neutronic conditions will be examined. Vital data for geological disposal scenarios will be determined, and new waste-minimizing routes for reprocessing of thorium based fuel will be

investigated. The very high Pu/TRU consumption in Th/Pu fuel will be validated by full core calculations and essential nuclear data for these core calculations will be determined experimentally.
In this project NRG collaborates with BNFL(GB), CEA (F), FZJ(D), IN2P3(F), JRC-IAM, JRC-IRMM, JRC-ITU, KWO(D). In the context of NRGs interest in reducing the waste problem with help of ADS, NRG takes also part in the IAEA-CRP on "Use of Th-based Fuel cycle in accelerator Driven Systems to incinerate Pu and to reduce Long-term Waste Toxicities". With the 3D Monte Carlo code MCNP different benchmark samples have been calculated and intercompared.

Nuclear data evaluation and validation

At NRG, considerable effort is devoted to the provision of nuclear data that are specific to ADS. The detailed engineering design of an ADS will require that the performance of the spallation target and all the problems related to the existence of high-energy particles can be predicted with sufficient accuracy. Some of the important quantities to assess are:

- intensity and spatial distribution of the spallation neutron flux outside the target,
- radiation damage and gas production in target, window and structural materials,
- radiotoxicity, activity and corrosion problems inside the target,
- required shielding to neutrons with much higher energies than in conventional reactors.

Provision of ADS related data is mainly done by means of theoretical development and computational implementation of nuclear models. For ADS, cross sections for the important materials need to be known for ALL possible reaction channels. This total amount of required information is so large that experiments alone can never cover the nuclear data needs. To fill this gap, the nuclear data must be simulated computationally, with the help of theoretical nuclear reaction models benchmarked against experimental data. The models can then be used in areas where no measurements exist. At NRG, novel optical models, pre-equilibrium, fission, direct and statistical models have been developed and implemented to give an enhanced description of nuclear reactions for nucleon energies up to 200 MeV. A particular improvement concerns the description of intermediate energy fission yields for actinides and sub-actinides. The results are used to generate nuclear data libraries up to 200 MeV, which can be used in combination with intranuclear cascade models that are better for the description above 200 MeV. Apart from this ongoing model and software development, NRG has already created 150 MeV data libraries for Fe, Ni and Pb (for JEFF) and for Cu (for ENDF/B-VI) using the Los Alamos code GNASH. The results at energies above 20 MeV have been combined with existing data below 20 MeV to come to one consistent final library. All the nuclear data have been stored in the common ENDF-6 format and have been checked according to a standard QA-system, which includes checking by the CHECKR, FIZCON and PSYCHE codes of Brookhaven National Laboratory and processing by the Los Alamos code NJOY. The enhanced description of nuclear reactions below 150 MeV is the result of a strong nuclear data collaboration with CEA, France and Los Alamos, USA.

The new data libraries have been directly tested on a simple iron benchmark, for a 43 MeV and 68 MeV neutron source, performed at JAERI. Our results confirm the findings at Los Alamos National Laboratory, namely that calculations with 150 MeV neutron data files give a drastic improvement of the description of macroscopic experiments, when compared with the intranuclear cascade code LAHET. A similar comparison for concrete is under study.
In the frame of the 4th framework programme of the European Community contributions have been given to the IABAT project (evaluation of Pb).

A revised evaluated data file for ^{232}Th is foreseen as an end product of the 5th framework programme. New neutron capture cross sections measured in Geel and Bordeaux will be combined with the other reaction channels to come to a new ^{232}Th file for JEFF. New intermediate energy region evaluation will be constructed for Pb, U8, Zr, and Fe. NRG takes part in the European 5th framework project HINDAS (High and Intermediate energy Nuclear Data for Accelerator-driven System).

Reactor Physics studies

The neutronics behaviour of an ADS is an important subject to be studied in the development phase. Detailed ADS design studies call for an accurate 3D geometry model. Preferably, these studies should therefore be performed with Monte Carlo codes. At NRG substantial effort was put into the use of the Monte Carlo code MCNP (and the special high-energy version MCNPX) for ADS applications. This work has a close connection with the nuclear data evaluation activity. Basic data libraries for high-energy ADS applications are converted at NRG for use in MCNP, so that the most recent and accurate data are used in the ADS neutronics analyses. This processing activity is a very specialistic job, which is performed with many years of experience. Extensively validated software is used, which leads to high-quality nuclear data libraries which are used for all Monte Carlo radiation transport analyses.

The code MCNP in its standard shape can only be used for static calculations. However, the time-dependent behaviour of an ADS requires full 3D burnup analyses. In order to make these analyses possible the NRG OCTOPUS code system was extended to include the possibility of performing 3D burnup analyses with MCNP and a burnup code (ORIGEN or FISPACT).

The version of MCNP which is used at NRG was substantially extended with important modifications for ADS applications. A useful addition for ADS analyses is the possibility to generate 3D reaction rates in a calculation. This allows the calculation of power distributions and flux distributions in a 3D Monte Carlo calculation. Thus, the insight in the neutronics behaviour of an ADS may be substantially increased.

Both the criticality analyses and the radiation shielding analyses of an ADS may be performed with MCNP. The NRG experience with radiation shielding analyses was partially obtained from a contribution to the European Fusion Technology Programme. A result of this work was the recognition of the importance of nuclear data for the calculated response. This led to a new evaluation of 56Fe nuclear data, which was extensively benchmarked. The evaluation experience is used for the generation of new nuclear data libraries for ADS as well (see above).

Finally, MCNP is used for the calculation of various nuclear data sensitivity coefficients. This is an important feature in the ADS design phase, as it allows parametric studies with a 3D Monte Carlo code. The results of sensitivity analyses may be combined with covariance data in order to calculate the uncertainty in calculated responses.

ADS related benchmark calculations with MCNP have already been performed in the framework of the IAEA-CRP on "Use of Th-based Fuel cycle in accelerator Driven Systems to incinerate Pu and to reduce Long-term Waste Toxicities". In the 5th Framework Programme of the EU MCNP calculations will be performed in a proposal on Accelerator Driven Design Analysis and in the subcritical neutronics experiment MUSE.

System studies and thermal-hydraulics analysis

In the PPT product group the activities related to Accelerator-Driven Systems are at the moment primarily thermal-hydraulics and safety studies. Up till now the work involved adaptation of the Light Water Reactor transient analysis code TRAC to be able to use the code with liquid lead coolant and a subcritical core. Several transient studies have been performed with the adapted TRAC code on the Fast Energy Amplifier concept proposed by Rubbia. For example, the beam-dump scenario has been studied, in which a full power proton beam is dumped instantaneously in the reactor, while the coolant is stagnant and at low temperature. Because of the high inertia of the liquid lead cooled, temperatures in the system raise significantly. For the 5th European Framework Program NRG takes part in a proposal on Accelerator Driven System design analysis, in which possible transients of several design options will be determined and evaluated.

As of next year also Computational Fluid Dynamics studies will be performed as part of the recently formed Benchmark Working Group on Heavy Liquid Metal Thermal-Hydraulics. Next to this, there are some ideas to perform structural mechanics calculations to evaluate the behavior of the window under the foreseen thermal and radiation loads.

Some effort has been spend on the subjects of economics of Accelerator Driven Systems. As soon as the ADS designs become more detailed NRG is able and willing to tackle regulatory issues and perform severe accident analysis as well as environmental risk assessments.

Accelerator-Driven Subcritical High Flux Irradiation Facility

The materials testing reactor in Petten (HFR) started operation in 1962. In 1984 a first replacement of the reactor vessel took place. The following vessel replacement is foreseen for the year 2015. This could be a suitable moment to perform changes in the reactor core as well. One of the possibilities is the conversion of the HFR to a subcritical system driven by a lead cooled spallation source in the centre of the core. This option should be considered if there are important advantages in performance leading to new fields of R&D and innovative applications. The minimum requirement is that the existing applications can be met.

The possible advantages are:
- excellent safety features due to subcriticality;
- more experimental positions due to reduction of control elements;
- reduction of overall fuel requirements;
- potential of very intense (fast) flux zone, which could be extended for experiments;
- flexible operation, e.g. extension of cycle length;
- possible auxiliary use of the accelerator.

A study is underway to check whether this important modification is feasible and whether there are advantages without and with optimisation. A conceptual design study is planned. Desired research is focused on target and target cooling.

Window material irradiation

The irradiation effects in structural materials is a critical issue for the integrity of the spallation target of an ADS. The irradiation damage consists of important atomic displacement rate (order 100 dpa/a in the window) and spallation elements production (h, He, P, S, Ti, V, Ca) that will result in significant hardening and embrittlement. To resist to such

severe irradiation conditions conventional martensitic steels (9Cr1Mo and 9Cr1MoVNb) and experimental clean martensitic steels of 7-9Cr1-2W are proposed. The 7-9Cr1-2W type steels are emerging as the most promising among the Low Activation Materials studied for the past 10 years in the various fusion programmes. In the 5th EU framework programme a research project has been established for further material testing taking into account the existing data and the production of spallation elements which is specific of the spectrum the target window will have to sustain. In this project the following institutions are working together: CEA(F), CIEMAT(E), CNRS(F), ENEA(I). SCK/CEN(B), RIT(S), PSI(CH), and NRG(NL).
At NRG low temperature (<400°C) up to 10 dpa irradiations of 7-9Cr1-2W in the Material testing reactor HFR will be performed with and without spectrum tailoring to evaluate the effect of dpa and He production. These irradiations will complete the scarce existing data on embrittlement on martensitic steels in a temperature range where the irradiation hardening and decrease in fracture toughness are expected to be particularly important. Metallurgy before irradiation and also Post Irradiation mechanical testing will be performed in the hot lab of NRG with related microstructure examinations. Transmutation calculations on the accelerator window have been performed with the codes LAHET, MCNP, MCNPX, and FISPACT.

3. KVI

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Who are we?

The Kernfysisch Versneller Instituut (KVI) is a national institute for both fundamental and applied nuclear and atomic research. Since the beginning of 1996 it operates a super-conducting cyclotron AGOR (Accelerator Groningen ORsay), build in collaboration with the Institut de Physique Nucleaire at Orsay, France. Agor is a versatile machine which can accelerate light and heavy particles: protons from 120 to 200 MeV and all heavier nuclei, depending on their charge state, from 6 to 100 MeV per nucleon. As a university institute the KVI is supported by the University of Groningen and by the Stichting voor Fundamenteel Onderzoek der Materie (FOM). It has a staff of scientific technical and administrative amounting to about 110 FTE.

What are we doing on the accelerator option?

The KVI has at its disposal a modern superconducting cyclotron capable of producing a variety of nuclear beams including protons with energies up to 200 MeV. In view of studying the 'accelerator option' for energy amplification we mention the KVI technical expertise on cyclotrons (Schreuder, Brandenburg et al.) and the scientific involvement of the institute with investigations of reactions on lead and actinide nuclei (van der Woude, Harakeh et al.) and with spectroscopy of actinide nuclei (van Klinken et al.).

KVI takes part in several EU 5th framework programmes, among which HINDAS (High and Intermediate energy Nuclear Data for Accelerator-driven System). The KVI is in charge of the measurements of the neutron- and light-charged particle multiplicities and participates in the measurements of double-differential cross sections. This work is performed in close collaboration with nuclear data group of NRG.

4. NIKHEF

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Who are we?

NIKHEF is the Dutch national institute for nuclear physics and high-energy physics. In the context of nuclear physics the Institute operates a 900 MeV linac and a storage/stretcher ring. The high-energy physics activity involves participation in experiments at CERN and DESY. The institute can contribute to the design and construction of parts of the accelerator.

5. IRI/TUD

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Who are we?

IRI is part of the Delft University of Technology, and is the Dutch national universities centre for education, training and scientific research on nuclear reactors and radiation applications. The fundamental research comprises reactor and neutron physics, radiochemistry, radiation chemistry, nuclear detection methods, as well as condensed matter physics. The institute's facilities comprise a 2 MW swimming-pool nuclear reactor, a 3 MV pulsed electron accelerator, positron beams, and two-phase boiling loops. The institute employs some 220 persons.

What are we doing on the accelerator option?

For ADS the emphasis is put on reactor physics (especially reactor dynamics and measurements of reactivity effects and kinetic parameters) and thermohydraulics. In the EU 5th framework programme a contributions will be given to the design analysis of an ADS and a theoretical and experimental contribution to the subcritical neutronics experiment MUSE at the Masurca Facility in Cadarache (France).

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