Technology and Components of Accelerator-Driven Systems

Workshop Proceedings
Mito, Japan
6–9 September 2016
Technology and Components of Accelerator-Driven Systems

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Foreword

The accelerator-driven system (ADS) is a potential transmutation system option as part of partitioning and transmutation strategies for radioactive waste in advanced nuclear fuel cycles. Following the success of the workshop series on the utilisation and reliability of the high-power proton accelerators (HPPA), the scope of this new workshop series on technology and components of accelerator-driven systems (TCADS) has been extended to cover subcritical systems as well as the use of neutron sources. The Third Workshop on Technology and Components of Accelerator-Driven Systems took place on 6-9 September 2016 in Mito, Japan, and was hosted by the Japan Atomic Energy Agency (JAEA).

This workshop, organised by the Nuclear Energy Agency (NEA) Nuclear Science Division, provided experts with a forum to present and discuss state-of-the-art developments in the field of ADS and neutron sources. A total of 42 papers were presented during the oral and poster sessions. The programme included a session of international and national programmes and five technical sessions addressing ADS experiments and test facilities, accelerators, simulation, safety, data and neutron sources. These proceedings include all the papers presented at the workshop. The opinions expressed are those of the authors only, and do not necessarily reflect the views of the NEA, any national authority or any other international organisation.

Acknowledgements

The OECD Nuclear Energy Agency (NEA) gratefully acknowledges the Japan Atomic Energy Agency (JAEA) for hosting the Third Workshop on Technology and Components of Accelerator-Driven Systems.
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<td>AAA</td>
<td>Advanced accelerator applications</td>
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<tr>
<td>ADANES</td>
<td>Accelerator-driven advanced nuclear system</td>
</tr>
<tr>
<td>ADS</td>
<td>Accelerator-driven system</td>
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<tr>
<td>AEC</td>
<td>Atomic Energy Commission</td>
</tr>
<tr>
<td>ANIMMA</td>
<td>Advancements in nuclear instrumentation measurement methods and their applications</td>
</tr>
<tr>
<td>ANL</td>
<td>Argonne National Laboratory</td>
</tr>
<tr>
<td>APT</td>
<td>Accelerator production of tritium</td>
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<tr>
<td>ATW</td>
<td>Accelerator transmutation of waste</td>
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<tr>
<td>BARC</td>
<td>Bhabha Atomic Research Centre</td>
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<tr>
<td>BBA</td>
<td>Beam-based alignment</td>
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<td>BOC</td>
<td>Beginning of cycle</td>
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<tr>
<td>BP</td>
<td>Burnable poison</td>
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<tr>
<td>BPM</td>
<td>Beam position monitors</td>
</tr>
<tr>
<td>BRAHMA</td>
<td>BeO reflected and HDPe moderated multiplying assembly</td>
</tr>
<tr>
<td>BT</td>
<td>Beam trip</td>
</tr>
<tr>
<td>BW</td>
<td>Beam window</td>
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<tr>
<td>BWLAP</td>
<td>Backward wave linear accelerator of protons</td>
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<tr>
<td>CAS</td>
<td>Chinese Academy of Sciences</td>
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<tr>
<td>CIEMAT</td>
<td>Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas</td>
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<tr>
<td>CLEAR</td>
<td>China lead-alloy-cooled reactors</td>
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<tr>
<td>CR</td>
<td>Control rod</td>
</tr>
<tr>
<td>CSS</td>
<td>Core support structure</td>
</tr>
<tr>
<td>CW</td>
<td>Continuous wave</td>
</tr>
<tr>
<td>DCCT</td>
<td>DC current transformer</td>
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<tr>
<td>DEMETRA</td>
<td>Development and assessment of structural materials and heavy liquid metal technologies for transmutation systems</td>
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<td>DHR</td>
<td>Decay heat removal</td>
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<tr>
<td>Abbreviation</td>
<td>Description</td>
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<tr>
<td>DN</td>
<td>Delayed neutron</td>
</tr>
<tr>
<td>EA</td>
<td>Energy amplifier</td>
</tr>
<tr>
<td>ECR</td>
<td>Electron cyclotron resonance</td>
</tr>
<tr>
<td>EFA</td>
<td>Experimental fuel assemblies</td>
</tr>
<tr>
<td>EFIT</td>
<td>European Facility for Industrial Transmutation</td>
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<tr>
<td>EMF</td>
<td>Electro-magnetic flow</td>
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<tr>
<td>EMP</td>
<td>Electro-magnetic pump</td>
</tr>
<tr>
<td>EOC</td>
<td>End of cycle</td>
</tr>
<tr>
<td>ESNII</td>
<td>European Sustainable Nuclear Industrial Initiative</td>
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<tr>
<td>ESS</td>
<td>European spallation source</td>
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<tr>
<td>EUROTRANS</td>
<td>European research programme for the transmutation of high-level nuclear waste in an accelerator-driven system</td>
</tr>
<tr>
<td>FA</td>
<td>Fuel assembly</td>
</tr>
<tr>
<td>FASTEF</td>
<td>Fast-spectrum transmutation experimental facility</td>
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<tr>
<td>FC</td>
<td>Fission chambers</td>
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<tr>
<td>FCG</td>
<td>Fatigue crack growth</td>
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<tr>
<td>FCT</td>
<td>Fast beam current transformer</td>
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<tr>
<td>FEM</td>
<td>Finite element method</td>
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<td>FMEA</td>
<td>Failure mode and effects analysis</td>
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<tr>
<td>FP</td>
<td>Fission products</td>
</tr>
<tr>
<td>FPGA</td>
<td>Field programmable gate arrays</td>
</tr>
<tr>
<td>FREYA</td>
<td>Fast Reactor Experiments for hYbrid Applications Project</td>
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<tr>
<td>GBEM</td>
<td>Grain boundary engineering materials</td>
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<tr>
<td>GUINEVERE</td>
<td>Generator of Uninterrupted Intense NEutrons at the lead VEnus Reactor Project</td>
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<tr>
<td>HLW</td>
<td>High-level waste</td>
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<td>HPPA</td>
<td>High-power proton accelerators</td>
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<tr>
<td>HPSIM</td>
<td>High-performance simulator</td>
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<tr>
<td>HWR</td>
<td>Half wave resonator</td>
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<tr>
<td>HXTR</td>
<td>Heat exchanger tube rupture</td>
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<tr>
<td>IAEA</td>
<td>International Atomic Energy Agency</td>
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<tr>
<td>ICSBEP</td>
<td>International Criticality Safety Benchmark Evaluation Project</td>
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<tr>
<td>IFMIF</td>
<td>International fusion materials irradiation facility</td>
</tr>
<tr>
<td>IHEP</td>
<td>Institute of High Energy Physics</td>
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<td>IMAC</td>
<td>Initial MYRRHA Accelerator Consortium</td>
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<tr>
<td>Abbreviation</td>
<td>Description</td>
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<td>--------------</td>
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<tr>
<td>IMP</td>
<td>Institute of Modern Physics</td>
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<td>INEST</td>
<td>Institute of Nuclear Energy Safety Technology</td>
</tr>
<tr>
<td>IPS</td>
<td>In-pile-sections</td>
</tr>
<tr>
<td>IQ</td>
<td>In-phase and quadrature-phase</td>
</tr>
<tr>
<td>IRPhEP</td>
<td>International Reactor Physics Experiment Revaluation Project</td>
</tr>
<tr>
<td>ISOL</td>
<td>Isotope separator online</td>
</tr>
<tr>
<td>ITER</td>
<td>Including the fusion reactor</td>
</tr>
<tr>
<td>IVFHM</td>
<td>In-vessel fuel-handling machines</td>
</tr>
<tr>
<td>IVFS</td>
<td>In-vessel fuel storage</td>
</tr>
<tr>
<td>IWF</td>
<td>Inter-wrapper flow</td>
</tr>
<tr>
<td>JAEA</td>
<td>Japan Atomic Energy Agency</td>
</tr>
<tr>
<td>J-PARC</td>
<td>Japan Proton Accelerator Research Complex</td>
</tr>
<tr>
<td>JSNS</td>
<td>Japan spallation neutron source</td>
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<tr>
<td>KALLA</td>
<td>Karlsruhe liquid metal laboratory</td>
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<tr>
<td>KIPT</td>
<td>Kharkiv Institute of Physics and Technology</td>
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<tr>
<td>KIT</td>
<td>Karlsruhe Institute for Technology</td>
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<tr>
<td>LANSCE</td>
<td>Los Alamos Neutron Science Centre</td>
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<tr>
<td>LBE</td>
<td>Lead-bismuth eutectic</td>
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<tr>
<td>LBECS</td>
<td>LBE conditioning systems</td>
</tr>
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<td>LEBT</td>
<td>Low-energy beam transport</td>
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<tr>
<td>LEDA</td>
<td>Low-energy demonstration accelerator</td>
</tr>
<tr>
<td>LEU</td>
<td>Low-enriched uranium</td>
</tr>
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<td>LFR</td>
<td>Lead fast reactor</td>
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<tr>
<td>LINAC</td>
<td>Linear accelerator</td>
</tr>
<tr>
<td>LLFP</td>
<td>Long-lived fission products</td>
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<tr>
<td>LMAD</td>
<td>Liquid-metal assisted damage</td>
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<td>LME</td>
<td>Liquid-metal embrittlement</td>
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<tr>
<td>LPC</td>
<td>Laboratoire de Physique Corpusculaire</td>
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<tr>
<td>LPSC</td>
<td>Laboratoire de Physique Subatomique and Cosmologie</td>
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<tr>
<td>LVRE</td>
<td>Lead void reactivity effect</td>
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<td>MA</td>
<td>Minor actinide</td>
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<td>MAXSIMA</td>
<td>Methodology, analysis and experiments for the “Safety in MYRRHA assessment”</td>
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<td>MC</td>
<td>Monte Carlo</td>
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MEBT Medium energy beam transport
MEGAPIE Megawatt pilot experiment
ML Measurement layer
MOC Middle of cycle
MOGA Multi-objective particle swarm genetic algorithms
MPV Multiple pressure variable
MSM Modified source multiplication
MYRRHA Multipurpose hybrid Research Reactor for High-Tech Applications
MYRTE MYRRHA Research and Transmutation Endeavour
NDT Non-destructive test
NEA Nuclear Energy Agency
NFSP New four-section procedure
OC Oxygen control
OLLOCHI Oxygen-controlled lead-bismuth eutectic loop for corrosion tests in high temperature
OS Oxygen sensors
P&T Partitioning and transmutation
PCGVS Primary cover gas and ventilation system
PEAR Pellet absorber rod
PHITS Particle and heavy ion transport code system
PIE Post-irradiation examination
PIV Particle image velocimetry
PNS Pulsed neutron source
PRS Pressure relief system
QWR Quarter wave resonator
R&D Research and development
RACE Reactor-accelerator coupling experiments
RAM Reliability, availability and maintainability
RAW Radioactive waste
RFQ Radio-frequency quadrupole
RRRFR Russian Research Reactor Fuel Return
RS Reuter-stokes
SCC Space-charge compensation
SCM Subcritical multiplier
<table>
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<td>SESAME</td>
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<td></td>
<td>metal cooled reactors</td>
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<tr>
<td>SGTR</td>
<td>Steam generator tube rupture</td>
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<tr>
<td>SINBAD</td>
<td>Shielding integral benchmark archive database</td>
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<tr>
<td>SNR</td>
<td>Signal-to-noise ratio</td>
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<tr>
<td>SNS</td>
<td>Spallation neutron source</td>
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<tr>
<td>SPT</td>
<td>Small punch tests</td>
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<tr>
<td>SR</td>
<td>Safety rods</td>
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<tr>
<td>SSC</td>
<td>Structures, systems, components</td>
</tr>
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<td>TCADS</td>
<td>Technology and components of accelerator-driven systems</td>
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<td>TEF</td>
<td>Transmutation experimental facility</td>
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<td>TH</td>
<td>Thermal hydraulics</td>
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<td>Thermal-hydraulics and ADS design</td>
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<td>Thermal hydraulics of innovative nuclear systems</td>
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<tr>
<td>TTF</td>
<td>Transit-time factor</td>
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<td>ULOF</td>
<td>Unprotected loss of flow</td>
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<tr>
<td>UNF</td>
<td>Used nuclear fuel</td>
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<td>USFM</td>
<td>Ultrasonic flow metre</td>
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<td>UVP</td>
<td>Ultrasonic velocity profiling</td>
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<td>WP</td>
<td>Work packages</td>
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<td>YSZ</td>
<td>Yttria-stabilised zirconia</td>
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Executive summary

Since 2010, the OECD Nuclear Energy Agency has organised a series of workshops dedicated to technology and components of accelerator-driven systems (ADS). The third workshop took place on 6-9 September in Mito (Japan). The main objectives of this workshop are to stimulate an exchange of information on the state-of-the-art developments in the field of ADS and neutron sources. The main topics of the workshop covered R&D status on ADS including accelerators, neutron sources and subcritical systems for current facilities and future experimental and power systems; technology, engineering and research aspects of the above components; system optimisation for reducing capital and operational costs and the role of ADS in advanced fuel cycles.

During the workshop, the following topics were discussed:

- International and national programmes
- ADS accelerator systems
- Neutron sources
- Design and technology of subcritical systems
- Current ADS experiments
- ADS data and simulations

ADS accelerator systems focused on the reliability and availability of systems looking at the continuous mode operation, operational control and challenges as well as the safety of systems/shielding of the accelerator and beam line. It also highlighted the development and tests of ADS-rated accelerator components. Neutron sources covered current and future intense neutron sources, spallation and non-spallation targets, the different design concepts and required technologies (such as coolant, materials, performance, instrumentation, etc.), operational characteristics and related technical issues. Design and technology of subcritical systems covered design concepts and performance parameters, ADS fuel/target design options, subcritical reactor physics (e.g. reactivity monitoring, etc.), coolant technology and materials, thermal-hydraulics and unique ADS auxiliary systems. Current ADS experiments considered low-power coupling experiments and zero-power physics simulators. The ADS data and simulations topic focused on nuclear and thermal-hydraulics issues, safety analysis and ADS transmutation fuel cycle (simulations of minor actinides (MA) content, final waste).

The workshop was scheduled with 12 invited talks and 32 oral presentations from 57 registered participants.

Welcome address and opening lecture

The welcome address was given by Y. Miura (JAEA) and S. Cornet (NEA). It was then followed by an opening lecture from A. Mueller (France) on the Progress and the Maturity of the field of TCADS where the history of ADS over the past 25 years was highlighted, mentioning related collaborative international
projects. From early conception and studies, the ADS technology has grown to maturity with large scale experiments and major achievements.

**SESSION 1: International and national programmes**

International perspectives from several national and international programmes were presented. *S. Okajima* (JAEA, Japan) described the status of transmutation technologies in Japan where transmutation of radioactive waste using ADS is promoted. R&D programmes on Partitioning & Transmutation (P&T) are being carried out. In Europe, several projects related to P&T led to the creation of the EUROTRANS integrated project. MYRRHA is progressing with several integrated EU projects. In particular, the MYRTE project was initiated to establish a full size injector and studies on lead-bismuth eutectic (LBE) fluid dynamics, LBE chemistry, ADS neutronics and Am-bearing oxide fuels. In China, the ADS development programme was launched in 2010 and phase 1 was successfully completed. Phase 2 is planned to start in 2017. The ADANES concept to transmute waste using gas-cooled/tungsten particle target ADS was introduced.

**SESSION 2: ADS accelerator systems**

Although ADS accelerator projects are considered as long-term projects, significant technical progress was presented. The Chinese ADS project is progressing in collaboration with several Chinese institutes (*Z. Wang, CAS and S. Wei, L. Bian, IHEP*): the fabrication and operation of a long spoke-resonator cryomodule and the pulsed mode operation of the C-ADS Injector I at 10-MeV, 10-mA were outlined. An overview of the progress in other areas such as controls and diagnostics was given. In Europe, the progress in the development of the MYRRHA injector is substantial. *F. Bouly* (LSPC, France) reported the construction and pre-commissioning of the LEBT for MYRRHA accelerators. *H. Saugnac* (INP, France) reviewed the design of the spoke cavities and their cryomodule for the MYRRHA Linac: excellent results on the prototypes were shown and the good safety margin on the specifications is an important asset for the reliability. *R. Garnett* (LANL, USA) described and analysed the common design features of the modern high-power accelerators. *A. Bogomolov* (IPPE, Russia) highlighted the features of a backward travelling wave (warm) Linac for high-power applications. The beam dynamic features are being considered for high-power applications. Finally, *M. Bourquin* (iThEC, Switzerland) gave an overview of the project iThEC for setting up an ADS experiment at INR Troitsk, with the objective to demonstrate waste transmutation and Th fuel cycle.

**SESSION 3: Neutron sources**

*E. Pitcher* (ESS) presented the progress and status of the European Spallation Source (ESS) project. The ESS has the mission to design, build and commission a powerful spallation source for neutron scattering research. It is currently under construction, and aims to deliver first protons to target by the end of 2019. The target station employs several innovative features, including a rotating tungsten target cooled by helium gas, a flat “butterfly” moderator, and beam expansion through raster scanning. *T. Sasa* (JAEA, Japan) described the latest status of a 250 kW LBE spallation target being designed for J-PARC. The target was designed to accept a focused 250 kW proton beam from the J-PARC Linac at a maximum design temperature of 500°C. *T. Wan* (JAEA, Japan) discussed the optimisation of target design for TEF-T LBE Spallation Target. The possibility of cavitation damage caused by pressure waves and turbulent LBE flow was investigated for the TEF-T LBE target through simulations. CFD analyses were carried out to study LBE flow pattern. However, some stagnant regions exist in the LBE for the original target design. To solve this problem, the target head was modified to reduce the stagnant region effectively and efficiently. The modified design showed an improvement of the safety margin of the TEF-T target. *L. Yang* (CAS, China) presented the progress of the granular target and the ADANES concept. The beam has been successfully
commissioned in: MEBT, TCM1, and TCM6 between 2014 and 2016. The 25 MeV LINAC was commissioned in 2016. Key sub-systems of granular target are being developed: mechanical lift, heat exchanger, filter unit, coupling of beam and target, grain erosion the different ADS core coolants were reviewed. The AIROX reprocessing process is being developed in order to operate with a closed fuel cycle. To close the session, J. Knebel (KIT, Germany) introduced the Multiple Pressure Variables (MPV) calculations method for liquid metal applications. A series of validation calculations were carried out on KIT design of HLM target for MEgawatt TARget: Lead-bIsmuth Cooled (META:LIC) and on the LBE-cooled beam window (BW) of MYRRHA.

SESSION 4: Design and technology of subcritical systems

G. Van den Eynde (SCK-CEN, Belgium) described the evolution of the MYRRHA design since 2005 as well as the requirements and main features: 100 MW maximum core power, passive decay heat removal, in-vessel fuel storage and fast fuel assembly unloading. Current design issues and further developments were presented: size of the primary system, polonium issue and oxygen concentration to mitigate corrosion. Current R&D programmes and supporting facilities were summarised. Y. Bai (INEST, China) talked about China strategic plan for nuclear power. China LEAd-based Reactor (CLEAR) was selected as the reference reactor for ADS project and for Lead cooled Fast Reactor (LFR) technology development. CLEAR-I (10 MW, subcritical and critical operation) project includes the following components: CLEAR-S (Lead-alloy cooled non-nuclear reactor integral test), CLEAR-O (Lead zero-power reactor integral test), CLEAR-V (Virtual reactor detailed simulator). Related R&D programmes and supporting facilities were presented. The investigation for sub-criticality adjustment mechanism of LBE-cooled Accelerator-Driven System was highlighted by A. Oizumi (JAEA, Japan). A new concept of sub-criticality adjustment mechanism using control rods (B4C and Ta materials) and burnable poison (Gd-Zr-H) was examined to maintain the same power level with a minimum beam power variation. Burnup calculations were carried out using the (ADS3D) three-dimensional ADS reactor analysis code system with a 70 energy-group nuclear data library based on JENDL-4.0 was used for the analyses. The analyses concluded that both options can be utilised to reduce the beam power variation. It was pointed out that minimising the beam power enhances the beam window design. N. Aizawa (Tohoku Univ., Japan) presented the study of reactivity control method for ADS using of burnable poison oxide. Different burnable poison materials (Gd-Zr-H, GD-Zr-D, and Gd₂O₃) were examined to maintain the same power level with a minimum beam power variation. Beryllium oxide (BeO) or Spinel (MgAl₂O₄) moderators were used. Oxide burnable poison prevents safety concerns related to H or D release from hydride at high-temperature. Different configurations were examined and evaluated. Results of the study showed that six or nine blocks of Gadolinium oxide with Beryllium oxide (98% BeO + 2% Gd₂O₃) can reduce the reactivity swing. J. Knebel (KIT, Germany) highlighted the LBE-cooled fuel assembly mock-ups recently tested at KALLA in nominal and accidental conditions expected in ADSs to investigate the effect of spacers and local blockages in order to provide feedback to the designers. Y. Bai (CAS, China) summarised the functions and specifications of the CLEAR facility, as well and the technical specifications of the prototype components and the results of the pre-test of heavy liquid metal technology based on KYLIN facilities for CLEAR-S. H. Obayashi (JAEA, Japan) presented the results of experiments using ultrasonic flow measurements techniques to monitor LBE flow using a flowmeter developed at JAEA. S. Saito (JAEA, Japan) outlined the status and progress of the development of the LBE corrosion OLLOCHI tests loop and described the various experiments performed with the loop. Liquid Metal Embrittlement is a process determined by the combination of material, stress and environment and requires numerous experiments to quantify the effect and J.B. Vogt (Uni. Lille, France) summarised the results obtained on the study of the mechanical behaviour of T91 martensitic steel in liquid lead-bismuth eutectic (LBE). T. Sugarawa (JAEA, Japan) introduced the R&D activities related to the measurement and control of the oxygen concentration in
LBE. Further improvement was done in order to guarantee adequate oxygen control. R. Sa (INEST, China) presented the study of LBE/steam interaction performed in KYLIN-II to support safety studies.

SESSION 5: Current ADS experiments

The KIPT (Kharkiv Institute of Physics and Technology) ADS facility is a accelerator-driven system designed to maximise its utility and minimise the time for replacing the target, the fuel and the experimental assemblies by using simple and efficient procedures. Lessons learnt from its design and construction were described by Y. Gohar (ANL, USA). Successful operation and accumulation of operation experience are expected. Q. Wu (INEST, China) presented the ongoing R&D activities for the construction of the first multifunctional lead-based zero-power physical testing facility (CLEAR-0). The European experimental programme FREYA Project was presented by A. Kotchetkov (SCK-CEN, Belgium). The programme achieved significant results (as highlighted by V. Bécares-Palacios [CIEMAT, Spain]) not only to support MYRRHA but also to expand knowledge of reactor physics aspects of subcritical system. Additional results are expected in the frame of the MYRTE project, the succeeding programme. Sub-criticality measurement technique is one of key technology for safe operation of ADS. Various subcriticality measurements results considering spatial and energy dependence performed in VENUS-F (T. Chevret and G. Lehaut, Uni. Caen, France) and BRAHMA (S. Bajpai, BARC, India) were presented.

SESSION 6: ADS data and simulations

Y. Dai (PSI, Switzerland) summarised the outcomes of the post-irradiation examination (PIE) of the collaborative MEGAPIE project. PIE was performed on the structural materials, T91 and SS 316L steels, irradiated in the lower part of the MEGAPIE target. No evident failure and detectable change in thickness in the beam window area were observed. No evidence of corrosion damage was detected on the inner surface of the beam window. J. Neuhausen (PSI, Switzerland) gave a summary of the results obtained within the SEARCH project, in particular in the work package WP6 dedicated to the study of the evaporation of volatile products from LBE and their capture from the gas phase. Results of experimental studies on the evaporation of polonium and mercury from LBE combined with density functional theory (DFT) calculations showed no indication of a stable phase in Pb-Bi-Po. D. Lopez (CIEMAT, Spain) presented the computational tool developed for transient and steady state analysis. MCNP and COBRA-4 were coupled for neutronic and thermal-hydraulics coupling analysis. The coupling code has been used in MYRRHA in Unprotected Loss of Flow (ULOF) transient mode less than 60 sec. S. Zhou (CAS, China) summarised the features of a Monte Carlo code called SuperMC designed for nuclear design and safety evaluation and concluded that the code need further V&V. F. Panza (INFN, Italy) presented a new hybrid fast-slow research ADS for material investigation, the spectrum and flux is designed for different purposes such as waste transmutation and training and education. M. Yang (INEST, China) reported on the development of digital virtual technology for China lead-based reactor (CLEAR).
Welcome address
On the progress and the maturity of the field of technology and components of accelerator-driven systems

Alex C. Mueller
Université Paris-Sud, France

Abstract

While many highly attractive features of accelerator-driven subcritical systems were quite obvious when their use was proposed, it also became quickly clear that major challenges existed for the development of the technology and its components in order to achieve a safe and viable new type of nuclear reactor. The last 25 years or so of research and development (R&D) on the various concerned topics have seen a number of quite spectacular advances which demonstrate the maturity to which the field has come. On selected examples, the author will try to show this historical progress to the present advanced status. For constraints of time and personal competence, it is asked for indulgence that the selection has some inherent bias.
Session 1: International and national programmes

Chair: T. Sasa
Japan Atomic Energy Agency, Japan
Status on transmutation technology and accelerator-driven systems in Japan

Shigeaki Okajima
Japan Atomic Energy Agency, Japan

Abstract

The previous presentation related to the title was carried out at the TCADS-1 in 2010 in Karlsruhe. In FY2008, Atomic Energy Commission (AEC) of Japan conducted a check and review of partitioning and transmutation (P&T) technologies. The results of it were shown in the presentation.

Since then the situation of the nuclear energy development policy in Japan has changed. One of the main reasons of the change was the Fukushima Daiichi nuclear power plant accidents which were induced by the Great East Japan Earthquake and the following tsunami in 2011. The status of the nuclear energy development took on a different posture in Japan.

This time the status of it, including the latest strategic energy plan issued by Government of Japan, and the ongoing programmes in Japan concerning the P&T technologies will be presented. The state of the art, not only in the development of the P&T technologies, but also in the development of ADS technologies, will also be shown.
European activities on accelerator-driven systems

Hamid Aït Abderrahim, Gert Van den Eynde
SCK•CEN, Belgium

Abstract

Presently, the European Union relies for 30% of its electric power production on generation II-III fission nuclear reactors leading to the annual production of 2 500 t/y of used fuel, containing 25 t of plutonium, and high-level wastes (HLW) such as 3.5 t of minor actinides (MA), namely neptunium (Np), americium (Am) and curium (Cm) and 3 t of long-lived fission products (LLFPs). This MA and LLFP stocks need to be managed in an appropriate way. The used fuel reprocessing (closed fuel cycle) followed by the geological disposal or the direct geological disposal (open fuel cycle) are today the envisaged solutions depending on national fuel cycle options and waste management policies. Required time scale for the geological disposal exceeds our accumulated technological knowledge and this remains the main concern of the public. The partitioning and transmutation (P&T) has been pointed out in numerous studies as the strategy that can relax constraints on the geological disposal, and reduce the monitoring period to technological and manageable time scales. Therefore a special effort has to be made to integrate P&T in advanced fuel cycles and advanced options for HLW management. Transmutation based on critical or subcritical fast-spectrum transmuters should be evaluated, in order to assess the technical and economic feasibility of this waste management option, which could ease the development of a deep geological storage.

From 2005, the research community on P&T within the EU started structuring its research towards a more integrated approach. This resulted during the FP6 into two large integrated projects namely EUROPART dealing with partitioning and EUROTRANS dealing with ADS design for transmutation, development of advanced fuel for transmutation, R&D activities related to the heavy liquid metal technology, innovative structural materials and nuclear data measurement. This approach resulted in a European strategy given in introduction based on the so-called “four building blocks” at engineering level for P&T that will result in identification of the costs and the benefits of partitioning and transmutation for European society.

The MYRRHA Project contributes heavily to the third building block of this European strategy and in this paper we will focus on the ADS programme in the EU through the MYRRHA Project and the associated FP7 and H2020 projects contributing to the progress of ADS in Europe namely: ARCAS, CDT, MARISA, MAX, FREYA, MAXSIMA and MYRTE
Introduction to accelerator-driven system activities in China

Wenlong Zhan
Chinese Academy of China, China

Abstract
After the intensive ADS R&D from 2011, the Chinese initial ADS (CIADS) conception design was approved, and is planned for construction from 2017 to 2022. CIADS has been designed with ~400MeV&10mA proton beam SCL, the dense granular flow target, lead-bismuth eutectic (LBE) cooling subcritical blanket, and other associate components and a total system thermal power ~10 MW. The new research centre includes a heavy ion accelerator facility which will be formed in co-operation with the nuclear power plant in Huizhou, Guangdong province. Most of the key techniques have been studied through an intensive R&D of which progress will be presented in this talk.

Besides traditional ADS R&D, new approaches have been proposed such as accelerator-driven advanced nuclear system (ADANES) for sustainable fission energy system which consists of the recycling of used nuclear fuel (UNF) and ADANES burner. Recycled fuel (oxidised or carbonised) is designed to remove ≥50% of fission products (FP) from UNF. ADS is an optimisation of the ADANES burner, which operates transmutation, breeds and produces energy from recycled fuel. By using accelerator driving subcritical core at the start, the ADANES burner could be safer, more redundant to burn the “raw” fuel (recycle fuel) and more cost effective. Therefore, the concept of ADANES to close the fuel cycle could sustain fission energy 100 times more and the nuclear waste <4% in quantity with a live time inferior at 500yr.
Session 2: Accelerator-driven systems

Chairs:

R. Garnett
Los Alamos National Laboratory, United States

A.C. Mueller
Université Paris-Sud, France
Beam commissioning activities of a high-power superconducting linac for China ADS

Zhijun Wang, Yuan Hea
Institution of Modern Physics, Chinese Academy of Science, China

Abstract

To develop the next generation of safe and cleaner nuclear energy, the ADS emerges as one of the most attractive technologies. Such a system is not only able to transmute the long-lived transuranic radionuclides produced as waste in the reactors of today's nuclear power plants into shorter-lived and less dangerous by-products but also to generate positive energy output at the same time. The prototype of the Chinese ADS (C-ADS) proton accelerator is comprised of two injectors, a driver linac and a 1.5 GeV, 10 mA CW superconducting main linac. The Injector Scheme II at the C-ADS demo facility at the Institute of Modern Physics (IMP) is a 10 MeV, 10 mA CW superconducting linac. The construction of Injector Scheme II started in 2014. The ECR-ion source, LEBT, RFQ and a test cryomodule with half wave resonator (HWR) cavities have been installed. Progresses of the beam commissioning of Injector Scheme II is reported in this paper.

Introduction

Application of ADS systems as one of the technologies for a cleaner nuclear power source has been proposed some time ago [1-3]. To satisfy the fast growing energy needs of China's economic development and to combat global warming, the next generation of safe and cleaner nuclear energy is needed. The ADS emerges as one of the most attractive cutting edge technologies to transmute the long-lived transuranic radionuclides produced as waste in the reactors of today's nuclear power plants into shorter-lived, less dangerous by-products and at the same time generating positive energy output. ADS systems are considered one of the vital measures at the forefront of the exploration for sustainable energy production. Therefore, the Chinese ADS (C-ADS) proof-of-principle project was launched in 2011 under the management of the Chinese Academy of Sciences [4]. The design of the C-ADS proton accelerator includes a low-energy injector linac and a 1.5 GeV, 10 mA CW superconducting main linac.

The accelerator will be developed through collaborations between the Institute of Modern Physics (IMP) and Institute of High Energy Physics (IHEP). The layout of the C-ADS design is shown in Figure 1. The demonstration project is divided into three stages. First, a research facility including a 250 MeV superconducting, 10 mA CW proton linac and a 10 MW test reactor will be built by 2022. At this stage, the many critical technologies will be tested and the proof-of-principle system will be synthesised. At stage 2, most of the challenges will hopefully be resolved and the energy of the linac will be increased to 1 GeV at 10 mA with a reactor power of 100 MW for an operational demonstration of the ADS system. Finally, a 1.5 GeV CW accelerator with beam current of 10-25 mA plus a 1 GW commercial prototype reactor system will be built for performance studies at daily operation.
To reach the goal of the first stage, a 25 MeV, 10 mA CW superconducting proton linac will be built to study the accelerator technologies. It consists of a 10 MeV injector linac and a superconducting main linac. The injector linac is designed to have two parallel 10 MeV injectors to obtain high availability. The Injector Scheme II being built at IMP is comprised of an ion source, low-energy beam transport line (LEBT), a 162.5 MHz radio-frequency quadrupole accelerator (RFQ), medium energy beam transport line (MEBT) and superconducting half wave resonator (HWR) accelerator section. The accelerator design and simulations are mainly carried out using the computer codes PARMTEQ [7] (for RFQ), TraceWin [5], TRACK [6] and BEAMPATH (mainly for LEBT) [8]. The layout of the Injector II is shown in Figure 2.

**End-to-end simulation of C-ADS Injector II**

Multiparticle beam dynamics simulations of Injector II were performed using the TraceWin code developed by CEA/Saclay. An initial water-bag beam distribution is used for the simulations and is extracted from the proton source and propagated through the main elements which include a RFQ cavity, the HWR superconducting cavities, the superconducting solenoids etc. In the simulation, the three-dimensional field maps of the elements are produced by third-party finite element codes, such as CST, Opero, etc. The space charge field is calculated using the Partran subroutine inside the TraceWin code. 100 000 macro particles were used to simulate a 10 mA proton beam from the ion source to the beam dump. The multiparticle simulation results are shown in Figure 3. Simulation results indicate that the particles can be accelerated and transported to the beam dump without beam loss at SC section even when including errors. A well-designed beam dynamics design is a prerequisite for successful high-power operation.
The beam commissioning of C-ADS Injector II demo facility

Installation of the beam line components of the C-ADS Injector II demo facility at IMP started in August 2014 and currently consists of the ion source, a LEBT, a RFQ, a MEBT, a HWR test cryomodule (TCM) and a movable beam diagnostics platform. The layout of the facility is shown in Figure 4. Beam commissioning activities at the demo facility have been ongoing for about two years and have undergone several stages. CW operation with full beam power is the most challenging beam commissioning issue because of lack of similar high power, low-energy superconducting linac operating experience at other facilities to draw from. To date, an 11 mA, CW proton beam at 2.7 MeV has been successfully demonstrated at the exit of the superconducting TCM cavity in February 2015 and 3.9 mA at 4.6 MeV at the exit of the six superconducting HWR cavities of CM1 (TCM6) in November 2015. The longest operating time of more than 200 minutes with 2.7 mA, 4 MeV beam was achieved in January 2016.

Figure 4: The photo of the online layout of the facility (ECR+LEBT+RFQ+MEBT+TCM6)

a) The proton source and the LEBT

The layout of the ECR proton source and the LEBT are shown in Figure 5. The LEBT has two focusing solenoids with horizontal and vertical orbit correction coils built inside [9]. There are a number of beam diagnostic instruments in both test chambers including a DC Current Transformer (DCCT) and horizontal and vertical beam slits. The proton source can be operated at 1 Hz in pulse mode or in CW mode. For
beam commissioning and measurement, the pulsed mode was usually applied. In addition, there are cone electrodes and a beam chopper to shape the beam transversely and longitudinally.

![Figure 5: Layout of proton source and LEBT](image)

The measured CW normalised transverse beam emittance at the entrance of the RFQ is about $0.13 \pi \text{mm-mrad}$ at 15 mA as shown in Figure 6, which satisfies the design goal. The estimated space charge neutralisation of the LEBT is about 0.87. The LEBT lattice is set according to the measurement results. The simulated acceptance of the RFQ is also plotted in Figure 6. The beam is well matched to the RFQ accelerator.

![Figure 6: The simulated RFQ acceptance (white area inside the black dots) versus the measured emittance (colour area) at the entrance of RFQ](image)

**b) The RFQ accelerator**

The RFQ cavity is one of the key technologies needed for CW operation. The Injector II RFQ was designed through collaboration between IMP and LBNL [10], and was fabricated at the IMP workshop. The physical parameters of the RFQ are conservative but still satisfy the beam quality requirements. A large aperture, low Kilpatrick value ($K_p$) and a low inter-electrode voltage were chosen to decrease the RF power and the
probability of RF sparking. The beam is accelerated from 0.035 to 2.1 MeV within 4.2 meters. The extraction energy of 2.1 MeV is lower than the activation threshold of copper. A four-vane structure is chosen and the total RF power is about 85 kW. The RFQ cavity has been operating in CW mode since June 2014. The longest operating duration without a trip has been 220 hours.

By measuring the beam energy spectrum at different RFQ input powers, the threshold of the RFQ input power for good quality beam can be found. From the results shown in Figure 7, the working input power of the RFQ should be above 93.6 kW. It corresponds to a design RFQ inter-vane voltage of 65 kV, which is measured by the x-ray spectrum. However, for normal operation the RFQ power was conveniently set to 100 kW to maintain stability of the RF system.

**Figure 7: RFQ beam current versus energy at different input power**

By utilising the movable beam diagnostics platform (D-plate) which includes an analysing magnet, the beam energy spectrum from the RFQ was measured at an RFQ input power of 110 kW. The obtained FWHM energy spread is 1.9% with the centroid energy of 2.135 ± 0.026 MeV, which agrees with the TOF measurement of 2.165 ± 0.005 MeV between BPM1 and BPM3. The measurements show the RFQ is functioning well compared to the design.

Test data for successful long-term operation at 10 mA, CW is shown in Figure 8. The proton beam ran successfully until the ECR tripped on 30 June 2014. The measured beam transmission is 96.9% at 10.5 mA.

**Figure 8: The history record of CW operation with 10 mA proton beam**
c) Beam commissioning of MEBT and TCM6

The MEBT [11], which is composed of seven quads and two room temperature Quarter Wave Resonator (QWR) type bunchers, is designed to match the beam from the RFQ cavity to the superconducting section in both transverse and longitudinal planes. A “double period structure” in the MEBT (as described in Huan JIA, HE Yuan et al., 2015) where each period is composed of a triplet and a QWR cavity is used to transport the beam between the two strong focusing lattices of the RFQ and TCM6. There are many beam diagnostic devices installed along the beam line to measure beam parameters from the RFQ cavity. The main beam parameters, including beam energy, emittance and beam current are measured in the MEBT to study the beam quality. These measured beam parameters are the foundation used to tune the downstream linac (TCM6). The superconducting section accelerates the proton beam from 2.1 MeV to 10 MeV using twelve SC HWR cavities. The $\beta$-value of the HWR cavity is 0.1, which keeps the transit-time factor over 0.5 along the acceleration energy range. The focusing period, which is composed of one solenoid and one resonator, can provide strong focusing at such low beam energy and is important in terms of minimising the core emittance growth and preventing the generation of beam halo. The aperture is chosen to be 40 mm to get a large acceptance to decrease the beam loss. The twelve periods are placed in two identical cryomodules, which break the periodic focusing lattice at the warm transitions between cryomodules. The phase advance along the linac is kept as smooth as possible as well as in the transition section. The first cryomodule housing six cavities has been commissioned with a 4 mA, CW beam.

The beam diagnostic devices are distributed in the MEBT section and include mainly Beam Position Monitors (BPMs) and transverse emittance measurements. The transverse emittance measurement device is composed of slits and wire scanners in both transverse directions. The emittance measurements are carried out with different quadrupole settings three times to verify the beam parameters out of the RFQ accelerator. Special lattice settings are chosen to keep the beam size to be 40 to 50 mm. The signal-to-noise ratio (SNR) of the wire-scanner electronics is high enough at these beam sizes but the power intensity is not high enough to damage the wire. The measured emittance located at the slit is tracked back to the exit of the RFQ accelerator with three different lattice settings using the TraceWin code. A comparison of transverse emittances between experiment and simulation are plotted in Figure 9. The Twiss parameter beta agrees very well between the experiment and the simulation results. There is much bigger difference for alpha because the beam size is not sensitive to the parameter alpha. The tails of the beam distributions in the phase space are almost the same. Self-consistent measurement results were achieved through multiple experiments. The ability to accurately measure Twiss parameters at the RFQ exit is one critical precondition to achieving commissioning at full beam power.

The ability to perform beam-based alignment (BBA) is critical to reduce beam loss caused by beam centroid displacement. The measurements of the BPM offset, which means the displacements between the magnetic centre of the quadrupole and the electric centre of the BPM are critical for BBA. The C-ADS linac uses two kinds of BLP configuration: 1) a room temperature (RT) BPM with quadrupole and 2) low temperature BPMs with superconducting solenoids. Measurements for both cases were carried out with a 1 Hz, 20 µs low current proton beam. The calibrated beam offsets of the BPMs are listed in Table 1. From Table 1, the room temperature BPMs are calibrated with quadrupoles using the null-comparison method [12]. The offset values are approximately 0.5 mm, which are close to the offline measurement value. While for the SC-BPM, which are operated at 4.5 K, the experimental data is not as expected for the three BPMs with values shown in red in Table 1. Additional analysis and more detailed study of the calibration of SC-BPMs will be done in the future.
Table 1: The calibrated beam offset of BPMs

<table>
<thead>
<tr>
<th>BPM index</th>
<th>X_offset(mm)</th>
<th>Y_offset(mm)</th>
<th>Operation temperature</th>
</tr>
</thead>
<tbody>
<tr>
<td>BPM1</td>
<td>0.449</td>
<td>0.559</td>
<td>Room temperature</td>
</tr>
<tr>
<td>BPM2</td>
<td>-0.026</td>
<td>0.432</td>
<td>Room temperature</td>
</tr>
<tr>
<td>BPM3</td>
<td>0.063</td>
<td>-0.325</td>
<td>Room temperature</td>
</tr>
<tr>
<td>BPM4</td>
<td>0.434</td>
<td>0.158</td>
<td>Room temperature</td>
</tr>
<tr>
<td>SC-BPM1</td>
<td>-0.04</td>
<td>-0.49</td>
<td>4.5K</td>
</tr>
<tr>
<td>SC-BPM2</td>
<td>5.58</td>
<td>1.61</td>
<td>4.5K</td>
</tr>
<tr>
<td>SC-BPM3</td>
<td>-0.03</td>
<td>-0.25</td>
<td>4.5K</td>
</tr>
<tr>
<td>SC-BPM4</td>
<td>3.14</td>
<td>-0.38</td>
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</tr>
<tr>
<td>SC-BPM5</td>
<td>13.23</td>
<td>1.05</td>
<td>4.5K</td>
</tr>
</tbody>
</table>

Figure 9: The comparison between simulation and experiment results with three different quadrupole settings
Superconducting HWR cavities are used to accelerate the proton beam from 2.1 MeV to 10 MeV. The geometric beta value of the cavity is 0.09 and the cavity covers the range of \( \beta \) values of proton beam from 0.056 to 0.145. For the low-energy superconducting cavities, one main physics issue is the mismatch between the beam velocity and the cavity geometric \( \beta \) value. The other issue is that the beam velocity changes inside the cavity, which means the simple “kick” model is not suitable for approximating the SC cavity case. A common cavity voltage calibration method is the phase scanning method. This method is also applied to the superconducting HWR cavity, even though for such a low-energy section, the transit-time factor (TTF) varies a lot with respect to the beam energy. The field map model is used to fit the experimental data to include the TTF effect. The results for the six cavities in TCM6 are listed in Table 2.

For the C-ADS Injector II, a comprehensive monitoring system is built to detect the beam performance during the beam power up to full beam power. The monitoring system includes the vacuum value of each section, the beam centroid position, sensors to measure the temperature rise along the beam pipe, sensors to measure the cooling water temperature of the beam dump, and so on. Many beam-commissioning activities on the SC accelerator were carried out since the June 2014. The CW operation activities are summarised in Figure 10.

### Table 2: The experimental \( E_{\text{peak}} \) of six cavities in TCM6

<table>
<thead>
<tr>
<th>Cavity index</th>
<th>( E_{\text{peak}} ) (TOF)</th>
<th>( E_{\text{peak}} ) (Diple)</th>
</tr>
</thead>
<tbody>
<tr>
<td>HWR01</td>
<td>17.25</td>
<td>17.88</td>
</tr>
<tr>
<td>HWR02</td>
<td>18</td>
<td>17.38</td>
</tr>
<tr>
<td>HWR03</td>
<td>18.48</td>
<td>17.86</td>
</tr>
<tr>
<td>HWR04</td>
<td>19.53</td>
<td>19.2</td>
</tr>
<tr>
<td>HWR05</td>
<td>19.53</td>
<td>18.78</td>
</tr>
<tr>
<td>HWR06</td>
<td>16.31</td>
<td>15.75</td>
</tr>
</tbody>
</table>

Figure 10: The dots in the figure represented the beam energy, which is the multiplication of beam power and operation time. The commissioning records are 28 kW of beam power and 200 minutes for operation time. More effort is needed to realise a high power, long stable operation.
Summary

The C-ADS Injector II demo accelerator has been constructed and the beam commissioning was successfully achieved in several stages. The beam commissioning philosophy consisted of three stages and was implemented in the high beam-power commissioning activities. The commissioning records of 28 kW in beam power and 200 minutes for operation time have been achieved. Significant experience in the operation and commissioning was acquired by the project team during the C-ADS Injector demo experiments. Some upgrade and improvement ideas are identified and will be carried out in the future. Some commissioning issues will be studied in further detail.

Acknowledgements

The authors appreciate the advice and discussions provided by our colleagues throughout the country and abroad regarding machine design and beam commissioning. Special acknowledgement is given to Andrei Shishlo (SNS) for his helpful discussions about the beam tuning. This work is supported by the CAS Strategic Priority Research Programme on Future Advanced Nuclear Fission Energy (accelerator-driven subcritical systems).

References

Design and implementation of FCT-based beam energy measurement system in C-ADS

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Abstract

The China accelerator-driven subcritical system (C-ADS) is a project to solve the current stored nuclear waste problem and the generated waste problem for nuclear power plants. A fast beam current transformer (FCT) system has been developed to monitor the variation of beam phase and energy during accelerator operations of the C-ADS. The FCT system developed includes the FCT detector, front-end electronics, main electronics, software and user interface. The system architecture, hardware and software design and implementation will be introduced in this paper. The application and experimental results of the FCT system will also be presented.

1. Introduction

C-ADS is the abbreviation for the China accelerator-driven subcritical system, which works under the principle that the spallation reaction between high-energy protons, produced by a linear accelerator, and heavy target nuclei generates dozens of neutrons per incident proton. This source of neutrons drives the subcritical blanket system to generate energy by maintaining chain reactions of the subcritical blanket system. The first stage of acceleration in the C-ADS accelerator is a radio-frequency quadrupole (RF Q). The RF frequency of 325 MHz is chosen for the Injector I RFQ, which is built by IHEP [1].

Beam energy is an important parameter for tuning the cavity, and the phase scan method using a FCT detector is employed to make this measurement. The beam time of flight through a fixed distance is measured between two FCT detectors, and then the beam velocity is calculated, from which the beam energy is derived.

The beam energy measurement system includes the FCT detector, signal processing electronics and a data acquisition system. The FCT detector is model FCT-082-xx which is produced by Bergoz [2]. The signal processing electronics are all developed in-house at IHEP and consists of two modules, one is the analogical front-end electronics being used for signal amplitude adjustment, the signal mixer and the signal filter; the second is the PCIe-based data acquisition electronics where the ADC operations and data processing are performed. The data acquisition system is an application programme developed for data acquisition, data processing, results storing and plotting run on a host computer.

The electronics implementation and the data acquisition system are introduced in this paper. The systems are carefully designed and optimised to meet a 0.1 degree resolution and better than
± 0.5 degree accuracy phase measurement requirement. A test device was first built to verify the design in the laboratory. The system has now been applied in the C-ADS injector project.

2. System principle

The phase scan method employed in the beam energy measurement system uses the phase difference of two different FCT detectors to deduce the beam energy variation. A schematic diagram of the system is shown in Figure 1. The signal from the beam that is modulated by the klystron RF is picked up by FCT1. The beam is drifted through the next downstream tank where the un-modulated beam signal is picked up by FCT2. The beam time of beam flight, corresponding to the phase difference of two FCT signals, is used to compute the beam energy [3].

![Figure 1: schematic diagram of FCT-based energy measurement](image)

The beam velocity is calculated from the flight time that beam takes to pass through the FCT1 and FCT2 detectors. If the distance that beam passes through FCT1 and FCT2 is L, and the flight time that beam passed through them is t, the beam velocity can be calculated using the simple equation:

\[ v = \frac{L}{t} \]  

(1)

The beam flight time can be measured from the phase difference between the two FCT signals, as shown in Figure 2.

![Figure 2: Phase difference of two Sinusoidal signals](image)

If the beam phase difference is denoted by \( \Delta \theta \), then:

\[ t = \frac{\theta}{2\pi}T = \frac{\theta}{2\pi f} \]  

(2)

The velocity of beam can then be deduced from (1) and (2):

\[ v = \frac{2\pi fL}{\Delta \theta} \]  

(3)
3. System hardware

3.1 System structure

The FCT-based energy measurement system function block is shown in Figure 3. The system includes two channels of identical electronics (one channel for each FCT detector). Each channel is composed of two parts, one is the front-end analogue hardware module, another is the ADC sampling module.

![Figure 3: System function blocks](image)

The basic concept of the system design is to interrogate the fundamental frequency of the FCT and down convert the fundamental frequency (325 MHz) and reference point frequency (325 MHz) signals to an IF (16.25 MHz) signal. The FCT and REF signal are sampled with 65 MHz IQ (In-phase and Quadrature-phase) frequency. The sampled data are transferred to the host computer to calculate $\Delta \theta_1$ (phase difference between FCT1 and REF) and $\Delta \theta_2$ (phase difference between FCT2 and REF). Finally, the phase difference between FCT1 and FCT2 ($\Delta \theta$) is acquired and used to calculate the beam energy.

3.2 Front-end analogue module

The front-end analogue electronics module consists of an analogue filter, a mixer and an amplitude adjustment. The filter function is used to remove non-fundamental frequency components from the signal. The mixer logic is used to obtain a 16.25 MHz IF signal by mixing the 325 MHz FCT signal with a 308.75 MHz reference signal. The amplitude adjustment logic is designed to satisfy the ADC input requirements. The function blocks are shown in Figure 4.

![Figure 4: Function blocks of the front-end analogue electronics module](image)
Band pass filters, composed of VLF-320 and BHP-300+ filters, filter the signals within a certain “band” without distorting the final input signal. The signal is further conditioned using an amplifier (ZRL-700) and an attenuator (VAT-3) to regulate the amplitude of the mixer input. After the mixer (Z-3+), an additional band pass filter (SIF 21.4) and an amplitude adjustment device (ZRL 1000 and VAT-3) are needed in the circuit to meet the system requirements. The front-end analogue electronics system is shown in Figure 5.

![Figure 5: Picture of the front-end analogue electronics](image)

### 3.3 ADC module

The ADC module developed at IHEP is a standard PCIe card. Its function block is shown in Figure 6. The interface signals include the two channel analogue inputs, a main clock input and a trigger signal input. The output clock frequency is 308.75 MHz, which is fed to the mixer located in the front-end electronics. The ΔI and ΔQ information is computed in an FPGA with ADC quadrature sampling data and the data is transmitted to a host computer with format (0XAA55, COUNT, ΔI, ΔQ) while the application software on the host computer acquires and processes the data to be displayed and stored. The ADC module is shown in Figure 7.

![Figure 6: Function block of ADC module](image)
3.4 Firmware

The FPGA chip selected for use on the ADC module is EP4CGX50DF27C8 (Cyclone IV). The main function block of its firmware is shown in Figure 8. The four main logic parts are the clock logic, the external trigger logic, the ADC data processing logic, and the PCIe interface logic. The clock logic provides the main clock for the FPGA inner logic and also provides the SPI configuration clock for the AD9512 and AD9518 converters. The external trigger logic is designed to start the data sampling process. The function of the ADC data processing logic is data sampling, Δl and ΔQ computing and data assembling. The PCIe module transmits the sampled data to the host computer via the PCIe bus.

4. System tests

The main application programme running on the host computer was developed with Visual C ++ and MATLAB tools and designed to meet the requirements for data acquisition and processing, including graphical display of the results and storage of all data. The test system built in the laboratory is shown in Figure 9. The trigger signal, reference clock signal and FCT signal were all generated using a signal generator device, two signal generators and two power splitters to test the system for signal synchronisation. All cables used in the system are coaxial and shielded.
Laboratory test results of one channel are shown in Figure 10(a), where the I0, Q0, I1, and Q1 are the raw data, and I0 and Q0 are sampled from the reference clock, I1 and Q1 are sampled from the FCT signal, the “phase 0” and “phase 1” are calculated from the reference clock and the FCT signal, respectively. There are four different values obtained because the starting point of the assignment to the variables I and Q is uncertain during data quadrature sampling, but this does not affect the measurement results. From the figure shown, the peak-to-peak phase difference (“phase 0” vs “phase 1”) ranges from 143 to 144 degrees.

Figure 10 (b) shows the two channel laboratory testing results: the channel 0 result is within 1 degree (221° to 222.5°), the channel 1 result is within 1.5 degrees (143° to 144.5°), and the difference of the two channels is within 1.6 degrees (35.9° to 37.5°). From the system tests, a 0.8 degree peak-to-peak resolution can be reached in laboratory.
In addition, an online test was performed on the C-ADS injector system. The result is shown in Figure 11. Approximately ± 2 degrees (162° to 166°) peak-to-peak resolution is demonstrated.
5. Conclusions

A beam energy measurement system based on Fast FCTs has been developed for C-ADS. After laboratory and online testing, a ±0.8 degree peak-to-peak laboratory resolution and about ±2 degree peak-to-peak online resolution was achieved. These results satisfy the system requirements. More research and development is needed in the future in order to get better system performance.

Acknowledgements

The system development is the synthesis of work of many people who work in Beam Instrumentation Group of Accelerator Center. Thanks to them all for their co-operation.

References


ADS Injector I cryogenic system in China

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Abstract

An accelerator-driven subcritical system in China (C-ADS) is planned for transmutation to minimise nuclear waste. As one of the important step towards a C-ADS facility, the Injector-1 Test Facility has been built at IHEP, CAS. Injector I has two cryomodules operating at 2K in a cryogenic environment to realise a 10 MeV proton beam energy. Each cryomodule includes seven spoke cavities and seven superconducting magnets. This paper describes the design and operation of the cryogenic system, including the cooling flow chart, the configuration of the operating cryomodules and operation progress of the cryogenic system.

Introduction

An accelerator-driven subcritical system in China (C-ADS) is planned for transmutation to minimise nuclear waste. The system uses a high-energy proton beam to bombard the metal target and generate neutrons to transmute the nuclear waste. The role of the proton accelerator in the ADS system is to produce a continuous, stable and reliable source of high-energy protons to the neutron-production target. These systems typically require a continuous wave (CW) high-power accelerator. Superconducting accelerating technology can solve the problem of power loss and cooling of the accelerator and is the generally accepted choice for ADS.

The Chinese ADS proton linear accelerator design has two 10 MeV injectors (injectors I and II) and one 10-1500 MeV superconducting linac. Injectors I and II are being developed and tested by the Institute of High Energy Physics (IHEP) and the Institute of Modern Physics (IMP), respectively. Different technical approaches are being adopted by the two institutes. Injector I uses two superconducting cryomodules, each having seven $\beta=0.12$ spoke cavities and seven solenoid magnets. In order to improve the operating performance of the superconducting cavities, the superconducting accelerator section operates at a 2 K cryogenic temperature. The cryogenic system for Injector I will not only meet the operation requirements of the superconducting cryomodules but also meets the vertical and horizontal test requirements for a variety of cavities. The horizontal test station, the vertical test station and the injector operating station can all be operated independently of each other. The cryogenic system also includes a high-pressure helium recovery, purification and gas storage system.

Profile of ADS Injector I cryogenic system

The ADS Injector I cryogenic system is a 2 K superfluid cryogenic system. It includes the refrigerator, cryogenic distribution valve boxes, superconducting devices, a 2K pumping system, a liquid nitrogen
system and a recovery and purification system. In order to meet the requirements, the refrigerator has two operating modes: refrigeration mode and liquefaction mode. According to the acceptance test, the capacity of the refrigerator is about 1 000 W @ 4.5 K in refrigeration mode and 300 L/h in liquefaction mode with LN$_2$ precooling. The cold box has three outputs; they are the 300 K and 40 K mixture output, the supercritical helium output and the two-phase helium output. The 300 K and 40 K mixture helium is for cooling down the cryomodules and the supercritical helium is for the helium shielding. The two-phase helium output is connected to the 3000-L Dewar.

The capacity of the 2 K pumping system is about 8 000 m$^3$/h @ 31 mbar. The operating pressure of the purifier is 20 MPa and the purification flow rate is about 105 m$^3$/h. The purity of the helium gas after purification is higher than 99.9995%.

**Cryomodules**

The C-ADS Injector I cryomodules consist of mainly the cold mass, two shields, the vacuum vessel and the high strength brackets, as shown in Figures 1 and 2. The superconducting cavities and solenoid magnets are arranged one by one and supported at the bottom by nearly adiabatic supports. There is an 80 K liquid nitrogen shield and a 5 K helium shield. The shields are also fixed to the supports. The cold mass and adiabatic supports are fixed to the adjustable room-temperature brackets and then connected to the vacuum vessel. Figures 1 and 2 are the two-cryostat configuration of the C-ADS Injector I where the two cryomodules, cryomodule 1(CM1) and cryomodule 2 (CM2), share one thermal-insulation vacuum. The cryomodules were assembled and tested in stages. The static heat load of one cryomodule is about 30 Watts according to recent tests.

*Figure 1: Spoke cavities and solenoid magnets distributed one by one in the cryostat*

*Figure 2: Assembly of the cryomodules*

The flow chart and mechanical design of the valve box and cryostat for the superconducting devices (cavities and magnets) are the core of the system design. The flow chart of the valve box and cryomodules is shown in Figure 3. The cooling down for the cryomodules has two steps. First, the superconducting devices are cooled by a mixture of 300 K and 40 K gaseous helium. The cool-down rate
must be controlled to avoid breakdown of the seals. Secondly, after the cryomodule temperatures are below 60K, 4.5 K liquid helium is used to directly cool the devices to 4.5 K. After that, 4.5 K liquid helium goes to 2.0 K by use of the heat exchanger, JT valve and the pumping system. When operating, the superconducting cavities and magnets are immersed in the 2.0 K superfluid helium.

The vapour pressure for 2.0 K saturated liquid helium is about 3×10^{-3} Pa which is achieved by a set of pumps. The pump unit forces helium gas from the vessels through the transfer line of about 50 metres length. The pumped-out helium gas goes directly to the suction of the main compressor and completes a closed cycle. When the system is cooling down or warming up, the returning gas can either go to the compressor or to the gas bag. Impure helium gas in the gas bag is purified and then returns to the cycle.

In order to improve the production efficiency of 2 K liquid helium, there is a phase separator and a heat exchanger in the 4 K to 2 K valve box. The two-phase helium which comes from the distribution valve box is separated into liquid and gaseous helium in the phase separator. Finally, 4.2-K saturated liquid helium goes through the heat exchanger and is cooled down to about 2.6 K by the returning 2 K helium gas and then throttled to the pressure 3×10^{2} Pa, which is realised by the pumps. The efficiency of producing the 2 K liquid helium can be more than 85%.

**Commissioning and operation**

The C-ADS Injector I cryogenic system started cooling down for 2-cavity, 2-solenoid cryomodule testing on 20 January 2015, and completed both the 4K and 2K commissioning on 26 January 2015 and 4 February 2015, respectively. Dozens of seals withstood the cryogenic test during the commissioning. Remote and automatic control of the main control valves was successfully demonstrated. Thermal-insulation vacuums of the cryogenic valve boxes and cryostats were all good. The 2K pumping system, the measurement and control system, the liquid nitrogen system, the recovery system and the safety protection system worked well.

In the following approximately one year, we successfully completed the cryogenic commissioning and operation of a 7-cavity, 7-solenoid cryomodule (CM1), and a vertical test station, followed by the combined two-cryomodule (CM1 and CM2) testing. This enabled the successful achievement of 10.1 MeV@10.03mA pulse beam energy of the C-ADS Injector I on 17 June 2016. The cryogenic system commissioning milestones are shown in Table 1 below.
Table 1: Commissioning milestones of ADS Injector I cryogenic system

<table>
<thead>
<tr>
<th>Date</th>
<th>Milestones</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 August 2014</td>
<td>Finished the acceptance test of the refrigerator</td>
</tr>
<tr>
<td>20 January 2015</td>
<td>Started cryogenic commissioning for the test cryomodule</td>
</tr>
<tr>
<td>26 January 2015</td>
<td>The test cryomodule reached 4.2 K</td>
</tr>
<tr>
<td>4 February 2015</td>
<td>The test cryomodule reached 2.0 K</td>
</tr>
<tr>
<td>28 August 2015</td>
<td>Started cryogenic commissioning for CM1</td>
</tr>
<tr>
<td>2 September 2015</td>
<td>CM1 reached 4.2 K</td>
</tr>
<tr>
<td>11 September 2015</td>
<td>CM1 reached 2.0 K, and started stable operation</td>
</tr>
<tr>
<td>28 September 2015</td>
<td>The vertical test(VT) station reached 4.2 K</td>
</tr>
<tr>
<td>29 September 2015</td>
<td>The VT station reached 2.0 K, and provided test environment for cavities</td>
</tr>
<tr>
<td>22 April 2016</td>
<td>Started cryogenic commissioning for CM1+CM2</td>
</tr>
<tr>
<td>30 April 2016</td>
<td>CM1+CM2 reached 4.2 K</td>
</tr>
<tr>
<td>4 May 2016</td>
<td>CM1+CM2 reached 2.0 K, and started stable operation</td>
</tr>
</tbody>
</table>

The cool-down process and stable operation of CM1 and CM2 are shown in Figures 4 and 5. It can be seen from Figure 4 (b) that the pumping process from 4 K to 2 K took about 2.5 hours. Figure 6 shows the measurement curves of the temperature, pressure and liquid level for CM1 and CM2 operating at 2 K over a span of 12 hours. During that period of time Injector 1 had full-power operation with proton beam. It should be noted that the observed pressure fluctuation was less than ±0.5 mbar, and the liquid level fluctuation was less than ±0.5%. This demonstrated that the cryogenic system was able to adapt to the dynamic heat loads during the Injector 1 beam commissioning. The pressure stability is realised through the control of the pumping system and the stability of the liquid level is realised through automatic control of the J-T valves.

![Figure 4: The first time cooling down curves for CM1 and CM2](image)

![Figure 5: Operation status for CM1 and CM2](image)
Figure 6: Operation measurement curves for CM1 and CM2

Acknowledgements

The cryogenic system will continue operation and machine studies for more optimisation and improvement. Thanks to all those that supported the design and construction of this project.

References


The proton linac for the MYRRHA Project

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Abstract

One major goal of the European MYRRHA Project is the demonstration of the feasibility of the transmutation of long-living radioactive nuclear waste. MYRRHA is designed as an accelerator-driven system (ADS) using a high-power cw proton accelerator coupled with a subcritical reactor with a thermal power of up to 100 MW. The linac has to deliver 600 MeV protons with a nominal beam current of 4 mA resulting in a beam power of 2.4 MW. The challenge of the linac development is the very-high reliability of the accelerator to limit the thermal stress inside the reactor. While parallel redundancy will be used in the injector, the required reliability in the main linac can be achieved by serial redundancy.

The injector consists of an ECR source, a 1.5 MeV 4-Rod RFQ and a chain of CH drift tube structures. All injector structures are driven by 176 MHz solid state amplifiers. The medium energy section between 17 and 85 MeV consists of superconducting single spoke cavities operated at 352 MHz. The main acceleration is provided by 2 groups of superconducting elliptical 704 MHz 5-cell cavities. The MYRRHA linac layout has a very robust beam dynamics design with low emittance growth rates to avoid excessive particle losses and machine activation.

The project will be staged in three phases: The construction of the 100 MeV linac with first experimental stations, the energy upgrade to 600 MeV and the construction of the reactor. The RFQ is presently under construction and will be tested with the already operational ECR source and LEBT section. In a further step the construction of the whole injector and medium energy section is foreseen. This paper describes the linac design, prototype tests as well as the status of construction.

Introduction

It is expected that the worldwide use of energy produced by fission reactors will increase significantly in the next decades. Practically all existing and the majority of planned fission reactors are based on a thermal neutron spectrum. One major drawback of these power plants is the synthesis of significant amounts of minor actinides (MA) by neutron capture and subsequent β-decay. Beside several plutonium isotopes ($^{238}\text{Pu}$, $^{240}\text{Pu}$, $^{242}\text{Pu}$) other long-lived MA isotopes ($^{237}\text{Np}$, $^{241}\text{Am}$, $^{243}\text{Am}$, $^{244}\text{Cm}$, $^{245}\text{Cm}$) are being
accumulated during the operation of thermal reactors. Although these isotopes represent only a small fraction of the total nuclear waste they are responsible for the majority of the radiotoxicity after a few hundred years of storage. Without any treatment of the burned fuel elements the storage time in an underground repository is in the order of 1 million years.

Partitioning of the nuclear waste and reuse of Uranium and Plutonium would reduce the volume significantly. But the storage time would remain in the same order. Additionally there is presently no approved repository worldwide. One very promising way out of this bottleneck of nuclear power generation is partitioning and transmutation (P&T) by accelerator-driven systems (ADS). An ADS couples a high-power proton accelerator with a subcritical reactor. The proton beam with energy between a few hundred MeV and a GeV hits a heavy metal target (Bi, Pb, Hg, Pb-Bi) to create a high flux of fast neutrons by spallation reactions. Each proton creates up to several 10 neutrons, depending on energy and target material. The spallation target is surrounded by the reactor core which is enriched with minor actinides and eventually with long-lived fission products. The problematic isotopes are transmuted either by fission or neutron absorption. The required storage time will be only in the order of a few hundred years assuming a reasonable portioning rate between $10^{-3}$ and $10^{-4}$. An ADS has to deliver a proton beam in the MW range. Together with a criticality $k_{\text{eff}}$ between 0.95-0.98 the thermal power of the reactor is typically 20-50 higher as the beam power.

The European MYRRHA Project (Multipurpose hYbrid Research Reactor for High-tech Applications) which will be realised at SCK•CEN (Mol, Belgium) aims to demonstrate the feasibility of large-scale transmutation. A 600 MeV, 2.4 MW cw proton linear accelerator will be coupled with a 85 MW$_{\text{th}}$ subcritical reactor ($k_{\text{eff}}$=0.95). MYRRHA will be an important milestone on the way to an European Facility for Industrial Transmutation (EFIT).

Beside transmutation MYRRHA can serve as powerful tool for many fields of fundamental and applied research as material sciences, isotope production or nuclear and atomic physics with rare isotopes.

The research on an European accelerator-driven system has started in 2001 with a study of an Experimental ADS (PDS-XADS, 2001-2005) within the 5th Framework Program EC project. This research period was followed by EUROTRANS (6th Framework Program, 2006-2009) and MAX (MYRRHA Accelerator Development and Experiment, 7th Framework, 2010-2014). Presently the latest steps towards the realisation of MYRRHA are prepared within MYRTE (MYRRHA Research Transmutation Endeavour, Horizon 2020, 2015-2019).

The MYRRHA proton linac: overview

The MYRRHA proton accelerator belongs to the family of high-power proton accelerators (HPPA). With an average beam power of up to 2.4 MW (600 MeV, 4 mA) it will be one of the most powerful proton linear accelerators (see Figure 1) [1].
The chosen values of beam current, energy and duty factor are a compromise between capital and operational costs, linac length and reliability issues. The design philosophy of the MYRRHA linac is based on the required high reliability and in parallel to be as efficient as possible. Every component will be operated well below their technological limits. The independently phased main linac has enough reserves to compensate failures in that section. An important issue is beam dynamics to provide the best possible beam quality. The guideline during the linac design was achieving maximum acceptance for the beam to avoid particle losses resulting in thermal breakdown of the superconducting cavities and activation of components. Table 1 summarises the main parameter of the MYRRHA linac.

The protons are produced in a 2.45 GHz ECR source with an extraction energy of 30 keV followed by a compact low energy beam transport (LEBT) with two solenoids. The first acceleration section consists of a 4-Rod RFQ (radio-frequency quadrupole) operated at 176.1 MHz. The 1.5 MeV beam is then injected into a room temperature (rt) CH-cavity section up to 5.9 MeV. In the present reference design the beam is then accelerated by six superconducting (sc) CH-cavities up to 17 MeV. Due to reliability reasons it is planned to use two of such injectors. In this low-energy regime it is very difficult to keep the beam on target in case of a component failure. Therefore a parallel redundancy scheme has been chosen (see Figure 2).
To provide high average beam power with moderate beam currents the whole linac operates at 100% duty factor (cw). To minimise the RF losses and operational costs and to keep the gradients at reasonable high gradients the major part of the linac is based on superconducting cavities. The intermediate energy sections consist of 48 sc spoke type cavities operated at 352.2 MHz while the high-energy sections consist of two groups of sc elliptical 5-cell cavities ($\beta = 0.5$, $\beta = 7$) operated at 704.4 MHz (see Figure 2).

A small fraction of the 600 MeV beam can be used for production of secondary beams using the ISOL (isotope separator onLine) method. In this case the beam hits a thick target producing rare isotopes which are then accelerated and identified by mass separators.

The main beam is guided through the high-energy beam line and finally hits the spallation target with a doughnut like beam footprint. As target material a lead-bismuth eutectic will be used.

### Table 1: Main parameters of the MYRRHA linac

<table>
<thead>
<tr>
<th>Particles</th>
<th>Protons</th>
<th>N/A</th>
</tr>
</thead>
<tbody>
<tr>
<td>Energy</td>
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<td>MeV</td>
</tr>
<tr>
<td>Current</td>
<td>4</td>
<td>mA</td>
</tr>
<tr>
<td>Beam power</td>
<td>2.4</td>
<td>MW</td>
</tr>
<tr>
<td>Duty factor</td>
<td>100</td>
<td>%</td>
</tr>
<tr>
<td>Beam stability</td>
<td>energy ±1%, current ±2%</td>
<td>N/A</td>
</tr>
<tr>
<td>MTBF</td>
<td>250</td>
<td>h</td>
</tr>
</tbody>
</table>

17 MeV injector

The low-energy section of the MYRRHA linac is the most critical part with respect to emittance growth and technological challenge. Because of the rapid velocity increase all RF cavities have a $\beta$-profile. As a result every RF cavity has a different design and different parameters. The design has been done very carefully because it is not possible to build and to test prototypes of each cavity. The beam dynamics is based on a constant negative synchronous phase (rt CH-section) and on EQUUS (EQUidistant mUlti gap structure) with constant cell length (sc CH-section). To minimise the emittance growth the longitudinal and transverse phase advance has been kept constant as far as possible. Figure 3 shows the schematic layout of the 17 MeV injector in the present reference Design [2].

Figure 3: Schematic layout of the 17 MeV Injector

**Proton source and LEBT**

As proton source an ECR source is from Pantechnik (France) is used. It is presently installed at LPSC Grenoble (IN2P3/CNRS) to test the performance of the source together with the LEBT section. The LEBT is
based on a short magnetic solution with two solenoids and is designed to maximise the transmission from source to the RFQ by considering space-charge compensation effects [3,4]. The extraction voltage of 30 keV is high enough for good beam transport and low enough to minimise the risk of HV breakdown and for effective bunching in the RFQ. Figure 4 shows a picture of the source-LEBT combination installed at LPSC [4].

The design of the LEBT has been done at IPNO (CNRS) with the goal to provide a maximum transverse beam emittance of $0.2\,\text{π-mm-mrad}$ (rms, norm.) at the entrance of the RFQ. Collimators are installed to remove unwanted ion species, especially $\text{H}_2^+$. Additionally a slow chopper system is installed in front of the RFQ to create a time structure with adjustable long beam holes for subcriticality monitoring of the MYRRHA reactor (typically 200 $\mu$s beam holes) and for average beam intensity regulation [5]. Presently simulations are being performed to optimise the LEBT parameters with respect to maximum RFQ performance (transmission, emittance growth, brilliance) [6]. This is an important issue because the RFQ is very sensitive to beam input parameters. After full characterisation of the system it is planned to be moved to Louvain-la-Neuve (Belgium) as injector for the RFQ accelerator and demonstration test bench for the first stage of the MYRRHA injector.

Figure 4: The source-LEBT combination installed at LPSC [6]

Radio-frequency quadrupole

The 176 MHz MYRRHA RFQ has to deliver 1.5 MeV protons with a maximum beam current of 4 mA to the drift tube cavities of the Injector. The RFQ-structure provides strong electric focusing, acceleration and longitudinal compression (bunching). As RF structure a 4-Rod RFQ has been chosen because of tuning possibilities, maintenance, lower capital costs and technological risk compared to a 4-Vane-RFQ.

The aim of beam dynamics design was to preserve excellent beam quality and to avoid the creation of halo particles especially in the longitudinal plane. Using the NFSP (New Four-Section Procedure) [7] with a soft and symmetric pre-bunching with full 360° acceptance it was possible to reach the requirements. The simulated transmission is close to 100% [8]. The electrode voltage has been chosen to 44 kV which gives enough transverse focusing but limits the required RF losses to about 25 kW/m.

Table 2 summarises the main parameters of the MYRRHA RFQ.
Table 2: Main parameters of the MYRRHA 4-Rod RFQ

<table>
<thead>
<tr>
<th>Parameters</th>
<th>protons</th>
<th>N/A</th>
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<td>Frequency</td>
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<td>MHz</td>
</tr>
<tr>
<td>$W_{in}$</td>
<td>30</td>
<td>keV</td>
</tr>
<tr>
<td>$W_{out}$</td>
<td>1.5</td>
<td>MeV</td>
</tr>
<tr>
<td>$U$</td>
<td>44</td>
<td>kV</td>
</tr>
<tr>
<td>$\varepsilon_{in}$ (trans, norm, rms)</td>
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<td>π mm mrad</td>
</tr>
<tr>
<td>$\varepsilon_{out}$ (trans, norm, rms)</td>
<td>0.21</td>
<td>π mm mrad</td>
</tr>
<tr>
<td>$\varepsilon_{out}$ (long, norm, rms)</td>
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<td>keV-deg</td>
</tr>
<tr>
<td>Length</td>
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<td>m</td>
</tr>
<tr>
<td>$R_o$</td>
<td>75</td>
<td>kΩm</td>
</tr>
<tr>
<td>RF losses</td>
<td>103</td>
<td>kW</td>
</tr>
<tr>
<td>$P_{tot}$</td>
<td>110</td>
<td>kW</td>
</tr>
<tr>
<td>Cell number</td>
<td>244</td>
<td>N/A</td>
</tr>
<tr>
<td>Transmission (sim)</td>
<td>98.6</td>
<td>%</td>
</tr>
</tbody>
</table>

One major concern is the cw operation with high thermal load. Assuming a shunt impedance of 75 kΩm the expected specific power is about 25 kW/m. During the last years a vigorous research programme has been performed. The design goal of the MYRRHA RFQ was an increased performance with respect to cw power levels compared to existing 4-Rod structures. The cooling of electrodes and stems has been improved significantly. This is important to minimise the frequency shift during operation and the geometrical distortions due to inhomogeneous heating, respectively [9].

For that reason a new technology has been developed. Cooling channels are milled into the massive copper parts and filled with dielectric material. Then the copper parts are copper plated with a very thick layer. Finally the dielectric material is removed. As a result we have a 3-dimensional structure of cooling channels. Figure 5 shows the geometry of a stem with optimised cooling channels.

Another important topic was the design of the RF contacts between tuning plates and stems. Formerly RF spring contacts have been used. But it turned out that the current density is sometimes too high for spring contacts. For the MYRRHA RFQ massive silver plates will be used. The required pressure between plates and stems is provided by special shaped shims. Additionally, the cooling of the massive tuning plates has been improved to avoid thermal expansion during operation.
A short prototype RFQ has been built [10] and tested to validate the promising results obtained by RF, thermal and structural mechanics simulations using CST Microwave Studio [11] and ANSYS [12]. Figure 6 shows the prototype RFQ.

Figure 6: MYRRHA RFQ prototype

The RFQ prototype has been tested at IAP Frankfurt using a 300 kW cw amplifier. After conditioning it was possible to transfer 46 kW cw without any problem into the prototype, corresponding to a specific power of 116 kW/m. The reflection at that power level was still below 1%. The obtained power level is about 4.5 times higher than required for MYRRHA and 2.5 times higher than ever reached in a 4-Rod RFQ (see Figure 7) [13].

The full RFQ cavity is presently under construction. It is expected that the RFQ will be available for RFQ-tests end of 2016.
CH-DTL-Section

The injector drift tube linac covers the energy between 1.5 and 17 MeV and is based on CH-cavities. The transition energy between room temperature and superconducting cavities has been set to 5.9-MeV. CH-cavities are efficient low and medium energy drift tube cavities for protons and ions. The operation mode is a TE\textsubscript{211} mode [14,15].

The room temperature section is formed by 7 CH-cavities with 3 to 9 gaps. Each cavity has a $\beta$-profile and a constant negative synchronous phase. The length and the energy gain of each cavity have been adjusted to keep the longitudinal emittance growth as small as possible [2]. To simplify the production of the cell length of each of the 5 superconducting cavities has been kept constant resulting in a small phase slip. Figure 7 shows the gap voltage, synchronous phase and energy gain along the CH-DTL.

Figure 7: Test of the MYRRHA RFQ prototype

Co-operation leads to high average thermal load in room temperature RF cavities. Therefore special care has to be taken on the cooling. Multi physics simulations have been performed to optimise field distribution, shunt impedance and the cooling system. The specific power in each cavity has been limited
to 30 kW/m. A 5-cell prototype cavity has been developed to validate the simulation results [13]. Figure 9 shows temperature distribution at full power (left) and a photograph of the copper plated cavity.

Figure 9: Temperature distribution of the rt CH-prototype and copper plated cavity

Figure 10: Test results of the superconducting CH-Prototype and CH-cavity

The section between 5.9 and 17 MeV is covered by superconducting CH-cavities which provide higher gradients than room temperature structures. A 7-gap 325 MHz prototype cavity has been built and tested in a vertical cryostat. The cavity is made from high RRR bulk niobium. The cavity has a length of 500 mm. The gradients which have been achieved at 4K are 8.5 MV/m and 14 MV/m at 2K, respectively (see Figure 10) [16]. This is by far more than the requirements for MYRRHA (3.5 MV/m). Additionally power couplers and frequency tuning systems have been developed. It is planned to test the whole system in a horizontal cryostat.

Superconducting main linac

The MYRRHA main linac is composed as an array of independently phased superconducting RF cavities. It consists of a 2-gap spoke section (17-81 MeV) and two groups of 5-cell elliptical cavities [17]. The gradients and energy gain per cavity are moderate. This scheme results in a large energy and phase acceptance. Additionally it gives plenty of tuning possibilities und provides the possibility to implement a dynamic fault tolerance scenario to meet the reliability requirements [18].
The beam dynamics layout is mainly driven by minimising emittance growth along the linac. The phase advance per lattice period is always kept well below 90° in order to avoid any structure or space charge driven resonance which can lead to emittance growth and halo formation [17].

**Spoke section**

The spoke section covers the energy between 17 and 81 MeV. It consists of 48 identical spoke cavities operated at 352.2 MHz. The optimum $\beta$ is 0.37 resulting in an accelerating cell length of about 15 cm. The cavities have been optimised with respect to minimum peak field ratios ($B_\text{p}/E_a=7.3$ mT/(MV/m), $E_p/E_a=4.3$), geometrical shunt impedance ($R_s/Q_0=217\ \Omega$) and minimum pressure sensitivity [17].

It has been decided to use short cryomodules housing two spoke cavities. Between the cryostats warm quadrupoles providing transverse focusing. Beside better access to components it simplifies fabrication and assembly of the linac. The nominal gradient for the MYRRHA spoke cavities is 6.2 MV/m and 8.2 MV/m in case of fault recovery mode [17].

**Figure 11: Spoke cryomodule**

In a next step a cryomodule has been constructed. Two spoke cavities fully equipped with 20 kW power couplers and tuner systems consisting of a slow and fast piezo based system have been successfully tested in that cryomodule. Figure 11 shows the cryomodule with cavities and auxiliary systems.

**Elliptical section**

The high-energy section accelerates the beam from 81 to 600 MeV using two groups of 704.4 MHz superconducting 5-cell cavities. Originally this cavity class has been developed for electron machines with $\beta=1$. Since the construction of the SNS linac [19] they are a common choice for superconducting hadron linacs above $\beta$ of 0.5. Due to the relatively low electric and magnetic peak fields elliptical cavities can be operated at high gradients. The number of cells in case of the MYRRHA linac is a compromise between energy gain per cavity, longitudinal acceptance and the possibility of fault tolerance.
The first group of elliptical cavities consists of 34 cavities covering the energy range between 81 and 184 MeV. Two cavities will be assembled per cryostat with warm quadrupoles in between. The geometrical $\beta$ of these cavities is 0.51.

**Figure 12: Horizontal prototype cryostat with $\beta=0.5$ elliptical cavity and test results**

A prototype cavity has been developed together with a horizontal cryomodule, cold tuner system and power couplers. The power couplers haven been successfully conditioned up to 62 kW cw [20]. The test with the whole system has delivered similar results as the vertical tests. Gradients of up to 14 MV/m have been achieved [20]. The nominal gradient is 8.5 MV/m and 10.5 MV/m in case of fault recovery mode. The measured pressure sensitivity was 350 Hz/mbar which is quite high for a superconducting cavity. This shows that low beta elliptical cavities are mechanically relatively soft. This might be a drawback with respect to stable operation and reliability. Beside further improvements of this cavity double spoke cavities from European Spallation Source (ESS) type could be an alternative. The high-energy part of the elliptical section consists of 60 5-cell cavities with an optimum $\beta$ of 0.705. Four cavities will be housed in one cryostat with warm quadrupole in between.

Table 3 summarises the parameters of all superconducting cavities of the main linac.
Table 3: Parameters of the cavity parameters of the main linac

<table>
<thead>
<tr>
<th>Section #</th>
<th>#1</th>
<th>#2</th>
<th>#3</th>
</tr>
</thead>
<tbody>
<tr>
<td>(W_{\text{in}}) MeV</td>
<td>17.0</td>
<td>80.8</td>
<td>184.2</td>
</tr>
<tr>
<td>(W_{\text{out}}) MeV</td>
<td>80.8</td>
<td>184.2</td>
<td>600</td>
</tr>
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<td>Cavity type</td>
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<td>Elliptical</td>
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<tr>
<td>Frequency MHz</td>
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<td>704.4</td>
<td>704.4</td>
</tr>
<tr>
<td>(B_{\text{p}}/E_{\text{a}}) mT/(MV/m)</td>
<td>7.3</td>
<td>5.5</td>
<td>4.6</td>
</tr>
<tr>
<td>(E_{\text{p}}/E_{\text{a}}) NA</td>
<td>4.3</td>
<td>3.3</td>
<td>2.5</td>
</tr>
<tr>
<td>(R/Q) (\Omega)</td>
<td>217</td>
<td>159</td>
<td>315</td>
</tr>
<tr>
<td>(E_{\text{a}}) (nom.) MV/m</td>
<td>6.4</td>
<td>8.2</td>
<td>11.0</td>
</tr>
<tr>
<td>(E_{\text{a}}) (max) MV/m</td>
<td>8.3</td>
<td>10.7</td>
<td>14.3</td>
</tr>
<tr>
<td># of cavities</td>
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<td>48</td>
<td>34</td>
</tr>
<tr>
<td>Synchr. Phase deg</td>
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<td>-36 to – 15</td>
<td>-36 to – 15</td>
</tr>
<tr>
<td>(Q_{L}) NA</td>
<td>(2.2\times10^6)</td>
<td>(8.2\times10^6)</td>
<td>(6.9\times10^6)</td>
</tr>
<tr>
<td>Aperture (\varnothing) mm</td>
<td>56</td>
<td>80</td>
<td>90</td>
</tr>
</tbody>
</table>

Reliability

Compared to other HPPA, many specifications are comparable (energy, beam current, required transmission, tolerable loss rate). The main difference is the required reliability with respect to beam trips. Only 10 beam trips longer than 3 s are allowed during a three month operation cycle [18]. There is a concern of a temperature drop in the reactor core after a trip which leads to thermal stress especially in the cladding of the fuel rods and eventually to material fatigue. Such a beam trip will lead to a shutdown of the reactor. The start-up procedure can take up to 20 hours. The limitation of beam trips is necessary to achieve the availability for the whole system of 80%. The required reliability level is up to 2 orders of magnitude higher than the achieved reliability of existing machines (see Figure 13) [18][21][22].

Figure 13: Comparison of number of allowed beam trips of MYRRHA and observed beam trips at SNS (average 2010-2013) [22]
There are different possibilities to increase the reliability of the MYRRHA linac. At first all components are designed conservatively and will be operated well below their technological limits. This will increase especially the life time of the components and increase the MTBF. The second approach is redundancy. A major component failure in the injector will likely lead to a beam trip. Therefore two injectors with parallel redundancy are foreseen. In case of the main linac a dynamic fault compensation scheme is planned (serial redundancy). In case of a detected cavity or RF system failure the phases and gradients of a number of neighbouring cavities are re-adjusted to keep the beam on target with nominal parameters. For that reason a certain margin in achievable gradients and power reserves of the amplifier is required [18][21].

Summary and outlook

Transmutation of high-level nuclear waste seems to be a promising alternative to long-term storage in underground repository. The MYRRHA Project aims to demonstrate the large-scale feasibility of transmutations. The MYRRHA proton linac has to provide a 2.4 MW cw beam with 600 MeV and 4 mA. Although is comparable to other HPPA reliability is a major concern. The linac design is very conservative with respect to beam dynamics and design values of components. Prototypes of each major component have been built and tested successfully. The proton source and LEBT are already in the test phase while the RFQ is under construction.

In a next step it is planned to build the linac up to an energy of 80 MeV and the construction of the RT CH-section of the injector could start in 2017.

Acknowledgements

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[12] www.ansys.com


The low-energy beam transport line for the MYRRHA accelerator

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Abstract

The MYRRHA Project aims at the construction of a new research reactor in Mol (Belgium) to demonstrate the transmutation feasibility with an accelerator-driven system (ADS). In its subcritical configuration, the MYRRHA facility requires a proton flux with a maximum power of 2.4 MW (600 MeV – 4 mA). Such a continuous wave (CW) beam will be delivered by a superconducting linear accelerator (linac) which must fulfill very stringent reliability requirements to ensure the safe ADS operation with a high level of availability. The proton beam will be injected in the superconducting linac with an energy of 17 MeV. The injector will enable to condition and pre-accelerate the proton beam from 30 keV to 17 MeV. The injector is composed of: the proton source, the low-energy beam transport line (LEBT), the Radio-frequency quadrupole (RFQ), and a series of CH-DTL cavities. Due to the very-high reliability requirements it is absolutely necessary to carry out prototyping of all the elements of the accelerator in order to optimize its operation. In this purpose a LEBT prototype has been built and is presently installed and operated at LPSC Grenoble (France). An experimental programme to optimize the tuning of the line, the beam transport and to study the space-charge compensation mechanism, is in progress. Once the line is commissioned it will be moved to Louvain-la-Neuve (Belgium) to be coupled with the RFQ. We here review the construction and the commissioning of the LEBT. Some experimental results are discussed, in particular the observed space-charge compensation effects.

Introduction

MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) is a fast spectrum nuclear facility which is planned to be built at SCK\textcdot CEN in Mol (Belgium) \cite{1}. This 100 MW\textsubscript{th} nuclear reactor is especially designed to demonstrate the ADS concept for high-level radioactive wastes transmutation.

The ADS subcritical core will be cooled by a lead-bismuth eutectic (LBE). The LBE will also be used as a spallation target to produce the necessary neutron flux to reach the criticality in the reactor. The facility therefore requires a continuous wave (CW) proton beam with a maximum power of 2.4 MW (600 MeV – 4 mA). In addition to the high-power beam requirement, the accelerator will have to operate with a high...
level of reliability. Indeed, frequently-repeated beam interruptions can induce high thermal stresses and fatigue on the reactor structures, the target or the fuel elements. In addition, beam interruptions will probably be systematically associated with reactor shutdowns that could also significantly affect the availability of the system, since the considered restart procedures could typically last about 20 hours [2].

To meet the reliability requirement, the present design of the MYRRHA accelerator is based on a superconducting linac solution [3], detailed in [4]. The 17-600 MeV MYRRHA main linac is composed of an array of independently-powered superconducting (SC) cavities. Three different cavity families are used to cover the energy range: a first section with 352.2 MHz Spoke 2-gap cavities ($\beta_{\text{opt}}=0.37$) and two following sections with 704.4 MHz elliptical 5-cells cavities ($\beta_{\text{opt}}=0.51$ & 0.70). The 17 MeV injector (Figure 1) will be composed of an ECR-ion source which will provide a 30 keV proton beam. Then the beam will transport through the low-energy beam transfer line (LEBT) and matched to the RFQ (Radio-Frequency Quadrupole) entrance. The 1.5 MeV bunched beam at the RFQ output will then be accelerated up to 17 MeV by CH-cavities [5].

![Figure 1: Schematic layout of the 17 MeV Injector [4]](image)

The conceptual design of this ADS-type proton accelerator had been initiated during previous Euratom Framework Programmes (PDS-XADS and EUROTRANS projects, 2001-2009) and it has been pursued in the frame of the MAX Project (Euratom FP7 programme, 2010-2014). To validate the made technological choices as regard to the reliability requirements it necessary to prototype each element that will compose the accelerator. In particular for the injector which is a crucial region of the linac. Indeed, it is in this area (LEBT + RFQ) that the beam is “shaped” for the rest of its acceleration in the linac. So the beam dynamics has to be understood and controlled to avoid beam losses in the superconducting linac. In addition, every element of the injector is unique and it is not possible to apply any series redundancy scheme to compensate the failure of one of them. In this purpose, one of the main focus of the MYRTE (MYRRHA Research Transmutation Endeavour, Horizon 2020, 2015-2019) project is the development and the construction of the RFQ and the commissioning of the LEBT.

The LEBT prototype has been built and is presently installed and operated at LPSC Grenoble (France). An experimental programme, to optimise the tuning of the line, the beam transport and to study the space-charge compensation mechanism, is in progress. Once the line is commissioned it will be moved to Louvain-la-Neuve (Belgium) to be coupled with the RFQ. We here review the design [6], the construction and the commissioning of the LEBT. Experimental results are discussed, in particular the observed space-charge compensation effects.

**LEBT function**

The LEBT represents three first metres of the MYRRHA accelerator. The purpose of this line is to ensure a reliable transport (with a minimum of losses) of the DC (Direct Current) 30 keV proton beam from the source to the RFQ. It also has to condition the beam to ensure its efficient acceleration in the rest of the
linac. Indeed, if the beam is not injected with proper parameters in the RFQ, this may cause losses in downstream parts of the accelerator. These losses have to be kept below 1 W/m, to limit activation and residual dose rate and to prevent deterioration of the superconducting cavity performances.

So, the goal of the LEBT transport line is to safely inject the proton beam inside the RFQ, by providing at the RFQ entrance a centred matched converging beam, with reasonable transverse emittances (ideally lower or equal to the RFQ design value of $\varepsilon_{\text{RMS.norm.proton}} = 0.2 \text{ mm.mrad}$) and the following Twiss parameters: $\alpha = 0.88$; $\beta = 0.04 \text{ mm/}(\pi \text{ mrad})$.

In addition the LEBT should enable to clean the proton from other species. Indeed, the ECR-ions source produces protons from the ionisation of a dihydrogen gas. Within this process other species than protons, such as $\text{H}_2^+$ and $\text{H}_3^+$, are also produced and extracted from the source. If these molecular ions are injected into the RFQ, they may create important parasitic losses. It is therefore absolutely necessary to intercept them in the LEBT.

The last function of the LEBT will be to create the beam time structure (Figure 2). In nominal operation the MYRRHA accelerator must produce a CW beam with periodical interruptions to monitor the subcriticality level of the reactor core, to adjust the mean beam power, and to feed in parallel and simultaneously an ISOL@MYRRHA facility by deviating up to 200 µA mean current from the MYRRHA beam by means of a fast kicker [7]. The beam “holes” will be created in the LEBT with a fast electrostatic chopping system.

![Figure 2: Proposed MYRRHA beam time structure for full power (~2.4 MW) nominal operation: Long pulses (2.28 MW) are sent to the reactor (duty cycle of 95%) while short ones (114 kW) are sent to the ISOL facility [7]](image)

Design studies

To fulfil all the functions required for the LEBT, several designs were studied. The final choice turned to a compact magnetic solution. It consists in a line which is about 3 meters long with two solenoids to guide and focus the beam to the RFQ input. On Figure 3, the proton beam dynamics in the LEBT, simulated with the TraceWin code [8], is presented. In this case, the electrostatic chopper, placed at the end of the LEBT, deflects the beam on the wall of the final collimation cone with an electrostatic voltage of 3.7 kV (about one-third the maximum chopper voltage). The chopper is the only electrostatic element in the LEBT. The number of electrostatic devices has been voluntary minimised, on a reliability basis, to avoid unwanted beam trips due to voltage breakdowns. A collimation system is placed in the middle of the line to clean the beam halo and to adjust the beam current. The collimation system also enables to intercept $\text{H}_2^+$ and $\text{H}_3^+$ particles. This process is described on Figure 4: according to tracking simulations one can expect to intercept about 99% of the $\text{H}_2^+$. About 80% of the $\text{H}_2^+$ will be lost in the aperture or intercepted by the collimation system. Most of the rest of $\text{H}_2^+$ will be collimated by the cone placed before the RFQ injection.
Figure 3: Multiparticle proton tracking in the LEBT with the TraceWin code [8]. The simulation input beam parameters are: \( \varepsilon_{\text{RMS.norm.proton}} = 0.2 \text{ mm.mrad} \), \( \alpha = -1.7 \) and \( \beta = 0.12 \text{ mm/(}\pi\text{.mrad)} \). An overall space-charge compensation of 90% was assumed except in the Chopper area (no compensation).

Figure 4: Multiparticle H\(_2^+\) beam simulation and percentage of beam lost along the line. The input beam parameters are the same than for the proton beam. An overall space-charge compensation of 90% was assumed.

Another solution to completely clean the proton beam from other species would have been to use a LEBT with a double achromatic deviation (dipoles), and thus in order to obtain energy separation of the ions. Nevertheless, with such a solution the LEBT would have been twice longer than the compact magnetic version. It would require more focusing elements (quadrupoles) and steering systems which makes the tuning procedure more complex than for the compact solution. In addition, the compact solution for the LEBT is cheaper than a longer version. Finally, a compact line enables to limit the space-charge compensation transient effects which has a strong influence on the beam dynamics, on the beam losses and consequently on the accelerator reliability.
High intensity and low-energy ion beams

Space charge and emittance growth

The possible sources of beam halo and emittance growth in a high-intensity injector are: the aberrations due to the ion source extraction optics, optical aberrations of the focusing elements, beam fluctuations due to ion source instability or power regulation and non-linearity of the electric field created by the beam space charge.

For high-intensity beams at low energy, the space charge is particularly strong. The electric field created by the space charge tends to defocus the beam and is strongly non-linear as it is induced by the non-uniform distribution of the charge particles of the beam. The space charge force is the result of two effects: a repelling force, due to the coulomb interaction of the beam charged particles, and an attractive magnetic force, induced by the movement of charged particles. In a first approximation, by considering a continuous cylindrical beam with an homogenous and uniform particles distribution, the resultant radial force – of these two effects – seen by a particle inside the beam is given by Equation 1. Where \( I \) is the beam current, \( q \) the beam particles charge, \( \varepsilon_0 \) the vacuum permittivity, \( r \) the radial position of the particle inside the beam, \( R \) the beam radius, \( c \) the light celerity, and \( \beta \) the reduced velocity of the beam particles.

\[
F_r = \frac{(1-\beta^2)}{\beta} \frac{q I}{2 \pi \varepsilon_0 c^2 R^2} \quad \text{(for } r < R) \tag{1}
\]

For the proton beam, the parameter \((1-\beta^2) / \beta\) is more than one hundred times higher in the LEBT (30 keV) than in than at the end of the MYRRHA linac (600 MeV). So, the low-energy beam transport is strongly dominated by this non-linear effect.

Space-charge compensation

The space-charge compensation (or neutralisation) occurs when a beam is propagating through the residual gas of the beam line. Its principle is described on Figure 5. The beam induces an ionisation of the residual gas, creating pairs of ions and electrons. In the case of a proton beam, the ions are repelled to the vacuum chamber walls, while the electron are trapped, by the positive potential of the beam, until a steady state is reached. In this way the global potential of the low-energy beam is neutralised. Thus, the space charge force is compensated; one could interpret this as an artificial decrease of the beam current in Equation 1.

Figure 5: Description of the space-charge compensation process [9]

The space charge compensation (SCC) can be of great advantage to limit non-linear effects in the beam transport through the LEBT, and optimise the injection into the RFQ. To enable SCC electrostatic elements must be avoided, otherwise the neutralising particles are attracted (or repulsed) by the electric field.
field induced by the focusing elements [10]. In the MYRRHA LEBT only the chopper is deflecting the beam with an electric field. This will induce transients that may cause losses in the RFQ or in the linac. These transients have to be measured experimentally. Although SCC have already been observed and used on several low-energy beam lines [11], it is a very complex physical phenomena to model that depends on many different parameters: the influence of the vacuum chamber walls, the beam transverse and longitudinal distributions, the different species (ions) inside the chamber, the residual gas interactions, the electro-magnetic fields from guiding elements, etc. The actual beam dynamics codes (such as TraceWin) are only based on rough approximation of this process. So it is of particular interest to experimentally study this process to develop and improve our models in order to optimise the commissioning and the operation of the accelerator. This reason was one the motivations to build the MYRRHA LEBT prototype.

The MYRRHA LEBT

The main objectives to build the LEBT prototype have been:

• to initiate the beginning of the MYRRHA linac construction and in particular its injector;
• to test and validate the adopted technical solutions for an accelerator which has to operate with an extremely high level of reliability;
• to experimentally study the SCC process in order to optimise the beam transmission to the RFQ and to improve low-energy beam transport models in view of the linac operation. Thus in order to develop a predictive tool which would enable to apply fine “online” LEBT tuning without perturbing the ADS operation.

The beam dynamics design was achieved within the MAX Project. The engineering design and the construction have been supported by SCK•CEN and the MARISA (MYRRHA Research Infrastructure Support Action, Euratom FP7 programme) project. The final design and the construction have been a collaborative work between LPSC (CNRS/IN2P3) and SCK•CEN teams. The ongoing commissioning and SCC experimental program is now carried out in the MYRTE Project. It mainly involves, LPSC, SCK•CEN and CEA, as well as COSYLAB [12] and ADEX companies for dedicated control system developments.

Figure 6: The MYRRHA LEBT at LPSC Grenoble

The LEBT presently installed and operated at LPSC, in Grenoble, is presented on Figure 6. The 2.45 GHz ECR (Electron Cyclotron Resonance) source was furnished by Pantechnik (France). Specific flat magnetic confinement configuration is provided by two permanent magnets, while a tapered axial RF
injection up to 1.2 kW is adopted. Beam is extracted from the plasma chamber by a multi-stage cascade of polarised electrodes at an Energy of 30 keV. This extraction voltage (30 keV) is high enough to ensure a safe beam transport (with SCC) and low enough to minimise the risk of high voltage breakdowns, as well as to ensure an effective bunching in the RFQ. The beam transport and focusing is ensured by two solenoid magnets – with integrated dipole steerers – provided by SigmaPhi (France). Collimation systems, located in the middle of the LEBT, are used for halo cleaning to improve the RFQ transmission, and several beam diagnostics can be inserted, allowing for interceptive beam current (Faraday cups), profile and transverse emittance measurements (Allison scanners). Before the RFQ injection flange, a short RFQ interface section hosts an electrostatic beam chopper, adopted to give the time structure to the beam delivery towards the future MYRRHA reactor.

Associated to its ion source, the LEBT is presently used to perform low-energy beam physics experiments on the space-charge compensation phenomenon, with the support of the MYRTE Project. Once fully characterised and commissioned the LEBT will be moved to Louvain-la-Neuve (Belgium) to be coupled with the RFQ: this will be a demonstration platform for the first stage of the MYRRHA injector.

Experimental results

Main objective

A dedicated SCC experimental test campaign have been carried out in 2015. The experimental goal of the experiment was to study the effects of the residual gas and of the solenoid on the beam properties in a steady-state regime. To do so, the beam properties (beam current, emittances, Twiss parameters) were measured after the first focusing solenoid. The beam diagnostics configuration is shown on Figure 6: a Faraday cup enables to measure the beam current right behind the first solenoid and two Allison scanners were used to measure the emittances and the Twiss parameters – in the horizontal and the vertical planes – at about 1.5 meters after the source extraction hole.

Analysis tool development

A measurement example of the beam particles distribution in the horizontal phase space is presented on Figure 7. Here the distribution of the different species are separated in the phase space. Indeed, they are not focused the same way by the solenoid, since their mass is different. The measurement enables to clearly identify, protons, $H_2^+$ and $H_3^+$. Nevertheless to extract the proton beam information, it is necessary to evaluate the background noise level, to select the ions of interest and to separate the distributions.

In this purpose a dedicated analysis code, TWISSGO, have been developed with MATLAB. The principle of this analysis tool is described on Figure 8. The first function of the tool enables to filter and smooth the background noise. It is also possible to “manually” remove the noise in some selected areas of the phase space. Then TWISSGO enables to separate and analyse the protons and the $H_2^+$ distributions. By choosing some signal levels the user can do a pre-separations of the distributions. Since the emittances distributions are crossing in the phase space, it is necessary to determine the amount of $H_2^+$ that are “wrongly” preselected in the protons distributions. This is achieved by interpolating the missing particles (hole) in the $H_2^+$ preselected distribution. According to this interpolation results the protons distribution can be corrected. Some final noise cleaning is then applied to the distributions and finally the statistical emittance and the twiss parameters are calculated.
Influence of the residual gas pressure

An extensive number of data have been collected: by injecting different type of gas in the line (Ar, He), and by varying the pressure level inside the beam line as well as the focusing strength of the solenoid. We present here, on Figure 9, one studied case showing the evolution of the proton horizontal emittance as function of the pressure level within the line. In this case the pressure is modified in the LEBT by Argon gas injection. The solenoid focusing strength was kept constant by maintaining a current of 69 A in the coil. The beam current was also maintained at a constant value at the emittance measurement location (I_{proton} = 8.5 mA).
Figure 9: Measurement of the proton horizontal emittance as function of the residual gas pressure in the middle of the MYRRHA LEBT

The measurement showed that the emittance is decreasing while the gas pressure is increased in the line. In addition, the beam divergence is also evolving as function of the pressure in the LEBT chamber. Indeed, Figure 10 shows the evolution of the Twiss parameter alpha as function of the pressure. Initially, when no argon is injected in the line, the beam is diverging ($\alpha < 0$). When the argon gas pressure is increased in the line, the value of $\alpha$ is also increasing. The beam starts to converge when the gas pressure reaches a value of $\sim 3 \times 10^{-5}$ mbar, and this without acting on the focusing strength of the solenoid. In additions it appears that the beam emittance distribution is less affected by non-linear effects when the gas pressure is increased.

Figure 10: Measurement the proton beam Twiss parameter $\alpha$ (horizontal plane) as function of the residual gas pressure in the middle the MYRRHA LEBT
Figure 11: Measurements of the beam particles density in the horizontal phase space for two different residual gas pressure. Argon injection with pressures of: 9.2 $10^{-6}$ mbar for pattern a, and 5.4 $10^{-5}$ for pattern b

According to these measurements one could make the hypothesis that the overall space-charge compensation degree [10] is modified. Nevertheless, within the explored pressure range in the LEBT a SCC degree of 95% to 99% is expected, such variation of the SCC degree may not explain such a diminution of the emittance. But it is also important to note that the SCC degree is probably not homogenous in the whole LEBT. It seems that inside the solenoids SCC is reduced. Inside the solenoids the electrons would not be able to move in radial direction with the same mobility than in a region without magnetic field. Most of the electrons would be confined near the axis of the solenoids: secondary electrons which are generated close to the fringe field would also be moved towards the axis [11]. By increasing the gas pressure this effect would be decreased, maybe due to a statistical increase of the electron population. The influence of the Allison scanner itself also has to be explored. When the diagnostics penetrates into the beam, secondary electron are emitted from the Allison scanner copper screen. This may locally increase the electron population and modify the SCC pattern.

Conclusions and perspectives

The low-energy beam transfer line of MYRRHA, based on a magnetic compact solution was built and have been operated at LPSC. The final steps in the commissioning of the line will be to install and operate the electrostatic chopper which will enable to create the beam time structure of the linac. The ECR source have been tuned and can produce a stable DC beam of 25 mA (total current).

In 2015, the experimental programme on the LEBT was focused on the experimental study of the space-charge compensation effect. The influence of the residual gas Type and pressure has been measured. Even though the SCC physical process need to be better understood, the experimental measurements showed that operating the LEBT by injecting Argon gas in the line, to keep a pressure around $5.10^{-5}$ – $1.10^{-4}$ mbar, could be a real advantage for the beam dynamics to minimise the beam losses probability in the linac. Indeed, the preliminary analysis showed that increasing the pressure in the line would help to decrease the emittance and non-linear effects, and thus to better control the injected beam in the RFQ. In addition, increasing the pressure in the LEBT will enable to reduce the SCC transient time when the beam will be chopped. This will also be an advantage to optimise the RFQ transmission especially during the beginning and the end of the beam macro pulse. Nevertheless, an increase of the residual gas pressure is attenuating the beam transmission through the LEBT; due to the electron capture of the protons. In addition, a two high pressure in the line may cause high voltage breakdowns of the
chopper system. So, it appears that the type and the residual gas pressure in the LEBT will be one of the most important parameters in the fine tuning of the LEBT. The better compromise will have to be found in order to minimise non-linearity in the beam transport with a maximum level of reliability.

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References

Design features of modern high-power proton linacs

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Abstract

The evolution of high-power linac projects and operations experience such as at Los Alamos reveals common design features that can be summarised as an integrated physics-design approach applicable to all high-power linac designs. This design approach leads to high performance, operational flexibility, low beam losses and high reliability and availability. Details of this approach will be presented along with relevant examples including past Los Alamos projects and several current projects such as the European Spallation Source (ESS) linac design, and others being developed in support of accelerator-driven systems (ADS). This work is supported by the United States Department of Energy, National Nuclear Security Agency, under contract DE-AC52-06NA25396.

Introduction

Over the last approximately 40 years, substantial contributions in beams physics and accelerator technology have been made that have impacted worldwide high-power linac designs. Many of these are seminal contributions made at Los Alamos. These include the development of accelerator design and simulations codes including the first applications of high-performance computing, advances in understanding beam dynamics of space-charge-dominated beams, emittance growth mechanisms and beam halos, the development of the RFQ, development of the EPICS Control System, and the use of superconducting RF (SCRF) accelerating structures for high-power beams.

Technology development at Los Alamos was supported by several projects including the Accelerator Production of Tritium (APT) Project [1] (1990-2001) and the Advanced Accelerator Applications (AAA) Program which included accelerator transmutation of waste (ATW) and accelerator-driven systems (ADS). It was during these projects that SCRF spoke resonator and elliptical cavities for low- and medium-velocity application, MW-class RF power couplers and high-power, high-energy RFQ technology to address space charge-related beam issues at low velocities in high-power linacs were developed.

Several technology demonstrations were achieved during the APT project which paved the way for future projects. The most significant of these was the fabrication and testing of the Low-Energy Demonstration Accelerator (LEDA) [2]. The LEDA was designed to demonstrate the front-end of the APT linac and consisted of a 75-keV proton injector followed by a 6.7-MeV, 100-mA, CW RFQ. The LEDA still currently holds the world record as the highest-average-power RFQ ever operated (670 kW). Demonstration of the LEDA was followed by beam halo experiments which used the LEDA RFQ beam coupled to a beam transport lattice to understand beam halo growth and associated beam-loss mechanisms to verify beam halo models and simulation code performance. Operations data from the Los
Alamos Neutron Science Center (LANSCE) linac while still operating at MW-levels was also used to do beam-loss benchmarking of design codes [3].

It was during the APT Project that a new design approach for high-power linacs was developed which became the basis for other designs that followed. This design approach was successfully applied to the Los Alamos design of the spallation neutron source (SNS) built at Oak Ridge National Laboratory in the US, operating at approximately 1-MW average beam power since 2010. Analyses of other recent high-power proton linac designs such as the ESS currently under construction, the Chinese ADS (C-ADS) design, the International Fusion Materials Irradiation Facility (IFMIF) and the Multipurpose hYbrid Research Reactor for High-Tech Applications (MYRRHA) 17-600 MeV SCRF linac all share common design features that originated with the earlier work on APT.

Since about 1995, all new high-power proton linac projects include SCRF accelerator sections. Progress in SCRF technology has made it more attractive and broadly applicable. Progress includes increases in accelerating gradients due to a better understanding of the surface physics of SCRF structures and improved cavity processing methods. Additionally, successful operating experience at facilities like the SNS has helped to further build confidence in the use of SCRF technology for high-power linacs. Development of high-power SCRF accelerator technology is now a worldwide effort. The ESS will be the next high-power proton linac to come online.

A SCRF linac provides many advantages for high-intensity applications. First, there is reduced operating cost through improved electrical efficiency. Also, as achievable accelerating gradients have increased over time, improved design solutions have been found to take advantage of these high gradients to realise significant construction and operational cost savings by reducing the number of accelerating cavities and associated RF and cryogenic cooling systems required. Reduced linac footprint implies reduced tunnel construction costs.

The use of SCRF cavities also offers improved operational flexibility. The typical short multi-cell cavities of a SCRF linac have larger velocity acceptance for the beam. Independent phasing of the cavities (or sections of cavities) can allow operation even with some RF-module failures. The larger cavity bore radius, acceptable for SCRF cavities, allows off-energy or poorly-focused beams after faults to be transported with minimal or controlled losses. It also allows less constrained transverse focusing that may be needed to balance beam-loss mechanisms in the linac. And reduced sensitivity to beam trips from thermally induced perturbations to cavity RF fields should result from the more stable operating temperatures resulting when operating at cryogenic temperatures.

Considerable ADS system design challenges still lie at the accelerator/subcritical multiplier (SCM) interface. Requiring the accelerator to meet reactor-level operational specifications or, likewise, requiring the SCM to be capable of accepting an erratic beam from an accelerator continue to be difficult design requirements to meet. Beam availability has been the usual requirement for accelerators rather than beam continuity/reliability (for example, greater than 90% availability has been achieved at SNS). Therefore, a major design goal for high-power ADS linacs continues to be how to substantially reduce beam interrupts so as to limit thermal stresses in the target. Reduction of beam-interrupt rates, as is presently required, continues to be very challenging.

An integrated design approach

All modern high performance, high-power linac designs require a balance of operational robustness, high reliability and availability and minimising the impact of uncontrolled beam losses. An integrated approach to the design of high-power linacs that leads to high performance, operational flexibility, low beam losses, high reliability and availability has been presented before [4]. The key elements of this design
approach include the use well-benchmarked simulation codes for design, beam dynamics and physics simulation, and estimating beam losses.

Physics-design goals include efficient acceleration and achieving high beam capture and low losses in all sections of the linac. Low losses can be achieved by minimising the impact of transitions in the linac through current-insensitive transverse and longitudinal focusing, controlling emittance growth and avoiding beam instabilities and resonances. The use of appropriate figures of merit (aperture-to-rms ratio, halo fraction, kurtosis, tune spread, equipartitioning ratios, etc.) are needed to quantitatively assess performance of the design, and error simulations are required to determine fabrication, alignment and operational tolerances.

The use of reliability, availability and maintainability (RAM) modelling, failure mode and effects analysis (FMEA), and fault tolerance studies also contribute to achieving the required high reliability and high availability. These studies inform the approaches needed to compensate for system faults and failures, including adding redundancy or modularity or implementation of real-time control and re-tuning methods. Several studies that developed new models and a base of previously-unavailable accelerator RAM data were compiled during the APT Project [5, 6] are also being applied to many current projects worldwide today.

Cost optimisation has historically not been widely used as part of the design process, as the main focus has been on meeting technical performance requirements. However, as the cost of large-scale accelerator facilities has increased, attention to cost optimisation has become more important and also needs to be considered. Cost optimisation is typically best understood through the use of parametric studies early on in the design of the fully-integrated linac system. For example, the selection of the final beam energies for APT and ATW were based on taking into account the rate of neutrons per beam watt of incident protons on the neutron-production target, as well as minimising overall cost [7]. Reliability data and modelling were also successfully applied to cost optimisation for APT [8]. More recently, design modifications have been made to reduce costs for the ESS while maintaining its performance requirements [9].

Understanding engineering and fabrication constraints early on in the design process leads to less design iteration and speeds overall optimisation. Emerging innovations in technology should be considered as the design evolves but associated technical and cost risks need to be evaluated before modifications to a design are considered.

This integrated approach has been successfully demonstrated in the past during the APT and SNS projects at Los Alamos. Most elements of this approach can also be found in recent modern high-power proton linac designs for spallation neutron sources such as the ESS [10] and ADS linac designs such as the MYRRHA 600-MeV proton linac [11] and the Chinese ADS project (C-ADS) [12]. Some important aspects of this design approach are discussed in the subsections below.

**Physics-design constraints to minimise beam halo and beam losses**

Minimising the impact of beam losses is important for all high-power linacs. Uncontrolled beam losses along the linac should not exceed 1 W/m at high energies for hands-on maintenance and for high availability (lower losses minimise radiation-induced equipment damage). There are several known beam-loss mechanisms including gas and magnetic field stripping (H+ only), scattering including intra-beam scattering, beam mismatch and space charge effects that increase beam emittance and generate beam halo. Others include system failure or malfunction, RF turn on transients, longitudinal and transverse field errors, and misalignments and steering errors that reduce or severely limit beam acceptances. The impact of some of these beam-loss mechanisms can be reduced or eliminated through good design choices.
Others can only be reduced by addressing operations-related issues. Beam halo is the most difficult to predict and to control.

However, proper selection of design parameters can minimise beam halo and beam losses. These include the use of SC cavities which provide large apertures without the penalty of high RF losses, strong focusing to maintain small beam sizes and control emittance growth, and current independent focusing lattices to smoothly transition the beam from one section of the linac to another. Current independence is accomplished by adiabatically varying transverse and longitudinal focusing parameters across the interface between two otherwise discontinuous sections of the linac to maintain a constant zero-current phase advance per unit length across the interface.

Providing strong beam focusing of space-charge-dominated beams in general will reduce emittance growth and result in smaller rms beam sizes. Results from the Particle-Core Model [13] indicate that maintaining a small rms beam emittance reduces the maximum halo amplitude and should, therefore, reduce beam losses. However, transverse and longitudinal tune depressions should stay above 0.3 to avoid inducing chaos within the beam that will increase the number of particles populating the halo. One approach to avoiding severe tune depressions is to reduce the number of particles per beam bunch. This is achieved by using high RF frequencies, another design choice. Recent work by Eshraqi and Lagniel [14] has developed equations that better link the basic linac parameters such as RF frequency, zero-current phase advance and emittance to minimise tune spreads since they are key parameters for halo formation and beam losses.

Several sources of free energy exist to drive space charge-induced emittance growth and production of beam halo. These include charge redistribution, thermal energy transfer, rms mismatch and structure resonances. Beams that are in equilibrium with the focusing channel of the linac experience no emittance growth. Good matching and a current-independent design are important but may not completely eliminate development of beam halo, although mismatch is perhaps the most common cause of emittance and halo growth. Envelope modes of mismatched bunched beams can drive particles to large amplitudes. The maximum resonant amplitude depends on emittance, beam energy and mismatch factor. Large mismatches drive particles to large amplitudes, however the maximum amplitude is damped at high beam energy.

The objective of beam matching at a transition in the linac focusing channel is to inject a beam with a phase-space ellipse (phase-space orientation) that conforms to the acceptance ellipse of the linac. Transverse current-independent matching is achieved by adiabatically (slowly) changing the quadrupole focusing gradients near the transition to make the phase advance per unit length the same for both adjacent focusing channels. Typically four quadrupoles must be adjusted (two on each side of the transition) to align the beam phase-space ellipse with the focusing channel acceptance ellipse. Current-independent longitudinal matching can be achieved by adiabatically adjusting the cavity RF phases and amplitudes near the interface to make the longitudinal phase advance per period the same for both channels at the interface.

**Fault compensation methods**

High availability (>80%) and high beam continuity (beam trips >3 sec must be very infrequent) are required to meet the extremely stringent requirements for ADS to minimise thermal stresses and material fatigue on the target window, reactor structures, and fuel assemblies [15]. This requires a combination of a robust design, redundancy and active fault compensation methods.

Approaches and methods of active fault compensation continue to evolve. As early as the APT project, simulation codes were used to study the robustness of the design and the inherent sensitivity of the linac performance to various types of operational faults or failures. Reliability studies in 2002-2004 in
support of MYRRHA within the PDS-XADS FP5 project also supported the need for fault compensation [16]. Additionally, reliability results from a 2015 study in support of the MYRRHA Project using SNS operations data have confirmed that the SCRF linac cavities and cryomodules, RF systems, and magnet power supplies and controllers most impact overall linac reliability and availability [17].

From this understanding, approaches to mitigate the observed performance degradation and associated beam losses have been developed. Local compensation methods for losses in transverse focusing and beam acceleration (loss of accelerating cavities) were explored and demonstrated during the APT project and during the design of a 600-MeV, 13.3-mA, CW linac design for a proposed ADS Test Facility in 2001 using beam simulations [18]. These simulation studies demonstrated the need to use multiple magnets and multiple cavities for local compensation of a component failure. It was also learnt that SCRF operating parameters need to be de-rated to re-establish the beam energy following the failure location in the linac without exceeding the available RF power in the compensating RF modules. A similar approach was later successfully demonstrated during beam operations at the SNS [19]. The SNS approach was also based on beam simulations, however included the develop of look-up tables used to quickly establish new RF phase and amplitude set points and re-establish operations.

The proposed 600-MeV MYRRHA linac design uses parallel redundancy of the 17-MeV injector section of the linac to improve reliability. The 17-MeV injector section includes an ion source, RFQ, beam matching and initial spoke-cavity section. The injector will be duplicated and the duplicate maintained as a hot spare. Upon failure of any component of the injector in use, a fast switching dipole will used to switch over to the hot spare. Fault compensation of the main linac relies on tuning flexibility, large beam acceptance and serial redundancy. Serial redundancy (local compensation) uses nearby beam line components (magnets and RF cavities) to compensate for a fault. This is the same approach as developed for APT and verified at the SNS. Local compensation has the advantage that a minimal number of RF cavity settings need to be modified, allowing quicker fault recovery.

The general fault compensation methods discussed so far rely on offline simulations to develop look-up tables for new magnet and RF cavity set points. Fault compensation is then achieved after fault detection via operator intervention using these results to determine the new operating settings. In general this process takes several minutes to achieve. Meeting the <3-sec requirement for ADS will require improved capabilities for fault diagnostics, controls systems developments and intelligent/automated fault compensation methods including fast automated beam restart and tuning. Towards that goal, local distributed computing approaches using fast Field Programmable Gate Arrays (FPGAs) coupled with genetic algorithms are being developed to improve fault compensation for the C-ADS linac [20]. Other approaches are also being developed worldwide in support of ADS.

**Advanced linac tuning methods**

Advanced tuning methods based on automated controls and optimisation methods will be needed to meet the challenging requirements of low beam losses and very-high reliability and availability for ADS. Several online simulators are being developed to help achieve this goal.

The ESS linac simulator has been under development for several years and is optimised for the EES linac [21]. It is written in an open XAL architecture that allows fast switching between different physics models, however does not yet use multiparticle simulations. This model is expected to be used for trajectory adjustments and establishing operating phase and amplitude set points during beam operations. Future development includes fast GPU-based particle tracking for improved beam modelling and increased computational speed. No plans for use of the simulator for active fault compensation or to minimise beam loss is apparent, however, extension of the presently envisioned implementation to include these capabilities would not be surprising.
Automated fast beam tuning methods have been proposed for ADS that will attempt to restore beam operations after a fault within 1-2 seconds. One such method is based on the use of fast controllers and automated feedback with a non-linear state space representation of the beam [22]. Optimised magnet set points have been successfully demonstrated coupling this beam representation method with chi-square minimisation of state variables that represent the beam envelope parameters and magnet set points.

A GPU-based online simulator has been developed at Los Alamos to provide near-real-time proton linac modelling and simulation [23]. This high-performance simulator (HPSIM) is designed to connect directly to the accelerator’s EPICS [24] control system. Coupling accurate beam simulation models such as HPSIM with modern real-time optimisation methods is expected be the next step in achieving the required fault recovery and reliability and availability goals for ADS [25]. This approach has the potential to quickly understand and optimise linac performance, including the minimisation of beam losses during stable operations, and to provide automated fault recovery during off-normal operating events.

Recent approaches to beam optimisation include the application of multi-objective particle swarm genetic algorithms (MOGA). These optimisation algorithms have been extensively applied to various accelerator design problems ranging from magnet and RF cavity design to global optimisation of an accelerator lattice [26] but application to high-power linac operations and beam-loss minimisation is only recent and still needs significant development. Multi-objective optimisation algorithms strive to find optimal trade-off possibilities between different and often conflicting objectives, making them appropriate for use in accelerator optimisation. Several optimisation methods have been investigated for use with the LANSCE linac that may be appropriate for ADS. Other new methods based on model-independent dynamic feedback may also be a viable approach for systems with limited real-time diagnostics and noise measurement data. These local methods in combination with the global optimisation capabilities of genetic algorithms have the potential to improve accelerator performance.

A simple, model-independent technique which uses a defined, measureable cost function to be minimised has also been demonstrated through a multiparticle simulation of the LANSCE linac low-energy beam transport region where the tuning of 22 magnetic quadrupoles and two RF buncher cavities was successfully optimised [27]. A major benefit of this model-independent approach is that its complexity does not grow as the number of parameters to optimise increases. However, this approach is a local technique, similar to other techniques like the gradient descent method, and may become trapped in local minimums. Applying a hybrid approach that combines model-dependent genetic algorithms for global optimisation with a model-independent approach for local tuning may overcome this limitation.

In addition to tuning optimisation and beam-loss minimisation, a future area of development needed for automated fault recovery is real-time monitoring of systems and their status, and being able to use this data to predict system or component failures in real-time. Presently this is being done offline using historical operations data and system/component failure analyses.

**Representative design examples**

Several design examples exist today that take advantage of what has been learnt since the early development of high-power proton linacs. As discussed earlier, the APT project established many aspects of the physics-design approach and included some early technology demonstration such as the LEDA. The SNS linac is the first working modern high-power linac operating at the MW-power level based on the APT design approach. The soon-to-be-completed ESS linac will be the next realisation of this design approach, embodying additional new advancements in design understanding and pushing the beam power limits to 5-MW. Several representative ADS designs have already been mentioned including MYRRHA, IFMIF, C-ADS and others. Since the 15-MW C-ADS design is rapidly advancing and represents the next leap in
beam-power performance, it will be compared to the SNS and ESS designs. A brief summary of each of
these three representative designs is given below and a comparison is made demonstrating the design
similarities between them in achieving their expected performance goals.

SNS

The SNS linac has been operational since 2006. Table 1 summarises the primary SNS parameters.
Figure 1 shows a schematic of the accelerator layout and beam pulse structure. Details of the final SNS
design, including predicted beam performance and studies performed to estimate losses and sensitivities
to resonances, etc. can be found in References 28, 29 and 30.

Operation of the SNS linac over the past nearly a decade confirms the predicted performance and
high availability resulting from good design choices. Measured operational availability has been as high as
> 95%. The combination of the design choices and implementation of modern controls allows the SNS to
quickly turn on after extended shutdown periods; days are required to bring the SNS back into full-power
operation as compared to older machines such as LANSCE that require weeks to months. The design
choices also allow for quick re-tuning of the linac (within minutes) upon loss of operation of a SCRF
cryomodule. Finally, the large beam apertures afforded by the use of SCRF cavities has allowed reduced
transverse focusing to mitigate unanticipated losses due to intra-beam scattering which depend on the
beam-gas interaction cross-sections while still meeting beam-on-target requirements.

Table 1: SNS primary parameters [28]

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Final beam energy, E (MeV)</td>
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</tr>
<tr>
<td>Repetition rate (Hz)</td>
<td>60</td>
</tr>
<tr>
<td>Frequency (MHz)</td>
<td>402.5/805.0</td>
</tr>
<tr>
<td>Uncertainty, ΔE (95% probability) (MeV)</td>
<td>± 15</td>
</tr>
<tr>
<td>SCRF cryomodule number</td>
<td>11+12</td>
</tr>
<tr>
<td>SCRF cavity number</td>
<td>33+48</td>
</tr>
<tr>
<td>Peak gradient, β=0.61 cavity (MV/m)</td>
<td>27.5 (± 2.5)</td>
</tr>
<tr>
<td>Peak gradient, β=0.81 cavity (MV/m)</td>
<td>35 (+2.7/-7.5)</td>
</tr>
<tr>
<td>Beam power on target (MW)</td>
<td>1.4</td>
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<tr>
<td>Pulse length on target (µs)</td>
<td>695</td>
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<tr>
<td>Chopper beam on duty factor (%)</td>
<td>68</td>
</tr>
<tr>
<td>Linac beam macro pulse duty factor (%)</td>
<td>6.0</td>
</tr>
<tr>
<td>Average macro pulse H current (mA)</td>
<td>26</td>
</tr>
<tr>
<td>Average linac beam current (mA)</td>
<td>1.6</td>
</tr>
<tr>
<td>Ring RF frequency (MHz)</td>
<td>1.058</td>
</tr>
<tr>
<td>Ring injection time (ms)/turns</td>
<td>1.0 / 1 060</td>
</tr>
<tr>
<td>Ring bunch intensity (10^-9)</td>
<td>1.5</td>
</tr>
<tr>
<td>Ring space charge tune shift, ΔQ_sc</td>
<td>0.15</td>
</tr>
</tbody>
</table>
Figure 1: Schematic layout of the SNS accelerator complex [31]

ESS

Construction of the ESS started in 2014 and is making steady progress towards its expected completion and first beam on target in 2019. The user programme is expected to begin in 2023 followed by full completion of the entire facility in 2025 [32].

To reduce cost of the ESS linac, in 2013 the final beam energy was decreased from the previous baseline of 2.5 GeV [33] to 2.0 GeV [34]. To maintain the required 5-MW average beam power on target the peak beam current was increased from 50 mA to 62.5 mA. Optimisation of the accelerator lattice also included changes to the transition energies between normal and superconducting sections of the linac, and increasing the required SCRF cavity accelerating gradients. The ESS linac will be the first SCRF linac to incorporate spoke-resonator cavities, although previous designs for waste transmutation as early as 1996 considered their use [35]. The availability/reliability goal is still >95% with an expected mean time between failures to exceed 22 hours. The SCRF sections of the linac will operate at 2K. The estimated mechanical time constant of the SCRF cavities is comparable to the 2.86 ms beam pulse length. As a result, static pre-detuning of the cavities as is done at the SNS will not suffice and dynamic tuning compensation using active piezo-electric tuners will be required. Table 2 below summarises the primary ESS linac parameters. Figure 2 shows a schematic layout of the optimised ESS linac.

Table 2: ESS primary linac parameters [34, 36]

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Final beam energy, E (MeV)</td>
<td>2000</td>
</tr>
<tr>
<td>Repetition rate (Hz)</td>
<td>14</td>
</tr>
<tr>
<td>Frequency</td>
<td>352.21/704.42</td>
</tr>
<tr>
<td>SCRF cryomodule number</td>
<td>13+9+21</td>
</tr>
<tr>
<td>SCRF cavity number</td>
<td>26+36+84</td>
</tr>
<tr>
<td>Peak gradient, β=0.5 cavity (MV/m)</td>
<td>9.0</td>
</tr>
<tr>
<td>Peak gradient, β=0.64 cavity (MV/m)</td>
<td>45.0</td>
</tr>
<tr>
<td>Peak gradient, β=0.86 cavity (MV/m)</td>
<td>45.0</td>
</tr>
<tr>
<td>Beam power on target (MW)</td>
<td>5.0</td>
</tr>
<tr>
<td>Pulse length on target (µs)</td>
<td>2,860</td>
</tr>
<tr>
<td>Linac beam macro pulse duty factor (%)</td>
<td>4.0</td>
</tr>
<tr>
<td>Peak macro pulse H+ current (mA)</td>
<td>62.5</td>
</tr>
<tr>
<td>Average linac beam current (mA)</td>
<td>2.5</td>
</tr>
</tbody>
</table>
The ESS design takes advantage of many of the approaches discussed in previous sections of this paper, in particular to reduce uncontrolled beam losses. The design goal is to keep fractional beam losses below $10^{-7}$. The most prevalent sources of beam loss are related to halo formation and include beam mismatches, high space charge and high tune depressions, the effects of non-linear fields, and capture losses during acceleration and transition between structures of varying RF frequency. To avoid the impacts of these sources of halo formation, the design of the ESS linac includes maintaining zero-current phase advances per period below 90 degrees to avoid envelope instabilities, smoothly transitioning and varying monotonically the phase advance per unit length to allow for nearly-current-independent matching and beam transitions between sections of the linac. The tune depression at any point along the linac is also required to remain $>0.4$ to minimise the impact of space charge [33].

Earlier designs of the ESS linac considered equipartitioning the beam to minimise emittance growth and associated beam losses due to energy exchange between transverse and longitudinal degrees of freedom [33]. Beam equipartitioning is difficult to achieve and maintain. Instead, the ESS SCRF linac is designed to have equal tune depressions in all three emittance planes (transverse and longitudinal) to minimise beam losses [10].

**Figure 2: Schematic layout of the ESS accelerator [34]**

![Image of Schematic layout of the ESS accelerator](image)

**C-ADS**

Since 2011, China has been preparing the design of a 15-MW driver linac for a 1000-MW thermal power ADS demonstration facility expected to be completed by 2032 [12]. The driver linac is defined to have a final beam energy of 1.5 GeV, operating in CW mode with a beam current of 10 mA. Table 3 below summarises the main C-ADS proton linac parameters. Figure 3 below shows a schematic layout of the C-ADS linac. Note the two different injector topologies shown.

**Table 3: C-ADS primary linac parameters [12]**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Final beam energy, $E$ (MeV)</td>
<td>1500</td>
</tr>
<tr>
<td>Repetition rate (Hz)</td>
<td>CW</td>
</tr>
<tr>
<td>Frequency (MHz)</td>
<td>(162.5) 325.0/650.0</td>
</tr>
<tr>
<td>SCRF cryomodule number</td>
<td>19+15+17+20</td>
</tr>
<tr>
<td>SCRF cavity number</td>
<td>38+60+51+100</td>
</tr>
<tr>
<td>Peak gradient, $\beta=0.21$, 0.40 cavity (MV/m)</td>
<td>32.5</td>
</tr>
<tr>
<td>Peak gradient, $\beta=0.63$, 0.82 cavity (MV/m)</td>
<td>39.0</td>
</tr>
<tr>
<td>Beam power on target (MW)</td>
<td>15</td>
</tr>
<tr>
<td>Pulse length on target (µs)</td>
<td>12 500</td>
</tr>
<tr>
<td>Linac beam macro pulse duty factor (%)</td>
<td>100</td>
</tr>
<tr>
<td>Average macro pulse H+ current (mA)</td>
<td>10</td>
</tr>
<tr>
<td>Average linac beam current (mA)</td>
<td>10</td>
</tr>
</tbody>
</table>
The C-ADS design also incorporates many of the design approaches discussed earlier to achieve high reliability and availability: making extensive use of component derating and redundancy to compensate for faults, making the design modular to allow for local re-matching, and making proper physics choices that improve the inherent fault insensitivity of the design. Recent testing of the Injector-1 configuration that includes a 325-MHz RFQ followed by two 325-MHz SCRF spoke-resonator cryomodules has been successfully completed [37]. Testing of the Injector  II scheme is expected in the future and a down-select to a single scheme will be made. The C-ADS design incorporates redundant injectors to mitigate loss of a major component at low beam energy where other compensation methods are difficult and unlikely to succeed.

The main SCRF linac lattice structure is compact and modular, incorporating the use of both SCRF spoke cavity and elliptical-cavity cryomodules. Transverse focusing in all sections is provided by solenoidal focusing compared to other designs such as the ESS, the MYRRHA and the ESS, all of which use quadrupoles. For the C-ADS, individual cavity or focusing magnet failures can be handled at all stages without introducing significant beam loss along the linac through local compensation and beam re-matching [12].

Transverse focusing in the main linac is designed to keep the zero-current phase advance per period below 90 degrees. This criterion is also met in the longitudinal phase-space plane through appropriately selecting the SCRF cavity fields and synchronous phases. Additionally, these values are optimised to avoid parametric and space charge resonances that can drive the halo and resulting beam losses and to approximately obey the conditions for equipartitioning [12]. Continuity of the zero-current phase advance per unit length at lattice changes are also maintained to reduce the current dependence of the design. Matching schemes have been investigated that maintain smooth RMS beam envelope changes to control emittance growth due to mismatch to less than 10% overall [37].

Figure 3: Schematic layout of the C-ADS linac showing the two different injector schemes [12]

Comparison of SNS, ESS and C-ADS Designs

Table 4 below summarises a comparison between the operating SNS, the now under construction ESS and the planned C-ADS. The first half of the table is a very brief summary of design and beam parameters for the purpose of demonstrating similarities between the requirements of the three linacs. The second half of the table compares the common aspects of the design approach used for each design.
Low-loss operation of the SNS linac demonstrates our design approach first developed during the APT project. A similar approach has been applied to the design of the ESS which builds on the experience and success of the SNS. It is expected that other future high-power linacs, including those for ADS, such as the C-ADS will also use a similar design approach.

Table 4: Comparison of SNS, ESS and C-ADS designs

<table>
<thead>
<tr>
<th>Parameter</th>
<th>SNS</th>
<th>ESS</th>
<th>C-ADS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Average Current</td>
<td>1.6 mA</td>
<td>2.5 mA</td>
<td>10 mA</td>
</tr>
<tr>
<td>Peak Current</td>
<td>26 mA</td>
<td>62.5 mA</td>
<td>10 mA</td>
</tr>
<tr>
<td>Rep Rate</td>
<td>60 Hz</td>
<td>14 Hz</td>
<td>CW</td>
</tr>
<tr>
<td>Pulse Length</td>
<td>1 ms</td>
<td>2.86 ms</td>
<td>12.5 ms</td>
</tr>
<tr>
<td>Beam Energy</td>
<td>65 keV-1000 MeV</td>
<td>75 keV-2000 MeV</td>
<td>35 keV-1500 MeV</td>
</tr>
<tr>
<td>Frequency (MHz)</td>
<td>402.5-805</td>
<td>352/704</td>
<td>325/650</td>
</tr>
<tr>
<td>Trans. Emittance (π-mm-mrad, norm.)</td>
<td>0.26</td>
<td>0.30</td>
<td>0.21</td>
</tr>
<tr>
<td>Tune Depression (\sigma_l/\sigma_0)</td>
<td>0.5-0.7, 0.3-1.25</td>
<td>0.5, 0.45</td>
<td>0.6-0.87, 0.73-0.85</td>
</tr>
<tr>
<td>(a_0/x_{rms})</td>
<td>8-10</td>
<td>&gt;10</td>
<td>&gt;10</td>
</tr>
<tr>
<td>Availability Goal</td>
<td>90% (95%)</td>
<td>95%</td>
<td>&gt;85%</td>
</tr>
<tr>
<td>Benchmarked Codes</td>
<td>√</td>
<td>√</td>
<td>√</td>
</tr>
<tr>
<td>Current Independent</td>
<td>√</td>
<td>√</td>
<td>√</td>
</tr>
<tr>
<td>Equipartitioning</td>
<td>-</td>
<td>√</td>
<td>√</td>
</tr>
<tr>
<td>Resonances</td>
<td>√</td>
<td>√</td>
<td>√</td>
</tr>
<tr>
<td>RAM Modelling</td>
<td>√</td>
<td>post design</td>
<td>-</td>
</tr>
<tr>
<td>Halos / Beam Loss</td>
<td>√</td>
<td>√</td>
<td>√</td>
</tr>
<tr>
<td>Errors</td>
<td>√</td>
<td>√</td>
<td>√</td>
</tr>
</tbody>
</table>

Summary and conclusions

Much has been learnt since 1995 when the rapid transition to using RF superconducting accelerator technology started worldwide. In addition to numerous significant technology advances, much has been learnt about linac beam dynamics, including the control of emittance and beam halo in high-intensity linacs. These advances have led to a relatively uniform approach worldwide for the design of high-power proton linacs for spallation sources and ADS applications. As a result, most modern high-power proton linac designs are now on a firm design basis and exhibit similar design features including inherent fault tolerance and low beam losses that allow hands-on maintenance and maintainability. In order to improve future linac performance and to meet the challenging operational goals for ADS, in particular for energy production, additional progress will need to be made in advanced real-time tuning and optimisation methods.

Acknowledgements

The author wishes to acknowledge the past members of the Los Alamos APT, ATW, ADS and SNS project teams who contributed significantly to the concepts presented in this paper.
References


Design of the spoke cavities and its associated cryomodule for the MYRRHA superconducting LINAC

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Abstract

The MYRRHA Project aims at the construction of an accelerator-driven system (ADS) at MOL (Belgium) for irradiations and transmutations experiments purposes. The facility will feature a superconducting LINAC able to produce a proton flux of 2.4 MW (600 MeV – 4 mA). The first section of the superconducting linac will be composed of 352 MHz (beta = 0.37) spoke type superconducting cavities housed in short two cavities cryomodules operating at 2K.

A first R&D phase on the Spoke cavities design and performances has been initiated in 2015. The first results on the performances of two prototypes designed for the MHYRRA LINAC, tested at 2K are presented.

The cryomodule has for main design constrains the availability and the reliability required by the operation in a nuclear plant environment. To fulfil these requirements several solutions have been taken into account and especially the so-called “Fault-tolerance” scheme that will be used to operate the proton beam. The second part of the paper describes the whole cryomodule design (spoke cavities, Cryostat, Power Coupler, Cold Tuning System...) by mainly focusing on the conceptual choices driven by the reliability and fault tolerance scheme requirements.

Introduction

At the beginning of 2016 the SCK•CEN decided to initiate the MYRRHA Phase 1 construction, which aims, concerning the accelerator part of MYRRHA, to the construction of a first LINAC section of 100 MeV and 2.5 mA continuous proton beam for 2023. A first prototyping period is starting on the basis of the previous Euratom R&D programmes as the MAX and MYRTE projects. The specificity of the MYRRHA LINAC comes from the strong requirements on reliability and availability imposed by the operation with a nuclear reactor. The number of beam interruptions longer than three seconds have to be minimised as much as possible in order to avoid thermal stresses on the reactor structures, the target and the fuel elements. The design of the accelerator features several ways to fulfil this constrain. The operation of the beam using the so-called “fault tolerance” scheme aims to maintain the beam in case of failure of some components on the LINAC, roughly, by increasing the power of adjacent cavities. This leads to a modular
design of the LINAC’s elements as well as taking great margins for the design. Efforts have also to be made on prototyping and experimental studies.

From the MAX Project a detailed design of a complete Spoke cryomodule, the accelerating component of the superconducting section from around 17 MeV to 100 MeV has been achieved. It has led to the manufacturing, with CNRS/IN2P3 and SCK/CEN common funding, of two superconducting spoke cavities (beta=0.37 at 352 MHz). These cavities are currently being tested under the MYRTE R&D programme. We present below the first cryogenic tests of these two prototypes as well as the cryomodule design and its associated components; Power Coupler, cold Tuning System. At least a description of the main tasks for the Spoke cryomodule prototyping and tests period for MYRRHA Phase 1, will be briefly presented.

Spoke cavities

The first superconducting section of the LINAC will be constituted of short two single bar spoke type superconducting RF cavities cryomodules operated at 2K. The cavity (Figure 1) is designed for an accelerating mode frequency of 352 MHz and an optimal beta of 0.37. Other design parameters are shown on Table 1.

**Table 1: MYRRHA spoke cavities main design parameters**

<table>
<thead>
<tr>
<th>Cavity Parameters</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>$V_{0,T}$ at 1 joule and optimal beta</td>
<td>0.693 MV/m</td>
</tr>
<tr>
<td>$E_{pk}/E_{acc}$</td>
<td>4.29</td>
</tr>
<tr>
<td>$B_{pk}/E_{acc}$</td>
<td>7.32 mT/MV/m</td>
</tr>
<tr>
<td>$Q_0@2K$ for $R_{res} = 20 \Omega$</td>
<td>$5.2 \times 10^9$</td>
</tr>
<tr>
<td>MAX accelerating gradient in nominal operation</td>
<td>$E_{acc,\text{max, nom.}} = 6.2$ MV/m</td>
</tr>
<tr>
<td>MAX accelerating gradient in ‘Fault Tolerance’ operation</td>
<td>$E_{acc,\text{max, fault}} = 8.2$ MV/m</td>
</tr>
<tr>
<td>RF losses @2K in nominal operation mode</td>
<td>$P_{loss,\text{nom.}} = 8$ W</td>
</tr>
<tr>
<td>RF losses @2K in “fault tolerance” operation mode</td>
<td>$P_{loss,\text{fault}} = 16$ W</td>
</tr>
</tbody>
</table>

The RF optimisation and the $E_{pk}/E_{acc}$ and $B_{pk}/E_{acc}$ ratios, representing respectively the field emission and the power dissipation versus the accelerating gradient, are typical for simple bar Spoke cavities. The design was strongly orientated to simplify the cavity shape and avoid manufacturing risks and cavity preparation (etching and clean room preparation) uncertainties.

The mechanical design of the cavity itself, made from pure niobium (RRR> 200) and its helium vessel made from pure titanium, was done in parallel with the cold tuning system design. The aim was to reach a high cavity mechanical stiffness (reduction of the external mechanical perturbations effects on the cavity fundamental mode frequency) while keeping a good cavity frequency adjustment with the cold tuning system. The goal is to limit the operation time of the slow and the fast tuning system to reach a high tuning system reliability.
Two cavities were manufactured and tested from the end of 2015. The measured curve of $Q_0$ versus $E_{acc}$ (Figure 2) shows very good performances, for the maximum accelerating gradient and for the $Q_0$ value (dissipated power on the cavity walls).

The high measured margin, far above the designed goal, may have some impact on some final design choices especially concerning the very-high $Q_0$ measured value that would, if this is confirmed on other prototypes, reduce the required cryogenic power for the LINAC cryoplant.

Concerning the mechanical behaviour of these two prototypes some discrepancies were measured between the simulated and the measured values. The sensitivity to mechanical loads (bath pressure variation, Lorentz forces), leading to instabilities on the RF fundamental mode frequency, is higher than expected mainly due to a lack of mechanical stiffness of the cavity and its helium vessel.

**Spoke cryomodule**

The cryomodule, designed in the framework of the MAX R&D programme, is a short two cavities cryomodule operating at 2K. Short cryomodules scheme for superconducting LINAC, typically leads to an increase of the LINAC length as well as an increase of the cryogenic power consumption and as a consequence to higher investments and operation costs. But in the case of MYRRHA where the reliability
and the availability is the key point for the LINAC design, short cryomodules scheme was chosen as it is fully coherent with the modularity required by the “Fault Tolerance” strategy adopted for the beam control. As an example, for MYRRHA spoke section, if two adjacent cavities, i.e. one full cryomodule and its associated cryogenic filling valve box, is off, whatever the cause of the fault (RF, Cryogenics, Vacuum) the beam can still be operated.

Figure 3: MYRRHA Spoke cryomodule

The cryomodule (see Figure 3), around 2 200 mm length and 1 200 mm diameter, features typical design concepts; only one thermal shield cooled with gaseous Helium at around 60K, several heat interceptions at 60K and 4K/10K for the power coupler external conductor and the cavity supporting arms; 30 Layers of Multilayer Insulation at 60 K and 10 layers at 4/2K.

A double skin passive magnetic shield, made from CryoPerm® or similar material with a high magnetic permeability at low temperatures, surrounds the cavities helium vessel and protects the cavities walls from external magnetic fields (terrestrial and magnetised surroundings parts). During the cavity cool-down process, the shield is precooled with a dedicated cryogenic loop to reach the optimal temperature of the shield material (typically below 60 K) before the cavities walls start being superconducting; at around 9K; and trap the external magnetic flux.

The power coupler is of SNS type with only one cold window. It was designed for a maximal power of 20 kW in CW mode, knowing that the maximum RF power required for the spoke cavity is 16 kW in fault tolerance mode and the nominal RF Power is 8 kW.

The cold tuning system design is mainly based on the ESS cold tuning system manufactured for the double bar spoke cavities of the ESS project LINAC. It ensures, with a lever arms mechanical system driven by a stepping motor, a slow tuning with a range of about 170 kHz and an effective resolution of around 0.2 Hz for a cavity bandwidth, in operation, of around 150 Hz. Piezoelectrics actuators are integrated in the system to provide fast tuning to compensate fast perturbations with small amplitudes (Lorentz forces, microphonic) in a range of around 500 Hz with a time response of around 1 kHz. A specific requirement for the MYRRHA cold tuning system, imposed by the fault tolerance scheme, is to completely detune and
retune the cavity on a range of 10 kHz, in around 1 second. It would allow to have the fault cavity transparent for the beam and perform the fault recovery process (switch-off fault components, adjust adjacent components and beam parameters) in less than 3 seconds. Preliminary studies show that this could be obtained by over dimensioning the stepping motor without modifying the global mechanical design of the tuning system.

Figure 4: MYRRHA Spoke cryomodule assembly

The conceptual cryomodule global design leads to have the cavity train (2 cavities, connecting tubes, power couplers and extremities valves) put on a table cooled at 60K. The train is maintained on the so-called cavity supporting arms and maintained with a clamping system using needle roller bearings and springs, similar to the design concept used for the XFEL cryomodule. This enables to adjust the cavity train axis at room temperature after its assembly inside the clean room (ISO class 4) and keep the position once the cavities are cooled at 2K. The final cryomodule axis alignment would then be performed by adjusting, if necessary, the supporting table posts or the whole cryomodule. This design choice was made to simplify the cryomodule assembly (see Figure 4) in order to reach reproducibility during the cryomodules manufacturing.

Future work

The definition of the budget and the schedule of the Spoke cryomodule for the MYRRHA Phase 1 R&D period has started. This first period has to lead to a complete Spoke Cryomodule test without beam but close to the nominal LINAC operations; at 2K, with full RF Power (20 kW in CW mode), with low level RF control system and Fault Tolerance scheme experiments. The first definition of the tasks has been done and the working packages shared, for the moment, between several labs (SCK\textcdot CEN, CNRS/IPNO, CNRS/LPSC, CNRS/LAL) and external collaborating firms (ACS, ADEX). The main cryomodule components and their assembly (vacuum vessel, thermal shield, cryogenic tubing, supporting table) call for tender is about to start. Meanwhile other components and topics requiring to be finalised as the power couplers, the cold tuning system, the RF amplifiers, the low level RF, the “fault tolerance” scheme implementation, the cryomodule cryogenic valve box, the test bench installation, will be studied and manufactured in parallel with the cryomodule manufacturing which could lead to a first cryogenic test in 2018.
Backward wave linear accelerator of ions (T~300 K) for subcritical nuclear reactors

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Abstract

Backward wave linear accelerator (Linac) of protons (BWLAP) options with energy up to 10 GeV have been researched and designed for accelerator-driven systems (ADS). BWLAP-method is a new technological platform for proton (ion) accelerators on high-performance linear structures at room temperature, with a strong backward spatial harmonic of RF-electro-magnetic field propagating against the emf energy flux. The topologically similar cells of linear structures are implied for all field energy application. The advantages of the BWLAP (T~300 K) over the SC linacs (T~2 K) while considering the value of total energy conversion efficiency \( \eta = \frac{P_{beam}}{P_{AC}} \) are being proved. There would be advisable to build and test the BWLAP industrial prototype as a compact powerful driving force for subcritical nuclear reactors for transmutation of radioactive waste (RAW) and minor actinides (MA), as well as for the production of electrical and heat energy, formation of neutron and other secondary beams. The BWLAP can operate both in a low duty cycle and in continuous modes.

Introduction

At present, it is more important than ever to develop the so-called "Fifth Generation" (Waste Free) of Nuclear Power with an inexhaustible energy supply (electricity and heating) from non-expensive and reliable and unlimited sources (\(^{238}\)U, \(^{232}\)Th), while at the same time utilising accumulated amounts of radioactive waste and therefore eliminating fears associated with the production and consumption of nuclear energy.

Nuclear plants which employ the reactors with self-sustaining chain reactions – the only present technology in the world – arouse wariness among most people including professionals. As a result, the nuclear power engineering industry has been under uncertainty now. New power plants based on this particular technology have not been constructed in the leading nuclear countries since the 1980s. Revival of the nuclear power engineering and expectations of its quick development are related to the hope for radical changes in the domain of modern power engineering initiated by transition of nuclear plants to another technological platform – to subcritical reactors with induced nuclear fission of nuclei – to the accelerator-driven systems (ADS). Unfortunately, in Russia those expectations are hardly associated with development of subcritical reactors, since the main national programme effort has been concentrated on the reactors-breeder, i.e. on the technology which potentially unacceptable for wide-spread use under the non-proliferation regime, and this way even self-destructible for nuclear power.

In the subcritical reactors powerful beams of high-energy protons can be generated by only linacs in a fully controlled, secure and efficient way. This fact constitutes fundamental difference between the
subcritical reactors and the present-day operating ones, and completely excludes any reactor failure or accident similar to the one in Chernobyl or Fukushima.

Among many papers on the matter, we would like to quote the report by Dr J.P. Revol, published in the Russian Physics-Uspekhi (Advances in Physical Sciences) [1], in which he described the present state of nuclear power engineering, along with a scenario of its transition to the subcritical technology, as follows: 

The real question facing scientists today is: Is it possible to change nuclear energy production in such a way as to make it more acceptable to society? Nuclear energy is a domain that has essentially seen no significant fundamental R&D since the end of the 1950s when the first civil power plants came into operation' and “Progress in particle accelerator technology makes it possible to use a proton accelerator to produce energy and to destroy nuclear waste efficiently. The energy amplifier (EA) proposed by Carlo Rubbia and his group is a subcritical fast neutron system driven by a proton accelerator. The EA could be particularly efficient for destroying, through fission, transuranic elements produced by presently operating nuclear reactors.”

Unfortunately the well-timed and informative Revol’s report, pretending be a thorough summary of the achievements in this research area, draws our attention only to Carlo Rubbia activity, and doesn’t refer to any pre-discoveries made before and after 1990 in the Soviet Union by Dubovsky, Marchuk, Vassylkov, Goldansky, Barashenkov and others. Having made broader estimate with strengthening inevitable versatile application in other areas could have been only amplify this point of view – topicality and urgency of the actual next steps towards the subcritical pre-industrial prototype reactor development. Though all research activity within the domain of subcritical reactors led by Carlo Rubbia created a new wave which may finally give some results, we shouldn’t forget previous research in this area in Russia and other countries since the 1950s. And by this time it has become obvious for everyone that cyclotrons cannot compete with linacs in ADS industrial applications.

But what is of a principle importance for us is that the report also contains a leading and guiding statement, widely spread and shared among the whole global community of accelerator designers, and echoed from one paper to another: ‘In the warm structures of linacs the energy conversion efficiency is small, and small aperture (diameter of the accelerator tube) creates problem with beam loss, which by that reason cannot be localised’ [1].

For instance, in 1994 the authors [2] made it clear (based on the traditional scheme analysis Gun-RFQ-DTL-high-energy section with Zsh=35.4 MOhm/m) that “warm” linacs became absolute and non-competitive with SC linacs.

Such statements have encouraged the majority of linac designers to work only on the development of the superconducting (SC) accelerating structures.

As the result, since the middle of 1990s there have been hardly found any research, serious analysis or publications devoted to the development of super-power accelerators on linear accelerating structures at room temperature (~300K).

Present-day attitude towards the research and papers referring to the accelerators based on the warm structures is described in [4] with a trenchant word non-starter, meaning: not expected to be implemented – “background”, as opposed to the “winning” technology of linacs based on the superconducting structures (SCS) at the temperature T ~ 2K.

As a matter of fact, in all projects over the last 20 years the designers of linacs concentrated their efforts on the opportunities provided by the SCS. For example:

- USA (2006) constructed the most powerful linear proton accelerator in the world – spallation neutron source (SNS with proton energy 1GeV, beam power 1.56 MW, beam length 258 m);
• USA (2008-2018) has undertaken the construction of a multipurpose ion linac (Project X) with energies up to 8 GeV [4] (total beam length 692 m);
• European Community (2010-2018) is developing the project of the neutron complex ESS that will use proton linear accelerator with energy 2.5 GeV and beam power up to 7.5 MW;
• China, India, Japan and South Korea have been financing programmes based on the construction of linear accelerators for fundamental and applied research, determining the future of the nuclear power systems in those countries.

In all of the above mentioned projects, linacs are based upon the only and the same technology – warm RFQ and Alvaretz schemes as the front-end, and SCS for high-energy sections. Billions of dollars have been spent to build them, while construction of the "warm" linear accelerators (for E > 1 GeV) has not been studied nor developed, or even nor considered. The last serious discussion of “warm” structures (known to the authors) took place at EPAC-96 [3].

In this report there is no any intention to defend the well-known linacs based on Standing Wave RFQ and Alvaret’z DTL normal conducting structures. We wish to draw everybody’s attention to the new method of ion acceleration [8-10] applied in the Backward Wave Linear Accelerators of Protons named “Ugly Ducling”, constructed in the 1981 for the first time in the world.

**Backward wave linear accelerators (BWLAP) and their advantages**

Our proposal to return to design studies on the non-superconducting structures (T~300K) is based on the actual advantages of the new ion accelerating method – BWLAP technology – over the SCS technologies (with all our respect to the achievements in SCS area) in respect of:

• efficiency of beam acceleration (electronic efficiency), and overall efficiency;
• total trajectory of accelerating structure (length of AS) is used to accelerate and focus ions;

![Figure 1: Comparison of DTL (Alvarez) and BWLAP schemes](image-url)

• compactness construction;
• flexibility in complying with customer requirements;
• practical absence of beam loss (operating safety);
• accelerator reliability (including maintenance simplicity) during long-term continuous operation (up to 60 years or more);
• technological quality (manufacturability);
• general expenses to accelerate beams;
• overall costs (including operating costs).

more than 50 years of R & D [8-27] in order to determine the optimal shapes and regimes of BWLAP in the energy range up to 10'000 MeV (from protons to multi-charged ions of $^{238}$U) resulted in some options (Table 1), which differ in:

- Layouts (technical and technological solutions) and level of compactness:
  - one-dimensional (linear) – 1D;
  - two-dimensional (twisted in a plane) – 2D – ABC2D;
  - three-dimensional – 3D-ABC3D;

- current, energy and power characteristics of the accelerated beams;

- frequency of the RF power supplies, acceleration rate $\Delta W / \Delta z$, number of the accelerating sections.

The Table 1 shows the results of computer simulation (based on the accumulated Data Base of the 50 years experiments with structures) of two- (Figure 2, Option 7) and single frequency (Figure 3, Option 9) BWLAPs on linear structures with energy 1 000 MeV and current 300 mA. In the 7th option there is only duty cycle of 1/40 mode ($l_{\text{main}} = 7.5$ mA) for the level of heat discharged in high-energy sections. Continuous wave (CW) mode of operation (the 9th option, Table 1) at low frequency (433 MHz) makes the linac reliable, heat removable, but in this case the 1016-MeV BWLAP-accelerator length is 837 m (or 160 m x 60 m in 2D-option). The optional BWLAPs will use MBK of THALES ED – TH2120, 433 MHz – in a single frequency pattern (BWLAP-CW); and TH2120, TH2104U, 1 300 MHz, Russian klystrons 2 600 MHz and "hypothetical" klystrons 3 900 MHz in pulsed (duty cycle ~ 1/40).
Table 1: BWLAP optional parameters

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Option 1</td>
<td>433</td>
<td>0.2–143.3</td>
<td>295.1938</td>
<td>42.24</td>
<td>6x8+4</td>
<td>96</td>
<td>91</td>
<td>39</td>
</tr>
<tr>
<td>Option 2</td>
<td>1300</td>
<td>0.34-175.7</td>
<td>303.2447</td>
<td>53.18</td>
<td>8*8+4</td>
<td>48</td>
<td>82</td>
<td>35</td>
</tr>
<tr>
<td>Option 3</td>
<td>1300</td>
<td>0.2–18.8</td>
<td>306.9249</td>
<td>5.77</td>
<td>2x4</td>
<td>23</td>
<td>86</td>
<td>37</td>
</tr>
<tr>
<td>Σ</td>
<td>1300</td>
<td>18.8–60.5</td>
<td>12.8</td>
<td>5x4</td>
<td>18.6</td>
<td>85</td>
<td>34</td>
<td></td>
</tr>
<tr>
<td>Option 4</td>
<td>1300</td>
<td>0.2–18.8</td>
<td>306.9249</td>
<td>5.77</td>
<td>2x4</td>
<td>23</td>
<td>86</td>
<td>37</td>
</tr>
<tr>
<td>Σ</td>
<td>1300</td>
<td>18.8–134.6</td>
<td>35.54</td>
<td>5x10</td>
<td>29</td>
<td>79</td>
<td>34</td>
<td></td>
</tr>
<tr>
<td>Option 5</td>
<td>1300</td>
<td>30.3–18.8</td>
<td>304.56</td>
<td>41.3</td>
<td>58</td>
<td>52.0</td>
<td>82</td>
<td>35</td>
</tr>
<tr>
<td>Option 6</td>
<td>1300</td>
<td>0.34-175.7</td>
<td>303.18</td>
<td>53.18</td>
<td>8*8+4</td>
<td>48</td>
<td>82</td>
<td>35</td>
</tr>
<tr>
<td>Σ</td>
<td>1300</td>
<td>175.7–1005</td>
<td>251.39</td>
<td>500</td>
<td>112.9</td>
<td>69</td>
<td>30</td>
<td></td>
</tr>
<tr>
<td>Option 7</td>
<td>1300</td>
<td>0.35-193.9</td>
<td>307.177</td>
<td>59.56</td>
<td>8x10</td>
<td>46.7</td>
<td>82</td>
<td>35</td>
</tr>
<tr>
<td>Σ</td>
<td>1300</td>
<td>193.9-273.6</td>
<td>24.48</td>
<td>2x10+5</td>
<td>8.8</td>
<td>71</td>
<td>31</td>
<td></td>
</tr>
<tr>
<td>Option 8</td>
<td>1300</td>
<td>0.35-93.45</td>
<td>310.2032</td>
<td>28.9</td>
<td>4x10</td>
<td>24</td>
<td>82</td>
<td>35</td>
</tr>
<tr>
<td>Σ</td>
<td>1300</td>
<td>93.4-633</td>
<td>167.6</td>
<td>26x10</td>
<td>74.5</td>
<td>71</td>
<td>31</td>
<td></td>
</tr>
<tr>
<td>Option 9</td>
<td>433</td>
<td>0.2–1016</td>
<td>311.77</td>
<td>316.7</td>
<td>42x8+4</td>
<td>837</td>
<td>90.7</td>
<td>38</td>
</tr>
<tr>
<td>(CW) Σ</td>
<td>433</td>
<td>1016</td>
<td>340</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Notes: Option 7 heat removal \( (F=40) \Delta P_h/\text{section}<143\ kW \)
Option 9 heat removal \( (CW) \Delta P_h/\text{section}<150\ kW \)

Figure 2: Two-frequency layout (1300-3900 MHz) of BWLAP-1.0 GeV

Linac – BWLAP - 1026 MeV

Sections S1- S15 1300 MHz
Sections S16- S23 3900 MHz 10 MW
Sections S24- S52 3900 MHz 30 MW
Energy W=1026 MeV
Current Iimp = 307 mA
Duty factor 2.5%
Electronic efficiency η = 58%
Total active length L = 92 m
Accelerator floor (49 m x 3 m)
Max. heat removal Ph/dz < 106 kW/m

Figure 3: Single-frequency layout (433 MHz) of BWLAP-CW-1.0 GeV

BWLAP-1016 MeV
(Pulsed or CW mode at 300K)

Sections S1- S180 433 MHz (CW-mode)
Energy E=1016 MeV
Current Iimp = 307 mA
Electronic efficiency η = 92%
Active length L = 830 m (60m x 160m)
Max. heat removal Ph/dz < 43 kW/m (CW)
In order to use the maximum supplied power, the RF power scheme applied active and virtual [19,20] klystrons. The every second section is powered by 3-dB adder or by klystrons alternately (Figure 4).

Every second section of the BWLAP is powered by a 3-dB adder, which total RF power is accumulated from fourth previous sections and next 4 MW RF supply. Each adder acts like a virtual klystron.

**Figure 4:** Scheme of the RF power supply for the backward running RF-wave in accelerator sections

**Figure 5:** Bunch position in the separatrix (for a 10^7 particle simulation of the BWLAP beam)

The particles capture and formation of protons in a clearly defined bunch is being completed in the first section (at W <4 MeV and \( n_{\text{capt}} > 90\% \)). Application of a longitudinal axial-symmetric magnetic field Bz for focusing in BWLAP has solved the problem of particles loss in the accelerating structures (production and significance – [30, 31], and decision – [24-26]).

Formed in an increasing Ez (z)-field the bunch (with its "coat") is located in the centre of the separatrix, providing a lossless particle beam in the further acceleration (Figure 5).

Once again emphasise: with accuracy up to 1 / 44mln-share of the value of the accelerated current (sensitivity to losses – the particle loss corresponds to 120 protons) – **there is no losses of accelerated protons in all BWLAP sections** (exception – the first section), and in all bending magnets along the line of the beam acceleration [18, 24-26] wherein the diameter of the beam is not exceeding 3 mm (Figure 6).

When designing the accelerator for subcritical reactor (BWLAP-CW-mode), much attention was paid to the high-frequency heat load (Figure 7) in the structures and possibility of removing the heat by flow of water [20, 23, 26, 27]. The chosen scheme of 433 MHz RF power and length of the accelerating structures let using all the 100% (4 MW) RF power (without loss) in two structures in series (Figure 4).
Undertaken research showed the possibility of construction of a continuous mode accelerator (BWLAP-CW) with proton energy 1.000 MeV and beam power up to 300 MW, where all accelerating structures (180 structures) are cooled down with water. With the existing RF-devices-klystrons (d.f. 12.5%), it is possible in such a scenario to obtain a proton beam with the power of 36 MW.

The main parameters of BWLAP-CW-1 GeV are shown in Table 2.

Table 2: Main parameters of BWLAP-CW
180 Sections, W=1016.015 MeV, I_{main}=306.9249 mA, L=837.5 m, 48 turns

<table>
<thead>
<tr>
<th>f MHz</th>
<th>Energy MeV</th>
<th>Power ΔP_{beam}, MW</th>
<th>RF-power (stated), MW</th>
<th>Length m</th>
<th>η_{e}=P_{e}/P_{RFused} %</th>
<th>η_{Σ}=P_{e}/P_{wall} %</th>
</tr>
</thead>
<tbody>
<tr>
<td>433</td>
<td>0.2–1016</td>
<td>311</td>
<td>46*8 + 4=372</td>
<td>837.5</td>
<td>91.8</td>
<td>39</td>
</tr>
</tbody>
</table>

BWLAP-1.0 GeV-short, which accelerates protons up to energies of 1 GeV, is the most compact one with active beam length 92 m (Table 3 and Figure 2), having the footprint of 50 m to 6 m.

Table 3: Main parameters of BWLAP-1.0 GeV-short
180 Sections, W=1026.19 MeV, I_{imp}=306.925 mA, L=92.1 m, 4 turns

<table>
<thead>
<tr>
<th>f MHz</th>
<th>Energy MeV</th>
<th>Power ΔP_{beam}, MW</th>
<th>RF-power (stated), MW</th>
<th>Length m</th>
<th>η_{e}=P_{e}/P_{RF} %</th>
<th>η_{Σ}=P_{e}/P_{wall} %</th>
</tr>
</thead>
<tbody>
<tr>
<td>1300</td>
<td>0.35–193.9</td>
<td>59.51</td>
<td>4*8</td>
<td>46.7</td>
<td>82</td>
<td>35</td>
</tr>
<tr>
<td>3900</td>
<td>193.9–273.6</td>
<td>24.46</td>
<td>3x10</td>
<td>8.8</td>
<td>71</td>
<td>31</td>
</tr>
<tr>
<td>3900</td>
<td>273.6–1016</td>
<td>231.1</td>
<td>15x30+15</td>
<td>36.6</td>
<td>58</td>
<td>23</td>
</tr>
<tr>
<td>Σ</td>
<td>1026.59</td>
<td>315.1</td>
<td>527</td>
<td>92.1</td>
<td>61</td>
<td>25</td>
</tr>
</tbody>
</table>
Energy conversion efficiency ($\eta$) of 1000 MeV accelerating structures, equal to the ratio of the beam power increase $\Delta P_{\text{beam}}$ to the total energy $\Delta P_{\text{AC}}$ used in order to obtain this $\Delta P_{\text{beam}}$, can reach $\eta = \Delta P_{\text{beam}}/\Delta P_{\text{AC}} \approx 40\%$ in BWLAP-CW. And energy conversion efficiency for the BWLAP-1.0 GeV-short is $\eta = \Delta P_{\text{beam}}/\Delta P_{\text{AC}} > 25\%$. Such a BWLAP, with proton energy equal to the one achieved in the most powerful modern SNS accelerator, will be 3.5 times shorter and will have 5 times higher the main beam power (7.5 MW), not to say of the 300 MW beam power in 1 ms pulse.

Electronic energy conversion efficiency – ratio of the accelerated beam $P_b$ power to RF-expenditure $P_{RF}$ – is given below (1). Based on this formula, it is easy to understand the efforts of the accelerator designers to increase $Z_{sh}$ (which is one of the most important RF-characteristics of accelerating structures) and $I_{\text{imp}}$. The “price” for the high gradient of acceleration (high $E$) is obvious also now: efficiency $\eta_e$ decreases while the $E$ increases.

$$\eta_e = \frac{P_b}{P_{RF}} \approx \frac{I_{\text{imp}}}{I_{\text{imp}} + \frac{E}{Z_{sh}}}$$ (1)

Figure 9 shows the efficiency of different accelerating structures (their shunt impedance $Z_{sh}(E)$ [MOhm/m]) for different values of energy of the accelerated protons [28,29].

(Figure 9 is taken from the report “Preliminary evaluation of the new structures for high-frequency linear accelerators, developed in the former Soviet Union”, prepared by the team of the US Army Space and Strategic Defense Command, February 7, 1994).
Superconducting 8 GeV project X and 1 GeV SNS versus BWLAP with water cooling

Here is a brief comparison of the most powerful SCS and Project X accelerators with NCS – BWLAP. Table 4 shows the main parameters of the superconducting accelerator SNS (Oak Ridge, USA, commissioned in 2007) and 8 GeV SC-Main Injector of Tevatron (FermiLab, Batavia, USA, planned to start operation by 2018/2020), in comparison with two different optional BWLAs with similar – calculated – parameters at room temperature (T~300 K) with water cooling structures.

The BWLAP data showed in Figure 5 is based on the experimental studies of over hundreds of models of accelerating structures in two-frequency ranges: 1 1818 and 915 MHz. There are basis for the BWLAs’ (433 and 1300 MHz) simulations and for comparison with the existing accelerators (SNS) and with project under development (Project X) on linear superconducting structures.

We emphasise that the superconducting accelerators have a cryogenic “plant” with the length comparable with the length of the accelerator itself. The “plant” maintains the required structure temperature 2K, but the removal of 1 W of the RF-heat power from the structure needs at least 1.2 kW of electric power [33].
Table 4: Comparison of the BWLAP with SNS and Project X accelerators

<table>
<thead>
<tr>
<th>Linac Parameters</th>
<th>SNS- Spallation Neutron Source</th>
<th>BWLAP/ ABC2D</th>
<th>8 GeV Injector</th>
<th>BWLAP/ ABC3D</th>
</tr>
</thead>
<tbody>
<tr>
<td>Accelerator energy</td>
<td>1 GeV</td>
<td>1 GeV</td>
<td>8 GeV</td>
<td>10 GeV</td>
</tr>
<tr>
<td>Type of particles</td>
<td>H</td>
<td>p⁺</td>
<td>H⁺, p⁺ or e⁺</td>
<td>p⁺</td>
</tr>
<tr>
<td>Beam power</td>
<td>1.56 MW</td>
<td>3 MW / 36 MW</td>
<td>2 MW</td>
<td>3 / 72 MW</td>
</tr>
<tr>
<td>Supply power</td>
<td>~15 MW</td>
<td>8 MW / 114 MW</td>
<td>12 MW</td>
<td>9 / 235 MW</td>
</tr>
<tr>
<td>Duration of beam pulses</td>
<td>1 msec</td>
<td>1 ms / 10 ms</td>
<td>1 msec</td>
<td>1 / 10 msec</td>
</tr>
<tr>
<td>Pulse current</td>
<td>26 mA</td>
<td>305 mA</td>
<td>26 mA</td>
<td>310 mA</td>
</tr>
<tr>
<td>Pulse frequency</td>
<td>60 Hz</td>
<td>12.5 Hz</td>
<td>0.6 – 10 Hz</td>
<td>0.1 / 2.5 / 12.5 Hz</td>
</tr>
<tr>
<td>Number of SC AS</td>
<td>81</td>
<td>-</td>
<td>384</td>
<td>-</td>
</tr>
<tr>
<td>Number of CryoModule/SCSolenoid</td>
<td>23 CM</td>
<td>92 SCS</td>
<td>48 CM</td>
<td>411 SCS</td>
</tr>
<tr>
<td>Number of klystrons</td>
<td>93</td>
<td>7 (1300)+75 (3900)</td>
<td>41</td>
<td>41(1300)+206(3900)</td>
</tr>
<tr>
<td>Number of Sections per klystron</td>
<td>1</td>
<td>2section / 1klystron</td>
<td>8 – 12</td>
<td>2section / 1klystron</td>
</tr>
<tr>
<td>Maximum surface field E</td>
<td>35 MV/m</td>
<td>32MV/m</td>
<td>45 MV/m</td>
<td>48 MV/m</td>
</tr>
<tr>
<td>Maximum accelerating gradient</td>
<td>16 MV/m</td>
<td>6 (1300)+18(3900)</td>
<td>22.5 MV/m</td>
<td>32 (3900) MV/m</td>
</tr>
<tr>
<td>Accelerator length (active)</td>
<td>258 m</td>
<td>120 m</td>
<td>692 m</td>
<td>578 m (60 m x 15m x 8m)</td>
</tr>
</tbody>
</table>

The total cost of electricity (total “wall power” PAC) needed to accelerate 2 MW proton beam in 8-GeV superconducting linear accelerator with 26 mA current and 1 millisecond current pulse (duration of RF-impulse 1.2 ms) is given by:

\[ P_{AC} = 2MW_{stdby} + (1MW / Hz * 10Hz) = 12MW \]

In the case of the beam power 36 MW, the frequency of the same duration pulses is equal to 180 Hz and \( P_{AC} \) increases to > 180 MW (if neglecting the power increase in low-energy RFQ+DTL resonators, used in non-superconducting part of the 8 GeV accelerator). More realistic estimation of the total loss leads to the conclusions that it could exceed 240 MW.

Let us emphasise: weak points of the complexes with superconducting accelerators are breakdowns in the accelerator cryogenic profiles and in the cryogenic plant.

Running of the superconducting accelerating systems also requires a significant number of staff and equally huge amounts of resources for its utilisation, as well as the spare time to undertake routine maintenance work and to repair potential malfunctions.

The construction of both SC structures and cryogenic plants constitutes significantly more than a half of the total costs of accelerating complexes based on superconducting accelerators.

In the NC-structures the heat is removed with the running water, which is quite a non-expensive solution.

The authors are convinced that the claims about the “low-energy conversion efficiency in the warm linear accelerators” were based on the much earlier studies of accelerator characteristics in comparison
with the SCS ones, and could refer to the electronic energy conversion efficiency of the Alvarets structures type (standing wave) of warm NC-accelerators only, especial when low current is accelerated.

Energy conversion efficiency achieved in the BWLAP-CW (conversion of the RF-filed energy into kinetic energy of the accelerated beam) reaches $\eta_e=\Delta P_b/\Delta P_{RF} \sim 90\%$ at room-temperature structures.

**Conclusions**

The research results leads to the following conclusions.

Characteristics of the BWLAP’s method and identified innovations in BWLAP-construction allow us to recommend the “warm” backward wave linear accelerators as serious competitors to the accelerators with all the superconducting structures in applications for the industrial programmes.

ADS complexes, if executed in the scheme of a backward wave accelerator with a system of water heat removal at room temperature, provide accelerated proton beams with necessary power and proton energy up to 10 GeV. In consequence, they will have huge advantages over the superconducting accelerators by virtue of:

1. Multiple benefits regarding the total energy conversion efficiency, beam power and pulse current.
2. Wide variety of functionality-specification of BWLAP design-of-choice, depending on application
3. BWLAP layout, providing seismic stability of the complex as a whole, and allowing to avoid “over-complication of the SCS linear accelerator’ caused by the necessity of considering in its construction the curvature of the Earth [4] and the Earth’s magnetic field (BWLAP – 10 GeV, beam 10 MW, set on the area 60 m x 24 m x 15 m).
4. Significantly higher technical and operational reliability, and maintainability.
5. Significantly lower (three to five times) costs of construction.
6. Potentially negligible activation both of the accelerator and of the whole ADS complex as a result of the absence of particle loss.
7. Significantly lower operating costs.
8. Future potential of BWLAP-1GeV: possibility to increase the beam power up to 300 MW.

Satisfactory results for the total energy conversion efficiency in the BWLAP were obtained due to:

- Method of backward wave acceleration (increase of $E_z (z)$ resulted in accelerating non-relativistic ions on the travelling – backward – wave).
- Achieved value of the accelerating high pulse current (> 300 mA).
- Development of a new RF power system for the accelerated structures in the travelling wave mode, which guarantees full usage of the installed RF power.
- Use of the external (in relation to the accelerating structures) focusing system with superconducting solenoids.
- Advantages from development and application of the accelerating structures with magnetic coupling between cavities-cells (structures with anomalous dispersion), due to high value of their shunt impedance (more than 100 MOm/m).
Acknowledgements

The authors are grateful to all the staff of the Division of accelerator technology and the leadership of the Institute of Chemical Kinetics and Combustion, Siberian Branch of the Academy of Sciences USSR and the leadership of the Academy of Sciences for many years of support and participate in the developing the new direction in the theory and practice of design and application of linear accelerators based on the original method of ion acceleration on the travelling backward wave. One of the authors (Bogomolov) is sincerely grateful to the staff of the 14th Department of the MEPhI, instilled in him the great love to accelerator technology (since 1962), and for their continued co-operation in resolving numerous issues for the past half a century. The authors are particularly grateful to General Designer of the Central Research Institute “Kometa” academician A.I. Savin for his continued support — in science, implementation of plans and in the financing of works.

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Session 3: Neutron sources

Chair: J. Knebel
Karlsruhe Institute of Technology, Germany
Progress and status of the ESS target

Eric J. Pitcher
European Spallation Source ERIC

Abstract

The European Spallation Source (ESS) project has the mission to design, build and commission the most powerful spallation source in the world for neutron scattering research. It is currently under construction in southern Sweden, with the aim to deliver first protons to target by the end of 2019.

Within the ESS is the target station, where neutrons are created via nuclear spallation using protons delivered by a linear accelerator. The powerful 5-MW proton beam produces nearly $10^{18}$ neutrons per second, some of which are slowed to speeds that are useful for conducting neutron scattering experiments, and then delivered to instruments. The target station employs several innovative features, including a rotating tungsten target cooled by helium gas, a flat “butterfly” moderator, and beam expansion through raster scanning.

The target station is being designed and manufactured within a European partnership that currently involves six scientific institutions in addition to the central team in Lund, Sweden. Most systems comprising the target station have entered final design and a small number have already entered the procurement phase. This paper summarises the target station design with a focus on some of its unique aspects.

Introduction

The European Spallation Source project has the mission to design, construct, and commission the most powerful spallation neutron source in the world to provide researchers with unprecedented neutron intensities for probing the structure of matter. The project is funded by a consortium of European countries, with Sweden and Denmark serving as the host countries. The project is on schedule to deliver first neutrons in 2019.

At the heart of ESS is the target station, which uses 2-GeV protons delivered by a linear accelerator to liberate neutrons from tungsten nuclei via nuclear spallation reactions. A small fraction of these liberated neutrons are slowed to speeds useful for studying the structural properties of matter using water and liquid hydrogen as moderating media. Of those neutrons that are slowed to useful speeds, a fraction leak from the moderators into the neutron guides that deliver them to neutron scattering instruments. This process defines the high-level functions of the target station:

- generate neutrons via the spallation process;
- slow the neutrons to speeds useful for neutron scattering;
- direct neutrons to neutron scattering instruments;
- operate safely with high reliability and availability.
About two-thirds of the target station project scope has been undertaken by partnering institutions throughout Europe. While this arrangement presents some challenges in terms of co-ordination and interface management, the project benefits greatly from the expertise brought to bear by these institutions on the design and manufacturing of some rather unique components.

Several unique features of the ESS target station distinguish it from its predecessors. These include a rotating spallation target, helium as the target primary coolant, flat moderators, proton beam expansion using raster magnets, neutron beam ports that allow viewing of either the upper or lower moderator, and a doubling of the number of neutron beam ports over the conventional number, among other innovations. Here we describe the salient features of the target station, which include the spallation target, the moderator-reflector system including neutron beam extraction, remote handling systems and the safety approach and features.

**Target station layout**

The target station occupies the target building, a cutaway view of which is shown in Figure 1. The target building is a large structure, about 130 m in length and 22 m in width. In Figure 1, the proton beam enters the Target Building from the left, passing through the proton beam transport hall and into the target monolith. The monolith, shown in Figure 2, is a large structure (11 m diameter by 6 m tall) consisting mostly of 3 000 tonnes of steel shielding. Embedded within the monolith is a 6-m-diameter vessel containing the major components involved in neutron production and delivery to instruments: the spallation target, the moderator-reflector system, the proton beam window and instrumentation. Some fraction of the proton beam’s power goes into liberating neutrons from tungsten nuclei\(^1\), but most of the beam’s 5 MW of power is deposited as heat in the target and surrounding structures. This heat is removed by a number of cooling systems whose principal components (pumps, heat exchangers, filters, etc.) are located in utility rooms downstream of the monolith. The monolith is the 11-m-radius, 8-m-tall steel and concrete structure surrounding the target-moderator-reflector system, which provides bulk shielding of the highly penetrating high-energy neutrons created by the spallation process. The target building also contains active cells, where the highly activated components from the monolith are moved via the high bay once they reach the ends of their service lives. There, components are stored for a short period (several months to several tens of months) while they decay, and are then dismantled (if needed) and packaged for transport to an approved off-site disposal facility.

**Figure 1: Cutaway view of the target building**

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1. Unlike fission, spallation is an endothermic nuclear reaction that converts kinetic energy into mass. Of the 5 MW of beam power incident on target, about 89% gets deposited as sensible heat, 9% is converted to mass, and about 2% is carried off by neutrinos.
Target

The ESS target design is inspired by the rotating tungsten target of the German SNQ project from the 1980s [1]. It features a 2.6-m-diameter wheel comprised of 36 sectors. The rotational speed is set such that each proton pulse strikes neighbouring sectors in succession. As the proton pulse repetition rate is 14 Hz, the wheel rotational speed is \((14 \text{ Hz})/36 = 0.39 \text{ Hz}\). The proton beam footprint on target is produced using raster magnets that scan a Gaussian-shaped beamlet over a 16-cm-wide by 6-cm-tall footprint (see). The resulting nominal time-averaged peak current density on target is 53 µA/cm².

The tungsten spallation material is loaded into the target wheel in the form of some 6 700 bricks, each 8 cm tall by 3 cm deep by 1 cm wide, for a total tungsten mass of 3.1 metric tons. Pure tungsten is used as it has shown better ductility retention than tungsten alloys under proton irradiation [2]. Over the 2.87-ms beam pulse, the rise in temperature due to adiabatic heating of the tungsten material within the peak heat zone is approximately 100°C. At steady state, the peak-heated tungsten then cools by 100°C before being subjected to another beam pulse 2.57 seconds later. The tungsten will be exposed to seven million thermal cycles each year, subjecting it to significant thermal fatigue. Thermal fatigue tests have demonstrated this to not be an issue for unirradiated tungsten, but the effects of irradiation on the tungsten bricks’ thermal fatigue properties will not be known until the first target wheel is removed from service and examined post-irradiation.

The ESS target will be the first medium- or high-power spallation target to employ helium cooling. Helium was selected because it does not react chemically with tungsten, which adds a margin of safety as compared to water coolant. Helium is an excellent choice from an activation standpoint, with essentially no direct activation by protons or neutrons. However, the helium will become activated due to its contact with tungsten. Tritium, produced in the target at an estimated rate of 500 TBq annually, will likely diffuse readily from the tungsten [3] at the expected operating temperature of 200°C to 430°C, and other elements may be ejected directly into the helium coolant via spallation product recoil [4]. In addition, a small amount of dust creation is anticipated, mostly from the stainless steel cassettes that hold the tungsten bricks in place, due to flow- and thermal-expansion-induced vibration of the bricks. Any dust will be removed by full-flow 5-µm mechanical filters placed directly downstream of the target wheel, and also by 0.5-µm mechanical filters in a 0.1% bypass flow to a purification system. In addition to fine dust removal, the purification system will employ getters to remove hydrogen (including tritium), halogens and other trace elements. Thus the helium loop is expected to have a low radioactive inventory outside of that collected and retained by the filters and getters.

The helium at a pressure of 10 bar flows through 2-mm-wide parallel channels between the tungsten bricks, removing 2.7 MW of heat from the tungsten that is deposited by the 5-MW proton beam. An additional 0.3 MW is removed by the helium from the internal stainless steel structures within the target, the stainless steel shroud that contains the high-pressure helium, and the target shaft, for a total heat removal of 3 MW. The helium mass flow rate is 3 kg/s, with an inlet temperature of 40°C and outlet temperature of 240°C.
Moderators

The ESS will be the first spallation source to employ flat, or low-dimensional, cold moderators [5]. This moderator design offers up to three times the cold source brightness as compared to traditional moderators with large viewed surface areas. With proper neutron guide design, for most neutron scattering instruments this factor of three gain in source brightness translates to an equal boost
in neutron flux on sample. The moderators also employ an innovative “butterfly” shape [6] that allows the placement of a water moderator above the peak neutron-production zone within the target, offering bright cold and thermal neutron beams to every beam line. Thermal and cold beams may be extracted simultaneously using bi-spectral guides [7].

The cold moderators use 20-K liquid hydrogen as the moderating medium. The liquid hydrogen flows in a loop and transfers the ~20 kW of nuclear heat load to a 16-K gaseous helium cryoplant. Included in the hydrogen loop is a catalyst that maintains a high para-hydrogen fraction (>99.5%) in the loop, which is needed for optimal neutronic performance.

The beam lines have an average angular spacing of six degrees, about 10% less than the smallest angular opening employed at currently operating spallation sources. In-monolith heavy shutters would be difficult to realise with such small angular spacing. If needed on a particular beam line, a heavy shutter may be placed external to the monolith, as is the practice at many sources today [8]. The small angular spacing allows the incorporation of 42 beam lines, offering options for instrument suite expansion well into the future without having to construct a second target station and dividing the proton beam power between two stations. There are two extraction ports provided at each extraction angle, one for viewing the upper moderator and the other for viewing the lower moderator, which provides flexibility in the placement of neutron scattering instruments. The initial moderator suite consists of a 3-cm-high moderator situated above the target and a 6-cm-high moderator below the target. A moderator-reflector plug is used to insert and extract the moderators into their operating position. They are expected to last at least one year before needing replacement due to radiation damage of the aluminium structures. The plug design offers the flexibility to employ different moderator configurations in the future should new moderator concepts be developed.

**Accelerator-target interface**

The scheme for transporting and expanding the proton beam to the target [9] is similar to that developed for the proposed materials test station project [10]. It consists of a raster magnet system based on the one designed for the accelerator production of tritium project [11], a neutron shield wall placed at the beam waist, and a long drift (~20 m) to the target. The horizontal and vertical 10-rms beam envelopes for the final 47 m of beam transport are shown in Figure 4. The proton beam is brought to a waist at the point where the horizontal and vertical deflections of the beam, induced by the raster system, cross over the beam axis, designated as the crossover position. Here, the beam transverse rms size is only 0.23 mm horizontal by 0.56 mm vertical. For 1 m upstream and 1 m downstream of the crossover, the beam pipe is reduced to 40 mm in diameter and embedded in a 2-m-thick shield wall. The purpose of the small pipe size and thick shield wall is to protect the raster and quadrupole magnets, and other beam line components, from activation by neutrons back streaming from the target. In this way, all accelerator components in this vicinity should be maintainable hands-on.

Downstream of the neutron shield wall and upstream of the monolith, the prompt radiation levels are quite high and the beam pipe and surrounding shielding in this area will become highly activated. All elements here are life-of-the-facility components that should never need to be maintained or replaced.
Safety

Safety is paramount in the design of the target station. Radioactive material is produced as a by-product of the spallation process and by the radiative capture of liberated neutrons in the surrounding structures. An overarching design goal for the target station is assuring that none or almost none of the tungsten is dispersed during postulated accidents, as most of the radioactivity is confined to the tungsten within the target, and the uncontrolled dispersal of this radionuclide inventory represents the highest risk of a significant radiation consequence to workers and the public. If a source of energy were to heat the tungsten to an unsafe temperature, the tungsten may oxidise, whereupon the oxide vapour could readily disperse. The onset for runaway oxidation of tungsten in the presence of steam occurs between 700°C and 800°C [12]. The radionuclide inventory in the tungsten will remain substantially intact so long as the tungsten temperature does not exceed 700°C, or is not exposed to water vapour or air if exceeding this temperature.

There are three sources of energy that can heat the tungsten during an off-normal event: the 5-MW proton beam, the liquid hydrogen in the cold moderators located next to the rotating target, and the residual decay heat within the tungsten. A highly reliable target safety system monitors process variables within the target station systems and prevents the beam from impacting the target should any parameter exceed acceptable limits. In this manner, the effects of any off-normal events that can lead to unsafe beam-induced heating are minimised. These measures, in combination with other safety systems, keep releases of radioactive material from the target station within acceptable levels. A detailed analysis of tungsten heating due to a release of the liquid hydrogen from the cold moderators shows the tungsten remains within a safe range [13]. The radioactive inventory within the ESS target produces 32 kW of decay heat. A prime motivation for employing a rotating target at ESS is the distribution of this decay heat over a much larger volume than would be the case for a stationary target. The 2.6-m diameter target wheel has a large surface area that radiates the decay heat to surrounding structures. Calculations indicate the ESS target design is able to transfer decay heat passively, relying solely on conduction and radiation, with the tungsten remaining in a safe temperature range [14].
Safe operation is assured through a rigorous hazards analysis process where potential radiological consequences of off-normal events are qualitatively assessed. Those postulated events with unacceptable consequences are identified as bounding events, which undergo more detailed quantitative analysis. Structures, systems, components (SSCs), or procedures that are credited with reducing the probability of an event occurring, or mitigate its radiological consequence, are deemed “important to safety.” Quality classes then apply to these SSCs that guarantee these elements meet strict standards in their manufacture, installation and testing for service. In addition, the “defines in depth” principle is applied to assure diversity and redundancy in the systems employed to assure safe operation of the target station.

Summary
With the help of key European partner institutions, most systems of the ESS target station are now in the final design phase. Procurement is under way of some components needed during civil construction of the Target Building. The target station incorporates many unique features that are adapted to high-power, long-pulse operation, such as a helium-cooled rotating tungsten target, a high-brightness flat moderator and beam expansion based on raster scanning. These innovations should produce a well-optimised design that operates safely, reliably and with low operational costs.

Acknowledgements
This paper summarises the work of many individuals across Europe supporting the ESS target station design. ESS-Bilbao is responsible for the target wheel design, Forschungszentrum Jülich for the moderator mechanical design and hydrogen loop, ESS ERIC for the neutronics design, and Centrum výzkumu Řež for the helium primary cooling system. The images in Figures 3 and 4 were created by Heine Thomsen of Aarhus University.

References


Design of 250 kW LBE spallation target for the Japan Proton Accelerator Research Complex (J-PARC)

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Abstract

The management of radioactive waste is one of the critical issues for sustainable nuclear energy application. In the latest strategic energy policy of Japan Express to enhance a research and development (R&D) to reduce the burden of long-lived nuclides in spent nuclear fuel. The Japan Atomic Energy Agency (JAEA) proposes to reduce the environmental impact caused from high-level radioactive waste by using partitioning and transmutation technology. Accelerator-driven system (ADS) is a candidate for effective transmutation of harmful nuclides, namely the minor actinides. To realise ADS, JAEA proposes to build the transmutation experimental facility (TEF) within the framework of J-PARC project. For the JAEA-proposed ADS, lead-bismuth eutectic alloy (LBE) is adopted as a coolant for subcritical core and spallation target. By using TEF in J-PARC, we are planning to solve technical difficulties for LBE utilisation to complete the data for the design of ADS. TEF locates at the end of LINAC, which is also important components to be developed for ADS, and share the proton beam with other experimental facilities in J-PARC. The 250 kW LBE spallation target will be located in TEF facility. The target will be used to prepare material irradiation database by both proton and neutron injection in the temperature range for typical LBE-cooled ADS. Various R&Ds for important technologies required to build the facilities are investigated such as oxygen content control, instruments development, remote handling techniques for target maintenance, and spallation target design. The large-scale LBE loops for 250 kW target mock-up and material corrosion studies are also manufactured and ready for various experiments. The latest status of 250 kW LBE spallation target optimisation were described.

Introduction

The Japan Atomic Energy Agency (JAEA) precedes research and developments (R&Ds) to reduce the radiological hazard of high-level radioactive wastes (HLWs) by Partitioning and Transmutation (P-T) technology [1]. As for the transmutation of hazardous nuclides, the accelerator-driven system (ADS), which combines a high-intensity proton accelerator and a fast subcritical core, is discussed from a viewpoint of transmutation efficiency and compatibility with a power generation cycle. Within the framework of the Japan Proton Accelerator Research Complex (J-PARC) project, JAEA is preparing to construct the Transmutation Experimental Facility (TEF) to study the minor-actinide (MA) transmutation by both MA-loaded fast reactors and ADS [2]. TEF is located at the end of LINAC, which is also important
components to be developed for future ADS, and shares the proton beam with other experimental facilities in J-PARC. A 250 kW spallation target is planned to install in TEF to prepare the irradiation database of various materials. A lead-bismuth eutectic alloy (LBE) was selected as a target material, which is also used as a spallation target and subcritical core coolant of JAEA-proposed ADS. R&Ds for important technologies required to build LBE spallation target are also performed, such as the optimisation of target design, target system design, application of LBE and so on. The latest design concept, and key technologies to establish LBE spallation target were described.

Outline of J-PARC TEF

As shown in Figure 1, TEF consists of two individual buildings: ADS Target Test Facility (TEF-T) [3] and Transmutation Physics Experimental Facility (TEF-P) [4]. Two buildings are connected by a beam transport line with a low power beam extraction mechanism using a laser beam. TEF-T is planned as a material irradiation facility which can accept a maximum 400 MeV-250 kW proton beam on a LBE spallation target. TEF-P is a facility with critical/subcritical assembly to study neutronics and controllability of ADS. Using these two facilities, basic physical properties of subcritical system and engineering tests of spallation target will be performed. R&Ds for several important technologies required to build the facilities are also underway, such as a remote handling method to spallation target vessel exchange, LBE loop components test, control of oxygen concentration in flowing LBE, material tests in non-irradiated LBE flow, and so on.

Figure 1: Transmutation Experimental Facility, TEF in J-PARC

The objective of TEF-T is to obtain the data to evaluate the accurate lifetime of proton beam window for full-scale ADS. TEF-T mainly consists of a spallation target, a cooling circuit and hot cells to handle the spent target and irradiation test pieces. A high-power spallation target, which will be mainly used for material irradiation of candidate materials for a beam window of ADS, is an essential issue to realise TEF-T. To set up the beam parameters, future ADS concepts are taken into account. In the reference case of the TEF-T spallation target, proton beam current density of 20 μA/cm², which equals to the maximum beam current density of JAEA-proposed 800MWth ADS, was assumed. The irradiation performance of the reference target case was evaluated around 8 DPA/year by 5 000 hours irradiation by 400MeV-250 kW proton beam injection. This value is about 20% of DPA considered in the beam window of JAEA-ADS. Further revision of the target design to control the LBE flow around the centre of the beam window to
decrease the thermal peak in window is underway to increase irradiation performance by improving the structural strength.

To evaluate a feasibility of a beam window of TEF-T spallation target, numerical analysis with a three-dimensional analysis model was performed. The analysis was done by considering a current density and a profile of incident proton beam into the target, and the thermal-hydraulic behaviour of LBE around the beam window as a function of flow rate and inlet temperature. The thickness of the beam window is also considered parametrically. After the neutronics and the thermal-hydraulic analyses, structural strength of the beam window is calculated to estimate a soundness of the beam window. As a reference case, a concave shape beam window was selected. The prototype design of the beam window for TEF target system was illustrated in Figure 2.

Figure 2: Reference design of LBE spallation target head for TEF-T

The trolley type target system is adapted to TEF-T LBE spallation target. The existing J-PARC spallation neutron source, 1 MW mercury target, is also mounted on horizontal trolley. The TEF-T target system is designed based on the mercury target system. Figure 3 shows a TEF-T target system with target trolley. To improve the gas tightness, several pillow-seal type joints are substituted to metal O-ring joints. In TEF-T, vacuum vessel is selected instead of safety hull and helium vessel, to catch irradiated LBE in the case of LBE leakage accident around target head. Inside the vacuum vessel is kept in deepest negative pressure in TEF-T building and then, acts as a barrier to confine radioactive reaction products in LBE. For effective cooling of the target system, pressurised water is selected as a secondary coolant. To prevent the accidents by contacting high-temperature LBE and water, double-annular tube is used for the heat transfer pipe for heat exchanger. Helium gas is selected as a filler gas between LBE and water. Basic design is underway and trial production of double-annular heat transfer pipe will be performed.
Basic R&Ds to realise LBE spallation target system

To operate LBE loop with high-temperature condition above 400°C, the mock-up of the primary circuit is manufactured and start operation since March, 2015. The mock-up loop named IMMORTAL (Integrated Multipurpose MOckup for TEF-T Real-scale TARget Loop), which illustrates in Figure 4, is aimed at developing the technologies for safe operation and reliable maintenance. Remote handling techniques for maintenance of irradiated LBE loop is also under development including replacement of spent target and other loop components with preheating systems. In order to operate the target in high temperature, it is difficult to use flanges to replace the spallation target, especially with the remote handling systems. The target replacement scenario by using pipe-cutter and automatic welding machine is under discussion and will be tested using IMMORTAL. After the establishment of TEF-T, IMMORTAL will be used to prepare the non-irradiated samples to clarify the effect of proton irradiation by comparing TEF-T irradiated samples, because all other operation parameters (temperature difference, LBE flow rate, oxygen concentration, etc.) can be adjusted with TEF-T irradiated samples.

Establishment of measuring techniques for LBE loop system is one of the key issues of the safe operation of LBE loop. The measurement and operation devices of the oxygen concentration in LBE are developed to suppress the corrosion by LBE. Two kinds of oxygen sensors, platinum electrode type and bismuth electrode type were tested and then, platinum electrode type was selected because of the higher output signals compared to the bismuth electrode type. In the case of platinum electrode type, open hole should exist to supply the air into the sensor. To prevent the leakage of irradiated LBE through the air hole, freeze seal design was installed in the sensor housing. Leakage test of the sensor housing was performed and confirmed that there are no LBE leakage even in the case of maximum design pressure of LBE cover gas. Figure 5 shows the platinum type oxygen sensor fabricated by JAEA.

Flow rate of LBE is another important parameter for both irradiation experiment and loop operation. However, the electro-magnetic flow metre, which is one of the most popular measuring methods, requires a complicated calibration procedure by lifting/draining LBE. Moreover, the output signal from electro-magnetic flow meter is highly affected by the wettability of the LBE to the electrode. By applying sodium-cooled fast-breeder reactor technology, we developed the ultrasonic flow meter to measure the flow rate of LBE, which can work with required high temperature range below 500°C. Durability test of the flow meter was successfully performed with temperature of 400°C up to 5,000 hours. Further improvement to increase the maintainability is underway.
The corrosion tests of the candidate structural material for TEF-T and ADS is also planned as a fundamental research to realise LBE-cooled ADS. A high-temperature material corrosion test loop OLLOCHI (Oxygen-controlled LBE LOop for Corrosion tests in HIgh temperature) were manufactured and installed in JAEA. OLLOCHI has three independent test sections that can change flow rate and temperature of LBE to correct material corrosion date efficiently. Construction of OLLOCHI was finished except the third test section, which plans to install special test device for corrosion with mechanical stress, and functional tests of the loop is underway. Appearance of the OLLOCHI is shown in Figure 6.

Summary
To perform the design study for the transmutation system of long-lived nuclides, the construction of TEF, which consist two buildings, TEF-T and TEF-P, is proposed under the J-PARC Project. According to the
current construction schedule, TEF-T will be built at the first phase and TEF-P will be constructed at the latter phase. Licensing procedures for TEF-P construction will be processed simultaneously with TEF-T construction. TEF-T is a facility to prepare the database for material irradiation in flowing LBE environment and accumulate operation experiences of proton-irradiated LBE loop system using 400 MeV – 250 kW proton beam. Design study of spallation target head and related R&Ds to establish the TEF-T LBE target system are underway.

References


Study on optimisation of target head design for the TEF-T LBE spallation target

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Abstract
The Japan Atomic Energy Agency (JAEA) has proposed an accelerator-driven system (ADS) for nuclear transmutation. To realise the future ADS target, the ADS Target Test Facility (TEF-T) will be constructed under the framework of Japan Proton Accelerator Research Complex (J-PARC). In TEF-T, pulsed proton beams (250 kW, 25 Hz, and 0.5 ms pulse duration) will bombard a lead-bismuth eutectic (LBE) spallation target to produce neutrons. LBE will be adopted as the coolant. To design the target, the verification of target structural integrity is the primary task.

For this purpose, firstly, cavitation damage caused by the negative pressure in LBE is an essential issue that needs to be considered. In the present study, the possibility of cavitation damage occurrence caused by pressure waves and turbulent LBE flow was investigated for the TEF-T LBE target through the numerical simulations. Results show that the maximum expansion ratio of cavitation bubble is only 1.2 due to the pressure waves, so that severe cavitation damage will not occur due to the pressure waves; the maximum negative pressure due to the turbulent LBE flow is only –4.5 kPa on a steady-state flow condition, which is too small to drive the growth of bubbles, so neither cavitation damage will occur due to the turbulent LBE flow.

Secondly, the LBE flow behaviour needs to be investigated because it determines the temperature distribution on the LBE target vessel, which affects the integrity of the target vessel. The CFD analyses have been carried out to study LBE flow pattern. However, some stagnant regions exist in the LBE for the original target design. To solve this problem, the target head was modified to reduce the stagnant region effectively and efficiently. The CFD analyses results showed that the stagnant region has been effectively reduced due to the modification of target head. As a result, thermal-hydraulic and structural analyses results showed that the maximum temperature on the LBE vessel is decreased by 35 °C, and the maximum thermal stress on the BW has been decreased by approximately 31 MPa. The safety margin of target has been improved.

Introduction
The Japan Atomic Energy Agency (JAEA) has proposed an accelerator-driven System (ADS) for nuclear transmutation [1]. To solve technical issues for ADS target, the ADS Target Test Facility (TEF-T) will be constructed under the framework of Japan Proton Accelerator Research Complex (J-PARC). In TEF-T, pulsed proton beams (250 kW, 25 Hz, 0.5 ms pulse duration) will bombard a spallation target to produce neutrons [2-3]. Lead-bismuth eutectic (LBE) will be adopted as the spallation material as well as the coolant. As the first step, the inlet temperature of LBE is 350 °C and the LBE vessel with a Beam Window (BW) will be made of 2 mm thick type 316 stainless steel [4]. During the operation, the service condition of
BW is extremely severe, as it should withstand high temperature, large stress loadings and various sorts of damages. Therefore, one major objective of the target design is to improve the strength, and increase the safety margin, of LBE vessel.

It has been reported that the pressure waves caused by the deposited energy of pulsed proton beams in liquid metal is a critical factor needs to be considered for spallation targets bombarded by pulsed proton beams. That is because the propagation of pressure wave will lead to negative pressure in liquid metal. Consequently, cavitation bubbles will expand due to the negative pressure in liquid metal, and the collapse of cavitation bubbles near the target vessel will impose cavitation damage on the target vessel especially at the BW area [5-6]. Besides the pressure waves, the turbulent flow of liquid metal also has the possibility to cause negative pressure in the LBE, and finally gives rise to the cavitation damage on target vessel. The cavitation damage is critical to the lifetime of BW. So it is necessary to investigate the possibility of cavitation damage occurrence for the TEF-T LBE spallation target, which has been performed through numerical simulations in the present study.

Furthermore, as the coolant material, the LBE flow behaviour determines the temperature distribution on the LBE target vessel, which is closely related to the soundness of the target vessel. However, our previous study [4] showed that some stagnant regions exist, in the inner tube and at the centre of BW, due to the special configuration of target vessel. Temperatures in these stagnant regions are very high. To increase the cooling performance of target, the studies on the optimisation of LBE flow is quite necessary. The sizes of stagnant regions have to be reduced as much as possible or at least be moved away from the centre of BW. For this purpose, the CFD analyses have been carried out in the present study as a continuous work of the target design [7]. Specifically, additional flow slits were added and the inner tube was modified to reduce the stagnant region in the inner tube, and un-parallel wing-type flow guides were added to move the stagnant region at the centre of BW. Subsequently, thermal hydraulic and structural analyses were performed to verify the effects of the modified design.

**Original target head design and proton beam conditions**

Figure 1 shows the schematic drawing of the original design of target head. The design has a BW, which is connected to a coaxially arranged annular flow channel between the outside vessel (here after named LBE vessel) and the inner tube. To return the flow, the BW owns a concave section with a curvature of 32 mm, which is joined with a concave section in the opposite direction with a curvature of 79.5 mm at the centre of BW. In the Phase-I operation of target, the BW and LBE vessel will be made of type 316 stainless steel with a thickness of 2 mm. The inner diameter of LBE vessel and inner tube are 150 mm and 105 mm, respectively. The plate-type irradiation samples are set in a sample holder that is installed at the front of inner tube. Totally there are 8 samples with a gap of 4 mm between each other in the X-direction. The size of each specimen is 40×145×2 mm. The rectification lattice with a size of 52×52×5 mm is installed at the front-end of the sample holder, and 49 square apertures with each size of 4×4 mm are made on the rectification lattice to flow the LBE to cool the irradiation samples. One slit of 2 mm in width is arranged along each side of the rectification lattice, through which the LBE flows to cool the side wall of the sample holder.

Figure 2 shows the heat deposition in LBE with various peak beam current densities. The heat deposition was calculated by using the PHITS code [8]. The proton beam conditions used in this study were summarised in Table 1. It has a Gaussian profile. The beam current, 625 μA, would be the maximum beam current during the TEF-T operation. The pulse duration of proton beam was set to be 0.5 ms. Considering the heat disposition in Y-direction is only 158 mm, the length of straight tube was set to be 300 mm in the following numerical calculations to reduce the model size and save the calculation cost.
Figure 1: Schematic drawing of original design of TEF-T LBE spallation target head

Figure 2: Heat deposition in LBE with various beam current densities: (a) in Y-direction, X=0 cm, (b) in X-direction, Y=15.8 cm

Table 1: Proton beam conditions

<table>
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<td>Average beam current (μA)</td>
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<td>Peak beam current density (μA/cm$^2$)</td>
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Numerical simulation models

Model to calculate pressure waves

Figure 3 shows the analytical model for calculating the pressure wave. The dynamic analyses were carried out by using the commercial finite element method (FEM) programme, LS-DYNA [9]. The explicit code of LS-DYNA was used for calculation. A half 3D model was built. The target vessel and LBE were divided into hexahedral solid elements. The minimum element size is ca. 2.0 mm. For the boundary condition, the maximum of Y-direction was a non-reflecting type. The contact of solid and liquid was set to an automatic contact type. To save the calculation cost, the sample holder and irradiation samples were not considered in this model. The temperature rising in liquid metal due to the injection of proton beams can be calculated as the followings:

\[
\Delta \Delta \Delta = -\frac{Q}{\rho C_p} t
\]  

(1)
Here, $\Delta T$ is the temperature rising of liquid metal; $QQ$ is the heat density in liquid metal; $\rho P$ is the density of liquid metal; $C_{pp}$ is the specific heat of liquid metal; $tt$ is the average time of each pulse of proton beam in one second.

**Figure 3: Schematic of the 3D model for calculating pressure waves**

**CFD analysis and structural analysis models**

The thermal-hydraulic analyses were carried out by using the commercially available CFD code STAR–CD, and the analyses feature is steady state. In the 3D models, the BW part of the target was built but the LBE vessel part was not, as shown in Figure 4. The quarter model owns tetrahedral meshes with a number of approximately 1.6 million, then the size of mesh is fine enough to obtain enough spatial resolution to simulate the LBE flow. The reference condition [4] of TEF-T target is focused on in the CFD analysis and thermal-hydraulic analysis. The beam current density is $20 \mu A/cm^2$. The inlet flow rate of LBE in the annual channel is $1 l/s$ with a temperature of $350 \degree C$. The inlet velocity around the annual channel was assumed uniform. LBE with this flow rate is competent to cool the BW down, and meanwhile avoid too fast LBE flow speed that capable of causing severe erosion/corrosion damage on the target vessel. The $Re$ number is more than $1.2 \times 10^5$ in the TEF-T LBE flow condition and hence turbulent flow is anticipated to be formed in the flow channel. The high $Re$ number $k$-epsilon ($k-\varepsilon$) turbulence model was adopted to simulate the main LBE flow in the flow channel. The physical property of LBE at 600 K was adopted for calculation, which is listed in Table 2. The gravity effect was considered with the direction in Minus-Z. For the boundary condition, the wall surface was given as a non-slip condition with zero velocity in the beginning. The pressure of the outlet boundary was set to a constant value of zero. The outer surface condition of BW is set to adiabatic, as the loss of heat by conduction and radiation from the BW to the surrounding vacuum environment is considerably small.

The temperature distribution on the BW can be obtained from the thermal-hydraulic analyses. Subsequently, the structural analyses for the BW was carried out by using the implicit code of a commercial finite element method (FEM) programme, LS-DYNA [9]. The mesh structure and the temperature of each element of BW were output from STAR-CD and then input to LS-DYNA for structural analysis. Physical properties of the structural material, including the young’s modulus, Poisson’s ratio and coefficient of thermal expansion were defined as temperature dependent.
Table 2: Physical property of LBE at 600 K adopted in the numerical simulations

<table>
<thead>
<tr>
<th>Physical property</th>
<th>Symbol</th>
<th>Unit</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Density</td>
<td>(\rho)</td>
<td>Kg/m(^3)</td>
<td>10327</td>
</tr>
<tr>
<td>Specific heat</td>
<td>(C_p)</td>
<td>kJ/(kg K)</td>
<td>146.5</td>
</tr>
<tr>
<td>Thermal conductivity</td>
<td>(\lambda)</td>
<td>W/(m K)</td>
<td>12.9</td>
</tr>
<tr>
<td>Viscosity</td>
<td>(\eta)</td>
<td>MPa s</td>
<td>0.00185</td>
</tr>
<tr>
<td>Kinematic viscosity</td>
<td>(\nu)</td>
<td>mm(^2)/s</td>
<td>0.177</td>
</tr>
<tr>
<td>Prandtl number</td>
<td>(Pr)</td>
<td>-</td>
<td>0.021</td>
</tr>
</tbody>
</table>

Results and discussions

**Possibility of cavitation damage**

Figure 5 shows an example of the time response of pressure at position A that was marked out in Figure 3. Position A is where the LBE contacted with the target vessel at the centre of BW. The peak beam current density is 20 \(\mu\)A/cm\(^2\). The high-frequency components appeared in the time responses attributes to the LBE impulsively contacting target vessel during pressure wave propagation. The maximum negative pressure peak of ca. -0.03 MPa appears at 1.6 ms, followed by a positive pressure of ca. 0.6 MPa at 1.8 ms.

The negative pressure caused by the expansion of vessel due to pressure waves. By following the expansion of vessel, LBE also expands, which induces the negative pressure in LBE. The amplitude and time duration of negative pressure is dependent on the proton beam condition, and significantly affects the expansion ratio of cavitation bubbles, which correlates with degree of cavitation damage. The Keller’s equation \[10\] was employed to calculate a single bubble response as a function of the time-dependent pressure as followings:

\[
(1 - \frac{R}{C_p}) R \dot{R} + (\frac{3}{2} - \frac{R}{2C_p}) R^2 = \frac{1}{\rho} \left(1 + \frac{R}{C_p}\right) \left(P_b[t] - P + \frac{R}{C_p} P_0\right) + \frac{R}{C_p} P_0 \dot{P}_t[t]
\]

\[
P_t[t] = \left(P_0 - P_V + \frac{2a}{R}\right) \left(\frac{R_0}{R}\right)^3 + P_V
\]

\[
P_b[t] = P_t[t] - \frac{2a + 4\eta R}{R}
\]

Here, \(RR\) is the bubble radius; \(RR_{00}\) is the initial bubble radius; \(CC_P\) is the specific heat of liquid metal; \(\rho\) is the density of liquid metal; \(22\) is the surface tension; \(44\) is the viscosity of liquid metal; \(PP_t\) is the pressure in the bubble; \(PP_{bb}\) is the liquid pressure beside the bubble; \(PP_{Vv}\) is the vapour pressure.
Figure 6 shows the time responses of expansion ratio of cavitation bubble at Position A calculated from Eq. (1). It is noted that the initial bubble radius, \( R_0 \) (core of the cavitation bubble) was set to be 10 \( \mu \)m and the bubble responses are shown as expansion ratio, \( R/R_0 \). In all the three cases, the maximum expansion ratio was not exceeds 1.2, which is a neglectable value. This is because magnitude of negative pressure is too weak and the duration time of negative pressure is too short to drive enough growth of cavitation bubbles. The cavitation density, \( DD \) has the following relationship with the expansion ratio of cavitation bubbles [11]:

\[
D = f\left(\frac{R_{\text{max}}}{R_0}\right)^3
\]

(5)

For the mercury spallation target in Japanese Spallation Neutron Source (JSNS), the expansion ratio of cavitation bubble reaches to more than 100 to drive the cavitation bubble growing big enough to impose cavitation damage on the BW [11]. According to Eq. (5), the damage intensity of LBE target is only several millionth of that of mercury target, so it is considered that such a small expansion ratio of cavitation bubble would not impose severe cavitation damage on the LBE vessel.

Figure 5: An example of the time response of pressure at position A; beam density: 20 \( \mu \)A/cm\(^2\)

Figure 6: Time response of expansion ratio of cavitation bubbles at Position A on various proton beam conditions; \( R_0 = 10 \mu \)m

Regarding of possibility of the cavitation due to LBE turbulence flow, the pressure distribution in LBE of the steady-state analysis for the original target design is shown in Figure 7. It can be seen that the maximum negative pressure of ca. -4.5 kPa exists downstream the rectification lattice. This attributes to
the pressure loss of LBE after passing through the narrow flow slits into a relative broader area. However, the amplitude of negative pressure in LBE is so small that the expansion of bubbles cannot be driven by the pressure. Therefore, it is considered that the cavitation damage would not be caused by the turbulence flow of LBE on a steady-state flow condition.

Figure 7: Pressure distribution contour in LBE for the original target design

![Pressure distribution contour in LBE for the original target design](image)

LBE flow pattern of original target head design

Figure 8 shows the velocity vector and velocity contour for the original design of target head. Clearly there was a “dead-flow” region in the LBE beside the centre of BW. From the velocity vector it can be seen that the stagnant region was formed due to the symmetric returned LBE flow from the radial directions. Also there are some stagnant regions exist in the upstream area in inner tube, which attributes to the occurrence of pressure loss caused by LBE flowing through the slits into the inner tube.

It is expected to control the temperature on the target vessel below 500 °C, for that temperature on target vessel above 500 °C leads to: (1), the allowable stress of 316 SS decreases; and (2), the development of erosion/corrosion damage of 316 SS due to contacting with LBE will be greatly accelerated. Figure 9 shows the temperature distribution on the BW, and a hot spot with a maximum temperature of 496 °C was formed at the centre of BW due to the existence of the stagnant region. The temperature is very close to 500 °C. On the other hand, although the structural integrity of BW can be verified from the viewpoint of maximum temperature and stress loadings [4], the compound effects of various sorts of damages on structural integrity was not taken into consideration. To more conservatively secure the safety margin for the target vessel, it is desirable to reduce the stagnant regions and decrease the temperature on target vessel.

Figure 8: Velocity vector and velocity contour of LBE flowing for the original design

![Velocity vector and velocity contour of LBE flowing for the original design](image)
Figure 9: Temperature distribution on the BW for the original target head design

Designs to reduce stagnant region in the inner tube

The stagnant region in the inner tube mainly attributes to the narrow flow slits with a width of 2 mm that located beside each side of the rectification lattice. The number and width of flow slits were modified to solve this problem. Figure 10 shows the schematic drawing of modification of target head to reduce the stagnant region in the inner tube. Two types of modifications of target head were made: a) two additional slits were added at each side of the rectification lattice to flow the LBE into the inner tube, and the width of each slit in the modified design (Design A) is 4 mm, 2 mm, and 2 mm, respectively; b) after adding the flow slits, there exist circular flow areas at the corner of inner tube, so the thickness of the inner tube corner was increased to remove the circular flow areas.

The LBE flow pattern of Design A was shown in Figure 11. The flow rate of LBE was increased from 1 L/s to 1.9 L/s, which is still under the capability of the target system pump. The stagnant region in the inner tube has been almost removed due to the modification, this will benefit to cool down the inner tube in this area. Moreover, the flow speed at the sample area was not changed significantly due to the modification. However, the stagnant region still exist at the centre of BW. In order to reduce or move the stagnant region at the centre of BW, flow guides were added to destroy the symmetric flow pattern at the centre of BW.

Figure 10: Schematic drawing of modification of target head
Designs to reduce/move stagnant region at the centre of BW

The influences of length, number and angle of plate-type flow guides on LBE flow behaviour have been systematically studied in our previous work [7]. On the basis of the experience obtained from the previous study, the plate-type flow guides were changed to wing-type flow guides to smoothen the LBE flow around the flow guides. Figure 12 shows the schematic drawing of model with the added wing-type flow guides (Design B). The angle of the flow guides with Y-direction is 67.5°. The guides were set vertically to the inner tube surface. Figure 13 shows the LBE flow pattern of Design B. The results indicates that the stagnant region still exists, but the size of stagnant region has been greatly reduced and been moved away from the centre of BW approximately 10 mm.

Temperature and thermal stress distribution

To verify the effects of Design B, the thermal-hydraulic analysis and structural analysis have been carried out. Figure 14 shows the temperature distribution contour on the BW of Design B. Compared with the temperature distribution contour on the BW of original design (Figure 9), the maximum temperature has been decreased by 35 °C. Figure 15 shows the corresponding thermal stress distribution on the BW for the original design and Design B. It can be seen that the maximum thermal stress has been reduced by
approximately 31 MPa as a result of the decreased temperature. In addition, the maximum thermal stress point has been moved away from the centre of BW. Consequently, the stress concentration on the centre of BW can be released. The results indicate that the optimised design up to now improves the safety margin of target.

![Figure 14: Temperature distribution contour on the BW of Design B](image)

![Figure 15: Thermal stress distribution contour on the BW for the original design and Design B](image)

(a) Original design  (b) Design B

**Conclusion**

The expansion ration of cavitation bubbles due to the negative pressure caused by pressure waves in the TEF-T LBE spallation target is too small to lead to severe cavitation damage on the target vessel; the negative pressure caused by turbulence LBE flow is not strong enough to drive the expansion of cavitation bubbles.

By adding flow slits beside the rectification lattice and increasing the thickness of inner tube, the stagnant region in the inner tube of TEF-T LBE spallation target has been almost removed; by adding un-parallel wing-type flow guides, the stagnant region at the centre of BW has been moved away from the centre of BW approximately 10 mm.

As a result of modifying the target head, the maximum temperature on the BW has been reduced by 35 °C, and the maximum thermal stress on the BW has been decreased by approximately 31 MPa. The safety margin of target has been improved through this study.

In the near future, experiments by using a particle image velocimetry (PIV) system that installed at a water loop will be carried out to validate the numerical simulation results.

**Acknowledgements**

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References


Numerical simulation of proton beam-induced pressure waves in spallation targets

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Abstract

Operation experience of the spallation neutron sources SNS and JSNS show that pulsed MW bombardment of liquid metal can result in pressure pulses impacting the structure of the target, thus severely limiting the lifetime of the target. For targets subjected to continuous proton beam bombardment, beam trips in particular two configurations should be considered in the design process. Firstly, an uncontrolled beam loss leading to cavitation in the beam interaction region and secondly start-up transients leading to sudden volumetric expansion and a high-pressure zone in the beam interaction region.

In order to allow for effective simulations during the design process and to gain a more detailed understanding the multiple pressure variables (MPV) method is proposed. The MPV approach is based on a single time scale multiple space scale asymptotic analysis derived for subsonic flow by an asymptotic series expansion in the Mach number.

The paper will introduce the MPV method for liquid-metal applications. Then a series of validation calculations for the META: LIC (MEgawatt Target: Lead-bismuth Cooled) target developed according to European Spallation Source (ESS) specifications will be used to demonstrate the advantageous features of the method emphasising that no time-step limitation beyond those enforced by the fluid flow calculation are imposed by the acoustics. Finally, the importance of the pressure pulse phenomena for ADS applications has been evaluated for the MYRRHA target.

Introduction

In short pulsed accelerator-driven neutron sources such as the Spallation Neutron Source (SNS) in Oakridge, TN and the Japanese Spallation Neutron Source (JSNS) heavy liquid metal is used as target material. The high-energy proton beam deposits energy into the liquid during a sub-microsecond time frame [1]. This energy deposition results in very-high heating rates. Since the timescale of the proton pulse is too short to allow the target material to expand during the pulse, the thermal expansion is impeded thus compressing the target material. This in turn results in the production of large amplitude pressure waves propagating through the target [2]. Once this pressure wave hits the target wall a negative pressure is induced due to the elasticity of the target wall. Moreover, if the short term tensile strength of the spallation material is met cavitation in form of cavitation bubbles will occur. Collapse of cavitation bubbles in the vicinity of the target wall is accompanied by the characteristic damage pitting.
In case of a long pulsed spallation neutron source the amplitude of the induced pressure wave due
to the proton beam heating is not as pronounced as for short pulsed sources. This is due to the lower
energy per single pulse corresponding to the longer pulse length. The increased timescale allows the
target material to partially adjust to the thermal expansion. Therefore, for a long-pulse target the
maximum pressure amplitude is no longer proportional to the total energy per pulse [3]. Instead, the
sudden change in heating rate at the beginning and end of each pulse becomes the source for pressure
waves. Yet all effects observed in short pulsed spallation targets may be relevant to long-pulse targets as
well. This includes induced pressure waves, cavitation and potentially pitting damage fortunately
occurring at a lower repetition rate. The target container suffers during normal operation and beam trips
substantial straining due the notable thermal volumetric expansion of the spallation material.

Even in continuous spallation sources using liquid-metal cooled targets beam trips can induce
pressure waves. A nuclear reactor application of a continuous source is proposed for the accelerator-
driven system MYRRHA (Multipurpose Hybrid Research Reactor for High-tech Applications). When the
proton beam restarts after a beam interruption an initial high-pressure zone is formed within the
spallation zone. This high-pressure zone then propagates outwards towards the target enclosure where
reflection results in a negative pressure zone. If the tensile strength of the fluid is exceeded, cavitation
and potentially pitting damage will occur.

The simulation of proton beam-induced pressure waves are numerically challenging. Since applicable
numerical time steps become extremely small, the transient phenomena of pressure wave propagation
need an exceedingly large number of time steps. According to the CFL criteria the time step is determined
by the ratio of minimum grid size and characteristic flow velocity. In simulations covering pressure wave
propagation compressibility of the fluid must be considered. Instead of the fluid velocity sound speed
enters the CFL criteria. Unfortunately, the ratio of fluid velocity and sound speed, i.e. the Mach number is
very small in target applications so that a pressure resolving simulation requires several orders more time
step compared to an incompressible simulation.

Aeroacoustics faces exactly the same problem. The Multiple Pressure Variable (MPV) approach was
developed in aeroacoustics to efficiently simulate weakly compressible flows. The MPV approach is
based on a single time scale multiple space scale asymptotic analysis derived for subsonic flow by an
asymptotic series expansion in the Mach number.

We develop a MPV method for liquid-metal applications and apply it to two target geometries,
specifically the META:LIC [4] and MYRRHA targets. For each target simulations are conducted which allow
estimation of proton beam-induced pressure waves.

**Multiple pressure variables method for liquid-metal application**

A detailed asymptotic analysis of the compressible Euler equations has been performed by Klainerman
and Madja [5]. Klein [6] extended this analysis to a multiscale asymptotic expansion for all physical
variables $\mathbf{w} = \rho, \mathbf{U}, p$ and $\mathbf{w} = f(\mathbf{x}, \xi, t) = f(M\mathbf{x}, t)$ in powers of the Mach number

$$\mathbf{w} = \mathbf{w}^{(0)} + M \mathbf{w}^{(1)} + M^2 \mathbf{w}^{(2)} + \cdots.$$ (1)

Here $\mathbf{x}$ is the local variable associated with convective phenomena, while $\xi = M\mathbf{x}$ is a large scale co-
ordinate and is associated with acoustic wave propagation. This asymptotic expansion is then substituted
into the non-dimensionalised governing equations. All terms with like power of the Mach number are
collected and separately equated to obtain a series of asymptotic limit equations. The asymptotic analysis
shows constant spatially leading order pressure $p^{(0)}$. However, $p^{(0)}$ varies in time and serves to satisfy the
equation of state 0, therefore is called thermodynamic pressure. Furthermore, the analysis clarifies that the
pressure term \( p^{(1)} \) is a function of the spatial variable. It is associated with acoustic waves hence it is called acoustic pressure. The second order pressure \( p^{(2)} \) depends on temporal and both spatial scales. It assures the compliance with the incompressible \( \nabla U = 0 \) contraint.

The characteristic Mach number of proton beam-induced pressure waves in liquid-metal spallation targets is due to the very low compressibility of liquid metals typically in the order of \( 10^{-4} \). Due to these very low Mach numbers an acoustic length scale becomes extremely long so distinction of the acoustic scale \( \xi \) is no longer applicable. Formally a numerically yet more efficient two pressure variable approach suffixes which was previously introduced by Park [8].

**Governing equations**

The compressible conservation equations for density \( \rho \), momentum \( \rho U \) and total energy per unit volume \( E \) are:

\[
\frac{\partial \rho}{\partial t} + \nabla \cdot (\rho U) = 0 \tag{2}
\]

\[
\frac{\partial (\rho U)}{\partial t} + \nabla \cdot ((\rho U)\otimes U) + \nabla p = \nabla \cdot \tau + \rho \mathbf{g} \tag{3}
\]

\[
\frac{\partial E}{\partial t} + \nabla \cdot [E U + p (U \cdot U)] = \nabla \cdot (\tau U) - \nabla \cdot \mathbf{q} \tag{4}
\]

where \( \tau \) is stress tensor, \( p \) pressure, \( t \) time and \( \mathbf{g} \) the gravitational acceleration.

The heat-flux \( \mathbf{q} \) is given by Fourier’s law \( \mathbf{q} = -\lambda \nabla T \), where the heat flux associated with temperature \( T \) and the thermal conductivity \( \lambda \).

Above system of equations is closed by the thermal and caloric equation of state of the fluid. The thermal equation of state for fluids is generally given in terms of pressure, temperature and specific volume. The caloric equation of state is provided in terms of internal energy or enthalpy [7]. To simulate general fluids, it is desirable to use a formulation of the energy equation in terms of pressure and enthalpy [9].

Neglecting without loss of generality thermal dissipation and hydrostatics the following equations are used to derive a multiple pressure variables formulation for liquid-metal applications:

\[
\frac{\partial \rho}{\partial t} + \nabla \cdot (\rho U) = 0 \tag{5}
\]

\[
\frac{\partial (\rho U)}{\partial t} + \nabla \cdot ((\rho U)\otimes U) + \nabla p = \nabla \cdot \tau \tag{6}
\]

\[
\frac{\partial p}{\partial t} + U \cdot \nabla p - p \cdot (\nabla \cdot U) = \nabla q + \rho \frac{\partial h}{\partial t} + \nabla \cdot (\rho U h). \tag{7}
\]

Equations (5) – (7) are non-dimensionalised so that the material properties appearing in the equations are replaced by dimensionless quantities. Therefore, this system of equations is valid for a class of physically similar problems. Reference length, velocity, density and pressure are used for non-dimensionalisation. The specific enthalpy is made dimensionless with the reference velocity squared. Dimensionless independent and dependent variables are denominated \( \hat{x}, \hat{t}, \hat{U}, \hat{\rho}, \hat{p}, \hat{h} \). The nabla operator in terms of non-dimensional spatial variables is \( \hat{\nabla} \). The equations read in non-dimensional notation:

\[
\frac{\partial \hat{\rho}}{\partial \hat{t}} + \hat{\nabla} \cdot (\hat{\rho} \hat{U}) = 0 \tag{8}
\]

\[
\hat{\rho} \left( \frac{\partial \hat{U}}{\partial \hat{t}} + (\hat{U} \cdot \nabla) \hat{U} \right) = -\frac{1}{M^2} \nabla \hat{p} + \frac{1}{Re} \Delta \hat{U} \tag{9}
\]

\[
\frac{\partial \hat{p}}{\partial \hat{t}} + \hat{U} \cdot \nabla \hat{p} - \hat{p} \cdot (\nabla \cdot \hat{U}) = M^2 \left( \hat{\rho} \frac{\partial \hat{h}}{\partial \hat{t}} + \hat{\nabla} \cdot (\hat{\rho} \hat{U} \hat{h}) + \frac{1}{Pr Re} \hat{\nabla} \cdot (\lambda \hat{\nabla} T) \right), \tag{10}
\]

\[
130
\]
where $M = U_{ref}/c_{ref}$ represents the global Mach number with $c_{ref}$ as the reference speed of sound. Reynolds and Prandtl numbers are denoted in suitable non-dimensional parameters Re and Pr, respectively.

The incompressible limit is as the Mach number tends to zero. The term $1/M^2$ multiplying the pressure gradient in the momentum equation shows singular behavior which is unfavourable for numerical methods. For simplicity hats denoting the dimensionless values are dropped below.

**Multiple Pressure Variables Approach**

The reduced MPV approach only considers the leading and second order pressure terms. When eliminating the first order pressure or acoustic pressure its longwave contribution is contained within the second order pressure $p^{(2)}$. The reduced MPV ansatz introduced analogue to [8] is given by

$$p(x,t) = p^{(0)}(x,t) + M^2 p^{(2)}(x,t) \quad (11)$$

Following the previous asymptotic analysis [5] [6] the leading order pressure is defined as the average pressure within the whole computational domain $\Omega$:

$$p^{(0)} = \frac{1}{|\Omega|} \int_{\Omega} p \, dV \quad (12)$$

Inserting the reduced MPV Ansatz Equation (21) into the dimensionless equations (Equations (8)-(10)) results in the following system of equations:

$$\frac{\partial \rho}{\partial t} + \nabla \cdot (\rho U) = 0 \quad (13)$$

$$\rho \left( \frac{\partial U}{\partial t} + (U \cdot \nabla) U \right) = \nabla p^{(2)} + \frac{1}{\rho(Re)_{Re}} \nabla \cdot \tau \quad (14)$$

$$M^2 \frac{\partial p^{(2)}}{\partial t} + \frac{\partial p^{(0)}}{\partial t} + (U \cdot \nabla) p - p \cdot (\nabla \cdot U) = M^2 \left( \frac{1}{PrRe} \nabla \cdot (\nabla T) + \frac{\partial H}{\partial t} + \nabla \cdot (U H) \right) \quad (15)$$

where $H = \rho \cdot h$ is enthalpy. This system of equations is implemented in our code.

**Spallation targets**

**META:LIC – MEgawattTArget:Lead bIsmuthCooled**

The META:LIC target [4],[11] is a liquid-metal target developed as comparative target solution for the ESS according to ESS specifications[12]. Lead-bismuth eutectic (LBE) is chosen as both target material and primary coolant. Figure 1 (left) shows the conceptual design of modular target system. The target system has three separately replaceable modules: a target module, a pump module and a heat exchanger module. These three modules are connected to a LBE pool. Heat exchanger and pump are submerged in the pool and the target module is attached to the pool. A containment which safely encloses the whole system is foreseen which is not shown in Figure 1. In order to replace the individual modules the whole META:LIC target system is moved on a trolley into hot cells where the containment can be opened. The anticipated trolley system is similar to the arrangements at SNS [1].

Figure 1 (right) depicts the target module which is double walled. As shown, the target module consists of a proton beam guide with a safety window (not shown), an inflow channel leading to a nozzle producing a uniform block velocity profile, a U-bend with an expansion chamber and spoiler enforcing
flow detachment to counteract the effects of the thermal expansion and pressure waves in the LBE due to the pulsed proton beam, and an outflow duct. The flow is pumped upwards into the inclined inflow channel, then accelerated by the nozzle. Next the LBE flows through the proton beam interaction zone which is inclined relative to the horizontal plane by 15 degrees. The LBE returns to the pool through an U-bend and outflow duct. A horizontal proton beam enters the liquid metal through a solid wall approximately 1.5 mm thick. The small inclination angle of the proton beam interaction zone provides almost coaxial LBE flow and proton beam. This results in a fairly uniform heating of the coolant so that a minimal coolant flow rate can be established. Moreover, the flow component perpendicular to the proton beam transports the fluid across the beam in a fairly short time. This is advantageous for pulsed beams, as successive beam pulses interact with fluid that was not subjected to the beam previously.

The META:LIC target suffers of both, pressure pulses due to the pulsed nature of the proton beam and the strain on the target container due to the thermal expansion of the LBE [3].

**Figure 1: Modular target concept of META:LIC (left), window target module (right)**

**MYRRHA – Multipurpose Hybrid Research Reactor for High-tech Applications**

The MYRRHA reactor is a flexible experimental ADS system currently under development at SCK•CEN, demonstrating the coupling of an accelerator, spallation target and subcritical core at a reasonable power level. Furthermore, it is planned as a flexible irradiation facility and its reactor is supposed to work both in critical and subcritical mode [13][14]. In an ADS the reactor runs in subcritical mode. The spallation neutrons provided from the accelerator and target are needed to maintain fission. When MYRRHA is operated in ADS mode, the central position of the core is occupied by the so-called spallation target assembly.

A sketch of the planned tube type spallation target is shown in Figure . As depicted in the figure the proton beam is guided by the beam tube into the central core region, where it enters the spallation zone through the proton beam window, which separates the accelerator vacuum from the target material. The proton beam window is cooled by the upward flowing primary coolant under forced convection supported by the primary pumps.
META:LIC target calculations

Results of the MPV calculations for the META:LIC target of a single proton beam pulse are shown for various spatial and temporal resolutions. Initial conditions correspond to fully developed flow. The caloric and thermal equations of state for LBE proposed in [15] are used.

Geometry, boundary and initial condition

The geometry used for the META:LIC simulations as well as the applied boundary patches are displayed in Figure 3. Due to symmetry considerations only half of the target width is simulated. According to specification the inflow velocity of 1.5 m/s. At the wall boundaries a no slip condition is applied. At the inlet and outlet wave transmissive boundary conditions reduce undesirable pressure wave reflections at both. The inlet temperature is set to 500K. Using the average speed of sound of LBE within the simulated geometry a global Mach number of $M = 0.0015$ is determined. For the calculations three different mesh resolutions were used, namely 0.37, 0.91 and 2.12 million cells. Five different time steps are analysed: 2, 0.4, 0.4, 0.02 and 0.01 ms. Five time steps are used to find the optimal balance between accuracy, resolution and dissipation of the numerical scheme.

The proton beam specifications are taken from [12] and displayed in Table1. For all presented calculations a single proton beam pulse is assumed. Figure 3 (right) displays the power density distribution of the proton beam in the liquid-metal target META:LIC.

<table>
<thead>
<tr>
<th>Table1: Proton beam specifications for ESS and MYRRHA</th>
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<tbody>
<tr>
<td><strong>META:LIC</strong></td>
</tr>
<tr>
<td>Time structure</td>
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<tr>
<td></td>
</tr>
<tr>
<td>Pulse length</td>
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<tr>
<td>Pulse repetition rate</td>
</tr>
<tr>
<td>Beam energy</td>
</tr>
<tr>
<td>Average beam current</td>
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<tr>
<td>Beam profile</td>
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</table>
Simulation results

The velocity and temperature distribution at the end of a single proton beam pulse is shown in Figure 4. There is no visible distortion of the velocity field due to the thermal heating of the LBE. Figure 4 (right) displays the total temperature rise of about 90 K during a single proton beam pulse.

Snap shots of the pressure during a single proton pulse on the coarse mesh for $\Delta t = 0.02$ ms is displayed in Figure 5. Figure 5 (a) shows the pressure in the target module at the beginning of the pulse when the proton beam first hits the liquid metal. During this stage a pressure amplitude of approximately 7 bar is attained. The pressure then rises during the continued energy deposition up to approximately 17 bar at the end of the pulse when the heat deposition stops, see Figure (b). Once the heat deposition stops a negative pressure zone is formed due to the inertia of the surrounding fluid which still conforms to the thermal expansion. Figure (c) depicts the pressure distribution approximately 2 ms after the heat deposition stops. A negative pressure of $-10$ bar is reached. In the algorithm strain is limited to $-10$ bar. In reality the physical limitation is given by an unknown tensile strength of LBE. Exceeding the limit leads to cavitation.

Figure 6 displays the temporal evolution of a single pressure pulse for different numerical time steps on the mesh with 0.37 million cells. The largest chosen time step corresponds to the proton beam pulse length and the smallest is 200 times smaller. The blue line in Figure 6 (left) displays the resulting pressure evolution if the pressure pulse is under resolved with a single time step. The red dashed line shows very low temporal resolution with a ten times smaller time step. Comparing the maximum pressure amplitude of these two calculations shows a difference of approximately 5 bar, the same discrepancy can be observed for the negative pressure after the heat deposition stops. Using a time step of 0.01 ms, results a pressure amplitude of approximately 14 bar and a negative pressure of 7 bar. From Figure 6 follows that the pressure peaks for intermediate temporal resolution. Small time steps increase numerical dissipation and thus the pressure amplitude. However, it can also be observed that increasing time resolution allows for fluctuations of the pressure during the pulse. These fluctuations are due to the now resolved pressure reflections at the wall boundaries. For conservative estimations we propose to use intermediate time steps.

Figure 6 (right) displays the influence of the mesh resolution on the pressure evolution. Here, the pressure as a function of time is shown for three different numerical grids and two different time-step length. The figure shows, that there is virtually no difference in the pressure evolution for the different meshes. This is the case for both time steps.
MYRRHA target calculations

Geometry, boundary and initial conditions

A simplified axisymmetric two-dimensional geometry is used for the simulations. The influence of the original hexagonal shape of the assembly on the resulting fluid flow are neglected for a first investigation of proton beam-induced pressure waves. Fixtures such as flow straighteners downstream of the proton beam interaction zone are disregarded for the same reason. Figure 7 (left) displays the simulated geometry as well as the applied boundary patches.

At the inlet boundary condition an inflow velocity of 0.45 m/s is chosen to limit the outlet velocity in accordance to the erosion limit. At the wall boundaries a no slip condition is applied. At the wall
boundaries a no slip condition is applied. At the inlet and outlet wave transmissive boundary conditions reduce undesirable pressure wave reflections at both. The inlet temperature is set to 543 K.

For the calculations the global Mach number is 0.0005. The behaviour of the pressure is evaluated at several sample points (P1 – P5) which are located along a straight line in the area with the highest energy deposition. P5 is in the region with the highest energy deposition and also closest to the proton beam window, point P1 is 0.155 m upstream of P5. Figure 7 (right) displays the position of the sample points.

The specifications of the MYRRHA accelerator are taken from 0 and listed in Table1. For the CFD simulations a peak current of 4mA is assumed. According to 0 the nominal time macrostructure of the proton beam features 200 μs beam holes at frequencies of 1- 250 Hz. Simulated frequencies are 100 Hz and 250 Hz. The intended sweeping of the proton beam is not considered in present calculations. This sweeping might result in slightly higher pressure wave amplitudes. Figure 7 (right) displays the power distribution in LBE in kW/cm³ per 1 mA.

**Figure 7: MYRRHA geometry and boundary conditions (left), power deposition and sample points (right)**

**Nominal proton beam operation at 250 Hz**

For this simulation a nominal macrostructure with 200 μs beam holes at a frequency of 250 Hz is assumed. Figure 8 displays the pressure history at sample point P5 (red dashed line) and P1 (blue continuous line) for a simulation time of 0.05s, which corresponds to 13 beam pulses. The figure shows that for normal operation conditions the pressure fluctuates by approximately ±0.2 bar at Point P5, the area with the highest energy deposition and ±0.15 bar at Point P1.
Figure 8: Pressure [bar] vs time [s] with nominal beam operation at 250 Hz

Beam trip results for 100 Hz and 250 Hz

The investigation of the influence of beam trip on the temporal behaviour of the pressure at Point P1 has been performed for a nominal macrostructure frequency of 100 Hz and 250 Hz. The simulated beam trips begin with normal proton beam operation as shown in Figure 8. Then the beam is interrupted for a predefined period. In the simulated case a beam trip of 0.01 and 0.001 s was observed. These short beam trips are not limited in occurrence according to the MYRRHA accelerator beam requirements given in [16]. Following a beam trip the proton beam returns to normal operation. The time-step width for the simulation is chosen to be 0.1*10^-3 s. Therefore, the beam trip with a duration of 0.001 s is discretised by 10 time steps and the second beam trip with 100 time steps.

Figure 9: Pressure [bar] vs time [s] for beam-trip durations of 0.01 s and 0.001 s for 100 Hz (top) and 250 Hz (bottom)
Figure 9 depicts both described beam trips for a nominal operation macrostructure of 100 Hz (left) and for an operation macrostructure of 250 Hz (right). For the shorter beam-trip duration depicted by the blue continuous line, peak pressure amplitudes of approximately 2 bar and 2.1 bar are calculated. For the second beam trip with a duration of 0.01s shown by the red dashed line the peak pressures reach about 1.75 bar. For both macrostructures and beam-trip durations the maximum pressure drop due to the sudden loss of proton beam heating is about 0.45 bar. The pressure oscillations following the peak pressure values are mainly due to wall boundary reflections. Small inlet outlet reflections are still observable and could be omitted by extending the computational domain. For an experimental facility these pressure pulses may be tolerable, but for long-term ADS operation fatigue should be addressed.

Conclusion

Liquid-metal spallation targets are subjected to time-dependent heat deposition for both continuous and pulsed operation. With a traditional CFD approach analysis of beam trips and the pulsed temporal structure requires extensive numerical simulations. In particular the acoustics limit forces extremely short temporal time steps. The simulation of the MYRRHA and META: LIC heavy liquid metal target using MPV method demonstrate that the restrictive time-step limitation can be overcome by the asymptotic approach. Both targets show pressure pulses with an amplitude of the order of bars which need consideration since pitting damage or fatigue can occur and should be controlled. Final verification of this damage requires additional experimental investigations on the tensile strength of the chosen liquid metal.

References


Session 4: Design and technology of subcritical systems

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MYRRHA design revision 1.6 and the phased implementation plan

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Abstract

Once started as a small irradiation facility, based on the accelerator-driven system (ADS) concept, the purpose of the MYRRHA Project has been extended to become a material testing reactor for material and fuel research, to study the feasibility of transmutation of minor actinides and finally to demonstrate at a reasonable power scale the ADS principle and technology. Being based on heavy liquid metal technology (lead-bismuth eutectic), it significantly contributes to the development of lead fast reactor (LFR) Technology. In the beginning of 2014, SCK•CEN has consolidated a coherent version of the primary system. This so-called version 1.6 of the primary system forms the basis of the work performed by an engineering consortium on the balance of plant activities. In the present paper, we detail the current design of MYRRHA and briefly present the current tracks under investigation towards the next release 1.8. In 2016, a phased implementation plan has been established with the aim of reducing the financial and engineering risk.

Introduction

MYRRHA (Multipurpose hYbrid Research Reactor for High-tech Applications) is the flexible experimental facility under development at SCK•CEN. MYRRHA is able to work both in subcritical (ADS) as in critical mode. In this way, MYRRHA should target the following applications catalogue [1]:

- To demonstrate the ADS full concept by coupling the three components (accelerator, spallation target and subcritical reactor) at reasonable power level (50-100 MWth) to allow operation feedback, scalable to an industrial demonstrator;
- To allow the study of the transmutation of high-level nuclear waste, in particular minor actinides that would request high fast flux intensity (Φ>0.75MeV = 1015 n/cm²s);
- To be operated as a flexible fast-spectrum irradiation facility for:
  - fuel development for innovative reactor systems;
  - material development for Gen IV systems;
  - material development for fusion reactors;
  - radioisotope production for medical and industrial applications by holding a backup role for classical medical radioisotopes and focusing on R&D and production of radioisotopes requesting very-high thermal flux levels.
The MYRRHA design originally started as a small irradiation facility (150 MeV, 1.5 MWth), having the production of radioisotopes for medical purposes as its single objective. In 1998, the purpose of the project has been extended to become a material testing reactor for material and fuel research, to study the feasibility of transmutation of minor actinides and to demonstrate the principle of the ADS at a reasonable power scale (above 50 MWth). Since then, the project is called MYRRHA.

Design evolution

MYRRHA consists of a proton accelerator delivering its beam to a spallation target coupled to a subcritical core. In 2005 MYRRHA consisted of a proton accelerator delivering a current of 5 mA at an energy of 350 MeV to a windowless spallation target coupled to a subcritical fast core of 50 MWth.

This 2005 design was used as a starting base within the EUROTRANS (EUROpean research programme for TRANSmutation of high-level nuclear waste in an accelerator-driven system) integrated project (2005-2010) of the European Commission in its 6th Framework Programme, which resulted in the XT-ADS [2] (eXperimental demonstration of the technical feasibility of transmutation in an accelerator-driven system) design, where a linear accelerator delivers a beam of 3.2 mA at an energy of 600 MeV into the spallation target. The reactor power of XT-ADS was 57 MWth.

The XT-ADS design was taken as a starting point for the work performed in the central design team [3,4] (CDT) project of the 7th Framework Programme. For a clear understanding of the design evolution through the different versions within the CDT project, we have decided to call the initial version “Release 1.0”, odd numbers being related to internal working versions and even numbers to reference, frozen versions. CDT has been concluded with a “Release 1.4” design, also called FASTEF (FAst-Spectrum Transmutation Experimental Facility).

Since the end of the CDT project in 2012, the MYRRHA team has performed some further engineering work in its core knowledge, the primary system and core design, leading to the current design version “Release 1.6” already briefly presented [5,6]. Recent updates on building design and plant layout have been published recently elsewhere [7].

In the present paper we go more in detail in the mechanical parts of the "Release 1.6" design (reactor vessel and internals, heat exchangers and fuel-handling machine). Next, we discuss the current investigations towards the next version 1.8. We conclude by explaining the phased approach which is currently being implemented.

General configuration

The main components/systems of the current “Release 1.6” design are still of the same type as the previous versions defined within the EUROTRANS project. The general design includes a primary circuit with lead-bismuth eutectic (LBE), a secondary circuit with water/steam and a tertiary circuit with air. The primary and secondary systems have been designed to evacuate a maximum core power of 100 MWth like the previous FASTEF version.

The power increase is justified to reach the objectives of the applications catalogue presented in the previous section earlier in this paper. All the components of the primary systems are optimised for the extensive use of the remote handling system during components replacement, inspection and handling.

Figure 1 shows a section of the MYRRHA reactor in the current design with its main internal components. Table 1 below summarises the main physical characteristics of the MYRRHA core and primary system. It is worth to note that the average hot plenum temperature is not 270+90 = 360°C, but a lower temperature (325°C) due to all bypass flows (lead-bismuth not flowing through a fuel assembly) in and out of the core.
Table 1: Main MYRRHA characteristics

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total reactor power</td>
<td>110 MW&lt;sub&gt;th&lt;/sub&gt;</td>
</tr>
<tr>
<td>Total primary mass flow rate</td>
<td>13800 kg/s</td>
</tr>
<tr>
<td>LBE mass inventory</td>
<td>7600 tonnes</td>
</tr>
<tr>
<td>Core inlet temperature at full power</td>
<td>270°C</td>
</tr>
<tr>
<td>Hot plenum temperature at full power</td>
<td>325°C</td>
</tr>
<tr>
<td>Average core temperature difference at full power</td>
<td>90°C</td>
</tr>
<tr>
<td>Cold shutdown temperature</td>
<td>200°C</td>
</tr>
<tr>
<td>Temperature of secondary cooling loop</td>
<td>200°C</td>
</tr>
<tr>
<td>Number of fuel assemblies in a critical reference case</td>
<td>108</td>
</tr>
<tr>
<td>Number of penetrations for experiments and other applications</td>
<td>55</td>
</tr>
<tr>
<td>Total neutron flux in first 6 experimental positions</td>
<td>2.6 10&lt;sup&gt;15&lt;/sup&gt; n/cm&lt;sup&gt;2&lt;/sup&gt;/s</td>
</tr>
<tr>
<td>Fast neutron flux in first 6 experimental positions</td>
<td>4.2 10&lt;sup&gt;14&lt;/sup&gt; n/cm&lt;sup&gt;2&lt;/sup&gt;/s</td>
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</tbody>
</table>

Reactor vessel and internals

The revision 1.6 of the MYRRHA reactor consists of a pool-type ADS with the ability to operate in critical mode. Consequently, all the primary systems are housed within the reactor vessel. The reactor is located in the reactor pit, which features a liner able to serve as secondary containment in case of a reactor vessel leakage or break. The reactor cover closes the reactor vessel and supports all the components.

The core unit consists of the core support structure (CSS) integrated with the silicon doping structures. The CSS includes the core barrel and the shielding jacket; the above core structure, which includes the tube bundle with the 55 multifunctional channels; the core restraint system. The silicon doping structures include the silicon barrels and the core cradle.

The reactor core consists of 211 positions. Such positions can be filled with fuel assemblies, reflector assemblies, control rods, scram rods, the spallation target unit, in-pile sections, reflector assemblies, instrumentation and surveillance capsules.

Because of the many structures above the core, the fuel and reflector assemblies have to be loaded from underneath. Thanks to the buoyancy, the fuel and reflector assemblies do not need locking devices to hold them on the core support structure. However, in order to avoid any small position change of the fuel assemblies inside the core during operation, a core restraint system fixes the radial position of the fuel assemblies. This core restraint system acts on the lower end of the fuel assemblies. For refuelling this core restraint system is pushed out towards the reactor bottom.

The diaphragm separates the cold, high-pressure LBE from the hot, low pressure LBE and contains the in-vessel fuel storages and the pump casings. The diaphragm is connected, together with the cover, to the reactor vessel. The diaphragm consists of two horizontal plates connected to each other by vertical shells allowing some components to reach the lower plenum. Due to this design, several volumes exist in between the two horizontal plates. Two of these volumes house part of the pumps and the heat exchangers. Each pump casing lodges one pump and two heat exchangers. Four volumes are dedicated for the in-vessel fuel storage (IVFS). These storages are cooled during operation mainly by the forced circulation of LBE imposed by the pumps, while during shutdown cooling is ensured by natural circulation.

Some new components (with respect to Revision 1.4) have been introduced in the design of the reactor, namely the Failed Fuel Detection Device and the Fuel Transfer Device. The former is able to detect whether a fuel assembly placed in there has some failed fuel pins while the latter is dedicated to the transfer of fresh fuel assemblies into the reactor and the removal of the spent assemblies.
Besides the above mentioned systems, two redundant LBE Conditioning Systems (LBECS) are foreseen to filter and condition the LBE to the necessary levels to ensure the chemical protection of the reactor components. Coupled to this system are also the extraction pumps, whose task is to extract the LBE from surface of the hot plenum where most of the impurities are expected to gather. The LBE is then pumped towards the LBECS. A Primary Cover Gas and Ventilation System (PCGVS) is foreseen to monitor and filter the cover gas of the reactor.

In the design revision 1.6, a system dedicated to the mitigation of an overpressure event in the primary system is also implemented. This is the Pressure Relief System (PRS) which is placed outside the primary vessel and, through a rupture disk placed on the reactor cover, limits the pressure in the primary cover gas to the maximum admissible pressure of 6 bara.

Due to the new safety requirements, a system able to ensure cooling of the reactor in case of severe accident has been implemented in the reactor design. This consists of the reactor top cooling system and the reactor vessel auxiliary cooling system, which together perform the Severe Accident Cooling. The LBECS, PCGVS, PRS and the pit sealing systems are external systems to the reactor vessel.

**Heat exchangers**

The main thermal connection between the primary and the secondary system is provided by the Primary Heat Exchanger (PHX). According to the design specifications, the PHX must fulfil three main functional requirements:

- Normal operation mode: during normal operation, the PHX must be able to remove the power generated by the reactor core and by all the other heat sources (pumps, polonium decay, IVFS). It has been designed for 110% of the nominal core power, in order to take into account all the additional heat sources. In this condition, the PHX operates in forced circulation regimes on both sides (LBE and water).
• Decay heat removal (DHR) condition mode: in case of accidental situation, the whole reactor (primary, secondary and tertiary systems) must be able to operate in passive conditions (natural circulation) in order to guarantee the DHR function. The combined secondary and tertiary system assumes then the role of DHR-1 system.

• Maintenance mode: during shutdown periods, once the decay heat power is low enough to be compensated by the thermal heat losses through the reactor primary vessel, it is necessary to provide power to the primary LBE in order to prevent freezing. This can be done through the LBE conditioning system and/or by heating the secondary water with an external power source and then transferring power to the primary LBE through the PHX operating in a "reverse" mode.

The PHX design (see Figure 2) chosen for MYRRHA is a counter-current shell-and-tube concept consisting of:

• 684 stainless steel (AISI 316L) tubes with a wall thickness of 1 mm;
• 2 tube plates (thickness: 80 mm);
• a double-walled central feed-water pipe connected to the secondary system recirculation line;
• a double-walled bottom head, collecting feed water and connected to the tube bundle;
• a top head, providing connection with the riser pipe of the secondary system;
• an external shroud separating LBE in the hot plenum from LBE flowing in the PHX.

![Figure 6: The MYRRHA heat exchanger](image)

All metallic surfaces separating primary LBE by secondary water, with the exception of the tube bundle for heat transfer coefficient efficiency reasons, have a double-walled structure: as a consequence, the bottom head and the feed-water pipe are double walled, while the external shroud and the top head maintain a single wall structure because no risk of interaction of LBE and water is involved in case of failure.
LBE from the hot plenum (~325 °C) enters the PHX from the inlet openings in the external shroud. The flow is then directed downwards, through the tube bundle, where the actual heat exchange takes place. Outlet openings, directing the LBE flow towards the primary pumps, provides the exit path for the cold (~270 °C) LBE.

On the secondary side, water at a pressure of 16 bar at nearly saturated conditions (~200 °C) flows down the central down-comer pipe into the PHX bottom head and then upwards through the tubes where it is heated by the counter-current flowing LBE, thus producing a water steam mixture with a final quality of ~0.3.

**In-vessel fuel-handling machine**

Two in-vessel fuel-handling machines (IVFHM), installed permanently in the reactor, handle the loading and the unloading of the fuel assemblies to the in-vessel fuel storages, which are integrated into the diaphragm. Due to the chosen manipulator concept, the most compact design (Figure 3) is obtained by using two independent machines, each covering the half of the core and serving the half of the in-vessel fuel storage positions.

Fuel-handling tasks will be performed in a liquid LBE environment inside the reactor at 200 °C. During reactor operations the IVFHM will remain in its storage position inside the reactor. At this time the LBE temperature will be 270 °C. Consequently motors, position sensors and other electrical components will be positioned above the level of the LBE on the reactor cover. Rotation of the guide arm is made directly at the reactor cover. Similarly raising and lowering of the gripper and guide arms is actuated at the reactor cover level.

Fuel assemblies are retrieved by the gripper, gripper rotation is actuated via mechanical transmissions at the reactor cover. The gripper (E) is normally locked, unlocking is actuated passively as the guide arm and gripper arm are brought close together. The gripper incorporates 11 ultrasonic sensors used to identify the fuel assembly using coded castellation on the lower edge of the fuel assembly inlet nozzle.

The guide arm (C) serves to prevent neighbouring fuel assemblies being withdrawn as the target fuel assembly is extracted. The guide also supports the upper half of the fuel assembly during movements in the horizontal plane.

During normal operations a calibration module occupies the access port (I). This allows the condition of the gripper to be monitored covering: rotation torque and accuracy; insertion/extraction force and accuracy; and ultrasonic identification system verification.

During a recovery scenario, special tools can be passed from the reactor cover level to the gripper via the access port. Rotational and translational seals will be used to form a boundary between the IVFHM and the reactor cover; however the primary containment between the IVFHM penetrations in the reactor cover and reactor hall will be the IVFHM cover.

The gripper and guide arm mast (R) represent a long structure liable to deformation under loading. The mast support provides structural support via a sleeve with plain bearing.

The mast support (G) is statically fixed to the barrel. The grid forms a partial barrier in the IVFHM chimney at the diaphragm lower surface level. This helps prevent loose items potentially including a loose fuel assembly to migrate into the IVFHM chimney.

The grid is perforated and maintains a 50 mm gap between itself and the diaphragm. Together the perforations and the gap allow LBE to flow during pump start, pump shutdown or pump failure where the
LBE level in the IVFHM chimney will change. A minimum distance of 50 mm is maintained between the IVFHM and the diaphragm to prevent contact during installation, operations and accident transients.

Towards a new revision

The current "Release 1.6" has answered the difficulties encountered in the previous versions [5]; however the price to pay is a dramatic increase in size and weight of all components. As example

- The reactor vessel diameter increases from 8.2 to 10.4 m (which is a challenge both for manufacturing and transportation);
- The LBE inventory increases from 4500 to 7600 tonne, which means a non-negligible impact on the initial investment cost;
- The rooms and equipment in the reactor building are increased in size as well to be able to handle the larger components of the reactor.

The current MYRRHA design is based on the assumption there is no interaction between Po and water or steam. In case of a steam generator tube rupture during maintenance, steam can directly escape to the reactor hall. Some LBE aerosols could be swept away but the radiological impact remains limited. The assumption was based on existing literature [8] and agreed upon by scientific community.

An R&D programme was identified and performed to prove this assumption. Unfortunately, an increased Po-volatility was observed. The mechanism behind this is not yet clearly understood. As a consequence, a safety analysis was performed and indicated that the containment can hardly or cannot cope with the increased source term.

**Figure 8: In-vessel fuel-handling machine**

A. Plug  
B. Barrel  
C. Guide Arm  
D. Gripper Arm  
E. Gripper  
F. Cover  
G. Mast Support  
H. Sleeve Support  
I. Access Port  
J. Access Port Support  
K. Barrel Rotation Motor  
L. Arm Rotation Motor  
M. Guide Arm Linear Actuator  
N. Guide Arm Linear Axis Actuator  
O. Gripper Arm Linear Actuator  
P. Grid  
Q. Guide Arm Upper Linear Guides  
R. Gripper and Guide Arm Mast
As a conclusion from this safety study, it is proposed to exclude the steam generator tube rupture by design using a monitored double wall design of the heat exchanger tubes. The monitoring of the heat exchanger tubes state is implemented by continuous gas pressure monitoring (e.g. Helium) between the two tubes. In case of a crack in one of the walls, the pressure in the double wall changes, indicating an anomaly in the system. This design change will result, however, in a significant size increase of the heat exchangers.

On the other hand to mitigate corrosion, the majority of international scientific experiments were conducted with an oxygen concentration of $10^{-6}$ wt-% in LBE in flowing conditions and at a higher temperature than the MYRRHA conditions. Because of the low shutdown temperature of 200 °C of MYRRHA, only a maximum of $10^{-7}$ wt-% of oxygen is achievable.

This was considered sufficient to mitigate corrosion and an experimental programme was initiated to confirm this (a lack of relevant experimental results in international literature). Recent results on liquid-metal corrosion in MYRRHA relevant conditions show unexpected severe localised corrosion damages, which questions the selected ranges of dissolved oxygen concentration.

It was decided to relax the oxygen concentration to $7 \times 10^{-7}$ wt-% ($10^{-6}$ wt-% with a safety margin) which is more in line with the existing international experiments and experience and thus easier to validate. This higher oxygen concentration requests an increase of shutdown temperature to 250 °C. The maximum allowed temperature of 466 °C is not relaxed; in contrary, it will be reduced to 450 °C, based on Russian experience feedback.

The shutdown temperature increase and the small maximum temperature decrease will result in a power decrease and as a consequence in a performance loss. To counteract this performance loss, the heat exchanger will be increased in size.

Both R&D issues, the increased Po-volatility and the severe localised corrosion, can be avoided or mitigated by design but the consequence is the size increase of the primary heat exchangers, which will have an important impact on the reactor vessel dimensions. At the same time the existing "1.6" design already resulted in a cost increase that had to be optimised. The contradicting requirements, increase of the size of the heat exchangers and reduction of reactor dimensions, resulted in a broad analysis of options. We have decided to investigate the possibility to obtain more compactness in the double-walled heat exchangers and in the fuel-handling machine.

The MYRRHA Project management has therefore decided that “Release 1.6” would not be the final one and that at least one more iteration is needed.

The phased approach

The outcome of several technical meetings and project meetings, including an evaluation by a panel of experts, was a decision by the SCK•CEN Board of Governors at the end of 2015 to go forward using a phased approach. One of the key risk factors of the whole MYRRHA Project that was identified was the technical risk in reaching the reliability request on the accelerator. This high reliability comes from the fact that long beam trips of the accelerator induced large temperature gradients in the spallation target and reactor structures. As a consequence, the start-up after such a beam trip is a non-trivial and time consuming task.

The Board of Governors of SCK•CEN decided that the accelerator reliability has to be demonstrated before a decision can be taken on the full project, hence a phased approach. In a first phase a part of the MYRRHA accelerator will be built and commissioned: a single injector (the MYRRHA accelerator has two injectors to improve the reliability; whether this will remain needed shall be the outcome of this first phase) coupled to a 100 MeV accelerator part.
In this first phase, we will focus on four items: accelerator design and R&D, target station design and R&D, accelerator balance of plant and licensing. Indeed, as this first accelerator part will deliver protons at interesting energies and currents, the intention is to construct a target station to make optimal use of these protons for science and engineering. In the coming years 2016-2022, this 100 MeV accelerator plus target station project will be designed, a licence file will be drafted and construction will start. The current planning is to start commissioning of this first phase accelerator in 2022 with full operation and demonstrated reliability in 2024.

A positive outcome of phase 1 will allow us to launch the construction of the 600 MeV accelerator part. This will be further subdivided into the design and R&D part for the 600 MeV accelerator and the 600 MeV accelerator balance of plant activities. The existing 100 MeV accelerator will be, of course, the initial part of the 600 MeV accelerator. The R&D required for this 600 MeV accelerator will be done in the years 2016-2024, in order to be ready for construction in 2025 after a green light of phase 1.

Finally, phase 3 will be the construction of the reactor part of MYRRHA and the coupling of the 600 MeV to this accelerator in order to complete the full MYRRHA Project. Also here, the objective by 2024 will be to be in a position to launch the construction of the reactor part. Therefore, the required R&D, design work, balance of plant and pre-licensing processes will continue from now up to 2024.

Conclusions

SCK•CEN is proposing to replace its ageing flagship facility, the material testing reactor BR2, by a new flexible irradiation facility, MYRRHA. Considering the international and European needs, MYRRHA is conceived as a flexible fast-spectrum irradiation facility able to work in both subcritical and critical mode.

Despite several non-obvious design challenges e.g. linked to the use of LBE, the new safety requirements in terms of seismic loading (consequence of Fukushima) or the choice of passive mode for decay heat removal in emergency conditions, we found no significant showstopper during the design development.

The mechanical design of the reactor has evolved with the different revisions. We currently are focusing in obtaining a more compact design by studying innovative components as double-wall heat exchangers and fuel-handling machines.

In order to reduce the financial and engineering risk, the SCK•CEN Board of Governors decided on a phased approach: in a first phase, the feasibility and reliability of a 100 MeV proton accelerator section should be demonstrated. Upon successful demonstration, the extension of the proton accelerator to 600 MeV and the construction of the reactor part can be started.

Acknowledgements

The basis for the current MYRRHA design has been done under two projects co-funded by the European Union, respectively the Integrated Project EUROTRANS (Ref. FI6W-CT-2004-516520, 6th Framework Programme) and the Collaborative Project CDT (Reference FP7-232527, 7th Framework Programme). Acknowledgement is also due to all the colleagues of the participant institutes for their contributions in many different topics associated with the XT-ADS and FASTEF design and operation.
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Design characteristics and research progress of China lead-based research reactor Clear-I

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Abstract

Chinese Academy of Sciences (CAS) launched an engineering project to develop an accelerator-driven subcritical system (ADS) for nuclear waste transmutation since 2011. China LEAd-based Reactor (CLEAR), proposed by Institute of Nuclear Energy Safety Technology (INESST), was selected as the ADS reference reactor, as well as for the technology development of the Generation IV lead-cooled fast reactor. In China ADS programme, CAS plans to develop ADS and the lead-based reactors through 3 stages. The first stage is ADS research facility, which consists of a 10MWth Pb-Bi cooled research reactor CLEAR-I, coupled with a proton accelerator (250MeV/10mA).

CLEAR-I was designed as a flexibility test platform for different operation mode and different nuclear fuel testing. So CLEAR-I had the critical operation capability without external spallation neutron source. According to the experimental object and implementation procedure, the deeply subcriticality and low power mode will be operated on the first step, and then the power will be increased by adding fuel assembly or increasing proton intensity step by step. The design technique principle including technology feasibility, safety reliability, experiment flexibility and technology continuity was performed in the CLEAR-I design procedure. The experiment functions including neutronics, thermal hydraulics, material and key components test can be conducted in this system, to provide the experience of design, construction and operation for ADS and lead-bismuth cooled reactor.

In this contribution, the design description and progress of CLEAR-I was presented including reactor core design, fuel assembly design, thermal-hydraulics design, reactor structure design, key components design and engineering safety features, etc.

Keywords: Lead-bismuth, Research reactor, Accelerator-driven system,
Investigation for subcriticality adjustment mechanism of LBE-cooled accelerator-driven system

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Abstract
The Japan Atomic Energy Agency (JAEA) has investigated 800-MWth lead-bismuth eutectic (LBE) cooled Accelerator-driven system (ADS) to transmute minor actinides (MA). One of the most important issues for the ADS development is the design of a beam window which is a boundary of the accelerator and the subcritical reactor. In the previous study, a feasible beam window concept, an ellipse shape with 2 mm thickness at the top, is presented. However, it is supposed that the ellipse shape beam window with thin thickness and high precision is difficult to fabricate. Thus, a hemispherical shape beam window is considered as a new concept for JAEA’s ADS. To realise the hemispherical shape beam window, it is required to decrease the maximum proton beam current. For this purpose, a new concept of subcriticality adjustment mechanism using control rod (CR) or burnable poison (BP) is investigated. The results of neutronics calculation show that the concepts with boron carbide (B$_4$C) and tantalum (Ta) CRs have good property to adjust the subcriticality. They have a possibility to maintain the proton beam current at 10 mA during the operation, which is a great benefit for the beam window design since the proton beam current becomes half of the previous design. However, it is required to consider CR drive mechanism and related apparatus. For the concept with BP, it is observed that the maximum proton beam current is 17 mA. This value is better than the reference case. Moreover, the BP concept is achieved by just introducing BP without driver mechanism. These concepts are useful to adjust the subcriticality and mitigate the design condition of the beam window.

Introduction
In order to transmute minor actinides (MAs) partitioned from the high-level waste, the Japan Atomic Energy Agency (JAEA) has investigated an accelerator-driven system (ADS), lead–bismuth eutectic (LBE) cooled subcritical reactor with 800 MW thermal power [1-3]. As a proton accelerator for the ADS, 1.5 GeV-30 MW linear accelerator (LINAC) is supposed to generate the spallation neutrons in the reactor. The proton beam is injected to the LBE spallation target at the centre of the subcritical core, where the spallation neutron source is produced. The core is able to transmute almost 250 kg MAs per year. It means that the transmuted amount corresponds to the amount discharged from 10 units of the light-water reactor with 1 GW electric power and 45 GWD/tHM burnup. In the JAEA’s ADS, as the effective multiplication factor ($k_{eff}$) with burnup, the burnup reactivity changes almost 3%. To keep the constant thermal power, it will be required to adjust the LINAC proton beam current to compensate the burnup swing. However, the increasing beam current puts an extra load on a beam window. In the previous study
[4], a feasible beam window concept which is an ellipse shape with 2 mm thick at the top is presented. However, it is supposed that the ellipse shape beam window with thin thickness and high precision is difficult to fabricate. Thus, a hemispherical shape beam window is considered as a new concept for JAEA’s ADS. To realise the hemispherical shape beam window, it is required to decrease the maximum proton beam current. The purpose of this study is to discuss a conceptual design of the LBE-cooled ADS using a subcriticality adjustment mechanism to reduce thermal load on the beam window. As the subcriticality adjustment mechanism, two types of concept, control rod (CR) and burnable poison (BP), are investigated.

**Investigation method and numerical analysis condition**

In the present study, calculation is made by three-dimensional reactor analysis code system for ADS, ADS3D [5], which is based on the multipurpose reactor analysis code system, MARBLE [6], with a 70 energy-group constant set based on the Japanese Evaluated Nuclear Data Library, JENDL-4.0 [7]. The calculated values of $k_{eff}$, proton beam current required to keep the 800-MWth, and peaking factor are compared between three different cases; reference, CR and BP cases and trade-off between the CR and BP cases are investigated. Here, the peaking factor is the peak-to-average ratio of the power density produced by fission.

Table 1 and Figure 1 show characteristics and an assembly arrangement of the reference ADS, respectively. In the CR case, we investigate two material types of control rods; boron carbide ($B_4C$) and tantalum (Ta). On the other hand, in the BP case, we choose a hydrides (H) of zirconium (Zr) - gadolinium (Gd) alloy (H-Zr-Gd) as the BP material. The $B_4C$ and Gd are widely used in nuclear reactor as the CR and BP, respectively. Moreover, CR made by Ta is used in a UK-built prototype fast reactor, PFR [8]. Figures 2 and 3 show the loading position of CR and BP assemblies, respectively. The fuel assemblies replaced by CR or BP ones are reloaded into the outermost fuel region in order to keep the number of fuel assemblies with reference case.

**Table 1: Characteristics of the reference ADS**

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Proton accelerator beam energy</td>
<td>1.5 GeV</td>
</tr>
<tr>
<td>Spallation target material</td>
<td>LBE</td>
</tr>
<tr>
<td>Coolant</td>
<td>LBE</td>
</tr>
<tr>
<td>$k_{eff}$ at BOC$^1$</td>
<td>0.97</td>
</tr>
<tr>
<td>Thermal power</td>
<td>800 MWth</td>
</tr>
<tr>
<td>Loaded MA mass at BOC$^1$</td>
<td>2.5 t</td>
</tr>
<tr>
<td>Fuel composition$^2$</td>
<td>(MA + Pu) N + ZrN</td>
</tr>
<tr>
<td>Operation period</td>
<td>600 EFPDs$^3$</td>
</tr>
<tr>
<td>Number of fuel assemblies</td>
<td>84</td>
</tr>
<tr>
<td>Number of fuel pins per assembly</td>
<td>391</td>
</tr>
</tbody>
</table>

In the CR case, Loading region and the number of CRs are determined so that the whole rod worth does not exceed $3\%\Delta k/k$ to prevent a criticality accident as shown in Table 2. As shown in Figure 2, the Ta assemblies are placed closer to the spallation target than the $B_4C$ ones because the capture cross section of Ta is smaller than the $(n,\alpha)$ cross section of B. Furthermore, the weight ratio of Zr and Pu in the MA fuel region is adjusted so as to set $k_{\text{eff}} = 0.97$ at beginning of cycle (BOC). Tables 3 shows the weight ratios of Zr and Pu in the MA fuel region of CR and reference cases. Moreover, Figure 4 shows the operation scheme for the CR in this calculation. It is assumed that the all CRs are fully inserted at the BOC and they are pulled out with a constant speed of 200 mm/100 days as the burnup proceeds.

Table 2: CR loading region, number of assemblies and whole rod worth

<table>
<thead>
<tr>
<th>Material</th>
<th>Loading region</th>
<th>Number of assemblies</th>
<th>Whole rod worth</th>
</tr>
</thead>
<tbody>
<tr>
<td>$B_4C$</td>
<td>Third ring</td>
<td>3</td>
<td>1.66%</td>
</tr>
<tr>
<td>Ta</td>
<td>Second ring</td>
<td>3</td>
<td>1.31%</td>
</tr>
</tbody>
</table>
Table 3: Weight ratios of Zr and Pu in MA fuel region of Reference and CR cases

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Reference case</th>
<th>B4C case</th>
<th>Ta case</th>
</tr>
</thead>
<tbody>
<tr>
<td>ZrN / (ZrN + PuN + MAN)*</td>
<td>0.388</td>
<td>0.322</td>
<td>0.331</td>
</tr>
<tr>
<td>PuN / (PuN + MAN)*</td>
<td>0.259</td>
<td>0.223</td>
<td>0.232</td>
</tr>
</tbody>
</table>

*: Weight ratio

Figure 4: Operation scheme for CR.

In the calculation of the BP concept, temporal behaviours of k_{eff}, proton beam current and peaking factor with respect to H content in the H-Zr-Gd, whose calculation conditions, case1, 2 and 3, are shown in Table 4.

Table 4: Calculation condition of BP cases

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Case1</th>
<th>Case2</th>
<th>Case3</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of assemblies</td>
<td>6</td>
<td>←</td>
<td>←</td>
</tr>
<tr>
<td>Loading region</td>
<td>2nd</td>
<td>←</td>
<td>←</td>
</tr>
<tr>
<td>H / (Zr + Gd)*</td>
<td>0.8</td>
<td>1</td>
<td>1.2</td>
</tr>
<tr>
<td>Gd / Zr*</td>
<td>0.050</td>
<td>←</td>
<td>←</td>
</tr>
<tr>
<td>ZrN / (ZrN + PuN + MAN)*</td>
<td>0.314</td>
<td>0.308</td>
<td>0.302</td>
</tr>
<tr>
<td>PuN / (PuN + MAN)*</td>
<td>0.259</td>
<td>←</td>
<td>←</td>
</tr>
</tbody>
</table>


Results and discussion

A concept of ADS with control rod

Figure 5 compares time evolution of k_{eff} between the reference and CR cases. The horizontal and vertical axes indicate the operation time in effective full-power days (EFPDs) and the k_{eff} values, respectively. In the reference case, the k_{eff} decreases as the burnup proceeds and becomes 0.945 at the end of cycle (EOC). On the other hand, in both B4C and Ta cases, the k_{eff} is maintained almost 0.97 from BOC to EOC. As described above, it is assumed that the CRs are drawn out constantly as 200 mm/100 day in this calculation. In the real operation, it is able to fine-tune the position of CRs. Thus, in the CR case, it is able to keep k_{eff} as 0.97 while operating the ADS.
Figure 5: Time evolution of $k_{\text{eff}}$ of the reference and CR cases

![Figure 5](image1)

Figure 6: Time evolution of proton beam current required to keep 800-MW thermal power of the reference and CR cases

![Figure 6](image2)

Figure 6 shows time evolution of the proton beam current required to keep the thermal power of 800 MW. In the reference case, the beam current increases from 10 mA at BOC to 18 mA at EOC as the burnup proceeds. In contrast, the concepts with CR cases succeed to suppress the increase of beam current for both the B$_4$C and Ta cases. Here, the beam current in the Ta case is larger than that in the B$_4$C case. This reason is that the Ta assemblies are closer to the spallation target than the B$_4$C ones as shown in Figure 2. It means that the Ta CR has more amount of neutron absorption the B$_4$C CR. Thus, to keep the 800-MW$_{th}$, the spallation target in the ADS with Ta CR requires to generate more neutrons than that with B$_4$C CR. Therefore, the proton beam current in the Ta case is larger than that in the B$_4$C case.

Figure 7 shows the time evolution of the peaking factors during the burnup. Since importance of spallation neutrons increases with increasing the proton beam current, the peaking factors also increase. Therefore, the temporal behaviour of the peaking factors are similar to that of the beam current. It can be seen from Figure 7 that, at BOC, the peaking factors of the CR cases are larger than that of the reference case. This is attributable to the decreased power density around the peak position by inserting
the CR assemblies. Also, it can be observed that the peaking factor of B$_4$C is larger than that of Ta at BOC. This is due to the larger $(n, \alpha)$ cross section of B than the capture one of Ta.

**Figure 7: Time evolution of peaking factors of the reference and CR cases**

![Graph showing time evolution of peaking factors.]

**A concept of ADS with burnable poison**

Figure 8 shows the time evolutions of $k_{\text{eff}}$ of the reference and BP cases. The time-step mesh of BP case is more finely than that of CR case because the reduction process of BP material by burnup is required to be taken into consideration. The $k_{\text{eff}}$ of BP case is closer to 0.97 than that of reference case at BOC. Since BP is able to reduce the initial excess reactivity, much fuel can be loaded than the reference case as shown in Tables 3 and 4. After the burn out of BP, a peak in the middle of cycle (MOC) appears immediately. As the H content in the H-Zr-Gd increases, the burnup reactivity between BOC and EOC becomes smaller. Moreover, the peak appearing in the MOC becomes larger and approaches the BOC as the H content increases. The reason is that the increase of the H content indicates softer neutron spectrum in the reactor. In such situation, a time of BP decreasing becomes shorter because the capture reaction rate of Gd becomes larger. Additionally, in the ADS with BP (1.2) case, the $k_{\text{eff}}$ peak reaches 0.98. If the case which the H content is 1.2% would be employed, the $k_{\text{eff}}$ at BOC should be adjusted to less than 0.96 from a criticality safety perspective. However, the proton beam current and peaking factor would become worse in that case.

**Figure 8: Time evolution of $k_{\text{eff}}$ of the reference and BP cases**

![Graph showing time evolution of $k_{\text{eff}}$.]
Figure 9 shows the time evolution of the proton beam current required to keep 800-MW$_{th}$ of the reference and BP cases. At the BOC, the beam currents of BP cases are larger than that of reference case because BP absorb the spallation and fission neutrons. After the burn out of BP, the beam currents of the BP cases are decreased and smaller than the reference case’s one. This reason is that the proton beam current depends on the $k_{eff}$. The $k_{eff}$ is increased after the burn out of BP as shown in Figure 8, and it means that the number of fission reaction is increased. Thus, the beam current is decreased in the BP cases after the burn out of BP. Furthermore, when the H content is increased, the beam current is decreased after the burn out of BP. It is because that more fuel material is loaded as the H content increase. The maximum beam current is almost 17 mA in the BP cases.

Figure 10 shows the time evolution of the peaking factors of the reference and BP cases during the burnup. For the same reason as the CR case, the temporal behaviour of the peaking factors indicates the similar tendency to that of the beam currents. As can be seen in Figure 10, the peaking factors of the BP cases are larger than that of the reference case at BOC. This is caused by the decreased power density around the peak position by inserting the BP assemblies.
Trade-off investigation

Table 5 summarises the investigation of the trade-off between CR and BP cases.

<table>
<thead>
<tr>
<th>Term</th>
<th>ADS with CR</th>
<th>ADS with BP</th>
<th>Trade-off</th>
</tr>
</thead>
<tbody>
<tr>
<td>Subcriticality adjustment</td>
<td>Good accuracy.</td>
<td>Better than reference case.</td>
<td>Operation</td>
</tr>
<tr>
<td>Max. beam current</td>
<td>10 mA (B$_4$C), 13 mA (Ta)</td>
<td>17 mA</td>
<td>Operation</td>
</tr>
<tr>
<td>Reactivity control</td>
<td>It is required to consider incorrect operation.</td>
<td>Simple. Just install it to the core.</td>
<td>Operation</td>
</tr>
<tr>
<td>Incidental equipment</td>
<td>Additional equipment is required: CR driving system, CR changer, etc.</td>
<td>Simple. Fuel changer can be used to change BP assemblies.</td>
<td>Cost</td>
</tr>
<tr>
<td>Safety design</td>
<td>It is required to take into consideration the reactivity insertion by incorrect drawing out.</td>
<td></td>
<td>Safety, Operation</td>
</tr>
<tr>
<td>Effect on the other equipment</td>
<td>It is required to put a restriction on fuel exchange procedure.</td>
<td>Nothing.</td>
<td>Cost, Operation</td>
</tr>
<tr>
<td>Maintenance</td>
<td>Regular check is required for CR driving system.</td>
<td>Maintenance-free.</td>
<td>Operation</td>
</tr>
<tr>
<td>Replacement (2 year = 1 cycle)</td>
<td>3 assemblies/cycle</td>
<td>6 assemblies/cycle</td>
<td>Operation</td>
</tr>
<tr>
<td>Fabrication of associated structure</td>
<td>It is required to give T91 fabrication technology examination.</td>
<td></td>
<td>Fabrication technology</td>
</tr>
<tr>
<td>Development test</td>
<td>There are several issues.</td>
<td>Nearly omissible.</td>
<td>Cost</td>
</tr>
</tbody>
</table>

We discuss 10 terms; subcriticality adjustment, maximum beam current, reactivity control, incidental equipment, safety design, effect of the other equipment, maintenance, replacement, production and development test. In the CR case, it is possible to reduce the beam current from 20 mA to 10 mA (B$_4$C) or 13 mA (Ta). However, there are the various issues in the CR case. For example, the additional equipment, maintenance and development test are required. For the development test of CR, the following issues are supposed; durability for CR driving system, LBE flow rate control, corrosion by LBE, earthquake resistance. On the other hand, the BP case is simple approach to control the subcriticality because the geometries of cladding tube and assembly for BP are same as the fuel ones. However, the fine adjustment of subcriticality using BP is more difficult than that using CR.

Summary

The new ADS with CR or BP, are investigated to reduce thermal load on the beam window. The concepts with CR/BP have a possibility to reduce the proton beam current from 20 mA to 10/17 mA, respectively. Therefore, the subcriticality adjustment mechanism can mitigate the design criteria of beam window. However, both concepts have their own issues. In the CR case, the additional equipment, maintenance and development test are required. For the development test of CR, it is not argued deeply here, but the following problems are supposed; durability for CR driving system, LBE flow rate control, corrosion by LBE, earthquake resistance and so on. If BP concept would be chosen, the further optimised design would be required. Moreover, the H content in the H-Zr-Gd BP has a large effect on $k_{eff}$, proton beam current and peaking factor. From the criticality safety perspective, it is important to take particular care to design the subcriticality adjustment mechanism by BP including H.
Acknowledgements

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References


Study of reactivity control method for accelerator-driven system by the use of burnable poison oxide

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Abstract
The present study is intended to reveal the applicability of burnable poison oxide to the reactivity control for accelerator-driven system (ADS) core based on the analysis results of the metallic hydride burnable poison-loaded ADS core. The target ADS design is 800 MWe lead-bismuth-cooled ADS proposed by JAEA. The reactivity control material is composed of the mixture of gadolinium oxide and moderator as beryllium oxide or spinel (MgAl2O4) and the shape of the control material for the core loading is assumed assembly-wise hexagonal block. The analysis results show that burnable poison oxide is able to use as the reactivity control material, and beryllium oxide is better as a moderator than spinel. The effective reactivity control is possible by the optimisations of the composition in the control block and the number of control block. As for the transmutation of minor actinide, the performance of the oxide is almost the same as the reference core of no control material loading, and superior to that of hydride.

Introduction
Accelerator-driven system (ADS) has been widely studied for the purpose of the reduction of the disposal of high-level radioactive waste by transmuting minor actinide and long-lived fission product. Research and development of ADS has been intensively performed to address various issues. The durability of the in-core structure materials is one of the concerns. The steady rated power operation is maintained in ADS by the increase of the beam current to compensate the burnup reactivity loss, and the boundary between the target surface and beam duct, “beam window”, is exposed to the more and more irradiation damage as the fuel burnup progresses. The reactivity control of an ADS is the promising method to reduce the beam current. Tohoku University has been studied the reactivity control method by the application of burnable poison (BP) to ADS core [1] [2]. The past studies showed that the effective reactivity control is possible by introducing the control block made of gadolinium hydride and deuteride. However, metallic hydride and also deuteride have the inherent characteristic that metallic hydride and deuteride dissociate hydrogen and deuterium in high-temperature environment, and it leads to the positive reactivity insertion and the critical accident. The control material with good chemical stability in high-temperature condition is desired, and the BP oxide is considered one of the candidate due to the better characteristic in high-temperature condition than that of the hydride and deuteride. The purpose of the present study is to examine the feasibility of BP oxide for ADS core. The next section reviews the outline of past studies.
of metallic hydride and deuteride-loaded ADS core design, and then, the BP oxide-loaded ADS core is designed. Finally, this paper is concluded.

Review of past study of BP-loaded ADS core

The reference core design employed in the past studies described below is 800 MWt class lead-bismuth cooled ADS core proposed by JAEA [3] [4]. Figure 1 shows the schematic view of the core, and the basic specifications are presented in Table 1. The innermost fuel assembly zone is called “Zone 1” and the outermost one is “Zone 4”. The past studies had performed the various optimisations of BP control block. We review the outline of the design study in the following.

<table>
<thead>
<tr>
<th>Plant / Fuel</th>
<th>Thermal Power</th>
<th>800 MWt</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial k&lt;sub&gt;eff&lt;/sub&gt;</td>
<td></td>
<td>0.97</td>
</tr>
<tr>
<td>Fuel</td>
<td>(Pu + MA) + ZrN</td>
<td></td>
</tr>
<tr>
<td>Coolant and Target</td>
<td>Lead-bismuth eutectic</td>
<td></td>
</tr>
<tr>
<td>Coolant inlet temperature</td>
<td>300 °C</td>
<td></td>
</tr>
<tr>
<td>Coolant velocity</td>
<td>2.0 m/s</td>
<td></td>
</tr>
<tr>
<td>Fuel Assembly pitch</td>
<td>233.9 mm</td>
<td></td>
</tr>
<tr>
<td>Flat to Flat</td>
<td>232.9 mm</td>
<td></td>
</tr>
<tr>
<td>Active fuel length</td>
<td>1000 mm</td>
<td></td>
</tr>
<tr>
<td>Pu ratio in (Pu + MA)N</td>
<td>36.2 vol%</td>
<td></td>
</tr>
<tr>
<td>ZrN ration of reference core fuel</td>
<td>67.1 vol%</td>
<td></td>
</tr>
</tbody>
</table>
Design study of gadolinium hydride-loaded ADS core

The applicability of gadolinium hydride as the burnup reactivity control material to ADS core was investigated in Ref. 1. The structure of the BP control block is based on that of fuel assembly and the block is composed of gadolinium-zirconium mixed hydride (Gd-hydride) pin and zirconium hydride (Zr-hydride) pin. An example of the control block is shown in Figure 2. The optimal specification was surveyed through the various optimisations such as a loading position, the position of Gd-hydride pin in a control block, the position of the number of control block, the number of Gd-hydride pin and the composition of Gd-hydride pin. The suitable specifications of the control block were as follows in terms of the reactivity control and safety parameters:

- pin pattern: Gd-hydride pins at outer rim of the BP assembly (other part is filled with Zr-hydride);
- loading position of BP assembly control block: Zone 2;
- number of the control block: six or nine (located at equal intervals);
- Gd ratio in Gd-hydride pin: 15-25% (changing by the number of Gd-hydride pin).

The burnup reactivity loss ranged from 0.75%\(\Delta k/k\) to 1.7%\(\Delta k/k\) with above block specifications, whereas the loss in the reference core was 4.3%\(\Delta k/k\). The issue occurred by the loading of BP hydride control block is the decrease of the amount of MA transmutation. The difference of MA transmutation amount between the BP hydride-loaded core and the reference core was about 30–40 kg because Pu burnup was relatively increased especially around the end of cycle by the moderated neutron generated by the hydride.

![Figure 2: Structure of BP hydride control block.](image)

Design study of ADS core with gadolinium hydride and deuteride

The study of reactivity control by the use of gadolinium deuteride was performed in Reference 2 to accomplish better reactivity control and the increase of MA transmutation amount. The purpose of the utilisation of metallic deuteride was to delay the burnup speed of gadolinium and to increase the transmutation amount of MA because the moderating power of deuterium is smaller than that of hydrogen and the decrease of relative Pu-burnup amount is expected. The structure of BP control block was based on the design in Reference 1. Six control block were located at Zone 2, and 66 Gd-deuteride pins with 20% gadolinium ratio were located at the outermost row in the control block as seen in Figure
2. The inner pins in the control block were composed of Zr-hydride and deuteride pins, and the numbers of Zr-hydride and deuteride pins were changed to control the moderation power of the block for better reactivity control. When all of hydrogen in Gd-hydride were substituted for deuterium, the reactivity control is less effective due to the weak moderating power. The combination of Zr-hydride and Zr-deuteride was necessary to achieve reactivity control and gadolinium burnup speed was slowed down. The time variations of k-effective with various control block configurations are compared in Figure 3. The burnup reactivity loss became 1.6%Δk/k at most with the use of Zr-deuteride inside of the 2 rows in the control block assembly. Almost the same amount of MA transmutation compared to that of reference core was possible to increase the number of metallic deuteride. However, the ability to control the reactivity tended to be small as the number of metallic deuteride increased. The control block with the use of Zr-deuteride inside of the 8 rows had almost the same transmutation amount of MA but the burnup reactivity loss was 2.2%Δk/k. Reference 2 concluded that the reactivity control was enabled by the employment of metallic deuteride but the burnup reaction loss became larger than that of metallic hydride and a certain amount of increase of the reactivity loss was necessary to improve the transmutation amount of MA.

![Figure 3: Time variation of k-effective in BP control block-loaded ADS core](image)
Three conditions are the same in the control block: 20% gadolinium ratio in Gd-hydride or – deuteride pin, 66 Gd-hydride or – deuteride pins at outer rim of the control block, and 6 control block in the core

**Design of BP oxide-loaded ADS core**

**Design procedure of ADS core**

The basic ADS core model employed in the present study was 800MWt lead-bismuth cooled ADS core which was the same in the past studies mentioned above. The material in the control block was composed of gadolinium oxide (Gd₂O₃) and the oxide moderator as beryllium oxide (BeO) or spinel (MgAl₂O₄). Beryllium oxide and spinel was being investigated as the candidate of moderator for the MA transmutation in a fast reactor core [5]. The structure of BP oxide control block assumed in the present study was assembly-wise hexagonal block, and the mixture of gadolinium oxide and a moderator was shaped into the hexagonal block with a 90% theoretical density. The pin-structural control block was not adopted in the present study because the moderation effect by the oxide moderator was weaker than that by hydride. The heat conductivity of BeO or MgAl₂O₄ was high enough that the small heat generation by the absorption of neutron in gadolinium was removed easily.
BP oxide-loaded ADS core was designed through two optimisations of the composition of the control block, the number of control block. The composition of the control block was changed by the ratio of gadolinium oxide and oxide moderator, and the ratio of gadolinium oxide was 0–20%. The gadolinium hydride block with assembly-wise hexagonal block shape was also investigated as a comparison core design. The loading zone of control block was fixed at Zone 2 on the basis of the earlier studies, and the number of the block was 3, 6 or 9. We evaluate the feasibility of the BP oxide-loaded core by the variation of k-effective and the burnup reactivity loss. The initial k-effective was set to 0.97, and therefore, k-effective had to be below 0.97 in the burnup period of 600 days. The inert matrix ratio in the fuel was changed to adjust the initial k-effective to 0.97. Finally, the core characteristics such as burnup characteristic and reactivity coefficients related to safety were surveyed on the optimal core design. A series of analyses was conducted by the use of MVP [6] and MVP-BURN [7] with the nuclear data library JENDL-4.0 [8].

Optimisation of BP composition and the number of assembly

The Gd content in the control block was set roughly to 0, 3, 5, 10 and 20%. The number of control block was 3, 6 and 9 at each analysis with various composition. Figures 4 and 5 show an example of the time variation of k-effective with BP oxide control block. These were the case of 6 BP oxide-loaded ADS core. The burnup reactivity loss decreased as Gd ratio became smaller except in the 0%-Gd-case, and the block using beryllium oxide showed good performance relatively while the block with spinel is less effective. This difference was resulted from the moderation power. As shown in Figure 6, the moderation power affected the neutron absorbability by gadolinium and finally, the variation of k-effective. The moderation power of spinel was so weak that spinel was found not to be suitable for a moderator in the ADS core, and beryllium oxide was selected as the best candidate for the moderator. As concerns the dependency of the number of control block (see Figure 7), the core with 3 control block was not effective for the reactivity control. The detailed surveys of the composition of the block were performed in the following with 6 and 9 control block-loaded core.

**Figure 4: Time variation of k-effective with 6 BP control block of beryllium and gadolinium oxide**
Figure 5: Time variation of k-effective with 6 BP control block of spinel and gadolinium oxide

Figure 6: Time variation of k-effective with 6 BP control block with 3% gadolinium oxide
The gadolinium ratio in the control block with gadolinium oxide moderator was changed from 0% to 5% for the detailed survey. Figures 8 and 9 show the results of k-effective variation with 6 and 9 control block-loaded core. The time variation of k-effective tended to flatten as gadolinium ratio decreased but in the case with the 1% Gd ratio, the k-effective started to increase from 50 days and exceeded 0.97, which was the design criterion in this study. The minimum burnup reactivity loss was obtained in the case of block with 2% gadolinium, and the reactivity losses were 1.7%Δk/k and 1.0%Δk/k with 6 and 9 blocks, respectively. As seen in the temporal transition of BP control block worth per block shown in Figures 10 and 11, the decrease of the worth was closely related to k-effective. The k-effective tended to decrease monotonically in the case with small worth change such as 3% and 5% gadolinium ratio. The k-effective variation showed wavy shape when BP control block worth changed noticeably. The appropriate content of gadolinium in the control block had a certain level of effect on the reactivity control, and by the optimisation of the BP control block composition, very effective reactivity control was possible. Besides, BP oxide control block had a potential to achieve the better control of the burnup speed of gadolinium than that by the hydride block. In a series of analysis, 2% gadolinium ratio was considered to be well-balanced composition in the moderation and the absorption by gadolinium. The core characteristics were analysed for the core loading 6 and 9 BP oxide control block with 2% gadolinium oxide in the next subsection.
Figure 8: Time variation of k-effective with 6 BP control block with 0-5% gadolinium oxide

Figure 9: Time variation of k-effective with 9 BP control block with 0-5% gadolinium oxide
Core characteristics of BP oxide-loaded ADS core

The core characteristics were analysed to make clear the influence of BP oxide loading. Doppler coefficient, coolant void reactivity and peaking factor were calculated as the representative safety parameter of the core. The transmutation characteristic was also calculated because the main role of ADS is TRU burner. Doppler coefficient was calculated by the use of k-effective in the normal operational condition and the state with 500 K increase from normal. In the calculation of coolant void reactivity, coolant was assumed to be voided in the fuel and whole core regions. The calculated characteristics were presented in Table 2.

The Doppler coefficients of BP oxide-loaded core were smaller than the reference core. However, the coolant void reactivities became worse than those of the reference core. The fuel region voided
coolant void reactivity was positive at each core models but those of BP oxide-loaded core was more than double than that of the reference core; the whole core voided reactivity was negative at all cases but those with BP oxide became from 1/3 to 1/5. The peaking factor was also worsened with the use of BP oxide due to the decrease of the inert matrix ratio. The tendency of the worsening of safety parameters was similarly seen in the BP hydride-loaded core. They were considered to result from the introduction of a moderator in a core and the change of fuel composition. The transmutation characteristic was improved by the use of BP oxide as compared to that of BP hydride. The total amount of MA transmutation became about 570 kg in BP oxide loading whereas that in the reference core was 524 kg and that in BP hydride loading was smaller by 30-40 kg from the reference core. The initial loading amount of MA and TRU in BP oxide-loaded core was larger than the reference core, but the transmutation efficiency was almost the same as the reference core’s one whereas the efficiency in BP hydride core was worse than that of the reference core.

The differences in the core characteristics among the core models discussed in this section came from the difference of the neutron spectrum. Figure 12 shows the neutron spectra of various core models. The spectra of BP control block were the spectra in a gadolinium compound. The trends of safety parameters in BP-loaded core were almost the same, but the difference of the spectra was considered to affect the burnup characteristic. The spectra of BP oxide were decreased gradually from 1 eV to 1 MeV and decreased rapidly below 1 eV; that of BP hydride was smaller in the range from 1 eV to 1 MeV and the slope below 1 eV is smaller than that of the oxide. The moderation power of the oxide is smaller than that of the hydride, and less thermal neutrons were generated in the oxide. The difference of the spectra affected the plutonium burnup characteristics in particular. Namely, the larger number of neutrons was consumed by plutonium in BP hydride-loaded core. Finally, these differences led to the transmutation characteristic.

### Table 2: Core characteristics of BP oxide control block-loaded ADS core

<table>
<thead>
<tr>
<th>Core type</th>
<th>BP oxide-loaded</th>
<th>BP hydride loaded*</th>
<th>reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of control block</td>
<td>6</td>
<td>9</td>
<td>6</td>
</tr>
<tr>
<td>Inert Matrix ratio [vol%]</td>
<td>59.4</td>
<td>55.2</td>
<td>57.6</td>
</tr>
<tr>
<td>Doppler coefficient [×10^{-4} Tdk/dT]</td>
<td>-6.75</td>
<td>-4.82</td>
<td>-4.08</td>
</tr>
<tr>
<td>Coolant void reactivity [×10^{-2} Δk/k]</td>
<td>0.642</td>
<td>0.824</td>
<td>0.669</td>
</tr>
<tr>
<td>Fuel region</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Whole core</td>
<td>-2.94</td>
<td>-2.40</td>
<td>-8.26</td>
</tr>
<tr>
<td>Peaking factor</td>
<td>2.19</td>
<td>2.22</td>
<td>2.38</td>
</tr>
<tr>
<td>Total amount of MA transmutation [kg]</td>
<td>571.0</td>
<td>569.5</td>
<td>489.4</td>
</tr>
<tr>
<td>MA transmutation efficiency [%]</td>
<td>19.0</td>
<td>17.8</td>
<td>15.6</td>
</tr>
<tr>
<td>Total amount of TRU transmutation [kg]</td>
<td>518.8</td>
<td>518.8</td>
<td>485.3</td>
</tr>
<tr>
<td>TRU transmutation efficiency [%]</td>
<td>11.9</td>
<td>10.4</td>
<td>9.86</td>
</tr>
</tbody>
</table>

* Number of pin: 180, Gd ratio : 15%, Burnup reactivity loss: 1.0%Δk/k (from Ref. 1)
Figure 12: Neutron spectra of various ADS core

To summarise the design of BP oxide-loaded ADS core, the effective reactivity control was possible by the use of beryllium oxide as a moderator and the appropriate optimisation. The safety parameters were almost the same as the BP hydride-loaded core, and the transmutation characteristic was improved drastically from the hydride one and comparable to the reference core.

Conclusion

The present study addressed the application of BP oxide to ADS core to achieve better reactivity control by the use of more stable material in high-temperature condition than hydride. The optimal BP control block specifications were investigated by changing the moderator material, the composition of control material and the number of BP oxide control block. The optimal specifications of BP control block were as follows:

- moderator material: beryllium oxide;
- composition of BP block: 98% beryllium oxide, 2% gadolinium oxide;
- number of BP control block: six or nine.

The burnup reactivity loss decreased to $1.7\%\Delta k/k$ and $1.0\%\Delta k/k$ with 6 and 9 blocks, respectively as compared to the reference core of $4.3\%\Delta k/k$. The core characteristic analysis was also performed to reveal the influence on the core by the introduction of BP oxide control block. The safety parameters of Doppler coefficient, coolant void reactivity and peaking factor were almost same as those of the BP hydride-loaded core, and the transmutation characteristic showed more significant improvement than BP hydride-loaded core. The present study concluded that BP oxide was proven to be feasible for the reactivity control.
References


Seismic performance analysis of China lead-based research reactor vessel with direct FSI method

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Abstract

Accelerator-driven subcritical system (ADS) has very attractive advantages such as its potential ability to achieve long-lived radioactive nuclear wastes transmutation, fission fuel breeding and energy production, etc. ADS consists of a subcritical reactor coupled with a proton accelerator and spallation target. Chinese Academy of Sciences (CAS) had launched the Strategic Priority Research Programme named “Advanced Nuclear Fission Energy-ADS Transmutation System” to develop ADS in China since 2011. Because of its good neutronics, thermal hydraulics and safety characteristics, China Lead-based Reactor (CLEAR) was selected as the reference reactor for CAS ADS project. China LEAd-based Research Reactor (CLEAR-I) is an integrative pool-type reactor, which uses liquid lead-bismuth alloy as the primary coolant.

The pool-type vessel is usually set up in lead-based fast reactor as the boundary of primary cooling system, and the primary coolant is filled and in direct contact with the vessel. The reactor vessel designed as large and thin shell vessel is subjected to high temperature and heavy coolant pressure in normal conditions. Especially in seismic loading, the structure vibration and coolant flow fluctuation may cause safety hazard, because of the fluid-solid coupling effect caused by heavy liquid metal. Thus, the seismic performance analysis of the reactor vessel is an important safety issue, where the support scheme of reactor vessel is a key design option. In this paper, top supported and bottom-supported schemes are considered for reactor supporting. Seismic performance analysis is accomplished by full 3D FEM model in ANSYS code and fluid-solid coupling effect are considered by using direct FSI method. As a general conclusion, it is feasible that both the bottom supported vessel and the top – supported vessel can satisfy the reactor design code. While the seismic displacement response of bottom-supported vessel at cover location is much higher than that of the top supported one, as a result, for ADS coupling requirement, the top – supported vessel is highly recommended for ADS system.

Keywords: CLEAR, ADS, Seismic Performance
Thermal-hydraulic experiments at Karlsruhe Liquid Metal Laboratory (KALLA) in support of ADS

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Abstract

Many ADS concepts rely on heavy liquid metals as primary reactor coolants. In the MYRRHA facility, conceived at SCK•CEN as an ADS which can operate in subcritical or critical modes, lead-bismuth eutectic (LBE) is used. In this context, extensive research activities are undertaken in Europe supporting the development of LBE technology. At the Karlsruhe Liquid Metal Laboratory (KALLA) of KIT, several key aspects of this technology, including materials compatibility, technology and instrumentation have been studied experimentally for almost two decades.

For the safety analysis and licensing of ADS a fundamental understanding of the thermal-hydraulic behaviour of main components and systems is necessary. Despite remarkable progress, generally applicable design and analysis guidelines are not available for complex geometries and phenomena. Reliable experimental data on prototypical components are essential for confirming the feasibility of a design and validating numerical simulation tools.

In this context, several LBE-cooled fuel assembly mock-ups were recently tested at KALLA in nominal and accidental conditions expected in ADSs. These experimental campaigns allowed investigating the effect of spacers (grid or wires) and local blockages, providing valuable feedback to the designers. This work presents the main experimental results, highlighting the accuracy of existing heat transfer correlations, and the relevance of hot spots based on local temperature distribution both at the wall and in the fluid. Moreover, the main open thermal-hydraulic issues for supporting ADSs which require further research are identified.

Introduction

In both open and closed nuclear fuel cycles minor actinides remain as long-lived, high-level radioactive waste (HLW). One of the most promising innovative technologies for handling the HLW accumulated over decades of operation at nuclear power plants and other installations are the partitioning and transmutation (P&T) processes, reducing the burden on geological disposal. This strategy can be optimised in a dedicated accelerator-driven system (ADS). Worldwide, extensive R&D programmes are undertaken in many countries considering several reactor designs [1]. In Europe, the reference design for an ADS demonstrator is the MYRRHA facility, developed at SCK•CEN in Belgium. This facility aims to demonstrate the ADS concept at a prototypical power by coupling the accelerator, target and subcritical core cooled by lead-bismuth eutectic (LBE) [2]. Moreover, MYRRHA is considered as a key support
infrastructure for the development of fast reactors within the European Sustainable Nuclear Industrial Initiative (ESNII).

The development of ADSs involves studies in several disciplines, as well as their interaction. This list includes accelerator and reactor physics, fuel, target, materials sciences, and thermal hydraulics (TH), among others. With a strong focus on the safety assessment of the reactor, TH studies play a key role during the design and licensing phases. In particular, sufficient cooling must be provided for maintaining the mechanical integrity of the fuel cladding as a containment barrier. To that end, the temperature must remain below certain threshold levels in nominal conditions as well as in postulated accidental scenarios.

In principle, extensive knowledge can be derived from previous experiences in liquid metal systems, e.g. in sodium fast reactors, although some differences are observed. On the one hand, the selection of LBE as coolant presents some relative advantages, for example the chemical inertness. Due to its high boiling point, the risk of void in the core, even in accidental scenarios, is greatly reduced. On the other hand, the thermal conductivity of LBE is lower than that of sodium, leading to larger temperature differences, particularly at low flow velocities. As a general rule, the previous works in sodium systems provide some valuable input for the basic TH design of LBE-cooled ADSs, but specific studies using LBE as coolant are necessary for detailed engineering purposes.

For evaluation and optimising the reactor design, engineers rely strongly on empirical correlations, system thermal-hydraulic and computational fluid dynamic codes. However, these predicting tools have a limited range of validity and thus should be used with caution. A major challenge for computing the turbulent heat transfer of liquid metals is that, as a consequence of their low-Prandtl number (Pr<<1), the temperature and velocity fields have a different statistical behaviour [3]. Common analogies assuming such similarity (e.g. the Reynolds analogy, typically used for fluids with Pr~1, such as water or air) cannot be applied to liquid metal flows and specific models must be developed and validated. Despite extensive progress in the past decades, summarised in [4], generally applicable guidelines have not yet been established. Thus, the safety assessment based on numerical predictions alone remains uncertain and additional information is requested for licensing.

In this context, TH experiments play a key role in the development and particularly the safety assessment of innovative ADS concepts, particularly for first-of-its-kind demonstrators. With this perspective, many European projects related to ADS development include large experimental activities on thermal hydraulics. An overview of the most relevant recent and ongoing European projects in the decade 2010-2019 covering TH experiments, including the following ones, is shown in Figure 1.
• EUROTRANS (2005 – 2010): European research Programme for the transmutation of high-level nuclear waste in an accelerator-driven system, completed, see [5].


• MYRTE (2015 – 2019): MYRRHA Research and Transmutation Endeavour, ongoing, see [7].

• SESAME (2015–2019): Thermal-hydraulics Simulations and Experiments for the Safety Assessment of MÉtal cooled reactors, ongoing, see [7].

The Karlsruhe Liquid Metal Laboratory (KALLA) of KIT has been and is active with experimental campaigns in all the above-listed projects. In this work, some previous and ongoing activities are presented in the following sections. Moreover, the main conclusions of these experiences and an outlook of planned future activities are presented in the final section.

**Experimental capabilities at KALLA**

For over fifteen years, KIT-KALLA has been investigating safety aspects in liquid metals for application in fast reactors and ADSs. The overall test program includes, among others, oxygen control systems, reliability and corrosion studies, electro-magnetic and mechanical pumps as well as heat transfer and flow measurements in prototypical rod bundle and spallation target setups.

Thermal-hydraulic experimental campaigns in liquid-metal systems are complex and time consuming, as evidenced by the duration of these projects, usually four years or even longer. In view of the large effort necessary for the construction of the experimental setup, these campaigns must be optimised for simultaneously fulfilling these two objectives.

• To investigate the safe operation of the reactor in the postulated TH scenario. For this, the mock-up geometry must have a prototypical size and power input, as well as representative operating conditions in terms of fluid velocity and temperature.
To provide reliable correlations and validation data for predicting models, essential for extrapolating the results, filling the gap between the experimental setup and prototypical reactor.

These two objectives lead to the combined need for large-scale infrastructure and accurate instrumentation. An overview of the capabilities at KALLA regarding both aspects is given below.

**Test facilities**

In general, each test facility at KALLA is optimised for studying specific phenomena. For example, static setups are used for studying oxygen sensors (KOSIMA, KOCOS), creep (CRISLA), fretting (FRETHME) and erosion (CORELLA). Further loop facilities, dedicated to studying long-term corrosion in flowing LBE (CORRIDA) and pure lead (TELEMAT), flow and heat transfer of sodium (ALINA), as well as lead-water interaction (SGTR) are not described here. The interested reader is referred to chapter 12 in the LBE Handbook [8].

On the other hand, loop TH facilities are designed to provide controlled boundary conditions (flow rate, inlet temperature and pressure and thermal power) to a test section. In this context, specific facilities are used for covering different ranges of these parameters. For studies related to ADS development the facilities listed in Table 1 are available at KALLA.

| Table 1: Main characteristics of thermal-hydraulic loop facilities for ADS studies at KALLA |
|---------------------------------------------|-----------------|-----------------|-----------------|
| Acronym         | THESYS           | THEADES         | GALINKA         |
| Full name       | Technologies for heavy metal systems | Thermal-Hydraulics and ADS design | Gallium-Indium loop Karlsruhe |
| Fluid           | LBE              | LBE             | Ga-In-Sn       |
| Start-up year   | 2001             | 2002            | 2008           |
| Maximum temperature, °C | 450 °C         | 450 °C         | 40 °C          |
| Maximum flow rate, m³/h | 16             | 47             | 2.4            |
| Maximum pressure head, bar   | 3               | 5.9            | 2.8            |
| Number of test ports | 2               | 4              | 1              |
| Maximum test section length, m    | 2.8             | 3.4            | 1.0            |
| Nominal pipe size | DN50 (2")       | DN100 (4")     | DN25 (1")     |
| Heating/cooling power, kW | 3 x 50          | 500            | 6              |
| Inventory, litres | 300             | 4000           | 5              |

The THESYS loop was first built in 2001 in a horizontal arrangement and later updated in 2005 in a vertical configuration. Its main original purposes were the development and testing of technologies adapted to an LBE environment, such as electro-magnetic pumps, instrumentation and electrical heaters. Upon completing these preliminary tasks, benchmarking heat transfer tests for the general validation of turbulent models used in CFD codes were performed. Currently, it is being updated with three independent power supplies and four parallel flow channels within the SESAME project.
For campaigns requiring higher thermal power and/or flow rates, the THEADES loop can be used, thanks to its larger air cooler and centrifugal pump. With these parameters, it is possible to test key components of LBE-cooled ADSs with prototypical dimensions and under reactor-like conditions of operating temperature, velocity and heat-flux density. Experimental campaigns in this loop include the investigation of spallation targets (with window and windowless) and rod bundles.

In recent years, a eutectic mixture of Ga, In and Sn (melting point: 11°C) has been used for small tests at room temperature. Although these studies cannot be directly extrapolated to ADS conditions, the GALINKA loop facility supports the development and testing of instrumentation and technologies. As the Prandtl number of Ga-In-Sn is similar to that of LBE, generic tests for validating turbulence models can be performed in this setup, exploiting the advantages of near room temperature operation.

**Instrumentation – know how**

For the reliable extrapolation of the experimental data to the reactor conditions detailed, accurate and reproducible measurements are required. This sets large constraints on the instrumentation. In particular, measuring techniques are adapted to the LBE environment, i.e. under conditions of high temperature, opaqueness, as well as large electric and thermal conductivity.

Both AC and DC electro-magnetic flow meters have been developed and tested in the THESYS loop. They were calibrated against momentum-based techniques, such as Vortex and annubar flow meters [9]. Local flow velocity can be measured with miniature Pitot and Prandtl probes. Ultrasonic methods and permanent magnet were also tested, see [10].

Local temperature measurements in heated tests are complicated due to the large average heat transfer coefficients normally found in liquid-metal systems, as a consequence of their large thermal conductivity. Large wall heat-flux densities are needed in order to attain measurable temperature differences, impacting on the design of the test section and its related instrumentation, given the limited space and steep gradients. At KALLA, miniature thermocouples are used for measuring local temperature at the fluid (0.25 mm) and at the heated wall (0.5 mm), calibrated to a relative accuracy of ±0.1°C. This level of accuracy is essential for the proper assessment of heat transfer correlations, as in the rod bundle tests described in this work.

**Selected previous experiences and results**

For pool-type reactors such as MYRRHA (and most liquid-metal based designs), safety-relevant TH studies are divided in two groups. On the one hand, pool TH analyses investigate the overall behaviour of the reactor in postulated scenarios, aiming to determining whether sufficient coolant flow rate is provided to prevent core failure. On the other hand, core TH studies are focused on the temperature distribution in the fuel elements under nominal and non-nominal conditions.

In line with the experimental infrastructure at KALLA, i.e. large loop-type facilities, work has been focused in-core TH studies, in particular on the temperature distribution in fuel assembly mock-ups. In this context, three campaigns of increased complexity were completed at KALLA in successive European projects. An overview of their main characteristics, compared to the current design of the MYRRHA fuel assembly is listed in Table 2. In all cases, high heat fluxes up to 1.0 MW m², representative of the reactor nominal operating conditions, have been applied.
Table 2: Comparison of LBE-cooled fuel assembly mock-ups tested at KALLA in completed European projects, and envisaged fuel element of MYRRA

<table>
<thead>
<tr>
<th>Rod bundle</th>
<th>DEMETRA</th>
<th>THINS</th>
<th>SEARCH</th>
<th>MYRRHA</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of rods</td>
<td>1</td>
<td>19</td>
<td>19</td>
<td>127</td>
</tr>
<tr>
<td>Rod diameter, mm</td>
<td>8.2</td>
<td>8.2</td>
<td>8.2</td>
<td>6.55</td>
</tr>
<tr>
<td>Rod heated length, mm</td>
<td>870</td>
<td>870</td>
<td>870</td>
<td>600</td>
</tr>
<tr>
<td>Pitch-to-diameter ratio</td>
<td>Annular (60 mm)</td>
<td>1.4</td>
<td>1.279</td>
<td>1.279</td>
</tr>
<tr>
<td>Spacer</td>
<td>Four-winged</td>
<td>Grid spacer</td>
<td>Wire (2.20 mm)</td>
<td>Wire (1.80 mm)</td>
</tr>
<tr>
<td>Max. bulk temperature</td>
<td>400°C</td>
<td>450°C</td>
<td>355°C</td>
<td>410°C</td>
</tr>
<tr>
<td>Max. power, kW</td>
<td>20</td>
<td>426</td>
<td>295</td>
<td>1449</td>
</tr>
<tr>
<td>Max. Reynolds number</td>
<td>237 000</td>
<td>120 000</td>
<td>48 000</td>
<td>48000</td>
</tr>
</tbody>
</table>

First, a single-pin arrangement was studied as a simplified but useful representation of a fuel bundle. Although sub-channel transport is not represented, peripheral variations can be neglected for widely spaced triangular bundles (P/D>1.2) and the equivalent annulus can be regarded and studied as a limiting case [11]. Detailed velocity and temperature profiles and turbulent spectra were measured at different flow rates and heat fluxes. With these measurements, a new empirical Nusselt number correlation for developing forced convective flows, and a criterion for the transition from forced to mixed convection have been proposed, see [12]. A sketch of the experimental setup mounted in the THESYS loop, and representative radial temperature profiles are shown in Figure 2.

Figure 2: Experiments on a single rod in an annular channel within the EUROTRANS/DEMETRA project. Left: setup. Right: radial temperature profiles

Within the European project THINS, a larger test section was installed in the THEADES loop. The geometry consists of a bundle of 19 rods in a triangular arrangement (with P/D=1.4) inserted inside a hexagonal channel. Such geometry is representative of hex-can fuel elements, albeit with a reduced number of rods, yet sufficient for representing the main flow features. Three grid spacers are used for
keeping the pitch constant and, at the same time, they have fine channels for placing thermocouples (38 in total, 0.25 mm in diameter), as shown in Figure 3.

**Figure 3: Side view and photos of the test section used in the THINS project: 19-rod bundle with three grid spacers, supporting thermos-couples for detailed temperature measurements**

Based on these detailed measurements, a local Nusselt number (Nu) is derived at each axial position. The experimental heat transfer results at each measurement layer (ML) are compared with the predictions of the empirical correlation recommended in [13], as shown in Figure 4 (left). It can be observed that the correlation is conservative, and at any given Pe, Nu is larger (30–40%) at the first position than further downstream. This result infers that the flow is in the developing region at x₁ and fully developed at x₂ and x₃. In order to confirm this condition, the third spacer was moved around its reference position, as shown in Figure 4 (right). As Nu does not change with the position (within the experimental uncertainties), the flow is fully developed. For further details, see [14].

For smaller pitch-to-diameter ratios, as considered for MYRRHA, wire spacers are preferred over grids. Within the European project SEARCH, a representative wire wrapped bundle was tested in the THEADES loop. For practical reasons the KALLA tests considered a lower number of rods (yet properly accounting for wall effects) and a larger rod diameter, see Table 2. All other parameters have been accordingly scaled, leading to the same P/D ratio and non-dimensional flow parameters, e.g. Reynolds number.

Detailed instrumentation could be implemented at the heated wall and selected sub-channels, thanks to the use of slightly larger pins. Three MLs in the heated region are defined, as shown in Figure 5. At each level, 18 TCs (0.5 mm) are inserted in grooves in the heater cladding and 5 TCs (0.25 mm) are placed at the centre of selected sub-channels. With a total of 69 thermocouples, detailed temperature profiles could be obtained.
Following both in situ calibrations and a posteriori corrections to the raw data, reliable results were obtained. An experimental Nu is derived at each ML and compared with several empirical correlations, as shown in Figure 6 (left). Disregarding the data at ML1 (not yet fully developed) best results are obtained with the correlation by Kazimi and Carelli [15], which over-predicts the data by 5.2% in average. Such experimental confirmation is of great value for the design team which must consider temperature thresholds during the irradiation of the fuel assemblies.

This correlation is actually suggested by the authors as a most conservative estimate for bare rod bundle (without spacers). The even lower Nu can be attributed as an effect of the wire spacer themselves, analysing the temperature profiles at a given measuring level, as shown in Figure 6 (right). The wires produce a directional sweeping and affect the turbulent mixing. In particular, the inner regions remain relatively hot, as they receive less cross-flow from the outer sub-channels. Thus, the mean wall temperature overheat $T_{wm} - T_b$ (marked as a black line in the colour legend) is higher, and Nu is lower than without spacers. Moreover, at each ML large temperature differences are observed. At the hottest spot (central rod), the wall overheat $(T_w - T_b)$ is almost twice its means value $(T_{wm} - T_b)$. 

Figure 4: Experimental results in the 19-rod bundle with grid spacers. Left: Nusselt number compared to the correlation by [14]. Right: Axial variation of Nu around the third position.

Figure 5: Side view sketch and photos of the test section used within the SEARCH project: 19-rod bundle with wire spacers, as a scaled setup for the MYRRHA fuel assembly.
Figure 6: Side view sketch and photos of the test section used within the SEARCH project: 19-rod bundle with wire spacers, as a scaled setup for the MYRRHA fuel assembly.

This experimental campaign allowed achieving two main objectives of the project. First, the validity of existing empirical correlations for the MYRRHA conditions was confirmed with a proper uncertainty rage. Second, reliable reference data was obtained, which can be used for validating simulations performed by the project partners. For further details, see [16].

Ongoing work

In 2016, three simultaneous TH experimental campaigns are ongoing at KALLA at different stages of planning, construction, measurements and evaluation. They have one major characteristic in common: the evaluation of core fuel assemblies in non-nominal conditions.

Within the MAXSIMA project, the effect of local flow blockages in fuel assemblies is studied. For that purpose, two blockage elements are inserted in the bundle with wire spacers used within SEARCH, at the locations indicated in Figure 7 (left). Both elements cover one sub-channel, either at the centre (C1) or edge (E1) of the bundle and are placed shortly upstream of the existing measuring levels, in order to obtain a direct comparison of the wake region. The worst-case scenario is given by a non-porous, poorly conducting blockage. In this experiment, the blockage elements are constructed as thin-walled (0.2 mm) stainless steel shell shaped as the sub-channel contour, filled with pourable ceramic with a low thermal conductivity (1.0 W m\(^{-1}\) K\(^{-1}\)). Support for additional thermos-couples in the wake region is provided at the blockage cover, see Figure 7 (right).

Figure 7: Test section with blockage elements studied within the MAXSIMA project. Left: schematic side view of the bundle and blockage locations. Right: steel shell of the blockage elements.
While worst-case blockage scenarios are studied within MAXSIMA, preliminary simulations indicate that blockage conditions can lead to clad failure, if the blockage is large enough. Although these results are yet to be validated with the upcoming experimental data, the expected blockage size remains unknown. Aiming to answer this question, the topic of blockage formation and growth is studied in a water test section as part of the MYRTE Project. In bundles with wire spacers, axially extended blockages are expected from the accumulation of particles [17], see Figure 8 (left). For the experiments, water is selected as a model fluid because optical instrumentation can be used. Such techniques have been tested with good results as shown in Figure 8 (right). In order to extrapolate the results to LBE conditions, non-dimensional similarity criteria, considering buoyancy, inertial and viscous forces, are taken into account. Furthermore, numerical predicting models are validated.

An additional approach towards the worst-case conditions considered in the safety assessment is to include the heat transfer across the wrapper tube in the analysis. In previous heated tests and simulations adiabatic boundary conditions are imposed as a conservative assumption, yet very penalising, particularly for edge-type blockages. In a reactor core, cooling across the wrapper tube wall is provided by flow between sub-assemblies, i.e. the inter-wrapper flow (IWF). Previous sodium experiments in Japan [18] have indicated that IWF can contribute significantly to the cooling of the core, particularly under decay heat removal conditions. In order to study the effect of this IWF, an experiment consisting of three parallel rod bundles and the flow channel between them is being constructed at KALLA in the frame of the European project SESAME, see Figure 9.

Figure 8: Left: process of blockage formation around the spacer in a rod bundle, from [17].
Right: Accumulation of particles (2.4 mm) in a sub-channel observed with an endoscope camera from inside a transparent rod bundle, as preliminary test for the MYRTE campaign.
For this new campaign, the THESYS loop is being upgraded with three independent power supplies for the bundles and four parallel flow channels. For sizing the test section, the MYRRHA core design is considered as a reference. The main objective of this campaign is to perform a parametric study on the effect of the IWF gap velocity and derive a novel correlation for the inter-subassembly heat transfer across the wrapper wall, which can be implemented in safety systems codes.

These three ongoing campaigns shall contribute to the safety assessment of fuel assemblies in LBE reactors and ADSs, particularly at accidental conditions. They consequently supplement the study of nominal scenarios in previous projects.

Conclusions and outlook

For the further development of ADS at a prototypical scale continuous studies on several disciplines are required. Thermal-hydraulic studies play a key role in the safety assessment for evaluating the temperature evolution in the core in postulated scenarios. Particularly for the introduction of innovative systems or components, such as using a heavy liquid metal as coolant, experimental investigations are a necessary step, supplementary to numerical simulations, because only limited lessons can be learnt from previous experiences.

The reference design for an ADS demonstrator in Europe is the LBE-cooled MYRRHA reactor, an initiative of SCK•CEN in Belgium. Many recent and ongoing European projects supporting the development of MYRRHA include large experimental activities on thermal hydraulic.

At KALLA, a comprehensive series or experimental campaigns for studying the heat transfer of LBE in scenarios relevant for ADSs were completed in recent years. They were part of international collaborative projects, fulfilling two objectives. First, the safe operation of the reactor component is investigated in a scaled mock-up at prototypical operating conditions. Second, reliable reference data is generated for the validation of numerical models, developed by project partners.

In this work, the main lessons learnt and selected results are presented. Within the completed DEMETRA, THINS and SEARCH projects, different fuel assembly representative geometries cooled by LBE at nominal conditions were tested. These results provided valuable feedback to the reactor designers, regarding the performance of empirical heat transfer correlations and observed hot spots in a rod bundle with different types of spacers.
Ongoing work in the MAXSIMA, MYRTE and SESAME projects extends this analysis to non-nominal conditions. Local blockages are a postulated scenario which can lead to clad failure, and this issue is investigated from different perspectives. First, the temperature distribution is measured for selected worst-case conditions in a 19-rod bundle installed in the THEADES loop. Second, the process of blockage formation and growth is studied using water as a model fluid, respecting all relevant non-dimensional similarity criteria. Third, the contribution of inter-wrapper flow cooling towards reducing hot spots is tested in a test section with four parallel flow channels in the upgrades THESYS loop.

In the upcoming years, in view of the rapid progress of prototypical ADS demonstration projects such as MYRRHA, specific thermal-hydraulic experimental investigations shall be needed for supporting the safety evaluation and licensing. Despite remarkable progress in the past decade, some key TH issues remain open and require further studies, including new experimental data. In the opinion of the authors future campaigns should consider the following fundamental and applied topics.

- Free surface and two-phase flows, relevant for understanding the dynamics of the cover gas in pool systems and postulated steam generator tube rupture accidental scenarios.
- Turbulent thermal mixing of jet flows, as found in the upper and lower plena.
- Natural convection heat transfer in pool geometries, including three-dimensional effects.
- Development of empirical correlations capable of predicting hot spots in addition to mean wall temperature as derived from a Nusselt number. While safety studies are focused on the hottest pin, existing correlations do not account for temperature profiles within a fuel assembly.
- Proof-of-concept and viability tests of key components, such as spallation targets, fuel bundles and systems of ultimate decay heat removal under postulated accidental scenarios.
- Study of transient cases, relevant for the conditions of decay heat removal.

New scaled experimental campaigns dealing with these open issues are necessary for further improving the predicting capability of numerical models, in turn necessary for extrapolating the results to the actual reactor geometry and conditions.

Acknowledgements

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References


Development of non-nuclear integrated test platform for China lead-based reactor

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Abstract

Liquid lead-based alloy is one of the most potential candidate coolant materials for fast reactor and accelerator-driven subcritical system (ADS) subcritical reactors due to its attractive nuclear, thermos-physical and chemical properties. In 2011, Chinese Academy of Sciences (CAS) launched the Strategic Priority Research Programme to develop ADS for nuclear waste transmutation, and China LEAd-based Reactor (CLEAR) was selected as the reference reactor for ADS project in CAS.

In the first step, the conceptual design and preliminary engineering design for China Lead-based Research Reactor named CLEAR-I with ~10 MW thermal power have been performed. The liquid lead-bismuth eutectic (LBE) was considered as the primary coolant material driven by mechanical pump in a pool type reactor (6.5 m in height and 4.8 m in diameter), and will make the heat exchange with pressurised single-phase water in the secondary loop, and air cooler as the final heat sink for CLEAR-I.

To provide an experimental platform for non-nuclear key technologies investigation and components performance test for CLEAR-I, a pool-type experimental facility named as CLEAR-S was designed and is under construction by FDS team. The height and diameter of the main vessel for CLEAR-S are about 1:1 and 1:2.5 to those of CLEAR-I, respectively. And more than 200 tonnes liquid LBE will be loaded in the facility. The maximum thermal power of this facility is about 2.5 MW. The main functions of CLEAR-S are as follows: 1) The thermal-hydraulic phenomena (such as thermal fatigue and thermal stratification, etc.) in pool-type heavy liquid metal reactor can be investigated; 2) The components prototype performance will be tested for CLEAR-I, such as mechanical pump, the heat exchanger, oxygen content measurement and controlling system for large volume LBE inventory, decay heat removal system, and so on; 3) Large-scale steam generator tube rupture (SGTR) will also be studied and tested, which may occur due to the heat exchanger immersed in high-temperature LBE with high pressured water as the secondary coolant. And CLEAR-S can also be an integrated test platform with international advanced level for engineering verification and basic research of liquid heavy metal cooled reactor technology. This contribution will present the functions and specifications of the whole facility, technical specifications of the prototype components, and also the pre-test of heavy liquid metal technology based on KYLIN series facilities for CLEAR-S.

Keywords: Lead-bismuth, Research reactor, Accelerator-driven system
Application of ultrasonic flowmeter for LBE flow

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Abstract

JAEA has been performing various R&D of ADS utilises lead-bismuth eutectic (LBE) alloy as a spallation target material and a coolant. J-PARC planning the construction of ADS target experimental Facility (TEF-T) as a preceding step before the construction of demonstrative ADS. Main purpose of TEF-T is post irradiation test of candidate construction materials of ADS by using a LBE spallation target. The flow monitoring system is one of the essential device to operate the target. The electro-magnetic flowmeter (EMF) is usually used for LBE flow. However, its instability and degradation of output were observed at several times. The purpose of this study is development of an Ultrasonic flowmeter for LBE. To secure the sufficient work of signal transmission/reception over 350°C, lithium niobate (LiNbO₃) crystal is used as an oscillator. LiNbO₃ has very-high Curie temperature. However, the piezoelectric efficiency is poor in comparison with PZT. The developed system simply uses the propagation time difference method hence it overcome the poor efficiency problem. To improve the poor wettability at wall-solid boundary, the sensor plug made by high chromium steel was installed to the test section, and several test was performed in LBE loop. In TEF-T, LBE drain and shutdown procedure is required for the maintenance of target system. In this situation, the change of wettability affects the decrease of signal amplitude. By the result of LBE drain/fill-up experiment, high chromium steel showed good wettability and the decrease of signal amplitude (P/P_max) was about 0.86. This value is enough to measure the flow rate. As a result of competitive test with a calibrated EMF, the measured value showed good agreement with EMF in each LBE flow rate. The correlation value was 0.9995. Finally, the long run test was performed by using developed method. Temperature of LBE was set to 350°C. This temperature is a planning inlet temperature of LBE target. Flow rate of test LBE loop was set to 26.8 litre/min, because to simulate same flow velocity of LBE flow channel of TEF-T target at the measurement section. To confirm the applicability to the operation time of LBE target in TEF-T, the total operation time was over 4 500 hr. As a result of this experiment, developed ultrasonic flow meter was successfully applied to flowing LBE, and provided stable measurement result with accuracy of less than 3%.
Introduction
To transmute radioactive wastes, JAEA has proposed an accelerator-driven system (ADS) which utilises lead-bismuth eutectic alloy (LBE) as a spallation target and a core coolant [1] [2]. JAEA is planning ADS target test facility (TEF-T) as a preceding stage of a demonstrative ADS [3]. The most important role of TEF-T is to obtain the irradiation data of several candidate materials in flowing LBE. Further, knowledge to be provided from the operation of TEF-T will establish the handling techniques for LBE system under high radiation environment. In TEF-T, LBE flow monitoring device is one of the indispensable components to specify the material irradiation condition. LBE flow measurement is too difficult because of its physical properties such as the opaqueness, high melting point and low Pr number. The planning irradiation temperature is 350-500 °C and the irradiation dose is about 8 dpa/year, hence the measurement device should work in the severe environment.

There are several LBE flow monitoring techniques. The electro-magnetic flowmeter (EMF) is usually used for liquid metal flow in high-temperature condition. However, between long-term experiments up to 3 000 hrs its instability and degradation of electrode were frequently observed in our experiences. This kind of immersion probes are especially disadvantageous because the wettability of LBE causes unstable measurement reproducibility and the obviously heavy flowing medium with high corrosiveness erodes the measuring part. Of course, the application to LBE requires excellent high heat resistance for the probe.

The ultrasonic flow measurement technique has been established in experimental applications in fluid dynamics and engineering [4] [5]. It is advantageous in that the ultrasonic signal makes its method applicable to opaque liquids such as liquid metals and it is applicable non-intrusively behind opaque walls. We focused on the ultrasonic technique and developed a flowmeter to achieve both long lasting and high-reliability flow monitoring for LBE target system in TEF-T. However, its application has several difficulties for LBE flow measurement such as the high-temperature condition, wettability and corrosiveness of LBE. Therefore, we performed R&Ds in order to resolve these issues.

Development of ultrasonic flowmeter

High-temperature transducer unit
We tried to apply the lithium niobate (LiNbO₃) crystal for the high-temperature transducer. LiNbO₃ is one of the applicable piezoelectric element to high-temperature condition (Tc = 1210 °C) and actually works as the transducer of ultrasonic flaw detector for high-temperature environment. It is also known that its efficiency is poor, therefore application is quite difficult to use of the weak echo signals such as an ultrasonic velocity profiling. This shortcoming is not such a big issue because our developed system simply utilises the propagation time difference of emitted signal.

In this study, we developed and employed the high-temperature ultrasonic transducer shown in Figure 1. The basic emission frequency of developed transducer is 4 MHz, and it is retaining its function up to 500 °C. The outline dimension of transducer is 32 mm in length, and 30 mm in diameter. The material of exterior casing is type 304 stainless steel. The active diameter of element is 20 mm. To increase the radiation resistance, all of this transducer consists of the inorganic materials by excluding the organic materials such as the adhesive bond. The transducer plug adheres to the transducer via the metallic couplant by using a fixing jig. The cylinder type of transducer plug is 163 mm in length, and 25.4 mm in diameter. This plug is immersed to LBE, and its front surface functions as the solid-liquid boundary for the signal emission and reception in the flow channel. In the problems of signal behaviour at the solid-liquid boundary, the wettability is the most serious problem in the ultrasonic flow measurement. The mirror finished treatment is applied to the front surface of the plug to improve the
wettability because LBE has poor wettability to the metallic material surface. In the poor wetted condition, the thin gas layer is formed on the immersed plug surface, and the exchange of the ultrasonic signal is refused. The high chromium steel shows a good wettability compared with the stainless steel. Therefore, we applied high chromium steel as a material of the plug to improve the wettability at solid-liquid boundary.

**Figure 1: Schematic illustration of high-temperature ultrasonic transducer**

![Schematic illustration of high-temperature ultrasonic transducer](image1.png)

**Figure 2: Schematic illustration of high-temperature ultrasonic transducer**

![Schematic illustration of high-temperature ultrasonic transducer](image2.png)

**Measurement principle**

There are several measurement method for ultrasonic flowmeter. We applied the propagation time difference method. This conventional method simply uses the propagation time of ultrasonic burst signal, so that it has two typical advantageous applying to actual environment in TEF-T. This method does not need the suspended tracer particles hence this advantage enables to exclude the uncertainties caused by the contamination of fine particles. The propagation time difference method resists noise because the relatively strong signal peak is acquired by measuring only the propagation time of emitted signal. Further, our developed system arranges two transducers to form a linear propagation path. Figure 2 shows the schematic illustration of measurement section. By using this simple configuration, it is clearly easy to detect the propagation time and to remove the influence of the multiple reflection in the plug because the emitted burst signal reaches a transducer on the reception-side in the shortest distance. In LBE, it has linear relationship to temperature. The sound velocity in molten LBE was already reported by Kazys [5]. However, this result was based on the experimental result on temperature condition less than 330 °C. Furthermore, we assumed that the sound velocity was affected by the contained impurities in used LBE. To apply the developed flowmeter, we took sample of used LBE from a test loop and measured the sound velocity in the temperature range 187-407 °C. In our application test, the following equation given by our experiment was employed.

\[
 c_{LBE} = 1801 - 0.2583T_{LBE}
\]  

(1)
**Figure 3: Experimental setup and photograph of measurement section**

**Result of application test**

**Experimental setup and conditions**

Figure 3 shows the schematic illustration of experimental setup. We performed several application test of the developed flowmeter by using JAEO Lead-Bismuth Loop #4 (JLBL-4). The LBE inventory of this loop is 20 litres. The electro-magnetic pump (EMP) is installed to drive LBE, and its maximum flowrate is 40 litres/min. The EMF is also installed to measure the reference flowrate. To mitigate the erosion/corrosion during the operation, this loop mainly consists of type 316 stainless steel. Two high-temperature ultrasonic transducers were installed to a π-shape measurement section, and the transducer located opposite to one another. The size of measurement section is 312 mm in length, and 39.4 mm in diameter. The distance between two transducers was 95.9 mm. The temperature of the loop and LBE were set to 350 °C. This temperature is same condition with the planning operation temperature of inlet side in TEF-T LBE target loop. To measure the flowing LBE temperature, a thermocouple was inserted into the nearest port located in the inlet side of the measurement section. Several measurement parameters such as the sound velocity were corrected by LBE temperature. The setting parameters of developed ultrasonic flowmeter is shown in Table 1. To measure the LBE flowrate, two transducers driven by an applied voltage of 200 V generated the ultrasonic burst signals. Its emission frequency of burst signal was 4 MHz, and the repetition frequency was 1000 Hz. To measure the signal propagation time, the wave number of burst signal was set to one cycle because the unnecessary reflection noise should be as few as possible for the detection of the signal peak. The time resolution to acquire single velocity data was 500 ms and the number of averaged velocity data was 256.
Table 1: Setting parameters of developed flowmeter

<table>
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<th>Parameter</th>
<th>Value</th>
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</thead>
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<td>Emission frequency of burst signal</td>
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</tr>
<tr>
<td>Nuclear</td>
<td>200 V</td>
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<tr>
<td>Renewable energy</td>
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<tr>
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<td>1000 Hz</td>
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<td>Greenhouse gas emission1</td>
<td>500 ms</td>
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<td>Nuclear fuel cycle</td>
<td>256</td>
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</tbody>
</table>

Evaluation of followability for flowrate

Figure 4: Measurement result of developed ultrasonic flowmeter vs EMF

The measured velocity obtained by the developed ultrasonic flowmeter was compared with the reference flow velocity measured by EMF. EMF was calibrated before and after this experiment and its validity was confirmed by the estimated theoretical output properties of EMP. Figure 4 shows the result of the application test of developed flowmeter. The horizontal axis shows the flowrate of LBE, and the vertical axis shows the velocity. Each data point are measured mean velocity along the ultrasonic signal path, and the solid line is the theoretical cross sectional velocity derived by measured flowrate obtained by EMF. Each bar shows the deviation of 256 data corresponding to the velocity fluctuation. In this experiment, the setting flowrate of LBE were 0, 10, 20, 30 litres/min. The experimental measurement result gave good agreement with the theoretical velocity. The correlation value between ultrasonic flowmeter and EMF was 0.9995. The deviation increased with increasing of flowrate and showed 0.034 m/s at the maximum. We have concluded that the cause for this trend can be result of formation of turbulent flow within a typical flow configuration such as the π-shape measurement section (see Figure 2) with increasing of flowrate.

Evaluation of durability

To evaluate the long lasting performance of developed ultrasonic flowmeter, we performed a durability test by same configuration in JLBL-4. The temperature of the loop and LBE were set to 350°C. To set LBE flowrate definitely, a calibrated EMF was applied at the onset of experiment and its validity was
monitored during the experiment by the estimated EMP output. In TEF-T target system, we propose the size of flow channel unlike JLBL-4. To simulate LBE velocity condition of TEF-T in the measurement section, the flowrate of LBE was set to 26.8 litres/min constant. The velocity was 0.366 m/s, and it was same velocity condition at the planning rated flowrate (85 litres/min) in TEF-T. The target time of total experiment time were set to 4500 hrs. This target time is an annual operation time of TEF-T target system. During the experiment period, the oxygen concentration in LBE was kept between $10^{-5}$ to $10^{-7}$ wt%.

**Figure 5: Experimental result of durability test**

Figure 5 shows the result of the durability test of developed flowmeter. The horizontal axis shows the elapsed time from measurement start, and the vertical axis shows the velocity. The operation time of JLBL-4 including two times of loop shutdown was 5880 hrs, and the total experiment time at a constant flowrate condition was over 4500 hrs. Each circle data point are measured mean velocity of approximately once a week and each square data point are received signal amplitude. The solid line shows the theoretical cross sectional velocity. During the experiment period, developed system provided enough signal amplitude to measure the flowrate, and the measured velocity data showed sufficiently stable output. The developed flowmeter successfully showed its effectiveness of long lasting performance for flowing LBE. The maximum deviation was 0.018 m/s, and the error of measurement data was less than 3% during the experiment period.

In TEF-T target system, we planning the several loop shutdown procedure other than the periodical maintenance in order to deal with the significant trouble. Therefore, the applicability of flowmeter in re-wetted condition is a key issue for the restart of LBE target loop. In the durability test, two times of loop shutdown was performed. After second loop shutdown, the signal amplitude decreased to around 87% for the maximum. However, it recovered to 98% 1700 hrs later after the restart of loop operation. The immersed surface condition seems to have improved due to removal of adhered impurities by flowing LBE. In any case, the developed flowmeter sufficiently worked in the durability test over 4500 hrs including two times of loop shutdown and showed excellent applicability by the measurement of LBE flowrate that simulated the condition of TEF-T target loop.
Summary

The realisation of LBE flow monitoring system is admittedly one of the key issue to obtain the purpose of TEF-T. In order to realise both long lasting and high-reliability flow monitoring technique for LBE target system in TEF-T, we developed an ultrasonic flowmeter. The conclusions of this study are as follows:

1. We developed the ultrasonic transducer to achieve the flow measurement in the high temperature and high-radioactive environment. To overcome both issues, we tried to apply LiNbO₃ element and realised the transducer sufficiently retaining its function up to 500 °C.

2. The developed flowmeter was successfully applied to measure the LBE velocity in the simulated operating conditions of LBE target system in TEF-T. The result of competitive experiment with calibrated EMF showed good agreement with the theoretical value in flowing LBE and the obtained correlation value was 0.9995. We have showed that developed flowmeter sufficiently retained its applicability with accuracy of less than 3% over 4 500 hrs in flowing LBE at 350 °C. The signal degradation in re-wetted condition was small and it was within the allowable range of the developed flowmeter. Further, the decrease of signal amplitude recovered under the flowing LBE condition approximately 1 700 hrs later.

References


Status of LBE corrosion test loop “OLLOCHI” and experiments at JAEA

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Abstract

An accelerator-driven system (ADS) for waste transmutation investigated in JAEA employs lead-bismuth eutectic (LBE) as spallation target material and core coolant. To realise future ADS and an ADS target experimental facility (TEF-T) planned for construction in J-PARC, there are many technical issues on LBE. In particular, corrosion data of relevant materials like T91 (Mod. 9Cr-1Mo) and SS316L steels at 400-550°C under oxygen concentration controlled and flowing condition are indispensable. JAEA has designed and built new LBE corrosion test loop named “OLLOCHI (Oxygen-controlled LBE LOop Corrosion tests in High temperature)”, to obtain the corrosion data at the higher temperature.

The OLLOCHI consists of an electro-magnetic pump (EMP), an ultrasonic flow metre, two heaters, two test sections, an expansion tank, a specimen exchange box, a cooler, oxygen sensors, an oxygen control system, a filter, a drain tank and a piping. The EMP is a linear inductive type with an annular channel. The ultrasonic flow metre has plug-in type transducers and it worked well at higher temperature (up to 500°C). The end piece of the test sections is in the expansion tank and the expansion tank is in the specimen exchange box, which will be kept in an inert gas atmosphere. It will be possible to exchange specimen holders without LBE drain to keep oxygen concentration of LBE. Oxygen sensors are set at both high-temp and low-temp section. Filter is installed at bypass line. The piping of high-temperature sections was made by T91 and the casing of heaters and the expansion tank are made by 2.25Cr-1Mo steel. The maximum temperature of these parts is 550°C. At low temperature sections, piping and components were made by SS316L and the maximum temperature is limited to 450°C.

The status for the OLLOCHI as of March 2016 is that the modification of heaters and sample exchange box and the conditioning operation without LBE have already finished. The oxygen sensors and the oxygen control system will be installed soon. After that, the conditioning operation with LBE and oxygen control test will be started. And more, additional flow metres will be installed in each test section until next March. In parallel with these tests, thermal-hydraulic analysis of the two test sections will be performed to identify the flow pattern in the specimen holders.
Introduction

An accelerator-driven system (ADS) for nuclear waste transmutation investigated in JAEA employs LBE (Lead-Bismuth Eutectic) as spallation target material and core coolant. To realise future ADS and an ADS target experimental facility (TEF-T) planned for construction in J-PARC, there are many technical issues on LBE. JAEA has built some apparatuses and developed LBE technologies. For corrosion/erosion examination, a flowing corrosion test loop (JLBL-1; JAEA Lead-Bismuth flow Loop-1) was installed in 2001 and 7 phases of corrosion tests (3 000 h X 6, 1,000h X 1) were performed in flowing LBE. Results of the corrosion tests were reported elsewhere [1-4]. However, the experiments in uncontrolled, low-oxygen-containing (-10-8wt%) LBE are simulation of the situation of a loss of oxygen or loss of oxygen control in an ADS system. To advance the next stage, JAEA has designed and built new LBE corrosion test loop named “OLLOCHI (Oxygen-controlled LBE LOop Corrosion tests in HIgh temperature)”, to obtain the corrosion data of relevant materials like T91 (Mod. 9Cr-1Mo) and SS316L steels at 400-550°C under oxygen concentration controlled and flowing condition. In this paper, status of the OLLOCHI and experimental plan at JAEA will be reported.

Design of the OLLOCHI

Objectives and performance requirement for the OLLOCHI

Objectives of the OLLOCHI are as follows:

1. fundamental study for future ADS development;
2. corrosion data collection for PIE on TEF-T irradiated materials;
3. development of filtering system.

To achieve the objectives, following performances are required:

1. It is possible to perform high temperature (>500°C, ΔT=100°C) corrosion test.Flow rate is over 1 m/s at test sections.
2. Oxygen sensor, oxygen concentration control system, purification system of LBE are equipped.
3. Multi test sections are essential. Test-piece should be exchanged without drain to keep oxygen concentration.
4. Non-contact type flow metre, like an ultrasonic flow metre, is preferable.
5. Number of flange should be decreased as possible.

The loop had been designed according to the above and the construction was finished in March 2016.

Configuration and main components

Figure 1 and Figure 2 show the photograph and the schematic flow of the OLLOCHI, respectively. The OLLOCHI consists of an EMP, two main heaters, two test sections, an expansion tank, a specimen exchange box, oxygen sensors and an oxygen control system, an air cooler, a filter unit, a flow metre, a drain tank and a piping. Thermocouples are placed in several points of the loop.

The loop has a hot leg running from the main heaters to inlet of the air cooler and cold leg running from the air cooler to inlet of the main heaters. The piping of the hot leg was made by T91 (Mod.9Cr-1Mo). Casing of the heaters and the expansion tank are made by 2.25Cr-1Mo steel. The
maximum temperature of the hot leg is 550°C. At cold leg, the piping and components were made by SS316L and the maximum temperature is limited to 450°C.

The EMP is a linear inductive type with an annular channel and sized to provide the loop with a maximum LBE flow rate of 20 L/min. corresponding to a velocity in the test sections of 1.5 m/s. The two test sections TS1 and TS2 containing specimens to be investigated are placed downstream of the main heater MH1 and MH2, respectively. The detail design of the test section will be mentioned later. The main heaters MH1 and MH2 can supply maximum power of 48 kW (24 kW x 2) and the power is sufficient to elevate LBE temperature by 100°C under the condition of flow rate of 20 L/min. The air cooler has a capacity to decrease LBE temperature by 100°C. The end piece of the test sections is in the expansion tank and the specimen exchange box, which will be kept in an inert gas atmosphere is placed on the expansion tank. It will be possible to exchange specimen holders without LBE drain to keep oxygen concentration of LBE. The expansion tank is placed at the highest position in the loop and there is a free surface of LBE. Oxygen sensors are set at both the hot leg and cold leg. For the hot leg, the oxygen sensor mount is placed between the expansion tank and the air cooler. For the cold leg, the mount is placed between the air cooler and the filter unit. The filter unit is installed at bypass line. As a filtering material, stainless steel mesh is employed. As a flow metre for LBE, an ultrasonic flow metre (USFM) was selected. The USFM has plug-in type transducers and it worked well at higher temperature (up to 500°C). Detail of the USFM was reported elsewhere [5].

<table>
<thead>
<tr>
<th>Heavy liquid metal</th>
<th>Lead-bismuth eutectic (LBE)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Maximum temperature</strong></td>
<td>550°C/ hot leg, 450°C/ cold leg</td>
</tr>
<tr>
<td><strong>Maximum flow rate</strong></td>
<td>20 L/min. (1.5 m/s at test sections)</td>
</tr>
<tr>
<td><strong>Electrical power</strong></td>
<td>24 kW (main heater) x 2</td>
</tr>
<tr>
<td><strong>Number of test sections</strong></td>
<td>2 (corrosion test)</td>
</tr>
<tr>
<td><strong>Oxygen control system</strong></td>
<td>Via gas phase</td>
</tr>
<tr>
<td><strong>Oxygen sensors</strong></td>
<td>2 in hot leg and 2 in cold leg</td>
</tr>
<tr>
<td><strong>LBE inventory</strong></td>
<td>100 L</td>
</tr>
</tbody>
</table>

Table 1: Operational parameters of the OLLOCHI
Figure 1: Photograph of the OLLOCHI

Figure 2: Schematic flow of the OLLOCHI

Detail design of the specimen holder

Corrosion data on main candidate and various steels under flowing LBE condition will be obtained by the OLLOCHI with small plate specimens. Figure 3 shows the design of the specimen holder. The specimen
holders were made of T91 tube with a pair of grooves at the inner surface. The plate-type specimens will be set in the grooves straight. Size of the specimen is 14 mm x 10 mm x 1 mm. The specimen holders are connected to the cap of the nozzle on the expansion tank and the holders will be easily exchanged in the specimen exchange box. As a future plan, a rod or pipe type specimen will be employed.

![Figure 3: The design of the specimen holder](image)

**Oxygen sensor and oxygen control system**

Oxygen sensors (OS) are set at both hot leg and cold leg. Some OS have been developed and tested [6]. From the results of pot tests, Pt/air type OS was selected as promising. To install the OS in piping of the loop, additional functions will be required when YSZ will break in the loop. To protect flow-out of YSZ debris, a stainless steel sheath with small holes was prepared. To protect LBE leak out, a freeze seal design was employed in the housing of the OS. Concerning to oxygen control (OC) device, the results and experience of pot and small loop (JLBL-4) tests were reflected to the design of the OC system for the OLOCHI. Measurement and potential control in flow condition will be started soon.

**Modification plan**

Some modifications are planned for the OLOCHI. To compare and guarantee the accuracy of measured flow rate, an electro-magnetic flow meter (EMF) will be installed in downstream of the EMP. And to measure and control velocity of LBE in the test sections, flow metres and valves will be installed in upstream of each test section. These modifications will be finished until next February. And more, as a future plan, installation of a mechanical testing machine in the 3rd test section of the OLOCHI is under consideration.

**Experimental plan**

Table 2 shows the schedule on corrosion study. TEF-T will be designed based on the existing corrosion data. Results of corrosion tests (<450°C and <500°C) will be reflected to the design of austenitic steel target and F/M steel target, respectively. Corrosion data on higher temperatures and PIE data on TEF-T irradiated samples will be reference of future ADS design.
Table 2: Schedule on corrosion study of the OLLOCHI

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</thead>
<tbody>
<tr>
<td><strong>TEF-T</strong></td>
<td>Design</td>
<td>Construction</td>
<td>Operation</td>
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<td></td>
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<tr>
<td>Max. temperature of TEF-T (°C)</td>
<td></td>
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<td></td>
<td></td>
<td>350</td>
<td>400</td>
<td>450</td>
<td>500</td>
</tr>
<tr>
<td>PIE on TEF-T irradiated samples</td>
<td>Cooling</td>
<td>PIE</td>
<td></td>
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<tr>
<td>Target material</td>
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<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Design and fabrication</td>
<td>Austenitic steel</td>
<td></td>
<td></td>
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<td></td>
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<td></td>
<td></td>
<td></td>
<td></td>
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</tr>
<tr>
<td>OLLOCHI Conditioning</td>
<td>Corrosion test (&lt;450°C)</td>
<td>Corrosion test (&lt;500°C)</td>
<td>Corrosion test (&lt;550°C)</td>
<td></td>
<td></td>
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<td></td>
<td></td>
</tr>
<tr>
<td>Modification</td>
<td>EMF USFM</td>
<td>3rd test section</td>
<td></td>
<td></td>
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**Acknowledgement**

We greatly appreciate the helpful comments and supports given by the members of Target Technology Development section of JAEE to this study.

**References**


Monotonic and cyclic mechanical behaviour of T91 in lead-bismuth eutectic: Is T91 reliable enough material for ADS?

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2Currently at Tsinghua University, Shenzen, China

Abstract

The paper summarises the results obtained at CNRS-Lille university on the mechanical behaviour of T91 martensitic steel in liquid lead-bismuth eutectic (LBE). Attention was paid on the chemistry of LBE, especially oxygen content which removal was suspected to affect the mechanical behaviour by decreasing the possibility of protective oxide layer to form. The conditions of loading were investigated in a temperature range from 200°C to 400°C by performing monotonic tests, Small Punch Tests (SPT) and tensile tests, and fatigue tests.

The results showed that T91 was ductile in most of the investigated situations but exhibited brittle fracture for a special set of conditions. The most critical parameter appeared to be the loading conditions. With SPT, (very) low loading rates gave rise to LBE embrittlement even when LBE was oxygen saturated. Under low cycle fatigue, in comparison with air, the fatigue crack initiation resistance was decreased by LBE and the fractography pointed an increased velocity of long crack. As well, the oxygen content played a lesser effect. The interaction between all these parameters is discussed and the similarities/differences in the LBE damage mechanism between monotonic and cyclic loading are pointed out.

Introduction

The concept of accelerator-driven systems (ADS) using heavy liquid metals such as lead or lead-bismuth alloy as a primary coolant and as a spallation target has driven a lot of scientists in the world to evaluate the most appropriate systems by thinking on neutronics, heat transfer, fluid mechanics, materials ... In France, national research programmes—the so-called GEDEON and GEDEPEON programmes have been supported by ministerial authorities (CNRS, ministry of education) and industrial partners (EDF, Framatome, Areva and CEA) in connection with European programmes (TECLA, Megawatt pilot experiment (MEGAPIE)-test, DEMETRA). The behaviour of structural materials has been examined due to the unusual environment in which some components are expected to work. This implicitly suggests that some metallic material families had to be proposed according to the component after assessment of their function and their prerequisite properties (mechanical, corrosion ...). Among them, the window which insulates the proton beam from the liquid-metal target appeared as a critical component. Indeed, the thickness of the window can be very small. In the MEGAPIE-test project, the thinner part was 1.5 mm [1]. Moreover, the window was exposed to a very severe environment – corrosion, irradiation, mechanical effects and coupling effect of these factors- which can lead to degradation of it. T91 martensitic steel was
selected for the window and for other in-core components whereas 316L austenitic stainless was preferred for out-core material components. T91 steel is a modified 9Cr1Mo steel developed to increase the mechanical strength by addition of Nb and V. It was selected because of its good swelling and creep resistance under irradiation up to 773K and because it is subjected to a reduced dissolution in lead or lead-bismuth due to the absence of nickel.

While the mechanical behaviour of T91 has been investigated in a large range of temperatures in ambient air, it was not documented for environments such as liquid metals.

However, materials scientists know that due to the contact of a liquid metal, a ductile solid material may see its mechanical properties deteriorated. Liquid-metal embrittlement (LME) and Liquid-metal assisted damage (LMAD) are two classical manifestations of the effect produced by liquid metals reported for a long time ago. Because the concept of immunity is no longer accepted, it was necessary to investigate in depth the mechanical behaviour of T91 to decide if the latter is a reliable material or not for ADS.

The present paper summarises the key results obtained after 10 years of research at CNRS-University of Lille, France, on the mechanical behaviour of T91 steel in LBE. These results together with results obtained from other institutes (PSI, SCK•CEN, KIT, UJV Řež) have been collected in chapter 7 of the handbook published by OECD in 2015 [2]. The starting point of the research was that LME or LMAD are in the range of environmentally assisted fracture or plasticity phenomenon as are stress corrosion cracking or hydrogen embrittlement for instance. Therefore, both variations in microstructure, in LBE chemistry, in mechanical loading have been considered. In addition, the effect of temperature which may act on each of the previous parameters has been considered.

**T91 martensitic steel**

All materials tested in the present paper had a chemical composition in agreement with the standard given in Table 1.

T91 steel usually receives the standard heat treatment which comprises an austenitisation at 1050°C for 1 h followed by air cooling and a subsequent tempering for 1 h at 750°C. The microstructure is a fully tempered martensite with an average hardness value of 278 HV.

<table>
<thead>
<tr>
<th>C</th>
<th>Cr</th>
<th>Mo</th>
<th>Nb</th>
<th>V</th>
<th>Mn</th>
<th>P</th>
<th>S</th>
<th>Si</th>
<th>Al</th>
<th>Ni</th>
<th>Fe</th>
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</thead>
<tbody>
<tr>
<td>Min</td>
<td>0.08</td>
<td>8.00</td>
<td>0.85</td>
<td>0.06</td>
<td>0.18</td>
<td>0.30</td>
<td>-</td>
<td>-</td>
<td>0.20</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Max.</td>
<td>0.12</td>
<td>9.50</td>
<td>1.05</td>
<td>0.1</td>
<td>0.25</td>
<td>0.60</td>
<td>0.020</td>
<td>0.010</td>
<td>0.50</td>
<td>0.04</td>
<td>0.40</td>
</tr>
</tbody>
</table>

**Experimental**

To study the behaviour of T91 steel in LBE, the small punch test technique was employed. It uses specimens with dimensions 10×10×0.5 mm i.e. thin specimens with a high ratio external surface to bulk. Since our research programme was connected with the window of ADS the thickness of which was about 1.5 mm, this was suitable. The SPT setup fitted in an electro- mechanical machine which allows performing tests in different environments (air, gas, liquid metals) and at temperatures ranging from 200°C up to 400°C and at cross-head displacement velocities between 0.5 mm/min and 0.0005 mm/min. Details of the set up can be found elsewhere [3].
The composition of LBE was 45 wt% Pb and 55 wt% Bi. It has been used with a saturated oxygen content or after purification. Flowing the LBE bath by a mixture of argon/hydrogen gas allowed decreasing the oxygen content up to $10^{-8}$ wt % at 450 °C.

Results

**Effect of microstructure**

The study of the effect of microstructure on LBE embrittlement was motivated by the fact that the harder a material is, a more susceptibility it exhibits. It aimed at considering situations where the hardness could change. Exposure to neutrons e.g. [4] or welding e.g. [5] modifies the microstructure of T91 steel.

To investigate the role of microstructure, T91 steel was tempered also at other temperatures lower than 750°C. Obviously, the microstructural changes resulting from heat treatment and irradiation are not the same. The modification in tempering temperatures essentially affects the precipitation state and the structures of dislocations but not the austenitic grain size which remained approximately at 20 µm. Special attention was paid on tempering temperature of 500°C which provided the highest value of hardness (about 446 HV). According to the literature [6, 7], Cr$_2$C and (Cr,Fe)$_{23}$C$_6$ particles are expected to form after the tempering at 500°C while VC precipitates and (Fe–C–Mo)$_{23}$C$_6$ (along lath boundaries and prior austenite boundaries) are promoted with the tempering at 750°C. The tempering is also known to induce microstructural changes in the dislocation structures produced during the martensitic transformation. Obvious rearrangement of transformation dislocations and initiation of martensite lath breakup is expected to occur after tempering for 1 h at 750°C but not after tempering for 1 h at 500°C.

It is observed that the T91 tempered at 750°C exhibited at 300°C a similar response in air and in LBE (Figure 1 a). However, for the T91 tempered at 500°C, the load-displacement curve with a large domain of plasticity recorded in air test was replaced by a quasi linear curve up to failure for tests in LBE. Fractographic analysis confirmed a brittle behaviour for the latter test as pointed out by the cleavage failure mode (Figure 1c) while a ductile fracture was observed after test in air (Figure 1b).

So, T91 steel appears to be an alloy sensitive to LME in case of specific heat treatment or hardening treatment.

![Figure 1: Small punch tests curves for T91 tempered at 500°C or 750°C after tests at 300°C in air and in LBE a), ductile fracture in T91 tempered at 500°C after test in air b), and brittle fracture in T91 tempered at 500°C after test in LBE c)](image)
Additional heat treatments involving other tempering temperatures have been investigated by SPT. A gradual change in the fractography with the tempering temperature and with the hardness of the material has been observed and is summarised in Figure 2.

**Figure 2: Evolution of the behaviour and of the fracture mode with tempering temperature and related hardness of T91 steel loaded by small punch test**

<table>
<thead>
<tr>
<th>Tempering temperature</th>
<th>500°C</th>
<th>600°C</th>
<th>650°C</th>
<th>700°C</th>
<th>750°C</th>
</tr>
</thead>
<tbody>
<tr>
<td>HV</td>
<td>396</td>
<td>374</td>
<td>318</td>
<td>285</td>
<td>256</td>
</tr>
</tbody>
</table>

**Effect of LBE chemistry**

The taking into consideration of the chemistry of LBE is driven by the corrosion properties of T91 in LBE. Indeed, it has been shown that several factors determine the corrosion mechanism and its more or less strong impact. These are the oxygen content in LBE, the temperature, the exposure duration and the fluid velocity. In oxygen saturated LBE, oxidation of the steel is observed with a typical duplex structure scale that consists of an outer magnetite sub-layer and an inner Cr-rich spinel layer. In low oxygen LBE, dissolution is the other type of corrosion.

Though most of the studies considered temperature far away from 350°C and for test durations higher than involved in SPT, some SPT at a loading rate of 0.5 mm/min were performed on T91 tempered at 750°C in oxygen saturated and low oxygen content LBE and at different temperatures. Oxygen was removed from LBE by flowing the liquid metal with a mixture of argon/hydrogen gas. This allowed decreasing the oxygen content up to $10^{-8}$ wt % at 450 °C. Then the purified LBE was transferred to the SPT setup installed in a test cell made of stainless steel where the atmosphere was controlled. Indeed, to protect the purified LBE from oxidation, the cell interior atmosphere was controlled thanks to a purification unit in order to remove water vapour and oxygen. This was based on flux sweeping and on use of reactive filters. The oxygen content and the water content in the cell interior could be decreased as low as 0.1~0.5 ppm and 5~20 ppm respectively [8].

Figure 3a gives the load-displacement curves of T91 tested at 200°C, 300°C and 400°C in purified liquid LBE and in oxygen saturated LBE. It seems that the curves obtained from tests performed in purified LBE and those obtained in oxygen saturated LBE do not exhibit very obvious differences. SEM observations of the SPT specimens after test in oxygen saturated LBE and in purified LBE revealed that the shape of the cracks was circular with some small radial cracks (Figure 3b). The fracture mode was ductile with dimples.

For these conditions, the chemistry of LBE had no impact on the mechanical response of T91 in the standard heat treatment.
Effect of loading rate

The role of loading rate was suggested by some procedures employed in stress corrosion cracking experiments where it is usual to perform slow strain rate tests to promote corrosion-deformation interactions. Though the corrosion mechanism in liquid metals differ from those observed in aqueous environments, it appeared interesting to decrease the loading rate to trigger such effect.

SPT were therefore performed at 300°C on T91 tempered at 750°C at two displacement speeds up to 0.005 mm/min and 0.0005 mm/min respectively in low oxygen LBE (Figure 4a) and oxygen saturated LBE (Figure 4b).

SPT curves show a strong dependence on displacement speed. In low oxygen LBE, the maximum load was strongly reduced as compared to the other test performed at high speed by a factor of two third.
Second, the displacement at maximum load was decreased in the same way. This abrupt decrease in mechanical properties was accompanied by a change in fracture mode as well at the macroscopic scale as at the microscopic one. The SPT specimens exhibited circular and radial cracks. The major circular crack separated the dome from the rest of the specimen. The decrease in the displacement velocity promoted radial cracking but in few quantities (Figure 5a). Both transgranular and intergranular brittle fracture was observed (Figures 5b and 5c). Therefore it is clear that T91 steel even in its recommended heat treatment can be embrittled by LBE and that the loading rate has a strong impact.

**Figure 5: SEM macro view (a) and brittle fracture (b, c) of the T91 steel tested in low oxygen LBE at 300°C and 0.005 mm/min**

It is possible to separate the possible effect of loading rate and of chemistry of LBE by analysing Figure 4b. It can be seen that the same behaviour can be observed in oxygen saturated LBE but for a loading rate 10 times lower than in low oxygen LBE. The macroscopic aspect of the dome after test in oxygen saturated LBE is similar to that obtained after test in low oxygen LBE (Figure 6a). A fully brittle fracture is observed for this condition (Figure 6b).

**Figure 6: SEM macro view (a) and brittle fracture (b) of the T91 steel tested in oxygen saturated LBE at 300°C and 0.0005 mm/min**

It can therefore be concluded that the most critical factor for occurrence of liquid-metal embrittlement of T91 by LBE is the strain rate and that the chemistry especially the oxygen content behaves as a catalyst.
Discussion

The present investigation shows that the conditions to embrittle by liquid LBE the T91 steel which in most of cases is a ductile material have been easily encountered. These results definitively reject the concept of immunity for a metal or metallic alloy to have their mechanical properties not affected by a liquid metal. However, reduction in toughness and/or acceleration in crack growth by LBE reported on T91 steel by using notched specimens e.g. [9] already suggested a certain degree of sensitivity to LBE embrittlement. This sensitivity is also suspected from the decrease of ductility on smooth specimen of T91 [3, 10] but it was necessary to modify the microstructure by changing the tempering temperature in the heat treatment of the steel or to decrease the oxygen content in the LBE bath. Indeed, LME is promoted in higher strength materials and requires the wetting of the bare material by the liquid metal. In general, oxidation prevents from wetting and avoids brittleness. Therefore, a high oxygen content dissolved in LBE and a rather soft material should have been favourable factors to overcome embrittlement of T91 steel by LBE.

The present investigation which employed smooth specimens tends to show that the situation is not as clear as one could expect. Indeed, changing the oxygen saturated LBE bath for the low oxygen LBE one did not modify very much the behaviour of the T91 in a temperature range between 200°C and 400°C when it was deformed at 0.5 mm/min [8, 11]. When the steel was deformed in presence of the purified LBE, slip bands have emerged from the surface and no or reduced oxidation was expected. Therefore, the absence of a protective oxide layer seems not to be detrimental and the initially considered role of corrosion has not occurred. Within this idea, the apparently weak effect of LBE chemistry is consistent with the brittle fracture observed in oxygen saturated LBE on T91 tempered at 500°C. Here, the ductile to brittle transition is associated with a decrease in the local toughness of the material by the liquid LBE. In the T91 tempered at 500°C, the plastic deformation was much localised and the slip bands less numerous than in the T91 tempered at 750°C. It results that the emerging slip bands at the external surface gave rise to stress raisers and exposed fresh metal which LBE could wet and where atoms of the liquid metal could adsorb. Thus, a decrease in the critical stress intensity factor is then expected as a consequence of a decrease in the surface energy by adsorbed atoms. Since the toughness of the T91 steel is already low when the material is tempered at 500°C, the additional contribution of adsorbed atoms allows reaching very easily the critical value of the stress intensity factor of the material.

To observe a brittle fracture in the soft T91 steel (tempered at 750°C), it is necessary to decrease the local toughness and to increase the stress intensity factor in order to trigger the brittle fracture at the external surface in contact with the liquid metal before the nucleation step of the ductile fracture in the bulk of the material. The decrease of the local toughness in the soft T91 steel is the same as in the hard material. Therefore, it is necessary to increase the stress intensity factor. Under plastic deformation and low loading rate or strain rate, a defect can initiate and grow up to a critical size in the slip band thanks to progressive rupture of atomic bonds. This becomes possible because atoms of liquid metal adsorb on fresh metal, decrease the surface energy and because the low strain rate increases dislocation mobility at crack tip allowing a large number of dislocations to maintain a non-oxidised part of the material, which in turn allows adsorption. The process being continuous, when the critical size of the defect is reached, it suddenly propagates in a brittle manner from the surface to the bulk before ductile fracture could start. This investigation leads us to believe that the low oxygen LBE did not modify the wetting conditions, e.g. by removing the oxide layer. However, the low oxygen content should promote adsorption effect and further entry in the material along interfaces from fresh deformation bands since their oxidation is much disfavoured.

The proposed explanation can also explain the decrease in fatigue resistance. The effect of loading mode has been investigated by performing low cycle fatigue tests on smooth specimens and fatigue crack propagation tests on three point bending specimens in air and in oxygen saturated LBE at 300°C. The
The effect of LBE on resistance to fatigue crack initiation can be appreciated in Figure 7 where the fatigue lives are shorter in LBE than in air. The density of short cracks was very high after tests in air while very low after test in LBE. This means that LBE promoted the overcoming of structural barriers such as grain boundaries. This is easily explained by the repetitive shear at the short crack tip which always enables or postpones the formation of an oxide layer and guides adsorption of atoms which allows rapid propagation into the bulk.

**Figure 7**: Low cycle fatigue resistance curves for T91 steel tested at 300°C in air and in LBE

The fatigue crack growth (FCG) rate in the Paris regime ($\Delta K > 10$ MPa.m$^{1/2}$) is approximately one order of magnitude higher in LBE than in air (Figure 8a). The increase in FCG rate was associated with a change in fracture mode. While ductile striations (Figure 8b) were observed in specimen fatigued in air, a mixed trans- and intergranular fracture was present in the specimen fatigued in LBE (Figure 8c). The change in fatigue resistance and in fracture mode even in LBE containing a high level of oxygen is easily explained by the repetitive shear at the long crack tip which always enables or postpones the formation of an oxide layer and guides adsorption of atoms. In other words, the effect of LBE on the behaviour of short and long crack seems to be similar.

**Figure 8**: Fatigue crack propagation rate vs cyclic stress intensity factor for the T91 tested at 300°C in air or in oxygen saturated LBE (a); SEM image of ductile striations in the TR750 material after fatigue in air (b) and after fatigue in LBE (c)

**Conclusions**

T91 steel exhibits in general a ductile response when loaded in various conditions even in presence of LBE.
Nevertheless, there exists a set of conditions which appears critical and for which LBE embrittlement is promoted and observed:

- a tempering treatment that leads to harden the steel;
- a very low strain rate.

Decreasing the dissolved oxygen in LBE appears as a catalyst factor.

The detrimental effect observed after low straining and cyclic straining seems to be similar.

Because even in the recommended heat treatment and in oxygen saturated LBE brittle fracture has been observed, T91 exhibits a limited reliability for ADS components.

Acknowledgements

The authors would like to acknowledge the financial support from the French CNRS research programme GEDEPEON and from the European Union for the TECLA, MEGAPIE-test and DEMETRA projects.

References


R&D activities on oxygen sensor and potential control for lead-bismuth eutectic

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²ATOX Co., Ltd. (Current affiliation)

Abstract

JAEE is planning to build a Transmutation Experimental Facility (TEF) for R&D on volume reduction and mitigation of harmfulness of high-level radioactive waste by using an accelerator-driven system (ADS). In the ADS target test facility (TEF-T) of TEF, a liquid lead-bismuth eutectic (LBE) target will be irradiated with a high-power (250 kW) proton beam, and irradiation effects on structural materials will be studied. LBE is corrosive, so it is necessary to control oxygen concentration in LBE adequately to protect structural materials from the corrosion. JAEA has tried to fabricate an oxygen sensor to measure the oxygen concentration and to control it. In this paper, current two R&D activities related to the measurement and control of the oxygen concentration in LBE are introduced. The first activity is the fabrication of the oxygen sensor. Another one is the control of the oxygen concentration in the LBE flow condition.

Introduction

The Japan Atomic Energy Agency (JAEA) is planning to build a Transmutation Experimental Facility (TEF) for R&D on volume reduction and mitigation of harmfulness of high-level radioactive waste by using an accelerator-driven system (ADS). In the ADS target test facility (TEF-T) of TEF, a liquid lead-bismuth eutectic (LBE) target will be irradiated with a high-power (250 kW) proton beam, and irradiation effects on structural materials will be studied. It is supposed that LBE is the promising candidate material as the ADS’s proton beam spallation target and core coolant because it has good neutron yield and is inactive chemically in comparison with other coolant materials (e.g. water, sodium). On the other hand, LBE is corrosive, so it is necessary to control oxygen concentration in LBE adequately to protect structural materials from the corrosion. In order to control the oxygen concentration in LBE, it is required to develop an oxygen sensor to measure the oxygen concentration.

JAEA has tried to fabricate the oxygen sensor and to measure the oxygen concentration in LBE. As the oxygen sensor, an electrochemical oxygen sensor which has been studied several studies [1-3], is employed. YSZ (yttria-stabilised zirconia) is used as a solid electrolyte. Theoretically, it is available to measure the oxygen potential by the electrochemical oxygen sensor, however, in the future use at TEF-T, the following functions are also necessary; a) protection of YSZ debris flow-out when YSZ breaks and b) protection of LBE leakage from the sensor when YSZ breaks. In this paper, a fabrication of the oxygen
sensor with these functions is introduced. The measurement and control of oxygen potential in LBE has been performed in the flow condition and its result is also introduced.

Theory

A principle for measurement of oxygen concentration using a solid electrolyte is presented in Figure 1. A solid electrolyte separates two regions which have different oxygen partial pressures. When the electrolyte is an oxygen-ion (O\(^{2-}\)) conductor, an electrochemical galvanic cell using a solid electrolyte is presented as follows: \(P_{O_2}(\text{reference})/\text{solid electrolyte}/P_{O_2}\), where \(P_{O_2}(\text{reference})\) is the oxygen partial pressure at the reference electrode (\(P_{O_2}\) high in Figure 1) and \(P_{O_2}\) is the oxygen partial pressure at the working electrode (\(P_{O_2}\) low in Figure 1). An electromotive force (EMF) is formed across the solid electrolyte between the different oxygen partial pressures. According to the Nernst equation, EMF, \(E\) is expressed as follows:

\[
E = \frac{RT}{4F} \ln \frac{P_{O_2}(\text{reference})}{P_{O_2}}
\]  

(1)

where \(R\) is the gas constant, \(T\) is temperature [K] and \(F\) the Faraday constant. When \(P_{O_2}(\text{reference})\) is already known, the oxygen partial pressure at the working electrode, \(P_{O_2}\) can be calculated by the measurement of \(E\) and Eq. (1). The EMF can be measured using a voltmeter. Figure 1 also shows a schematic diagram of an oxygen sensor and interface reaction in measurement of oxygen concentration in liquid LBE using YSZ as a solid electrolyte and Pt/gas reference electrode. The oxygen activity, \(a_o\) in equilibrium with an oxygen pressure \(P_{O_2}\) is written assuming that dissolution of oxygen into liquid LBE obeys the Henry’s law:

\[
a_o = \gamma_o C_o = C_o = \left(\frac{P_{O_2}}{P_{O_2}^s}\right)^{\frac{1}{2}}
\]  

(2)

where \(\gamma_o\) is an activity coefficient, \(C_o\) is the oxygen concentration in LBE, \(C_o^s\) is the saturated oxygen concentration in LBE and \(P_{O_2}^s\) is the oxygen concentration in the air in equilibrium with oxygen-saturated LBE. The activity \(a_o\) becomes unity when the oxygen dissolved in LBE attains the level of saturation (\(C_o = C_o^s\)). The saturated oxygen concentration in LBE is calculated using the following Orlov’s correlation:

\[
\log C_o^s = 1.2 - \frac{3400}{T}
\]  

(3)

where the unit of \(C_o^s\) is weight %.

Figure 2 shows a schematic diagram to measure oxygen concentration in liquid LBE using Pt/air type oxygen sensor. When measurement is performed in LBE, the air is used as the reference gas in Pt/air reference system. The SUS rod is used as an electrode immersed in LBE. Therefore, the system for measurement in LBE is represented by Pt/air/YSZ/LBE/304SS. The relationship between the EMF and the oxygen concentration in LBE has been calculated for the reference electrode sensor using standard Gibbs energy of PbO. The correlations derived by Courouau et al. [5],

\[
E_{sat} = 0.128 - 6.368 \times 10^{-5}T
\]  

(4)
$$E = -0.210 + 5.538 \times 10^{-5}T - 4.309 \times 10^{-5}T \ln C_o$$  \hspace{1cm} (5)$$

were used in this study.

**Figure 1:** Principle for measurement of oxygen concentration using solid electrolyte

![Figure 1: Principle for measurement of oxygen concentration using solid electrolyte](image1)

**Figure 2:** Schematic diagram of Pt/air type oxygen sensor

![Figure 2: Schematic diagram of Pt/air type oxygen sensor](image2)

**Fabrication of oxygen sensor**

**Pt/Air type sensor**

Figure 3 presents the appearance of the Pt/air oxygen sensors. The total length of the sensor is about 240 mm and the length of YSZ is about 100 mm. ZR-8Y which is a partially stabilised zirconia with 8 mol% yttria was employed as YSZ tube. There is an air hole to use the air as the reference.

Figure 4 shows the measurement result by the fabricated oxygen sensors under saturated oxygen condition. The black line in this figure corresponds to the theoretical value calculated by Eq. (4). It was confirmed that the measured values above 400 °C were almost same as the theoretical value.
Other functions

As mentioned above, when YSZ will break in the TEF-T loop, the following functions will be required; (a) protection of YSZ debris flow-out to LBE flow and (b) protection of LBE leakage from the sensor.

To protect the flow-out of YSZ debris, a stainless steel sheath shown in Figure 5 was prepared. The sheath had small holes (diameter = 0.8 mm, pitch = 1.55 mm) to contact LBE. In such situation, it was considered that PbO and other oxide might remain a space between YSZ and the sheath, then the output of the sensor might change. To confirm this point, the comparison of EMF with and without the sheath was carried out. Figure 6 shows the measurement result. It was confirmed that there was little difference for the EMF value with the use of the sheath.

For the protection of LBE leakage from the sensor, a freeze seal design was employed in the housing of the oxygen sensor. Figure 7 presents the conceptual diagram of the freeze seal design. In this figure, the left sensor is a simple configuration of the Pt/air oxygen sensor. In this design, the leakage of LBE will occur when YSZ breaks because there is no obstacle inside the housing. On the other hand, as the freeze
seal design, a narrow flow channel is prepared inside the housing. Due to the narrow flow channel, a pressure drop of LBE flow will increase and a velocity of the LBE flow will decrease. It means a time which LBE reaches to the air hole, will increase. This design expects that LBE will freeze before LBE reaches to the air hole.

To confirm the feasibility of the freeze seal concept, an experiment was performed. In this experiment, a housing itself was prepared and set as shown in Figure 7. A certain pressure was given to LBE and the LBE leakage was checked. The LBE temperature was 450 °C in all cases. Table 1 summarises the experimental results. The TEF-T loop is designed that the maximum pressure is 0.3 MPa and the pressure during the operation is 0.1 MPa. Through the experiments, it was confirmed that there was no leakage of LBE even if 0.5 MPa pressure was given. It means the freeze seal design can employ for the use in the TEF-T loop.

Figure 5: Stainless steel sheath to protect the flow-out of YSZ debris

![SUS sheath](image)

Figure 6: Comparison of EMF with / without SUS sheath

![Graph](image)
Measurement and potential control in flow condition

Measurement and potential control in flow condition were performed in JLBL-4 (JAEA Lead-Bismuth Loop). JLBL-4 is a small LBE loop for the experiment to measure LBE flowrate by UVP (Ultrasonic Velocity Profiling) method. Table 2 presents main parameters of JLBL-4. In this experiments, the LBE temperature was 350 °C and the LBE flowrate was 26 L/min. Another type of the oxygen sensor, which was Bi/Bi2O3 type made by SCK•CEN [3], was used in this experiment because the Bi/Bi2O3 sensor did not need air (no air hole). At that point, the freeze seal design was not realised, so the sensor which had no possibility of the LBE leakage even if YSZ broke, was used.

Figure 8 shows the experimental result in JLBL-4. Basically, the oxygen concentration decreases by the generation of an oxide coating in the structure such as piping. It is considered that the oxygen concentration during the TEF-T operation will maintain $10^{-5} - 10^{-7}$ wt% to prevent the corrosion. So, when the oxygen concentration deceased below $10^{-7}$ wt%, dry air was injected. As shown in Figure 8, the oxygen concentration was controlled between $10^{-5} - 10^{-7}$ wt%. However, as the future work, the oxygen concentration should be maintained as constant.

Table 2: Main parameters of JLBL-4

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Main material</td>
<td>316SS</td>
</tr>
<tr>
<td>Inventory</td>
<td>20 L</td>
</tr>
<tr>
<td>Max. pressure</td>
<td>6 bar</td>
</tr>
<tr>
<td>Max. electrical power</td>
<td>7 kW heaters</td>
</tr>
<tr>
<td>Max. design temperature</td>
<td>500 °C</td>
</tr>
<tr>
<td>Flow rate of LBE</td>
<td>Max. 40 L/min</td>
</tr>
<tr>
<td>Flow rate of observation</td>
<td>EMF</td>
</tr>
</tbody>
</table>

Table 1: Experimental results for freeze seal design

<table>
<thead>
<tr>
<th>Case</th>
<th>Pressure to LBE [kPa]</th>
<th>Pressurisation time [min]</th>
<th>LBE leakage</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>206.0</td>
<td>8</td>
<td>No</td>
</tr>
<tr>
<td>B</td>
<td>306.8</td>
<td>13</td>
<td>No</td>
</tr>
<tr>
<td>C</td>
<td>520.0</td>
<td>10</td>
<td>No</td>
</tr>
</tbody>
</table>

Figure 7: Conceptual diagram for freeze seal design
Summary

In an effort to use in the TEF-T loop, JAEA has been tried to fabricate the oxygen sensor. As the result, the Pt/air type oxygen sensor with the stainless steel sheath and the freeze seal design was developed. It was confirmed that output voltage of the sensors was adequate in a wide temperature range. It was observed that there was little difference for the EMF value by using the sheath. It was also confirmed that there was no leakage of LBE even if 0.5 MPa pressure was given. The developed sensor will be installed to the TEF-T loop.

The control of oxygen potential in LBE was also performed in the flow condition. Basic oxidation operation was performed and the oxygen concentration was maintained the target range, $10^{-5} - 10^{-7}$ wt%. As the future work, the oxygen concentration should be maintained as a constant.

References


High-pressure water Injecting into LBE: Pressure wave and transient temperature distribution phenomena of heat exchanger tube rupture on KYLIN-II-S facility

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Abstract

Lead-bismuth eutectic (LBE) has been proposed as the primary candidate spallation target and coolant materials for China LEAd alloy-cooled Reactors (CLEAR) with the use of high-pressure water as secondary loop coolant. However, the risk of leakage of water into lead alloy due to heat exchanger tube rupture (HXTR) will cause an intolerant pressure increasing and steam dragging in the reactor core.

Equipped with a water vessel (25MPa, 500°C), a reaction vessel (25MPa, 500°C) and other indispensable auxiliary systems, KYLIN-II-S facility was designed and constructed in the Institute of Nuclear Energy Safety Technology (INEST-CAS) and could be used to predict the multiphase thermal and flow phenomena of a HXTR for CLEAR.

After the commissioning stage of the facility and qualification of all measurement and data acquisition systems, KYLIN-II-S was operated to perform water (2MPa, 200°C) injecting into LBE (0.1MPa, 400°C), corresponding peak pressure due to water vapourisation and jet expansion was recorded by pressure transducers, but the maximum peak pressure, as detected in reaction vessel, never overcame the value of the injecting pressure. A three-dimensional temperatures distribution was also detected by hundreds thermocouples installed inside the reaction vessel. The NTC code was used to simulate the performed test and experimental data were in good agreement agreed with the calculated data.

Keywords: Lead-bismuth, CLEAR, HXTR
Session 5: Current ADS experiments

Chair: K. Tsujimoto
Japan Atomic Energy Agency, Japan
Design, construction, and commissioning of Kharkiv Institute of Physics and Technology (KIPT) ADS facility

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Abstract

As a part of the Russian Research Reactor Fuel Return (RRRFR) programme of the United States Department of Energy, Argonne National Laboratory (ANL) of United States and Kharkiv Institute of Physics and Technology (KIPT) of Ukraine are collaborating on the design and the construction of an accelerator-driven system (ADS). RRRFR is a trilateral initiative among the United States, the Russian Federation, and the International Atomic Energy Agency (IAEA) to repatriate high enriched uranium fuels (fresh and irradiated) to Russia. This ADS is designed with a proliferation-resistant low-enriched uranium (LEU) fuel and is driven by an electron accelerator. The facility design has been developed to maximise its utility and minimise the time for replacing the target, the fuel and the experimental assemblies by using simple and efficient procedures. The subcritical assembly is designed to obtain the highest possible neutron flux intensity with an effective neutron multiplication factor of <0.98. The facility will be used for producing medical isotopes, training young nuclear professionals, supporting the Ukraine nuclear industry, and providing capability for performing material research, and basic science experiments. This paper describes the design, the construction experience and the commissioning work for starting the facility operation.
Physical design of multifunctional lead-based zero-power reactor CLEAR-0

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Key Laboratory of Neutronics and Radiation Safety, Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences, China

Abstract

Liquid lead or lead-based alloy is a potential candidate coolant for fast reactors and hybrid systems because of its many unique nuclear, thermal-physical and chemical attributes. According to the latest roadmap of Generation IV reactors proposed by the GIF organisation in 2013, the lead-cooled fast reactor is expected to be the first Generation IV nuclear system to achieve industry demonstration and commercial application.

A multifunctional lead-based zero-power reactor is proposed to simulate the cores of various lead-based reactors such as critical fast reactors with different fuel loading, subcritical reactors driven by different neutron sources, thermal-fast-coupled critical reactors with different layouts. It requires the core configuration to be flexible with sufficient safe considerations. The fuel rods are cylindrical and they can be arranged in triangle or square. Generally, the fuel rods are surrounded with solid Pb-Bi, and the central area of core is occupied by external neutron source which is introduced from upper side of core, the $k_{eff}$ of the core is kept at 0.97. Several control rods are assembled in the active zone to adjust the $k_{eff}$ and avoid critical accidents. When the facility is used for experiments for other kinds of reactors, part of the surrounding and reflector material can be changed to water or graphite, the central area can also be filled with fuel rods and “coolant-simulator”, making the neutron spectra in core changed from fast spectrum to thermal spectrum.

Detail design and analysis for the facility is undertaking. With the flexibility, the facility can be used to perform zero-power experiments for numerous new reactor concept and providing data for their licence procedure.

Keywords: Zero-power reactor, Lead-based, nuclear design, core physics
Simulation of the MYRRHA core design in the zero-power coupling experiments of the FREYA Project

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5Institut de Physique Nucléaire d’Orsay, CNRS/IN2P3, Université Paris-Sud, France

Abstract

One of the main aims of the FP7 FREYA Project was to validate a methodology for online subcriticality monitoring to be used in the ADS MYRRHA. These experiments have been realised at SCK•CEN during the last five years in the zero-power ADS system consisting of the VENUS-F reactor coupled with the GENEPI-3C accelerator. In the period 2015 to March 2016, the experimental methods that were tested and selected in the first years of the project, were applied in a set of subcritical cores that simulated some peculiarities of the MYRRHA core design. These features mainly concern simulations of MYRRHA in-pile-sections (IPS) and reflector in the VENUS-F core. Before performing the measurements in the subcritical VENUS-F cores driven by the GENEPI-3C accelerator, similar critical VENUS-F cores were assembled and investigated. The rod drop measurements needed for the MSM method, which are applied as references for the subcritical measurements, as well as the core characterisation measurements were executed in all these critical cores. Reactivity effects as void effect of the coolant, fuel temperature, water ingress and fuel agglomeration effects were investigated to support the safety and licensing issues of the MYRRHA core design. The VENUS-F critical and subcritical configurations for the mock-up of the MYRRHA core, the experimental methods applied and general conclusions are presented.

Introduction

EU FP7 five-year FREYA Project [1] (Fast Reactor Experiments for hYbrid Applications) has been recently (March 2016) accomplished. All objectives of the project were fulfilled:

- Complete the experimental programme for the validation of the methodology for online reactivity monitoring initiated within the GUINEVERE [2] project in EUROTRANS (Work Package 1);
- Conduct experiments with a set of VENUS-F cores in support of the design and licensing of MYRRHA/FASTEF [3] (Work Packages 2 and 3);
- Conduct experiments in support of the design and licensing of lead fast reactors (Work Package 4).
In more details, the investigations concerning the first item (WP1) were related to different subcriticality levels definition for the nominal operation mode of an ADS from the point of view of methodology. This means that different subcritical configurations with $k_{eff}$ values in the range 0.95-0.97 were loaded and investigated with regard to the applicability of different measurement techniques. Also, specific configurations with a deeper subcritical level of 0.9 were investigated to study the determination of the subcriticality level during core loading operations. In FP5 project MUSE [4], the reflector effect was shown to be of significant influence on specific reactivity monitoring techniques. In FREYA in order to investigate the robustness of the methodology with regard to the reflector effect, several experiments with different reflector materials were performed. To complete the work, the robustness of the reactivity indicators with regard to the source position, a change of the vertical height of the source target was investigated to reproduce possible variations of the height of the spallation source in a real ADS. The first results of the project FREYA regarding the methodology for online reactivity monitoring were presented at the previous TCADS-2 meeting in Nantes [5-7]. The recent FREYA experiments devoted to the simulations of specific details of the MYRRHA core design in several critical and subcritical VENUS-F cores and the application of the methods to control reactivity that have been chosen at the first stage of the project are presented in this paper.

The second main objective was to support the design and licensing of the MYRRHA/FASTEF reactor, and to provide experimental data for validation of neutron transport codes and nuclear data libraries used in lead fast reactor development. This involves implementing core configurations in the VENUS-F reactor that simulate the neutronic characteristics of the MYRRHA/FASTEF design. Since the MYRRHA/FASTEF design is currently expected to have both critical and subcritical modes of operation, two work packages have been specifically designed for each of these modes of operation. Thus, WP2 aimed to investigate the subcritical mock-ups of the MYRRHA/FASTEF reactor while WP3 aimed to investigate the critical ones. Due to the demand of the safety authorities to have the critical core each time before the subcritical one, the order of WPs as had planned in the Project was changed: WP3 was realised before WP2.

WP3: MYRRHA critical mock-ups

Three core configurations characteristic of MYRRHA have been investigated within WP3. They have been denoted as CC5, CC7 and CC8. The most distinctive feature of the new cores with respect to the cores investigated in WP1 is the introduction of a new type of fuel assembly. The WP1 fuel assembly contained only 30% U metal fuel rodlets and lead blocks. In the new assembly, oxygen was inserted in the form of Al$_2$O$_3$ rodlets, in order to better simulate the spectrum of the current MYRRHA-FASTEF design, which envisages the use of MOX fuel. This new assembly contains 13 U metal rodlets, 4 Al$_2$O$_3$ rodlets and 8 Pb blocks, and thus has been designated as U13Al4Pb8 (Figure 1).
Critical configurations have been investigated

The CC5 configuration simulated the “clean” MYRRHA core, i.e. only contained fuel and lead elements. In the CC7 configuration, reactor grade graphite assemblies were introduced in order to simulate the MYRRHA reflector, planned to be made of BeO. Finally, in the CC8 configuration (see Figure 2), two additional assemblies were inserted to simulate the thermal spectrum in-pile sections (IPS) of MYRRHA, intended for Mo-99 production. Two different IPS mock-up assemblies have been designed. They both are made up of polyethylene blocks wrapped with Cd, but differ in the way the highly enriched uranium Mo-99 production targets are simulated. Thus, the first IPS mock-up assembly, placed at position (-4, 1), contains a 30 w/o uranium rodlet, while the second one, at position (4, 1), contains two aluminium-wrapped 93 w/o uranium foils.
Figure 2: CC8 VENUS-F critical core configuration with graphite simulated BeO MYRRHA reflector and in-pile sections (IPS)

<table>
<thead>
<tr>
<th>Type of assembly</th>
<th>Amount</th>
</tr>
</thead>
<tbody>
<tr>
<td>- PbA, lead assemblies</td>
<td>63</td>
</tr>
<tr>
<td>- PbA with holes for detectors</td>
<td>9</td>
</tr>
<tr>
<td>- GA, assemblies with graphite</td>
<td>20</td>
</tr>
<tr>
<td>- FA fuel assemblies (including EFA)</td>
<td>41</td>
</tr>
<tr>
<td>- CR control rods</td>
<td>2</td>
</tr>
<tr>
<td>- POAR rod drop</td>
<td>1</td>
</tr>
<tr>
<td>- SR, safety rods</td>
<td>6</td>
</tr>
<tr>
<td>- IPS</td>
<td>2</td>
</tr>
</tbody>
</table>
**Cores characterisation and reactivity effects measurements**

The list of the measurements performed in critical configurations of VENUS-F reactor in the WP3 of FREYA is presented in the Table 1.

**Table 1:** Summary of the experiments performed in the critical core configurations of VENUS-F investigated during WP3 of the FREYA Project

<table>
<thead>
<tr>
<th>CORE</th>
<th>CC5</th>
<th>CC7</th>
<th>CC8</th>
</tr>
</thead>
<tbody>
<tr>
<td>Control Rod worth</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>Axial traverses</td>
<td>U-235</td>
<td>U-238</td>
<td>Pu-239</td>
</tr>
<tr>
<td></td>
<td>U-237</td>
<td></td>
<td>Np-237</td>
</tr>
<tr>
<td>Radial traverses</td>
<td>U-235</td>
<td>U-235</td>
<td>U-235</td>
</tr>
<tr>
<td>Spectral indexes / Minor actinide responses</td>
<td>F28/F25</td>
<td>F28/F25</td>
<td>F28/F25</td>
</tr>
<tr>
<td></td>
<td>F25</td>
<td>F25</td>
<td>F25</td>
</tr>
<tr>
<td></td>
<td>F49/F25</td>
<td>F49/F25</td>
<td>F49/F25</td>
</tr>
<tr>
<td></td>
<td>F37/F25</td>
<td>F37/F25</td>
<td>F37/F25</td>
</tr>
<tr>
<td></td>
<td>F42/F25</td>
<td>F42/F25</td>
<td>F42/F25</td>
</tr>
<tr>
<td></td>
<td>F51/F25</td>
<td>F51/F25</td>
<td>F51/F25</td>
</tr>
<tr>
<td>Coolant voiding</td>
<td>---</td>
<td>Yes</td>
<td>---</td>
</tr>
<tr>
<td>Fuel temperature</td>
<td>Yes</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

The measurement of control rod worth is one of the most typical reactor physics experiments. Within WP3 of FREYA, the worth of the control rods was measured using the well-known positive period method. In this method, starting from a critical state of the system, one of the control rods is partially extracted so that a positive reactivity (<1$\%$) is introduced. In this way, the power of the system (and, consequently, the count rate in the flux monitors) will rise exponentially, and the period $T$ of this exponential can be related with the reactivity $\rho$ through the inhour Equation 1:

$$\rho = \frac{\Lambda_{\text{eff}}}{\Lambda_{\text{eff}} + T} + \frac{T}{\Lambda_{\text{eff}} + T} \sum \frac{\beta_{\text{eff},i}}{1 + \lambda_iT} \approx \sum \frac{\beta_{\text{eff},i}}{1 + \lambda_iT}$$

(1)

where $\beta_{\text{eff},i}$ are the effective delayed neutron fractions of the $i^{th}$ group with decay constants $\lambda_i$.

Figure 3 illustrates the differential curves of control rods worth obtained in the CC5 core using Equation 1. Because of the limitation of this method (if one rod is completely withdrawn the second one is still in the core) part of the curve is obtained by polynomials extrapolation of the measured data.
Figure 3: Differential curves of the control rods worth in the CC5 core
Height 0 mm corresponds to the lower position of CRs

In an axial traverse experiment, fission rates in a number of isotopes are measured with 8 mm fission chambers (FC) alongside an experimental channel of fuel assemblies EFA-1 and EFA-2 or in channel of lead, IPS and graphite assemblies (see Figure 2). Typical axial traverse shape in the core (position (-3,1)) with U-235 and U-238 FCs and specific shape of U-235 traverse in the IPS mock-up (containing polythene and U-235 90% enriched foils) are shown in Figures 4 and Figure 5.

Figure 4: U-235 and U-238 axial traverse in the CC8 core position (-3,1)
Height 0 cm corresponds to the lower position of the fuel
Radial traverse measurements were performed in experimental channels in different assemblies placed at different positions within the CC5, CC7 and CC8 core configurations. The positions measured experimentally in every core and their distances from the core centre are shown in Table 2 and Table 3. All measurements have been performed at the core midplane. Notice that for the case of the CC5 and CC7 cores, radial traverses have been performed only in one side of the reactor, while in CC8 core measurements have been performed in both sides of the reactor. Furthermore, for the case of CC8, measurements have been performed with both U-235 and U-238 fission chambers (FC). The radial traverse obtained with U-235 FC in the CC8 core is shown in Figure 5. The presence of polythene in the IPS and graphite in reflector are the reasons of the complex shape of this traverse.

Table 2: Radial traverses positions measured in the CC5 and CC7 cores

<table>
<thead>
<tr>
<th>Position</th>
<th>B1</th>
<th>C1</th>
<th>(-6,1)</th>
<th>(-5,1)</th>
<th>(-4,1)</th>
<th>(-3,1)</th>
<th>(-2,1)</th>
<th>(1,1)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Material</td>
<td>Lead</td>
<td>Lead</td>
<td>Lead/Graphite</td>
<td>Lead</td>
<td>Lead</td>
<td>Fuel</td>
<td>Fuel</td>
<td>Fuel</td>
</tr>
</tbody>
</table>

Table 3: Radial traverses positions measured in the CC8 core

<table>
<thead>
<tr>
<th>Position</th>
<th>B1</th>
<th>C1</th>
<th>(-6,1)</th>
<th>(-5,1)</th>
<th>(-4,1)</th>
<th>(-3,1)</th>
<th>(-2,1)</th>
<th>(1,1)</th>
<th>(4,1)</th>
<th>(5,1)</th>
<th>(6,1)</th>
<th>B2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Material</td>
<td>Lead</td>
<td>Lead</td>
<td>Lead</td>
<td>Graphite</td>
<td>IPS</td>
<td>Fuel</td>
<td>Fuel</td>
<td>IPS</td>
<td>Lead</td>
<td>Graphite</td>
<td>Lead</td>
<td></td>
</tr>
</tbody>
</table>
In a spectral index measurement, the ratio between count rates in fission chambers made of different isotopes is obtained. From these ratios it is possible to extract information related to the neutron energy spectrum within the assembly. For the measurements of the spectrum indexes and the MA fission rate ratios, small FCs were used. All these FCs have 4 mm outer diameter and small deposit mass (20-200 μg). FCs with the following deposits were used for the measurements: U-235, U-238, Pu-239, Pu-240, Pu-242, Np-237 and Am-241. For the measurements these FCs were placed in the same specific experimental fuel assemblies (EFA) as used for the axial traverses, which have a special stainless steel guiding tube instead of the standard FA element in the 5×5 FA structure or in others, not fuel assemblies with appropriate holes in the middle plane of the core. In the CC7 core the measurements were performed only in one specific position and not with all FCs while in the CCS and CC8 cores many positions along the core radius and all available FCs were used. In this way, an extensive range of spectral conditions were investigated, see Figure 6. Comparison of the spectral index measurements with Monte Carlo and deterministic calculations was published in [8]. Next, updated version of this comparison will be presented in the ND-2016 Conference in Bruges in September 2016.
Some reactor operation parameters, such as temperature coefficient and void effect of the coolant, were determined in CC7 core. These two important reactivity effects are always the subjects of safety reports of a new installation and should be investigated.

For the Lead Void Reactivity Effect (LVRE) five voided fuel assemblies were prepared for the experiments. They were loaded into the core centre one by one, replacing the standard FAs, to simulate the separate and cumulative effects of the coolant voiding. In all cases, after introduction of the voiding in the core, the core then was made critical by moving two CRs together. The LVRE was estimated using the CRs calibration curves. The results of these estimates are presented in Figure 7. Analysing the measurement results, one can state that LVRE with one to the four central voided FAs has a positive reactivity effect, with a value of about 5 pcm per assembly, and that it is accumulative. But after adding one additional voided FA, in addition to these central four, the reactivity effect is negative.

Temperature effect was observed as a side-effect during a power run of the VENUS-F reactor dedicated to the measurement of the spectrum index F28/F25 with activation foils. The reactor was operated at a power of about 100 W during 6.5 h. A small increase of temperature by about 4°C was detected with thermocouples placed in the middle of the core. This small increase of temperature caused a negative reactivity effect that had to be compensated by the extraction of the control rods by about 2.4 mm to keep the reactor critical. Thanks to the precisely calibrated control rods, the reactivity effect of the temperature variation was determined to be (2.9 ± 0.3) pcm. The uncertainty of the reactivity change is mainly determined by the uncertainty of the control rod height.
WP2: VENUS-F subcritical configuration for design and licensing of MYRRHA

The objective of Work Package 2 of the FREYA Project was to perform an experimental programme in support of the design and licensing of the MYRRHA subcritical core. The facility was made to work in ADS mode by coupling the subcritical VENUS-F reactor with the GENEPI-3C accelerator.

SC-8 VENUS-F subcritical MYRRHA mock-up

The subcritical core configuration SC8 (see Figure 8) at the VENUS-F reactor was prepared on the base of the previous critical VENUS-F core CC8 (Figure 1) which simulated the MYRRHA critical core design. The SC8 core, as well as its forerunner CC8, simulated the following peculiarities of the MYRRHA design: BeO reflector (using graphite) and in-pile sections (IPS). The principal difference between CC8 and SC8 cores is the presence of the vertical beam line of GENEPI-3C accelerator in the centre of the SC8 core. To arrange this vertical line, four central assemblies were moved to the border of the core making this core subcritical with $k_{eff}$ about 0.96. This subcritical level was thoroughly determined with several methods have been chosen in WP1 of FREYA as candidates for reactivity control in the future ADS designs and especially in MYRRHA. The results of such criticality obtained with two methods (BTM – beam-trip method, SJI – source jerk integral) as usual for zero-power experiments, were compared with reference MSM method values [9].

Reactivity effects in MYRRHA subcritical mock-up VENUS-F cores

Since the objective of WP 2 of the FREYA Project was to perform an experimental programme in support of the design and licensing of the MYRRHA subcritical core, reactivity effects because of water ingress in the core and fuel agglomeration were simulated in the SC8 VENUS-F core and measured with different methods. Based on preliminary calculated results, three modifications of the reference SC8 core were chosen for reactivity effect measurements:

- For fuel agglomeration effect a special fuel assembly was designed which simulated the situation when the fuel melts in the upper part of the core and agglomerates in the bottom part. This special fuel assembly replaced standard FA in position (-1,4).
Figure 8: SC8 VENUS-F subcritical core configuration

<table>
<thead>
<tr>
<th>Type of assembly</th>
<th>Amount</th>
</tr>
</thead>
<tbody>
<tr>
<td>- PbA, lead assemblies</td>
<td>59</td>
</tr>
<tr>
<td>- PbA with holes for detectors</td>
<td>9</td>
</tr>
<tr>
<td>- GA, assemblies with graphite</td>
<td>20</td>
</tr>
<tr>
<td>- FA fuel assemblies (including EFA)</td>
<td>41</td>
</tr>
<tr>
<td>- CR control rods</td>
<td>2</td>
</tr>
<tr>
<td>- POAR rod drop</td>
<td>1</td>
</tr>
<tr>
<td>- SR, safety rods</td>
<td>6</td>
</tr>
<tr>
<td>- IPS</td>
<td>2</td>
</tr>
</tbody>
</table>
- For water ingress increasing neutron leakage the lead reflector assembly PbA in position (-1,5) was replaced with assembly containing polyethylene (PE).
- For water ingress increasing moderation the standard FA in position (-3,4) was replaced with FA containing 2 PE plates instead of Pb plates.
- All these reactivity effects were measured with the methods cited above. The results obtained relatively to the unperturbed core are shown in Table 5, together with MCNP5 calculations.

### Table 5: Reactivity effects relative to the reactivity of the reference core without perturbation obtained with different experimental methods and MCNP calculations

<table>
<thead>
<tr>
<th>Effect</th>
<th>MCNP</th>
<th>MSM</th>
<th>BTM</th>
<th>SJI T=20 ms</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel agglomeration</td>
<td>+13±9</td>
<td>+57±33</td>
<td>+88±51</td>
<td>+43±3</td>
</tr>
<tr>
<td>Value, pcm</td>
<td>± pcm</td>
<td>± pcm</td>
<td>± pcm</td>
<td>± pcm</td>
</tr>
<tr>
<td>Water ingress leakage</td>
<td>-366±9</td>
<td>-282±34</td>
<td>-288±51</td>
<td>-358±13</td>
</tr>
<tr>
<td>Value, pcm</td>
<td>± pcm</td>
<td>± pcm</td>
<td>± pcm</td>
<td>± pcm</td>
</tr>
<tr>
<td>Water ingress moderation</td>
<td>+331±10</td>
<td>+288±37</td>
<td>+301±51</td>
<td>+312±8</td>
</tr>
</tbody>
</table>

### Conclusions

Within the FREYA Project, several fast lead critical and subcritical VENUS-F cores were assembled and investigated to support the safety and licensing issues of the MYRRHA core design.

A large number of measurements as control rod worth, axial and radial traverses, spectral indices as well as reactivity effects as coolant (lead) void, fuel temperature, water ingress and fuel agglomeration were performed to provide experimental data for validation of neutron transport codes and nuclear data libraries.

Several experimental techniques were investigated as candidates for subcriticality monitoring to be used in the ADS MYRRHA. The most promising ones are the beam-trip method and source jerk integral. The subcriticality levels of many VENUS-F core configurations were determined using these methods. The results agree with the reference MSM method within the uncertainties in most of the cases.

Further investigation of the subcriticality monitoring techniques is planned within the MYRTE Project [10].

### Acknowledgements

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References


Reactivity determination and monitoring in FREYA Project subcritical cores:
Assessment and correction of spatial and energy effects

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Abstract
During the EU FP7 FREYA Project, reactivity monitoring experiments, including pulsed neutron source (PNS) and beam trip (BT) experiments, have been performed in a number of subcritical configurations of the VENUS-F zero-power reactor in the SCK•CEN. Starting from the basic SC1 core, consisting of a Lead Fast ADS mock-up with U₃O₈ fuel and a lead reflector, a number of different cores have been implemented by modifying the (D,T) neutron source height, by introducing stainless steel in the reflector, and by simulating some features of the projected MYRRHA facility (thermal islands, graphite reflector, introduction of Al₂O₃ in the core). Furthermore, the variation of the control and safety rod position has allowed investigating an even wider range of reactivity levels.
This work investigates the capability of Monte Carlo simulations to assess the spatial and energy effects that affect the reactivity monitoring techniques based on the point kinetics model (area ratio, source jerk, prompt decay constant) that are commonly applied to extract the reactivity from the results of PNS and BT experiments. The MCNP 6.1.1 code has been used for this purpose. Furthermore, it is studied the possibility to apply these Monte Carlo simulations to obtain general relationships between the reactivity and the measured parameters that can replace the point kinetics ones to provide accurate reactivity results when spatial and energy effects are relevant. Finally, the applicability of this methodology for analysing and correcting the spatial and energy effects that can affect reactivity monitoring techniques in the future MYRRHA facility is discussed.

Introduction
The integration of accelerator-driven subcritical systems (ADS) in the nuclear fuel cycle is a widely recognised option to transmute the minor actinides present in the spent nuclear fuel and thus reduce the hazards posed by high-level nuclear waste. However, prior to the deployment of industrial scale ADS, a good knowledge of the physics of accelerator-driven subcritical systems and the development of accurate techniques for determining the subcriticality level of such systems is required in order to guarantee its safe operation and to maintain the safety margins to prevent the system to become critical.

With these purposes, a number of zero-power experiments have been conducted in the last years, such as MUSE [1], RACE [2], KUCA [3] and Yalina [4]. In particular, under the FREYA experiment of the
7º EU framework programme [5], a wide range of subcritical configurations of the VENUS-F reactor of SCK•CEN in Mol (Belgium) have been investigated. The VENUS-F reactor is a zero-power mock-up of a lead-cooled fast ADS and for the FREYA experiment, it has been coupled to the GENEPI-3C (D,T) neutron source, developed by the CNRS. An important feature of GENEPI-3C is its capability to operate both as a pulsed neutron source (PNS) and as a continuous source with beam trips.

In this work, it is presented a summary of the analysis of the reactivity measurement experiments performed during the FREYA Project by CIEMAT. This work is structured as follows. First, the core configurations that have been experimentally measured are presented. In the next two sections, the performance of the two main online reactivity monitoring techniques (area ratio/source jerk and prompt neutron decay constant) is discussed. Finally, some hints are given about the predicted performance of these techniques (specifically, the area ratio/source-jerk technique) in the MYRRHA facility [6], current in the design stage.

The analysis of the data of the FREYA Project is currently being carried out by different project partners, in fact, some other related contributions are presented in this same conference. It must be remarked that the huge amount of data gathered will still require a long time to complete the analysis. Hence, this work intends to present the status of ongoing work rather than final results. Therefore the results presented here must be regarded as preliminary.

FREYA core configurations

The VENUS-F reactor is made up of a 12x12 grid that can be filled with different types of elements: fuel elements, control elements, or elements made of lead or other materials to simulate a reflector or other structures. The entire assembly is placed within a cylindrical lead reflector. Control elements include two $B_4C$ control rods and six safety rods that have the composition of a fuel element in their lower part (at the core level during normal operation) and $B_4C$ in the upper part (at the core level when inserted). Furthermore, a pellet absorber rod (PEAR) used for MSM experiments was also present. The core configurations investigated during the FREYA experiment are sketched in Figures 1 and 2. In the subcritical configurations, the four centremost positions were replaced by the accelerator tube.

These core configurations investigated during the FREYA Project can be classified into four groups:

First, there are three reference cores SC1, SC2 and SC3 that contain only fuel and lead elements. The fuel element type used in these configurations is made up of 30% enriched $U_{enr}$ rodlets alternated with lead blocks. These cores differ in the level of subcriticality ($k_{eff}$ 0.95-0.98), which have been achieved by changing the number and distribution of fuel and lead elements loaded. For the case of the SC1 configuration, additional reactivity levels have been obtained by modifying the position of the two control rods (note that the control rod position 479.3 mm corresponds to the critical position in the equivalent critical core).

The second group of core configurations (Task 1.2) corresponds to deep subcritical cores. The deep subcritical configurations include those obtained by introducing a different number of safety rods (four or six) in the SC1 configuration and a purpose-designed deep subcritical configuration (SC4) obtained by loading a reduced number of fuel elements.

The third group of configurations were investigated to study the impact on the reactivity monitoring techniques of the source position and the presence of different materials in the reflector. These configurations retained the number and distribution of fuel elements of the SC1 configuration, but had elements made up of polyethylene and stainless steel introduced in the reflector. Furthermore, measurements were also performed changing the length of the accelerator tube, obtaining a variation in
the target location. The position could then change from the core midplane to the top of the fissile core (in the figures, these configurations are denoted as “normal thimble” and “short thimble”).

Finally, the SC8 configuration was intended to simulate the subcritical MYRRHA core. For this configuration, a new type of fuel element was designed that introduces $\text{Al}_2\text{O}_3$ with the purpose to simulate the oxygen in the MOX fuel intended for MYRRHA (it must be remarked that MOX fuel was not available for the FREYA experiments). Furthermore, some graphite elements were also introduced to simulate the BeO reflector and two elements made up of polyethylene with HEU mock-up targets in order to simulate the in-pile sections (IPS) of MYRRHA, intended for Mo-99 production.

In Table 1, the calculated kinetic parameters ($k_{\text{eff}}$, $\beta_{\text{eff}}$ and $\Lambda_{\text{eff}}$) for all these configurations are presented. They have been obtained using the MCNP 6.1.1 code and the JEFF-3.1.1 library; for the case of $\beta_{\text{eff}}$ and $\Lambda_{\text{eff}}$ the KOPTS option has been used. Notice how the value of $\beta_{\text{eff}}$ (essentially determined by the fuel) is very similar in all configurations but the value of $\Lambda_{\text{eff}}$ (essentially determined by the moderator) in the SC8 configuration is about three times larger than in the other configurations.

The detectors used to monitor the reactivity have been mostly U-235 fission chambers, placed at the core midplane in different positions in core and the reflector, although a U-238 FC has also been used in some configurations to monitor just the fast contribution of the spectrum.
Figure 1: Core configurations investigated during the FREYA Project

Fuel element (Umet 30% + Pb)
Lead element

Control elements
- Safety rod
- Control rod
- PE1let Absorber Rod
Figure 2: Core configurations investigated during the FREYA Project

Fuel element (Umet 30% + Pb)
Fuel element (Umet 30% + Pb + Al₂O₃)
Lead element
Graphite
Stainless Steel
Polyethylene

Control elements
- Safety rod
- Control rod
- PEllet Absorber Rod
Table 1: Calculated kinetic parameters of VENUS-F cores (MCNP6.1.1 and JEFF-3.1.1)

<table>
<thead>
<tr>
<th>Core</th>
<th>C.R. position</th>
<th>$k_{eff}$</th>
<th>$\Lambda_{eff}$ (µs)</th>
<th>$\beta_{eff}$ (pcm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>SC1</td>
<td>600 mm</td>
<td>0.97168 ± 0.00002</td>
<td>0.513 ± 0.026</td>
<td>705 ± 14</td>
</tr>
<tr>
<td></td>
<td>479.3 mm</td>
<td>0.97022 ± 0.00002</td>
<td>0.493 ± 0.014</td>
<td>728 ± 15</td>
</tr>
<tr>
<td></td>
<td>240 mm</td>
<td>0.96597 ± 0.00002</td>
<td>0.469 ± 0.008</td>
<td>725 ± 14</td>
</tr>
<tr>
<td></td>
<td>0 mm</td>
<td>0.96309 ± 0.00002</td>
<td>0.467 ± 0.007</td>
<td>707 ± 14</td>
</tr>
<tr>
<td>SC2</td>
<td>479.3 mm</td>
<td>0.95301 ± 0.00002</td>
<td>0.503 ± 0.014</td>
<td>729 ± 15</td>
</tr>
<tr>
<td>SC3</td>
<td>479.3 mm</td>
<td>0.97749 ± 0.00002</td>
<td>0.469 ± 0.009</td>
<td>695 ± 14</td>
</tr>
<tr>
<td>Deep SC1, 4 S.R. in</td>
<td>479.3 mm</td>
<td>0.90852 ± 0.00002</td>
<td>0.532 ± 0.020</td>
<td>717 ± 14</td>
</tr>
<tr>
<td>Deep SC1, 6 S.R. in</td>
<td>479.3 mm</td>
<td>0.87374 ± 0.00002</td>
<td>0.547 ± 0.021</td>
<td>719 ± 14</td>
</tr>
<tr>
<td>SC4</td>
<td>479.3 mm</td>
<td>0.89893 ± 0.00002</td>
<td>0.601 ± 0.009</td>
<td>705 ± 14</td>
</tr>
<tr>
<td>SC1+IPS, short thimble</td>
<td>479.3 mm</td>
<td>0.96695 ± 0.00002</td>
<td>0.504 ± 0.018</td>
<td>729 ± 15</td>
</tr>
<tr>
<td>SC1+IPS, normal thimble</td>
<td>479.3 mm</td>
<td>0.96622 ± 0.00002</td>
<td>0.488 ± 0.007</td>
<td>749 ± 15</td>
</tr>
<tr>
<td>SC1+IPS + SS</td>
<td>479.3 mm</td>
<td>0.96614 ± 0.00002</td>
<td>0.482 ± 0.007</td>
<td>724 ± 15</td>
</tr>
<tr>
<td>SC1 + SS</td>
<td>479.3 mm</td>
<td>0.96865 ± 0.00002</td>
<td>0.475 ± 0.008</td>
<td>714 ± 15</td>
</tr>
<tr>
<td>SC8</td>
<td>347.7 mm</td>
<td>0.96266 ± 0.00002</td>
<td>1.340 ± 0.016</td>
<td>733 ± 14</td>
</tr>
</tbody>
</table>

Reactivity measurement with the area-ratio technique

According to the area ratio (Sjöstrand) technique for measuring the reactivity for the results of a PNS experiment, the reactivity of a subcritical assembly is related with the ratio between the areas under the curves corresponding to the prompt ($A_p$) and delayed neutrons ($A_d$) by the equation:

$$\rho = -\beta_{eff} \frac{A_p}{A_d}$$

However, this relationship is obtained in the point kinetics model and does not take into account the high-order spatial and energy modes that are present in actual systems. Several methodologies have been proposed to deal with these effects; a common one is the calculation of detector-specific area-ratio correction factors $C_{det}$ in the shape:

$$\rho = -C_{det} \frac{A_p}{A_d} \Rightarrow C_{det} = -\frac{\rho}{A_p/A_d}$$

These correction factors can be obtained with the help of neutron transport codes and then be applied to correct the experimental results. A further extension of this technique has been proposed in [7], and consists in replacing these expressions by a generalised relationship between the area ratio and the reactivity, i.e. $\rho = f(A_p/A_d)$. These generalised relationships can in principle have any functional shape (not necessarily linear) and can be obtained with the help of neutron transport codes. For this purpose, a number of fictitious reactor configurations are generated by modifying a set of parameters in the reference model of the system. For every one of these “perturbed” configurations, the area ratio and the reactivity are computed. Then, the set of pairs of values $(A_p/A_d, \rho)$ thus obtained are fitted to a certain model to obtain a relationship between the reactivity and the area ratio.

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The advantage is that if the obtained relationship between the reactivity and the area ratio can be considered univocal, then it can be applied to the measured area-ratio values for obtaining the reactivity. Furthermore, an estimate of the systematic error due to the uncertainties in the geometry and composition of the facility and the nuclear data can be obtained and included in the uncertainty given for the reactivity, in addition to the statistical one. Notice that, as it is the case of the correction factor methodology, because of the aforementioned spatial and energy effects, the relationships thus obtained are specific of the type and position of detector used.

The \((A_p/A_d, \rho)\) pairs obtained for U-235 fission detectors at different positions thus obtained with MCNP for the configurations of VENUS-F investigated during the FREYA experiments are presented in Figures 2 and 3. For the SC1 to SC4 configurations, the MCNPX 2.7.0 and the JEFF-3.1.1 library have been used; for the case of SC8, the MCNP 6.1.1 code and the ENDF/B-VII.1 library have been used instead. The parameters that have been chosen for variation are lead density (from 10 to 12 g/cm\(^3\)), the uranium density (from 17.5 to 19.5 g/cm\(^3\)) and the uranium enrichment (from 29% to 31%), starting from the descriptions of the SC1 (either the “standard” SC1 configuration and the deep subcritical variant obtained by inserting four safety rods in the core) and the SC8 configurations.

It can be observed (Figure 2 (a), (b) and (c)) that for the obtained pair of points \((A_p/A_d, \rho)\) for a detector in the core in the SC1 configuration are very close to the point kinetic relationship (represented by a solid line, considering \(\beta_{eff} = 720 \pm 15\) pcm in all cases) but that there is a tendency to depart from this line when detectors in reflector are considered, the difference increasing as the detector position moves outwards. This effect is more relevant as the subcriticality level increases, which is compatible with the fact that the point kinetics is an exact approximation for a critical reactor. With these results, the generalised relationships that have been considered are two linear functions \(\rho = a + b(A_p/A_d)\), with different sets of parameters \(a\) and \(b\) in the subcritical \((A_p/A_d < 9)\) and deep subcritical \((A_p/A_d > 9)\) ranges. These linear functions are represented as dotted lines in the figures.

In Figure 2 (d), (e) and (f), the impact of the VENUS-F concrete building in the shape of the generalised is presented. The \((A_p/A_d, \rho)\) pairs obtained without considering the building in the MCNP geometric model are plotted in a dimmer colour. Notice that the effect is rather negligible for the two innermost reactors, but it becomes relevant in the outermost one.

Figure 3 (a), (b) and (c) shows the pairs of points \((A_p/A_d, \rho)\) corresponding to the SC2, SC3 and SC4 configurations, as well as for the configurations implemented during Task 1.3 for studying the reflector and source effect. It can be observed that they are in line with the values obtained for the SC1 configuration (represented in grey), and therefore, the same generalised relationships have been used in these configurations.

Finally, in Figure 3 (d), (e) and (f), the pairs of points \((A_p/A_d, \rho)\) for the case of the SC8 configuration are shown. They show a similar behaviour that in the SC1 configuration, being close to the point kinetics results for low subcriticalities but departing as the subcriticality increases. However, it is remarkable that when a U-238 detector is considered in one of the outermost experimental positions (Figure 3 (f)), the points remain close to the point kinetics relationship. This indicates that the spatial effects in the outermost detectors are due to thermal neutrons backscattered from the externals.

Figures 4 to 7 present the experimental \(k_{eff}\) results obtained with both the point kinetics relationship \((\rho = -\beta_{eff} A_p/A_d, \text{ with } \beta_{eff} = 720 \pm 15\) pcm) and with the generalised relationships \(\rho = a + b(A_p/A_d)\) obtained in the way described in this section. It can be observed that the values of \(k_{eff}\) obtained with detectors in different positions tend to provide very similar values for \(k_{eff}\) even when the simple point kinetics relationship is applied, except for the outermost ones, and, to a lesser extent, the detector located in the core, the closest to the neutron source. Focusing in the outermost detectors, this
difference can be explained by the presence of the concrete building. Indeed, the generalised relationship shows that when the presence or absence of the concrete building is considered (shown as solid or dotted lines), the range of variation of the results obtained with these outermost detectors is large enough to encompass the results obtained with the innermost ones.

In any case, it is observed that in all configurations the generalised relationship tends to shift slightly downwards the $k_{\text{eff}}$ results, when compared with the point kinetics. In Figure 4, where the results obtained with these techniques are compared with MSM results [8], it can be observed that the generalised results closely match the MSM results that the point kinetics ones. In any case, it is observed that, with independence of the presence of biases in the experimentally obtained $k_{\text{eff}}$ results, the technique is clearly capable to detect the differences of reactivity caused by the change of configuration or the movement of the control rods.

For the case of the SC8 configuration, it is also observed that the detector in position (4,4) also provides $k_{\text{eff}}$ results significantly larger than the other detectors. The reason is currently unknown.

As a final remark, it is worth noting that source-jerk results from beam-trip experiments, which should be equivalent to the area-ratio results, show a large dispersion of the reactivity results obtained with different detectors, at least for some configurations, even when attempting to correct the results with the generalised methodology. The reasons for this behavior are still being investigated.
Figure 2: Generalised area-ratio/reactivity relationships obtained with MCNP for the SC1 configuration

(a) Detector in (1,-3) (core)  
(b) Detector in (6,-2) (internal reflector)  
(c) Detector in A1 (external reflector)

(d) Detector in (1,-3) (core)  
(e) Detector in (6,-2) (internal reflector)  
(f) Detector in A1 (external reflector)
Figure 3: Generalised area-ratio/reactivity relationships obtained with MCNP for the SC1, SC2, SC3, SC4 and SC8 configurations (S. T.: short thimble; N. T.: normal thimble)

(a) Detector in (1, -3) (core)
(b) Detector in (6, -2) (internal reflector)
(c) Detector in A1 (external reflector)
(d) Detector in (4, 4) (internal reflector)
(e) Detector in A1 (external reflector)
(f) Detector in C2 (external reflector, U-238 F. C.)
Figure 4: area-ratio $k_{\text{eff}}$ experimental results for the SC1 configuration and different control rod heights

\[ \rho = \beta_{\text{eff}}(A_p/A_d) \]

\[ \rho = \alpha^2(A_p/A_d) \]

Figure 5: area-ratio $k_{\text{eff}}$ experimental results for the SC1 configuration with 4 or 6 safety rods inserted, SC2 and SC3 configurations

\[ \rho = \beta_{\text{eff}}(A_p/A_d) \]

\[ \rho = \alpha^2(A_p/A_d) \]
Figure 6: Area-ratio $k_{\text{eff}}$ experimental results for the variants of the SC1 configuration intended to study reflector and source effects

\[ \rho = |\lambda_{\text{eff}}(A_j/A_o)| \]

Figure 7: Area-ratio $k_{\text{eff}}$ experimental results for the SC8 configuration and different control rod heights. Detector in C2 was a U238 FC

\[ \rho = |\lambda_{\text{eff}}(A_j/A_o)| \]
Reactivity measurement with the prompt decay constant technique

The second major technique to obtain the reactivity from the results of a PNS (or beam trip) experiment is the prompt decay constant technique. With this technique, the reactivity of the subcritical assembly is obtained from the exponential decay constant  of the prompt neutrons after the pulse from the external source, using the equation:

\[ \rho = \alpha \Lambda_{eff} + \beta_{eff} \]

However, as it is the case with the area-ratio technique, the prompt neutron decay constant technique is also derived from the point kinetics model, and therefore, it is also affected by spatial and energy effect in actual systems. In particular, these effects may result in the presence of high-order modes that mask the fundamental one.

In principle, the generalised methodology described for the area ratio technique can also be applied to attempt to correct these effects. In this case, what is searched for is a relationship between the prompt neutron decay slope (determined from a fit of the PNS to an exponential decay in some time range after the pulse) and the reactivity that replaces the above point kinetics relationship. However, in a previous work [9] it has been shown the difficulty to find such relationships for the FREYA cores; in particular, it has been found that as the subcriticality level increases, the obtained values of \( \alpha \) appear to become independent of the reactivity.

A possibility to overcome this problem that is now being investigated is the application of the Fourier transform to the results of a PNS experiment. A sketch of the procedure followed is shown in Figure 8. The motivation to attempt this procedure is that the effect of the high-order modes in frequency space may be limited to certain frequency ranges while leaving other frequency ranges unaffected where the fundamental mode value of \( \alpha \) can be obtained. Notice as well that the value of \( \alpha \) may be obtained from both the real or the imaginary part of the Fourier transforms. Furthermore, if the plots of \( 1/Re[F(\omega)] \) vs. \( \omega^2 \) and \( \omega/Im[F(\omega)] \) vs. \( \omega^2 \) are considered, this value of \( \alpha \) can be obtained from fits to simple linear models.

Figures 9 and 10 show the results of the application of this technique for two detectors of the SC1 configuration, one in the core and the other in the reflector, for the different reactivity levels obtained by varying the control and safety rods position. The theoretical results, obtained from the values of \( k_{eff}, \beta_{eff} \) and \( \Lambda_{eff} \) in Table 1, are also plotted as solid lines, for comparison.

It can be observed that for the case of the detector in the reflector, the experimental curves are close to the theoretical ones although there is some shift. In any case, it is important to remark that the difference in the shape of the curves corresponding to different reactivity levels becomes very apparent, especially in the case of the subcritical configuration. This contrasts with the results obtained when it was attempted to determine the value of \( \alpha \) by fitting the PNS experiments to exponentials, as above mentioned. For very low values of \( \nu^2 \), however, the experimental results tend to underestimate the theoretical ones. This may be an effect of the presence of the high order, low-decaying modes. On the other hand, as the value of \( \nu^2 \) increases, the experimental values tend to overestimate the theoretical ones. A complete explanation for this effect has not been found yet.
Figure 8: Fourier transform applied to a PNS experiment. H(t) represents the Heaviside step function.

For the detector in the core, however, the difference between the theoretical and the experimental results is much larger. The reasons for this behaviour remain to be explained. In any case, again the difference in the shape of the curves in the frequency space for different control and safety rods position is very apparent, even for the deep subcritical configurations. Hence, the generalised methodology may still be applied to obtain a relationship between some parameter of the shape of the experimental curves $1/\text{Re}[F(\nu)]$ vs. $\nu^2$ and $\nu/\text{Im}[F(\nu)]$ vs. $\nu^2$ and the reactivity. This remains to be investigated at this moment, however.
Figure 9: Fourier Transform of a PNS experiment applied to a detector in the reflector

Detector in (-6,6) (reflector)

Figure 10: Fourier Transform of a PNS experiment applied to a detector in the core

Detector in (1,-3) (core)
Application to MYRRHA

Quite obviously, the final objective of zero-power experiments such as FREYA is to develop reactivity monitoring techniques that are applicable to industrial scale systems. Here, the performance of the area-ratio technique MYRRHA facility is investigated with simulations performed with the MCNP code. A sketch of the MYRRHA facility showing the different types of elements present is presented in Figure 11. The MCNP model of the facility has been taken from [10]. A number of detector positions ranging from the spallation target in the centre to the externals have been investigated; they are also shown in Figure 11. In all cases, U-235 fission chambers have been considered.

The area ratio results obtained with these detectors is presented in Table 2, alongside with the correction factors calculated as $C_{det} = -\rho/(A_p/A_d)$. Notice that the values of these correction factors are very close to the value of $\beta_{eff}$, the difference being the largest for the outermost one and the one in the spallation target area. This allows expecting that the area ratio will be a strong estimator of the reactivity for the MYRRHA facility, although spatial effects will be noticed for detectors placed very near of the spallation target or in very external positions.

Figure 11: Schematics of the MYRRHA core and the detector positions used for area-ratio calculations

Table 2: Calculated area ratio, correction factors and kinetic parameters of the MYRRHA facility

<table>
<thead>
<tr>
<th>Detector no.</th>
<th>Area ratio</th>
<th>$C_{det}$ (pcm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>#0</td>
<td>17.6 ± 0.5</td>
<td>240 ± 7</td>
</tr>
<tr>
<td>#1</td>
<td>13.7 ± 0.3</td>
<td>309 ± 8</td>
</tr>
<tr>
<td>#2</td>
<td>12.9 ± 0.3</td>
<td>329 ± 9</td>
</tr>
<tr>
<td>#3</td>
<td>12.2 ± 0.3</td>
<td>348 ± 9</td>
</tr>
<tr>
<td>#4</td>
<td>13.0 ± 0.8</td>
<td>320 ± 20</td>
</tr>
<tr>
<td>#5</td>
<td>12.1 ± 0.7</td>
<td>350 ± 20</td>
</tr>
<tr>
<td>#6</td>
<td>12.4 ± 1.2</td>
<td>342 ± 33</td>
</tr>
<tr>
<td>#7</td>
<td>11.3 ± 1.9</td>
<td>372 ± 61</td>
</tr>
<tr>
<td>#8</td>
<td>12.4 ± 1.3</td>
<td>342 ± 36</td>
</tr>
<tr>
<td>#9</td>
<td>9.7 ± 1.1</td>
<td>438 ± 48</td>
</tr>
</tbody>
</table>
Conclusions
During the FREYA Project, PNS and beam-trip experiments have been performed in a wide range of subcritical configurations of the VENUS-F reactor. The area-ratio reactivity estimator has been shown to be a robust estimator of the reactivity when applied to detectors placed over a wide range of positions within the assembly, although the results with the detectors placed in the outermost part of the reflector show a bias with respect to the innermost ones. On the other hand, the application of the equivalent source-jerk technique to beam-trip experiments is still under investigation. The so-called generalised version of the area ratio technique has been applied to investigate the spatial effects and to obtain relationships between the reactivity and the area-ratios. According to simulation results, the area ratio technique is also expected to perform well in MYRRHA. Finally, some results about the application of the Fourier transform to obtain the prompt neutron decay slopes are presented.

Acknowledgements
The work presented here has been partially supported by the FREYA Project of the 7th EU Framework Programme and the ENRESA-CIEMAT agreement on “Transmutación Aplicada a los Resíduos de Alta Actividad”. The authors also wish to acknowledge the support and dedication of VENUS and GENEPI teams during the realisation of the experiments.

References

Spatial effects during beam interruption experiments for ADS subcriticality monitoring using a Monte Carlo alpha-mode expansion of the neutron flux


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Abstract
In the framework of the current international efforts to study the feasibility of accelerator-driven systems (ADS), experiments are being carried out at the GUINEVERE facility located at the SCK•CEN centre in Mol, Belgium, in particular for the evaluation of subcriticality monitoring techniques. The facility is composed of the subcritical lead fast reactor VENUS-F coupled to the GENEPI-3C accelerator providing the external neutron source through (d,t) fusion reactions. One of the reactivity determination methods investigated is the analysis of beam interruption experiments, during which the continuous beam ensuring a constant reactor power is periodically stopped, leading to a decrease of the neutron population in the reactor. The evolution of the neutron flux can be studied thanks to several detectors located in the reactor, and the reactivity extracted using an analysis based on the point kinetics (PK) equations. However, PK assumptions are no longer valid for the subcriticality range investigated ($k_{eff}$ around 0.96). As a result, the reactivity values extracted from experiments depend on the detector location. An approach based on a modal expansion has been implemented in order to investigate the space-energy effects responsible for the faulting of the PK equations. This approach consists in using the alpha-basis corresponding to the alpha-eigenvalue problem, in order to recompose and study the time-dependent neutron flux in the reactor. This basis is obtained from the diagonalisation of a Markov Transition Rate Matrix using a space-energy-direction discretisation of the phase space. Here, we will present the implementation of this method in the Serpent 2 Monte Carlo transport code, and the results obtained for a simplified model of the VENUS-F reactor. The computation of space-dependent correction factors for the raw results of beam interruption experiments using this modified version of Serpent 2 is also briefly discussed.

Introduction
The GUINEVERE (Generator of Uninterrupted Intense NEutrons at the lead VEnuS REactor) Project [1] was launched in 2006, within the 6th Euratom Framework Programme IP-EUROTRANS [2], in order to study the
feasibility of transmutation in accelerator-driven subcritical systems (ADS). This facility hosted at the SCK-CEN site in Mol (Belgium) was then used in the follow-up FREYA Project (7th European FP) [3] which was partially dedicated to the investigation of techniques of online subcriticality monitoring.

The facility couples the fast lead-moderated reactor VENUS-F, whose reactivity can range from deep subcritical to critical thanks to its high modularity, with an external neutron source provided by the deuteron accelerator GENEPI-3C via $T(d,n)^4\text{He}$ fusion reactions occurring at the reactor core centre. The reactivity measurement during beam interruption experiments was one of the reactivity monitoring techniques studied within the FREYA Project. Based on the analysis of detector count rate decreases when the beam is off, this technique can suffer from rather strong spatial and energy effects, precluding the correct reactivity from being estimated, depending on where the detectors are positioned in the reactor. Although these effects can be rather well taken into account in order to get reasonable reactivity values for almost all the detectors using direct Monte Carlo simulations of the beam interruption experiments [4], it is important to characterise and understand them as well as possible. This is the reason why modal expansion capabilities were added to the Monte Carlo neutron transport code Serpent 2 using an approach based on a Markov Transition Rate Matrix [5].

After presenting the GUINEVERE facility, the method used here to extract reactivity using the time evolution of neutron population after a beam interruption is described. To illustrate the spatial effects at work in the subcritical VENUS-F reactor, histograms of detector count rate evolutions over time are given for detectors representative of two extreme cases: one fission chamber in the core, the other one in the outer part of the reactor reflector. Then the principles of the Markov Transition Rate Matrix method are briefly introduced and results of the Serpent simulations on a simplified model of the VENUS-F configuration SC3 are discussed.

The GUINEVERE facility
The VENUS-F fast reactor is contained in a cylindrical vessel of approximately 80 cm in radius and 140 cm in height. A 12x12 matrix surrounded by a parallelepiped stainless steel casing can receive up to 144 elements of $\approx$8x8 cm$^2$ in section which can be fuel assemblies, lead assemblies or specific elements for accommodating detectors or absorbent rods. The remaining room in the vessel is filled with semi-circular lead plates, which act as a radial neutron reflector. In addition the core is equipped with top and bottom 40 cm thick lead reflectors. Each fuel assembly (FA) contains a 5x5 pattern, filled with 9 fuel rodlets and 16 lead bars, surrounded by lead plates. The fuel is 30 wt. % enriched metallic uranium provided by CEA. Among the set of FAs, six are actually safety rods (SR) made of boron carbide and fuel followers with the absorbent part retracted from the core in normal operation. Two control rods (CR) made of natural boron carbide can be positioned at various locations in the 12x12 grid. They can be moved vertically from 0 mm (fully inserted in the core) to 600 mm (fully retracted). Another absorbent rod, whose reactivity worth is very small, called pellet absorber rod (PEAR), is available for performing rod drop experiments.

Various configurations of the reactor in terms of reactivity can be studied thanks to the modular shape of the core. The so-called SC3 configuration is represented in Figure 1. 97 FAs (in blue for the regular ones, in light blue for the SRs with fuel followers) were arranged in a way to create a pseudo-cylindrical core. The two CRs (in red) were located at the core periphery and retracted at 479.3 mm in height and the PEAR rod (in green) was fully inserted. The reactor was equipped with 10 fission chambers (FC) working in pulse mode. Four different types of FC were used, either Photonis CFUF34, CFUL01 and CFUM21 [6], or GE reuter-stokes (RS). They contain 1 mg, 1 g, 10 mg and 100 mg of deposit, respectively. The main deposit nuclide is $^{235}\text{U}$. 

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As can be seen in Figure 1, four FAs are missing in the centre of the reactor core to allow inserting the GENEPI-3C deuteron accelerator [7]. Built by a collaboration of CNRS-IN2P3 laboratories, GENEPI-3C accelerates deuterons up to an energy of 220 keV towards a tritiated titanium target located at the reactor core centre. The fusion reactions at core midplane generate a quasi-isotropic field of ~14-MeV neutrons which are subsequently multiplied by the surrounding subcritical reactor. The GENEPI-3C can operate in pulsed mode, in continuous mode and also in continuous mode with short beam interruptions. Each mode was used in the FREYA Project but only the latter will be discussed in this paper.

**Beam interruptions experiments**

**Principle of the method**

The measurement of reactivity during short interruptions of a continuous beam was one of the reactivity monitoring techniques studied at the GUINEVERE facility within the FREYA Project. During these beam interruptions, as the reactor is subcritical, the neutron population decreases over time. Using the point kinetics theory, one may link the behaviour of the evolution of the neutron population during the interruption to the reactivity of the reactor.

More precisely, the inverse point kinetics equation states that, provided that the source intensity has been constant long enough for the reactor to be in a steady-state regime [8]:

$$\rho_\delta(t) = 1 + \frac{1}{n(t)} \left[ \frac{\Lambda_{\text{eff}}}{\beta_{\text{eff}}} \frac{dn}{dt} - \frac{n_0}{\beta_{\text{eff}}} \sum_j \beta_{\text{eff}}^{\text{jeff}} e^{-\lambda_j t} \sum_j \lambda_j \beta_{\text{eff}}^{\text{jeff}} e^{-\lambda_j t} \left[ \int_0^t n(t') \exp(-\lambda_j t') dt' \right] \right]$$  \hspace{1cm} (1)

where is the reactivity of the system in dollars, $n(t)$ the neutron population, $n_0$ the neutron population before the interruption, $\Lambda_{\text{eff}}$ the neutron generation time, $J$ the number of precursor groups, $\beta_{\text{eff}}^{\text{jeff}}$ the effective delayed neutron fraction of precursor group $j$ and $\lambda_j$ the corresponding precursor decay constant.

The neutron population as a function of time can be obtained in different locations in the reactor using the FCs described above, assuming that the FC count rates are proportional to the neutron flux.
Regarding the kinetic parameters needed for this equation, they have been obtained using the deterministic neutron transport code ERANOS [9].

**Results of the experiments**

During the experiments, 2-ms beam interruptions were performed every 25 ms (40-Hz frequency). Due to the low flux in the reactor, several beam trips (a few hundred thousand) were performed to increase detector statistics. For each FC, the data consist in one histogram obtained by summing all the detector time responses as a function of the time elapsed after each interruption.

Typical shapes observed for two different detector locations are shown in Figure 2, as well as the shape predicted by point kinetics theory. The CFUF34 FC is located in the core, while the CFUL01-658 FC is in the outer reflector, outside of the SS casing. For both FCs, a prompt decay of the neutron populations is observed just after the interruption. Then, the neutron population seems to tend to its delayed neutron level more or less rapidly depending on the detector location: the farther from the centre of the reactor the detector is, the more different from point kinetics (PK) the neutron population evolution is. This indicates that some spatial effects, not taken into account by PK theory, are present in the reactor.

![Figure 2: Comparison of the experimental count rates obtained in configuration SC3 for the detector CFUF34 in the core (blue) and the detector CFUL-658 in the outer reflector (red). The curve from point kinetics is also shown (green).](image)

Neutron transport simulations carried with MCNP [10] have shown that the distorted shapes of the experimental count rates for the detectors in the outer reflector are due to the presence of the concrete walls around the reactor. Indeed, neutrons leaking through the external vessel may scatter on the walls and get back to the reactor. As the concrete walls contain light elements such as hydrogen or oxygen, neutrons are likely to lose a great part of their energy by scattering. The FC deposit being made of $^{235}$U, these neutrons induce significant and delayed counts in the FCs. This effect is much more important in the outer reflector than in the core, because the probability for these low-energy neutrons to reach a detector before being absorbed decreases as they travel in the reactor.

Consequently the reactivity extraction using Equation (1), whose result depends on whether or not the experimental count rates reach the delayed neutron level, is strongly affected by the presence of the walls. A successful correction for this effect has been applied using spatial dependent correction factors obtained with MCNP simulations [4]. However the main drawback of such a global approach is that all the distortions from PK originating from different physical effects are treated altogether, precluding any simple way of disentangling the various contributions to the correction. This is the reason
why a method involving a modal expansion of the flux was implemented in order to qualitatively characterise the impact of the walls on the dynamical behaviour of the reactor during a beam interruption experiment.

**Modal expansion using an \( \alpha \)-mode basis**

Let us start with the Boltzmann equation governing the behaviour of the neutron population in the reactor which reads:

\[
\frac{\partial \varphi}{\partial t} = F_p \varphi(r, E, \Omega, t) - M \varphi(r, E, \Omega, t) + \sum_j \frac{\chi_j}{4\pi} \lambda_j C_j(r, t)
\]

\[
\frac{\partial C_j}{\partial t} = F_{ij} \varphi(r, E, \Omega, t) - \lambda_j C_j(r, t)
\]

Where \( \varphi \) is the neutron flux, \( C_j \) the precursor concentration of group \( J \), \( \nu \) the neutron speed, \( \chi_j \) the delayed neutron emission spectrum for the precursor group \( j \), \( M \) the migration operator combining leakage, scattering and collision, \( F_p \) the prompt fission operator and \( F_{ij} \) the delayed neutron operator for the precursor group \( j \).

Assuming that the time dependence of the neutron flux and the precursor concentrations can be separated from their energy and space distributions, and describing the time evolution by an exponential with a decay constant \( \alpha \), we get:

\[
\varphi(r, E, \Omega, t) = \varphi(r, E, \Omega) \exp(\alpha t)
\]

\[
C_j(r, t) = C_j(r) \exp(\alpha t)
\]

Introducing Equation (3) into Equation (2) finally leads to the alpha-mode form of the Boltzmann equation:

\[
\frac{\alpha}{\nu} \varphi(r, E, \Omega) = F_p \varphi(r, E, \Omega) - M \varphi(r, E, \Omega) + \sum_j \frac{\chi_j}{4\pi} \lambda_j C_j(r)
\]

\[
\alpha C_j(r) = F_{ij} \varphi(r, E, \Omega) - \lambda_j C_j(r)
\]

The \( \frac{\alpha}{\nu} \) term corresponds to an additional absorption rate (resp. production rate) for a supercritical (resp. subcritical) reactor [11]. The solutions to this equation are eigenvalues associated with their eigenfunctions \( \varphi_{bi} \), known as the – modes of the Boltzmann equation. The first mode \( i=0 \) is called the fundamental mode, and is related to the reactivity of the system by:

\[
\alpha_0 = \frac{\rho - \beta_{\text{eff}}}{\Lambda_{\text{eff}}}
\]

The actual temporal, spatial, angular and energetic evolution of the flux may then be obtained using the basis formed by these modes, that is:
where the values of the amplitude coefficients $T_i(t)$ depend on the source definition.

Using the bi-orthogonality property of the direct and adjoint fluxes and defining $\gamma_i$ as:

$$\gamma_{ni} = \left\langle \varphi^\dagger_{\alpha n}, \psi_{\alpha i}^{-1} \psi_{\alpha i} \right\rangle = \gamma_{ni} \delta_{ni} = \gamma_{ni}$$

the coefficients $T_i(t)$ of Equation (6) can be computed for any neutron source. The $\langle \rangle$ notation denotes the integration over the phase space, and the symbol $\dagger$ denotes the adjoint related quantities. In order to simulate a beam interruption experiment, we need to define a continuous source $S(r, E, \Omega, t)$ being interrupted at a given time $t_{int}$:

$$0 < t \leq t_{int} : S(r, E, \Omega, t) = S_0(r, E, \Omega)$$
$$t > t_{int} : S(r, E, \Omega, t) = 0$$

Before the beam interruption, it can be shown [11] that the coefficients $T_i(t)$ read:

$$0 < t \leq t_{int} : T_i(t) = \frac{\langle \varphi_{\alpha i}, S_0(r, E, \Omega) \rangle}{\alpha_i \gamma_{ni}} \left[ \exp(\alpha_i t) - 1 \right]$$

Then, once the source is interrupted at $t = t_{int}$, the temporal evolution of each mode is simply described by an exponential decrease driven by the eigenvalue $\alpha_i$ corresponding to this mode. That is:

$$t > t_{int} : T_i(t) = T_i(t_{int}) \exp(\alpha_i t)$$

Betzler and collaborators developed a method to compute the $-\text{mode}$ basis using Monte Carlo calculations [5][11]. Unlike a lot of other methods, either deterministic or stochastic, developed for calculating the $-\text{modes}$ of a reactor, it is not limited to simplified systems or to the dominant modes of the problem and higher modes of the $-\text{mode}$ equation for any kind of problem can be computed. Basically, a Markov Transition Rate Matrix (MTRM), using a space-energy-direction discretisation of the phase space, is computed during the Monte Carlo random walk. This matrix $Q$ corresponds to the adjoint problem of Equation (4):

$$Q = \begin{bmatrix} \nu(-M + F_p)^\dagger & \nu F_d^\dagger \\ (\chi \lambda)^\dagger & -\lambda \end{bmatrix}$$

and the direct problem is obtained by transposing the matrix $Q$:

$$\begin{bmatrix} \nu(-M + F_p) & \chi \lambda \\ \nu F_d & -\lambda \end{bmatrix} = Q^T$$
The rates of the different reactions (fission, capture, scattering) and of the leakage in each
discretised element of the phase space are evaluated during the random walk. Then, the matrix Q and its
transpose are diagonalised in order to obtain the eigenvalues and eigenfunctions of both adjoint and
direct fluxes and precursor concentrations.

Rigorously, the terms appearing in the Q-matrix are weighted by the adjoint neutron flux. However
the different rates calculated during the random walk are actually not weighted by the adjoint flux, not
readily obtainable in Monte Carlo codes, but by the neutron population instead. Since the choice of the
weighting function tends to become unimportant when the phase space gets more and more finely
discretised, this approximation is expected to impact more or less the results depending on the degree of
phase-space discretisation used.

We have implemented this method in the Serpent 2 Monte Carlo neutron transport code [12], which
has been modified in order to provide the different terms of the matrix Q. After validation on simple
1D problems, we now propose to test the ability of this approach to reproduce and describe neutron
dynamics in a simplified model of the VENUS-F reactor.

**Modelling of the VENUS-F reactor**

**Modelling without the walls**

To investigate the impact of the concrete walls around the reactor, a simplified three-dimensional
geometry of the VENUS-F reactor in configuration SC3 was modelled first without including the concrete
walls (see Figure 3 without considering the walls in red). FAs were homogenised and the core (in blue in
Figure 3) was implemented as a cylinder of 100 cm in diameter and surrounded by a lead reflector (in
yellow) of 160 cm in diameter, split in two parts (inner and outer reflector) by a 3-cm-thick stainless steel
casing (in orange). The multiplication factor was evaluated by Serpent 2 to be 0.97363(7), to be compared
with the actual keff value of SC3 of 0.97090(100) measured by the MSM method.

**Figure 3: Geometry of the modelling with walls created in Serpent 2 (x-y plane on the left,
x-z plane on the right).**

This simplified modelling did not include control and safety rods, nor any axial heterogeneous
structures, in order to use a two-dimensional discretisation (along x and y) in the calculations. The phase-
space discretisation parameters are given in Table 1. As mentioned above, the discretisation mesh must
be fine enough for the forward-weighted approximation of the matrix terms to be reasonable, but it also
impacts the size of the matrix which is limited by the time needed for its diagonalisation. Unfortunately,
with the parameters shown in Table 1, a huge computer-time is already needed.
After diagonalising the Q-matrix, several eigenvalues below 1 s\(^{-1}\) were obtained, which correspond to some alpha modes driven by delayed neutrons. The first prompt eigenvalue was found to be \(\alpha_{p0} = -153704\) s\(^{-1}\). This value can be compared to the prompt alpha-value of \(\alpha_{p0,\text{SERPENT}} = -166217\) s\(^{-1}\) calculated from Equation (5) using the adjoint weighted parameters \(\beta_{\text{eff}}\) and \(\Lambda_{\text{eff}}\) calculated by Serpent 2. The two values are close, and would get even closer if the discretisation were improved: this shows the limitation related to the approximation made to obtain the terms of the matrix Q.

Table 1: Discretisation of the phase space used for the modelling without concrete walls

<table>
<thead>
<tr>
<th>Position</th>
<th>N=900</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>30x30 Cartesian grid in the X-Y plane, from – 100 to 100 on both axis</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Direction</th>
<th>M=4</th>
</tr>
</thead>
<tbody>
<tr>
<td>Energy</td>
<td>G=3</td>
</tr>
<tr>
<td></td>
<td>[0, 100 keV], [100 keV, 1 MeV], [1 MeV, 20 MeV]</td>
</tr>
</tbody>
</table>

The shapes of the first three prompt alpha-modes for the spatial flux are shown in Figure 4, for the lower (bottom) and higher (top) energy groups. One may notice that the shapes of the lower energy groups are similar to those of the higher energy one but spread outwards because of the stronger absorption of low-energy neutrons in the core.

Figure 4: Shapes of the first three prompt alpha modes for the modelling without concrete walls

The external neutron source \(S_0(r, E, \Omega, t)\), was modelled as a spatial Gaussian distribution with a maximum at the reactor centre (\(\mu=0\) cm, \(\sigma=5\) cm), emitting isotopically neutrons in the higher energy group. In order to ensure the stabilisation of the concentration of the neutron precursors before the interruption of the continuous beam, the source was on from \(t = 0\) to \(t = t_{\text{int}} = 1000\) s.
Figure 5 shows the neutron flux constructed from the alpha eigenvectors as given by Equation (6) for $t = 0$, when the source is switched on in a reactor free of neutrons, and $t = t_{int}$, just before the beam interruption. The coefficients $T_i(t_{int})$ of the decomposition of the flux at $t = t_{int}$ on the alpha-mode basis are given in Figure 6. The modes are ordered by decreasing eigenvalue: the higher the mode number, the lower the absolute value of the corresponding eigenvalue. One can see the complexity of the decomposition and the need of higher order modes to adequately reproduce the neutron flux.

![Decomposed flux at t=0](image1) ![Decomposed flux at t=t_{int}](image2)

**Figure 5: Neutron flux recomposed using the alpha-mode basis at t=0 (left) and just before the interruption (right).**

Modelling with the concrete walls

The geometry described above was then modified by introducing the concrete walls, as show in Figure 3. In the actual reactor, the walls are several metres away from the vessel. Implementing such a distance in our simplified model would require a huge Cartesian mesh for getting accurate results, an
the resulting matrix would not be diagonalisable in a reasonable amount of time. Consequently, 10-cm thick concrete walls (in red in Figure 3) separated from the reactor vessel by 10 cm only were modelled instead. So what was done is a qualitative rather than quantitative study of the impact of the concrete walls on the reactor dynamics during beam interruptions. Serpent 2 calculations gave a $k_{\text{eff}}$ value of 0.97557(7) for this configuration, the small difference with the previous modelling being due to the walls. Using the adjoint weighted parameters $\beta_{\text{eff}}$ and $\Lambda_{\text{eff}}$ calculated by Serpent 2, one gets $\alpha_{0, \text{Serpent}} = -130420 \text{ s}^{-1}$.

We now turn to the calculation of alpha-modes using the MTRM. Since neutron thermalisation in the concrete walls leads to a softer spectrum in the reactor, a finer energy discretisation had to be considered. The number of energy groups was thus increased to 6, and the spatial mesh was decreased to keep a reasonable matrix size (see Table 2).

Table 2: Discretisation of the phase space used for the modelling with concrete walls

<table>
<thead>
<tr>
<th>Position</th>
<th>N=529</th>
</tr>
</thead>
<tbody>
<tr>
<td>23x23 Cartesian grid in the X-Y plane, from –100 to 100 on both axis</td>
<td></td>
</tr>
</tbody>
</table>

| Direction | M=4 |
| Energy    | G=6 |
| [0, 1 eV], [1 eV, 100 eV], [100 eV, 10 keV], [10 keV, 100 keV], [100 keV, 1 MeV], [1 MeV, 20 MeV] |

The prompt fundamental eigenvalue given by the diagonalisation of the Q-matrix was found to be $\alpha_{0} = -5572 \text{ s}^{-1}$, a value about 25 times smaller than the one given by Equation (5) or the one obtained using the MTRM but without the walls. As can be seen in Figure 7, the shapes of the first three prompt modes are largely changed compared to those of the wall-free case. This difference is related to the presence of the walls which soften the neutron energy spectrum in the reactor: lower neutron speeds result in lower transition rates in the matrix and consequently in lower alpha eigenvalues (in absolute). Of course the strength of the alpha-mode shapes is shifted towards the walls from where the lower-energy neutrons originate.

Figure 7: Shapes of the first prompt modes obtained with the modelling including the concrete walls
The local flux at a given position can be calculated using the decomposition shown in Equation (6). In order to obtain count rates that might be compared to those given by the detectors in the VENUS-F reactor, this flux must be convoluted by the $^{235}$U fission cross section in the different energy groups used for the discretisation. These cross-sections were obtained with Serpent 2 by evaluating the quantities:

$$\Sigma_{\text{f},\text{g}} = \int_{\Omega} \int_{E} \phi(r, E) \sigma(E) \, dr \, dE$$

(13)

where the integration is performed on the volume of the discretised spatial element $n$ and on the energy group $g$.

Two positions were considered: one in the core ($x = -26$ cm, $y = 0$ cm) and one in the outer reflector ($x = -69.5$ cm, $y = 0$ cm). The evolution of the detector count rates during a beam interruption, obtained by convoluting the neutron flux given by Equation (6) at the positions of interest with the multi-group $^{235}$U fission cross sections given by Formula (13), are shown in Figure 8.

**Figure 8: Evolution of detector count rates obtained using the $-\$ mode basis, in the core (blue) and in the outer reflector (red)**

Despite the simplicity of the modelling of the geometry, the qualitative agreement with the experimental results shown in Figure 2 is striking: the count rate for the detector in the core rapidly reaches the delayed neutron level, while a slower decrease is observed for the detector located in the outer reflector. However, even for the outer reflector case, the delayed neutron level is reached within the 2-ms interruption in, which is not observed experimentally. This is not surprising since the walls are much closer to the vessel in the simplified model than in the actual reactor.

**Contribution of the modes**

The distribution of the coefficients $T_i(t_{\text{int}})$ of the modes entering the decomposition of the flux shows a complexity level similar to the one shown in Figure 6 for the model without concrete walls. The temporal evolution of the flux could be obtained for the local count rates in the two positions investigated. However we propose here an alternative representation based on the contributions of the modes. The contribution of a mode can be seen as the count rate variation that would be observed if this mode were not considered in the modal expansion. A high contribution shows that a mode has a major impact on the count rate. For each position investigated, we chose to represent in Figure 9 the evolution of the contribution of the three most significant modes during the first millisecond after the interruption.
As can be seen, the temporal evolutions of the mode contributions strongly depend on the position investigated. In the core, the high-order mode number 818 is dominant for the first 10 µs, and quickly decreases as the contribution of the delayed fundamental mode increases. The prompt fundamental mode $\alpha_{p0}$ never impacts the count rates above 2%. On the other hand, the count rate in the outer reflector is dominated by the prompt fundamental mode for about 200 µs. Then, the delayed fundamental mode is the most significant one. The mode number 818 starts at 4% then only decreases.

Interestingly, the shape of the prompt mode number 818 is almost identical to the fundamental mode without the walls (see Figure 10), and its associated eigenvalue, $\alpha_{818} = -136146 \text{ s}^{-1}$ is quite close to the prompt fundamental eigenvalue calculated by Serpent 2. It appears that this mode is actually representative of the dynamical behaviour of the reactor on its own, independently of the walls.

As a result, another interpretation of the contributions observed in Figure 9 is that the count rate decrease in the core after the interruption is mainly driven by the reactor itself, and the concrete walls have very little impact on it. On the other hand, the count rate in the outer reflector is mainly ruled by the neutrons coming back to the reactor after losing their energy by scattering on the walls.

This approach gives insight on the neutron kinetics in the reactor since the effects of the walls on the evolution of detector count rates can be associated to some modes and distinguished from the behaviour of the reactor itself.

**Figure 9: Contribution of the three most significant modes (prompt fundamental mode $\alpha_{p0}$ in magenta, delayed fundamental mode in black and prompt mode number 818 $\alpha_{p818}$ in orange) to the count rates in the core and in the outer reflector**
Figure 10: Prompt mode number 818 with the concrete walls and prompt fundamental mode without the walls

Conclusion and prospects

Beam interruption experiments carried out at the GUINEVERE facility showed that the concrete walls surrounding the reactor affect the extraction of reactivity using the point kinetics theory. For fission chambers in the outer reflector, count rates are indeed impacted by low-energy neutrons coming back from the walls. A method using a Markov Transition Rate Matrix to expand neutron fluxes on the mode basis was successfully implemented in the Monte Carlo neutron transport code Serpent 2 to investigate the effect of these walls. Beam interruption experiments were simulated using this basis, and detector count rate evolutions similar to experimental ones were calculated despite the simplicity of the model geometry used in Serpent 2.

A promising result was obtained as the contribution of the walls to the count rates could be associated to a particular mode, and distinguished from the behaviour of the reactor itself. Not only this method allows one to get more physics insight about the space-energy effects at work than direct Monte Carlo simulations of beam interruption experiments, but should also provide a way to disentangle the various contributions to the space and energy dependent correction factors applied to raw reactivity estimates from detector count rate curves to extract the actual reactivity of a reactor configuration and consequently to assess their reliability and robustness for subsequent transposition studies for other accelerator-driven systems.

Presently, the discretisation used for the method is being limited by the time needed to compute the eigenvalues and eigenfunctions of the matrix. Further work will include the parallelisation of the diagonalisation process to speed-up the calculation, allowing to improve the discretisation mesh and to study geometries corresponding to actual configurations of the VENUS-F reactor and their heterogeneities.

References


Study of FREYA pulsed neutron source experiments using a multi-group time-dependent diffusion equation


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Abstract

One of the challenges in the accelerator-driven subcritical reactors is the online subcriticality monitoring during operation. Different experimental methods are being evaluated in the FREYA Project in order to establish a robust methodology. The experimental facility is made of a versatile (different beam time patterns) deuteron accelerator GENEPI-3C that produces 14-MeV neutrons by fusion reactions on a tritium target at the centre of the VENUS-F subcritical fast core. The neutron population in the core can be studied thanks to a set of fission chambers spread in the reactor.

One method to measure the reactivity of ADS is to carry out pulsed neutron source experiments. In that case, the so-called area method allows one to extract the reactivity using the ratio of the prompt decay area to the delayed neutron area. However this method is based on point kinetics and numerous experimental results show that depending on the detector type and position, the area method result must be corrected from spatial effects to get the right answer. In this work we propose to go beyond point kinetics theory in order to study spatial effects in the VENUS-F reactor using a 1-dimensional multi-group diffusion equation with a simplified geometric model of VENUS-F. The solutions of this equation are explored in the modal approach and the diffusion parameters are estimated by Monte Carlo simulation. This model allows one to understand and evaluate PNS spatial correction factors for the VENUS-F core in a various set of configurations with $k_{eff}$ ranging from 0.9 to 0.97. Results show that correction factors are driven by simple geometry and material effects.

Introduction

One of the challenges in the use of accelerator-driven subcritical reactors (ADS) is the online subcriticality monitoring during core operation and core loading phases. For this purpose, various experimental methods are being evaluated at the GUINEVERE facility located at the SCK/CEN site in Mol (Belgium). The facility couples the versatile deuteron accelerator GENEPI-3C and the modular fast neutron subcritical
reactor VENUS-F. GENEPI-3C can deliver pulsed as well as continuous beams of deuterons. Those impinge on a tritium target located at the reactor core centre, creating a pulsed or continuous 14-MeV neutron source by fusion reactions.

One of the reactivity measurement techniques tested in the framework of the FREYA Project of the 7th Euratom Framework Programme [1] for core loading procedures, is the analysis of pulsed neutron source experiments with the so-called area (or Sjöstrand [2]) method.

Results of the analysis performed on data measured by ten fission chambers (SFCs) showed that raw reactivity estimates need to be corrected from space-energy effects which obviously are not taken into account by reactor point kinetics [3]. Although such correction factors can be successfully calculated using either a Monte Carlo [4, 5] or a deterministic neutron transport code [6, 7], it is difficult to get much physics insight from them since all the neutron physics processes are intertwined in such complex codes. This is the reason why we have developed a semi-analytical approach. It uses a time-dependent multi-group diffusion equation to describe a multi-zone but single-dimension fast reactor, which is solved using a modal expansion of the neutron flux.

In this paper, we first give a short description of the GUINEVERE facility and we present the main characteristics of the VENUS-F subcritical reactor as well as those of the GENEPI-3C accelerator. The principle of the area method for analysing PNS experiments and for calculating the reactivity are recalled. Results inferred from the experimental data measured with ten fission chambers spread throughout the entire reactor while operating in the SC1 configuration are shown and discussed. The need for correction factors is emphasised. This leads to the presentation of our time-dependent multi-group diffusion approach of the problem. We introduce the simplified, one-dimensional, multi-zone model developed to describe the subcritical configuration of VENUS-F. This model is employed to reproduce the pulsed neutron source experiment by using a time-dependent multi-group diffusion equation (diffusion model). The rough calculation of the multi-group parameters obtained from simulation performed a simplified Monte Carlo transport code is briefly explained. Finally, the main steps for solving the diffusion problem by means of the decomposition of the neutron flux on α modes are presented. This allows us to calculate the reactivity value given by the area method as a function of the detector position inside the core. It is shown that, as in the case of the experiments, the results depend on the fission chamber location. To solve this problem, position-dependent correction factors are calculated and then applied to the experimental data of different cores allowing an excursion on reactivity from ~4.1$ to ~17$. The corrected reactivity values such obtained are in agreement with the reference reactivity value inferred from MSM method [8]. Finally future improvements of the model are discussed.

The GUINEVERE facility

The VENUS-F core

The VENUS-F fast reactor takes place in a cylindrical vessel of approximately 80 cm in radius and 140 cm in height. A 12x12 grid contained inside a square stainless steel casing can receive up to 144 elements of ≈8x8 cm\(^2\) in section, which can be fuel assemblies, lead assemblies, or specific elements for accommodating detectors or absorbent rods. The remaining room in the vessel is filled with semi-circular lead plates, which act as a radial neutron reflector. In addition the core is reflected by top and bottom 40 cm-thick lead-reflectors. Each fuel assembly (FA) contains a 5x5 pattern, filled with 9 fuel rodlets and 16 lead bars, surrounded by lead plates. The fuel is 30 wt. % enriched metallic uranium provided by CEA. Among the set of FAs, six are actually safety rods (SR) made of boron carbide and fuel followers with the absorbent part retracted from the core in normal operation. Two control rods (CR) made of boron carbide
can be positioned at various locations in the 12x12 grid. Another absorbent rod, whose reactivity worth is very small, called PEAR, is available for performing rod drop experiments. Various configurations of the reactor in terms of reactivity can be studied thanks to the modular shape of the core. In this paper, we focused on four subcritical configurations of VENUS-F, named hereafter SCx (x from 1 to 4). These configurations are presented in Figure 1. The FAs (in blue for the regular ones, in light blue for the SRs with fuel followers) are arranged in a way to create a pseudo-cylindrical core with a central hole to allow inserting the GENEPI-3C deuteron accelerator [10]. The two boron-carbide CRs (in red) are located at the core periphery. The reactivity of each configuration was measured using the MSM method [8], in order to have a reference reactivity value with which the results of PNS experiments can be compared. As shown in Figure 1, the Venus-F reactor was equipped with 10 fission chambers working in pulsed mode. Three different types of FC were used, either photonis CFUL01 and CFUM21 [9], or GE reuter-stokes (RS), whose specifications are listed in Table I. In order to help localising the various assemblies and detectors, an arbitrary co-ordinate system is used in the 12x12 grid: the upper left corner is labelled (-6,6) and the lower right one (6,-6), there is no (0,0) element. Outside the 12x12 grid, six cylindrical cavities bored in the outer reflector can receive experimental devices. They are labelled, from left to right: A1, B1, C1, A2, B2 and C2.

**The GENEPI-3C accelerator**

Since the VENUS-F reactor is subcritical, an external neutron source is needed to drive the neutron multiplication inside the core. This external source was created at the centre of the VENUS-F core by deuterons interacting through the tritiated target. The deuterons ions were accelerated up to an energy of 220 keV by the GENEPI3C particle accelerator [10] built by a collaboration of CNRS-IN2P3 laboratories. The fusion reactions at core mid-plane generate a quasi-isotropic field of ≈14-MeV neutrons. The GENEPI-3C can operate in pulsed mode, in continuous mode and also in continuous mode with short beam interruptions. In pulsed mode, the GENEPI3C accelerator provides 1 microsecond pulses of ≈40 mA peak current. The neutron source intensity in this mode is around 1-2×10^6 neutrons/pulse. The pulse frequency can vary from 50 to 1000 Hz. For the work presented herein, a frequency of 200 Hz was chosen.
Figure 1: Mid-plane cross section of the four configuration of VENUS-F core, from SC1 to SC4 FAs are in blue. Lead elements are in yellow. Detector names and positions are indicated by arrows.

Table I: The ten fission chamber used for the pulsed neutron source experiments. There is a different position for SC1.

<table>
<thead>
<tr>
<th>Name</th>
<th>Main deposit</th>
<th>Approximate mass (mg)</th>
<th>Location</th>
</tr>
</thead>
<tbody>
<tr>
<td>CFUL01-659</td>
<td>235U</td>
<td>1000</td>
<td>(-6,6)</td>
</tr>
<tr>
<td>CFUL01-658</td>
<td>235U</td>
<td>1000</td>
<td>A1 (only for SC1)</td>
</tr>
<tr>
<td>CFUL01-653</td>
<td>235U</td>
<td>1000</td>
<td>A1 (C2 for SC1)</td>
</tr>
<tr>
<td>RS-10071</td>
<td>235U</td>
<td>100</td>
<td>(6-6)</td>
</tr>
<tr>
<td>RS-10072</td>
<td>235U</td>
<td>100</td>
<td>(6,6)</td>
</tr>
<tr>
<td>RS-10074</td>
<td>235U</td>
<td>100</td>
<td>(-6,-6)</td>
</tr>
<tr>
<td>RS-10075</td>
<td>235U</td>
<td>100</td>
<td>C1</td>
</tr>
<tr>
<td>CFUM21-667</td>
<td>235U</td>
<td>10</td>
<td>(6,-2)</td>
</tr>
<tr>
<td>CFUM21-668</td>
<td>235U</td>
<td>10</td>
<td>(-2,-6)</td>
</tr>
<tr>
<td>CFUF34</td>
<td>235U</td>
<td>1</td>
<td>(-1,-3)</td>
</tr>
<tr>
<td>CFUL01-673</td>
<td>238U</td>
<td>100</td>
<td>C2</td>
</tr>
</tbody>
</table>
The area method for pulsed neutron source experiments

When dealing with pulsed neutron source (PNS) experiments, the area method (also referred to the Sjöstrand method) allows one to determine in a straightforward way the reactivity (in dollar units) of a subcritical nuclear reactor with no input from theoretical calculations, as long as the assumptions of the neutron point kinetics hold in the reactor. This technique is based on the analysis of the time response of detectors placed in the reactor, after a source neutron pulse. The evolution of the detector count rates strongly reflects that of the neutron population over time. Indeed, assuming that neutron point kinetics can represent the neutron population evolution $n(t)$ over time, the equation of its time decrease after a pulse (considered as a Dirac peak) within the one-delayed neutron group approximation reads:

$$n(t) = n_0 \left( e^{-\frac{-\rho + \beta_{\text{eff}}}{\lambda}} + \frac{\lambda \beta_{\text{eff}} \Lambda}{(-\rho + \beta_{\text{eff}})^2} e^{-\frac{-\rho \lambda}{(-\rho + \beta_{\text{eff}})}} \right)$$  \hspace{1cm} (1)$$

where $\lambda$ is the one-group decay constant, $\rho$ is the reactivity, $\beta_{\text{eff}}$ is the effective delayed neutron fraction and $\Lambda$ the mean generation time. In equation (1), we can distinguish a fast component due to prompt neutrons, and a slow component, due to delayed neutrons. The integration of the prompt component over time gives the prompt area $A_p$:

$$A_p = n_0 \frac{\lambda \beta_{\text{eff}} \Lambda}{(-\rho + \beta_{\text{eff}})}$$  \hspace{1cm} (2)$$

whereas the integration of the delayed component gives the delayed area $A_d$:

$$A_d = n_0 \frac{\beta_{\text{eff}} \Lambda}{\rho(-\rho + \beta_{\text{eff}})}$$  \hspace{1cm} (3)$$

Then, the ratio of these two areas gives directly the value of the reactivity in dollars:

$$\rho_s = \frac{\rho}{\beta_{\text{eff}}} = -\frac{A_p}{A_d}$$  \hspace{1cm} (4)$$

Two conditions must be verified before application of the area method:

- the period must be long enough to ensure the disappearance of the population of prompt neutrons;
- the precursor population must have reached a steady state from one pulse to another.

Experimentally, for a set of pulses repeated with a fixed frequency, a single pulsed neutron source (PNS) histogram is constructed by summing the fission chamber time responses as a function of the time elapsed after the neutron pulse. The analysis consists in separating in this histogram the prompt neutron contribution from the delayed neutron one. After integrating the time spectrum to get the surfaces $A_p$ and $A_d$, the anti-reactivity of the reactor configuration can be calculated using Eq. 4. Typical PNS histograms are presented in Figure 6. for various detector positions: CFUF34 in the core, CFUM21667 at the core-reflector interface, RS-10071 in the corner of the 12x12 grid and CFUL01-658 inside the outer part of the reflector. These histograms were built by adding up at least one million pulses for a beam frequency of 200 Hz and they are normalised to the same maximum. The PNS histogram ranges from time 0 up to the inverse of the pulse frequency. Except for the CFUL01-658, the PNS time spectra exhibit almost the same shapes, which however depend on the detector position inside the reactor. The fast component of the neutron population decay corresponds to the prompt neutron driven decrease. It is
also observed that the closer to the reflector is the FC, the slower this fast component decay. This behaviour signs the presence of spatial effects, which are not predicted by the point kinetics model. Except for the two detectors located inside the outer lead reflector, beyond 2 ms, a quasi-constant level, referred hereafter to as the delayed neutron level $L_d$, is reached after $\approx 2$ ms. This constant component is the sum of the contributions of the delayed neutrons originating from the successive pulses. The constant level of the delayed neutrons $L_d$ is first obtained by calculating the average count rate on a domain ranging from $t_{\text{min}}$ to $t_{\text{max}}$. The latter is simply the period $T$ between two beam pulses, which is the inverse of the pulse frequency. The former was chosen to be $t_{\text{max}}-0.5$ ms.

Then, the area $A_d$ is nothing else than the rectangular area of width $T$ and height $L_d$. The total area $A_{\text{tot}}$ being the sum of all the counts in the PNS histogram, the prompt surface is calculated from the following difference: $A_p = A_{\text{tot}} - A_d$. Finally, the reactivity is calculated as:

$$\rho = \frac{A_{\text{tot}} - A_d}{A_d}$$  (5)

The reactivity values extracted from PNS histograms according to Eq. 5 are represented by solid black dots in Figure 2 for configurations SC1, SC2, SC3 and SC4. The error bars were calculated by taking into account the statistical errors as well as the systematic ones. This systematic error accounts for the deviation from a flat distribution between $t_{\text{min}}$ and $T$. A dispersion in the results is (clearly) noticeable, which seem to depend on the detector location in the reactor and more specifically on the distance of the FC to the reactor centre.

**Figure 2:** Reactivities extracted from the different detectors for configurations SC1, SC2, SC3, SC4

Raw results are in black, results corrected with the diffusion model without walls are in blue and those corrected with a model including walls are in red. The dashed area correspond to the MSM reference value with its uncertainties.
Result wise, three groups of detectors can be identified. The first one contains only the CFUF34 detector, which is the only one located in the reactor core. It is also the only one from which the reactivity value obtained with the area method is in very good agreement with the reference value given by the MSM method. The second group gathers six (RS-10074, RS-10071, CFUL01659, CFUM21-667, CFUM21-668 and RS-10072) or even seven (RS-10075) detectors, which are located either at the core-reflector interface or in the corners of the 12x12 grid, in the inner part of the reflector. The last detectors (RS-10075, CFUL01-653 and CFUL01-659) form the third group. They are located rather far away from the core, in the outer part of the reflector, outside the stainless steel casing. Clearly the area method fails at providing the correct value of the reactivity when the FCs are not in the inner part of the stainless steel casing. The effect seems to be stronger when the detector is farther from the core. In the case of the third group, one just needs to look at the upper part of Figure 2 to observe that the neutron population does not decay as predicted by neutron point kinetics. Furthermore the neutron population does not even reach the delayed neutron level within the time window corresponding to the period between the beam pulses. In these conditions, the area $A_d$ is overestimated, which leads to an underestimation of $A_r$ and the reactivity value extracted is wrong. If one wants to extract the real reactivity from any detector, some correction factors must be computed. This was successfully achieved using the Monte Carlo neutron transport code MCNP [11] in [4, 5]. In that case, the reactivity given by a detector was corrected by a multiplication factor $f_{MCNP}$ calculated as:

$$f_{MCNP} = \left(\frac{\rho_{KCODE}}{\beta_{eff}}\right) \left(\frac{R-R_p}{R_p}\right)$$

(6)

where $\rho_{KCODE}$ is the reactivity of a VENUS-F MCNP model which was computed with a stabilised, iterative fission source (the so-called KCODE calculation). $R$ and $R_p$ are the position-dependent FC fission rates with and without delayed neutrons, respectively. They were calculated for a fixed-source mode with a 14-MeV neutron source located at the centre of the MCNP model of VENUS-F.

Use of 1D time-dependent diffusion model

1D description of VENUS-F

Results of the area method showed that raw reactivity estimates need to be corrected from space-energy effects which obviously go beyond reactor point kinetics. As seen in Figure 2, the spatial dependence of the results exhibits a rather simple behaviour as a function of the detector position. This points to the possibility of a simple approach which would take advantage of reactor symmetries and would provide a way to identify the most relevant physical phenomena affecting a Pulsed neutron source experiment carried out in the VENUS-F reactor. To go beyond reactor point kinetics, VENUS-F was roughly described by an infinite cylinder, taking advantage of its quasi-cylindrical shape and that source and detectors are all located in the middle plane of the reactor. However, depending on the detector considered, two different 1D models must be considered in order to mimic the 3D actual core at best (one in the X-direction and one in the XY direction). Indeed, when dealing with FCs CFUF34, CFUL01-653, RS-100-75, CFUM21-668 and CFUM21-667, we will use the X model, whereas the XY model will be used for FCs CFUL01-659, RS-10072, RS-10071 and RS-10074. In terms of materials, the 1D models are also simplified compared to the actual composition of the reactor. The core region is made of a homogeneous mixture of enriched uranium, lead and iron which replaces the heterogeneous FAs made of enriched uranium rodlets, stainless steel plates, lead plates and lead bars. The stainless steel casing composition is reduced to iron. The inner and outer parts of the reflector are made of lead only.
The diffusion model

The space, energy and time evolution of the neutron flux is modelled by the 1D time-dependent multi-group diffusion equation which reads:

\[
\begin{align*}
\left\{ \begin{array}{l}
\frac{1}{c} \frac{d}{dt} \psi(r,t) = \left( M + (1 - \beta)F_p \right) \psi(r,t) + \sum \lambda_i X_{di} C_i(r,t) + S(r,t) \\
\frac{d}{dt} C_i(r,t) = -\lambda_i C_i(r,t) + \beta_i F^T_i \psi(r,t)
\end{array} \right.
\end{align*}
\] (7)

where the components of the \( \Psi(r,t) \) vector are the neutron fluxes of the different energy groups whereas \( C_i \) denotes the population of precursors of delayed neutrons. The matrix \( M \) gathers the operators of leakage, absorption and scattering. The matrix elements read:

\[
\begin{align*}
M_{g,g} &= D_g \Delta r - \Sigma a_{g,g} - \sum \Sigma_{g' \rightarrow g} p_{g' \rightarrow g} \Sigma_{s,g} \\
M_{g,g'} &= \sum \Sigma_{g' \rightarrow g} p_{g' \rightarrow g} \Sigma_{s,g'}
\end{align*}
\] (8)

\( D_g \) refers to the diffusion coefficient in the energy group \( g \), \( \Sigma_{a(g)} \) is the macroscopic absorption (scattering) cross section of the medium for the neutrons in the energy group \( g \). \( p_{g' \rightarrow g} \) is the probability of transfer from group \( g' \) to \( g \).

\( \beta \) is the total delayed neutron fraction and \( F_p \) is the prompt fission operator:

\[
F_{p,g,g'} = \chi_{g} \nu \Sigma_{f,g'}
\] (9)

where \( \chi_{g} \) is the part of the prompt neutron energy spectrum which is in the range of group \( g \), and \( \nu \Sigma_{f,g} \) is the average number of neutrons emitted by fission times the macroscopic fission cross section of the medium for neutrons in the \( g \) energy group. Each component of the vector \( X_{d,i} = (\chi_{d,i,g} = 0, \ldots, \chi_{d,i,g} = G) \) corresponds to the part of the delayed neutron energy spectrum which is in the range of group \( g \) for the delayed neutron (DN) group \( i \). \( \lambda_i \) and \( \beta_i \) are the radioactive constant and the delayed neutron fraction associated with the DN group \( i \), respectively.

Since the reactor model is made of different zones corresponding to different media, the diffusion model has to satisfy the continuity condition at each interface:

\[
\left\{ \begin{array}{l}
\psi_-(r = \text{interface}) = \psi_+(r = \text{interface}) \\
D_- \frac{\partial \psi_-}{\partial r} \bigg|_{r = \text{interface}} = D_+ \frac{\partial \psi_+}{\partial r} \bigg|_{r = \text{interface}}
\end{array} \right.
\] (10)

For the boundary with the reactor exterior we consider a null incoming neutron current:

\[
j_-(r_{or}) = \Sigma_s \psi_-(r_{or}) + \Sigma_s \frac{\partial \psi_-}{\partial r} \bigg|_{r = r_{or}} = 0
\] (11)

For the work described hereinafter, eight delayed neutron groups were used [12] and neutrons were split in three energy groups: \( g = 1 \) corresponds to the neutrons with \( E > 100 \text{ keV} \), \( g = 2 \) to neutrons such that \( 10 \text{ eV} < E < 100 \text{ keV} \), and \( g = 3 \) to neutrons with \( E < 10 \text{ eV} \).
Estimation of the macroscopic parameters

In order to estimate very rapidly the macroscopic parameters entering the diffusion model, a simplified Monte Carlo neutron transport code was developed. The neutrons are transported stochastically in a spherical homogeneous medium made of enriched uranium, natural lead and iron, as to mimic the VENUS-F FAs. The continuous energy microscopic cross sections are directly retrieved from the ENDF database [13]. Fission, absorption, elastic and inelastic reactions are the only processes considered. The latter is treated only roughly. The number of reactions of each type is incremented after each occurrence during the random walk. In addition, the neutron energy flux is calculated by dividing the total reaction rate by the total macroscopic cross section.

The flux in an energy group is defined as:

\[
\psi_g = \int_{E_g} dE \frac{R_{\text{tot}}(E,r,t)}{\Sigma_{\text{tot}}(E,r)}
\]  

(12)

where \( R_{\text{tot}}(E,r,t) = \Sigma_{\text{tot}}(E,r)\psi(E,r,t) \) is the number of interactions at point \((r,t)\) of the test particles of energy \(E\).

The macroscopic cross section of reaction \(q\) in group \(g\) is computed as:

\[
\Sigma_{q,g} = \frac{1}{\psi_g} \int_{E_g} dE \; R_q(E,r,t)
\]  

(13)

and the transfer cross section from group \(g'\) to group \(g\) as:

\[
\Sigma_{g}^{g'\rightarrow g} = \frac{1}{\psi_g} \int_{E_g} dE \int_{E_{g'}} dE' p(r,E' \rightarrow E) R_s(E,r)\]  

(14)

\( p(r,E' \rightarrow E) \) is the probability of transfer from energy \(E'\) to \(E\).

We use the P1 approximation (of the scattering term) [14] for calculating the diffusion coefficient in group \(g\):

\[
D_g \approx \frac{1}{3(\Sigma_{t,g} - \mu \Sigma_{s,g})}
\]  

(15)

where \(\Sigma_{t,g}\) is the total cross section in group \(g\) and \(\mu = 2/(3A)\). The part of the prompt fission spectrum falling into group \(g\) is:

\[
\chi_g = \int_{E_g} dE \frac{2\pi}{(\pi T)^{3/2}} \sqrt{E} \; e^{-E/T}
\]  

(16)

Finally the average inverse velocity is calculated as:

\[
\frac{1}{v_g} = \frac{1}{\psi_g} \int_{E_g} dE \frac{R_{\text{tot}}(E,r,t)}{v \; \Sigma_{\text{tot}}(E,r)}
\]  

(17)

\(\lambda\)-mode and \(\alpha\)-mode solutions

Steady-state solutions of Eq. 7 can be found by solving the eigenvalue equation:

\[
\tilde{M} \Phi_\lambda(r,t) = -\lambda \left( \sum_i \beta_i \tilde{F}_{di} + (1 - \beta) \tilde{F}_p \right) \Phi_\lambda(r,t)
\]  

(18)

\[
\Phi_\lambda(r,t) = \exp\left[-\int_{-\infty}^{r} dR \frac{1}{v} \frac{R_{\text{tot}}(E,R,t)}{\Sigma_{\text{tot}}(E,R)}\right]
\]  

(19)
There are an infinite number of $\lambda_n$ eigenvalues. Each $\lambda_n$ is associated with one $\Phi_{\lambda,n}(r)$ eigenvector. Unfortunately these eigenvectors cannot be directly used to solve Eq. 7 because they do not linearise the equation and they do not form a suitable basis for decomposing a fast neutron source. Indeed $F$ is a non-diagonal matrix operator when using more than one energy group.

However the $\lambda$ modes can be used to compute important static parameters associated with the associated critical reactor. The main parameter that characterises a nuclear reactor is the neutron multiplication factor $k_{\text{eff}}$ which is also the inverse of the smallest eigenvalue $\lambda_0$ associated with the eigenvector $\Phi_{\lambda_0}(r,t)$ which is called the fundamental mode. $\lambda_0$ is a factor that multiplies the neutron production so that it compensates exactly the neutron disappearance in order to obtain a critical reactor. Figure 3 shows the neutron fluxes of the fundamental mode for the reactor models $X$.

Figure 3: Fundamental flux obtained with the VENUS-F model and the VENUS-F+building model in dashed line. Right part zoom in the outer reflector for the low-energy group

Also, removing the delay part in Eq. 19 gives the prompt factor $\lambda'_0 = k_p^{-1}$ and leads to the effective delayed neutron fraction $\beta_{\text{eff}}$ using the relationship $\beta_{\text{eff}} = 1 - \frac{k_p}{k_{\text{eff}}}$ [15]. Going back to the time-dependent Eq. 7, we now turn to the $\alpha$-mode equation. This equation is based on the following ansatz:

$$\Phi_\alpha(r, t) = e^{-\alpha t} \Phi_\alpha(r) \rightarrow \frac{d}{dt} \Phi_\alpha(r, t) = -\alpha \Phi_\alpha(r, t)$$

(19)

As a first step, we can introduce the relationship 19 in Eq. 7 but restrict ourselves to the prompt problem:

$$-\alpha \left( \frac{1}{\tau} \right) \Phi_\alpha(r) = \left( \hat{M} + (1 - \beta) \hat{F}_p \right) \Phi_\alpha(r)$$

(20)

In that case, the first eigenvalue ($\alpha_0$) is related to the fundamental $\lambda'$-mode by:

$$\alpha_0 = \frac{(k_p - 1)(1 - \beta) \Lambda}{k_p}$$

(21)

which gives a mean generation time $\Lambda = 0.36 \ \mu s$. 

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There is an infinite number of \( \alpha \) eigenvalues corresponding to an infinite number of eigenvectors \( \Phi_\alpha(r) \). Each mode, characterised by the number of nodes of the eigenfunction, exists for each energy group. In order to have the same representability of each energy group, we select for our \( \alpha \)-basis the same number of state in each group. These properties will allow us to compose an external neutron source as well as the delayed neutron source on the \( \alpha \)-state basis.

Let \( P_\alpha \) be the projector on the basis states \( \{ \Phi_\alpha \}_N \). The neutron flux \( \Psi(r,t) \) can be expanded as:

\[
P_\alpha \Psi(r,t) = \sum_n a_n(t) \Phi_\alpha_n(r)
\]

(22)

and the neutron source as:

\[
P_\alpha S(r) = \sum_n s_n \Phi_\alpha_n(r)
\]

(23)

As an example, the spatial distribution and the alpha-mode composition of the fast neutron source used to simulate PNS experiments in the X model is shown in figure 4. As expected, there is no significant source contribution in the low-energy group and consequently almost all the low \( n \) modes (for which the low-energy group component dominates) have a negligible contribution whereas the high \( n \) modes (for which the high-energy group component dominates) are strongly populated.

Now let us include the precursors to solve the time-dependent full problem. Since each \( \alpha \)-mode gives birth to a precursor population which can be considered as immobile, we can write for each \( \alpha \) mode:

\[
\tilde{c}_\alpha(r) = F^r \Phi_\alpha_n(r)
\]

(24)

\[
\frac{d}{dt} C_i(r,t) = \tilde{c}_\alpha(r) \frac{d}{dt} c_{i,\alpha}(t) = (-\lambda_i c_{i,\alpha}(t) + \beta_i) \tilde{c}_\alpha(r)
\]

(25)

and the feedback in the neutron equation is:

\[
P_\alpha \left( \frac{1}{v} \right) \cdot X_{d,i} C_i(r,t) = c_{i,\alpha}(t), P_\alpha \left( \frac{1}{v} \right) \cdot \tilde{F}_{d,i} \Phi_\alpha(r)
\]

(26)

Figure 4: Left: amplitude of the different modes to compose the source
Right: spatial shape of the neutron source

The time-dependent solution

The projection of Eq. 7 on the \( \alpha \) basis leads to a coupled system:
\[
\begin{pmatrix}
\frac{d}{dt}a_0(t) \\
\vdots \\
\frac{d}{dt}c_{i,\alpha=0}(t)
\end{pmatrix} = \mathbf{H}(t) \begin{pmatrix}
a_0(t) \\
\vdots \\
c_{i,\alpha=0}(t)
\end{pmatrix}
\] (27)

\(\mathbf{H}\) contains all the coupling terms between the vectors \(\Phi_\alpha(r,t)\) and the precursor concentrations \(c_{i,\alpha}(t)\). In the case without any external neutron source, the time dependence in \(\mathbf{H}\) vanishes. We can diagonalise \(\mathbf{H}\), and use the resulting eigenvector \(\mathbf{V}_\epsilon\) and the \(\epsilon\) eigenvalue as new solutions. Since, in a PNS experiment, the external neutron source can be considered as a Dirac pulse \(\delta(t)\), the previous solution can be used by setting \(a_\alpha(t = 0) = s_\alpha\), and by allowing the different exponentials involved in the solution of Eq. 27 to decay freely. Once the diffusion model has been solved to get the neutron flux, it is easy to calculate a quantity representing an “area” of prompt neutrons:

\[A_p(r) = \sum_{n}^{N_\alpha} \frac{s_{n,\alpha}}{\alpha_n} \Phi_\alpha(r)\] (28)

Associated with the corresponding total “area”:

\[A_{tot}(r) = \sum_{n}^{N_\alpha (1+N_d)} \frac{s_{n,\epsilon}}{\epsilon_n} \mathbf{V}_\epsilon(r)\] (29)

and with the corresponding delayed neutron “area” \(A_d = A_{tot} - A_p\). As in the case of the PNS experiments, a value of reactivity can be calculated as:

\[\rho_{\text{area}}(r) = -A_p(r)/A_d(r)\] (30)

To get closer to experimental count rates given by the FCs, the flux was convoluted with the U235 fission cross section. Then, correction factors similar to those calculated with a neutron transport code can be calculated as:

\[f_{\text{diffusion}}(r) = \frac{\rho^s_{\text{area}}(r)}{\rho_{\text{area}}(r)}\] (31)

where \(\rho^s_{\text{area}}\) was given by the \(\lambda\)-mode approach.

**Results**

The results of the area method performed on the VENUS-F 1D model within the framework of the diffusion model are shown in Figure 5. In the left part of the Figure, the reactivity \(\rho_{\text{area}}(t)\) calculated using relationship 30 is given as a function of the detector position in the reactor and is compared to the true value of the reactivity given by the \(\lambda\) mode analysis for the configurations studied.
The ant reactivity calculated using the area method is predicted to be overestimated when the detector is located close to the external neutron source and underestimated when the detector is farther away. This leads directly to a correction factor which is below one at small distance and larger than one at larger distance from the source and which gets stable in the reactor reflector. This behavior shows that theoretically one could find a detector location free of correction. However this position might likely turn out to be inaccessible for technological reasons in a real reactor. This is the reason why it is important to be able to correct results for detectors located anywhere in the reactor. The correction factors $f_{\text{diffusion}}(r)$ were then applied to the raw experimental values. The final reactivity values are shown in Figure 2 (in blue) together with the reference reactivity value of SC1 given by the MSM measurements [8]. The corrected values of reactivity are compatible with the MSM reference value for all the detectors except those located the farthest from the reactor centre. This was expected since it has been shown in [16] that these two detectors were strongly influenced by the reactor surroundings (concrete walls, reactor vessel flanges) which are not taken into account in the present VENUSF 1D model. To investigate this effect we included two additional zones in our diffusion model: a layer of air and another made of concrete. The effect of this new geometry is easily seen on the fundamental flux (dashed lines in Figure 3). The contribution of the low-energy neutron is increased outside of the Stainless Steel casing due to the presence of the concrete wall. Unfortunately, the use of the correction factors calculated with this new geometry do not improve the results for the detectors located outside the stainless steel casing, as can be seen in Figure 2 (red points). To further investigate these results, the evolution of the count rate of one FC located in the outside reflector (CFUL01-658) was compared with the count rate evolution predicted by the diffusion model including walls. As can be seen in Figure 6, the experimental curve does not reach the delayed neutron level within the experimental time window contrary to what is predicted by the diffusion model. Although a lot of ingredients of the model were modified to try to mimic the experimental curves, the model failed to reproduce the spectacular effect observed in the data.
Conclusion

In accelerator-driven subcritical reactors, the neutron flux is known to exhibit higher modes due to the presence of an external source which induces deviations from reactor point kinetics. This is the reason why a time-dependent multi-group diffusion equation was used to describe Pulse Neutron Source experiments in the subcritical lead fast neutron reactor VENUS-F. The complexity of the description was deliberately reduced to minimum (one-dimensional, a few homogeneous zones) so that solving the diffusion model using an $\alpha$ mode decomposition of the neutron flux be straightforward and almost fully analytical and consequently very fast. The solutions of the diffusion model were used to compute correction factors to be applied to reactivity values calculated using the area method on experimental PNS histograms built for a set of fission chambers spread in the reactor. Interestingly, for most detectors, the results turned out to be very similar with those obtained using full Monte Carlo neutron transport simulations in a thorough geometrical description of the VENUS-F reactor. But the diffusion model fails to reproduce the temporal evolution properly and consequently to correct the reactivity bias for the detectors in the outer reflector. But it can be noticed that the time integration washes out the failure of time evolution of the model and allows us to have spatial corrections for detectors which reach delayed neutron level. Unfortunately, due to the statistics needed to perform this experiment, it is too time consuming to decrease the beam frequency in order to wait for the reaching of the delayed neutron level.

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References


Zero-power coupling experiments in support of the MYRRHA ADS in the frame of the MYRTE Project

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Abstract

Zero-power coupling experiments in support of the MYRRHA ADS project have been planned in 2017 in Working Package five (WP5) of the MYRTE (MYRRHA Research and Transmutation Endeavour) HORIZON2020 project. These experiments in the VENUS-F zero-power reactor – GENEPI-3C accelerator coupled system are the prolongation of the recently accomplished (February 2016) experiments in the FREYA Project that were realised at SCK•CEN during the last five years. The results that were obtained within FREYA will be used for the pre-licensing phase, but need to be completed with some additional experiments. These investigations will include simulations of in-pile-sections (IPS) inserted inside the VENUS-F core for stainless steel testing, of the influence of the reflector type and thickness, and enable the optimisation of the type and position of the detectors for reactivity monitoring. Finally, specific questions may arise from the safety authorities during the licensing of the MYRRHA facility that will require the execution of dedicated experiments. Void effects of the coolant for example will be simulated in the VENUS-F core for Pb and Bi. All these experiments aim at validating calculation tools and nuclear data for MYRRHA, and at providing reliable data with regard to the specific safety aspects of the facility. The plans and the tasks of WP5 of the MYRTE Project are presented.

Introduction

The goal of the MYRTE European Union Project [1] that has started in April 2015 with a duration of 4 years is to perform research to support the development of the MYRRHA research facility [2], which aims to demonstrate the feasibility of high-level nuclear waste transmutation at industrial scale. The MYRTE consortium involves 27 European organisations including universities, research institutes and industrial corporations. Most MYRTE consortium members have been involved previously in major projects concerning the development of Generation IV LFR and ADS technology, see Figure 1. The goal of MYRTE is to perform the necessary research in order to demonstrate the feasibility of transmutation of
high-level waste at an industrial scale through the development of the MYRRHA research facility. Within MYRRHA as a large research facility, the demonstration of the technological performance of transmutation will be combined with the use for the production of radioisotopes and as a material testing for nuclear fission and fusion applications. Numerical studies and experimental facilities are foreseen to reach this goal.

Besides co-ordination and dissemination activities, the MYRTE Project contains five technical work packages (WP). These technical work packages cover the priority research domains for the further development of the MYRRHA Project from a safety and reliability point of view of the facility. The content of the proposed work packages was defined based on the outcome and current state of the art of the different European projects CDT, MARISA, SEARCH, MAXSIMA, MAX, FREYA, ESNII+, THINS, FAIRFUELS, PELGRIMM and national programmes.

The first work package deals with the general co-ordination of the consortium. The second work package is devoted to the realisation of the injector part of the MYRRHA accelerator to demonstrate the feasibility and required reliability of this non-superconducting part of the accelerator. Also, some activities related to the overall design of the MYRRHA accelerator are foreseen. Those two activities have been identified at the end of the MAX-projects as the two main relevant topics to be addressed in the area of the linear accelerator for an ADS. These activities also build further on the efforts related to the setup of the Initial MYRRHA Accelerator Consortium (IMAC) centralised in the MARISA Project.

The third work package addresses the main outstanding issues in liquid-metal reactor thermal hydraulics by numerical simulation and experimental validation. Based on the results of the FP7 THINS project and the Belgian National programme and the interaction with the Belgian Safety Authorities in the frame of the MYRRHA pre-licensing phase, pool thermal hydraulics and fuel assembly thermal hydraulics were identified as the main focus of this WP. In these fields, the project aims at extending the knowledge and validation base by experimental research for the subsequent development and validation of numerical approaches, establishing best practice guidelines and final application to the MYRRHA system.
A dedicated WP4 on chemistry of volatile radionuclides was foreseen as it has been highlighted by the Belgian Safety Authorities as a major focus point within the pre-licensing phase. In this WP, the evaporation from LBE, capture and deposition of Po and fission products will be studied in detail starting from the results obtained in SEARCH, MAXSIMA and the Belgian national programme.

A special WP5 on experimental reactor physics is also foreseen to carry out supplementary experiments after the FREYA Project at the VENUS-F zero-power reactor GENEPI-3C accelerator coupled facility [3] to address questions that can arise from the safety authorities. The details of this WP are presented in this paper.

In the last technical WP, advanced studies on Americium-bearing oxide fuel are carried out to yield essential safety data to demonstrate the capability of developing minor-actinide fuel for transmutation.

**WPS: Experiments in support of the MYRRHA design evolution**

**Objectives**

To perform additional experiments to validate the reactivity monitoring methods in complement to the ones achieved during the FREYA Project. To carry out additional integral measurements to study the effect of fuel, reflector and structure material changes in the specific MYRRHA core design, and to investigate the impact of dummy IPS on the reactivity for code validation. As specific questions from the safety authorities during the licensing of the new MYRRHA/FASTEF facility are expected, dedicated experiments have to be foreseen.

**Work description**

The determination of the subcriticality level is a major safety aspect of an ADS such as MYRRHA. Therefore several methods for the determination of the reactivity of such subcritical systems have to be investigated in zero-power experiments.

For ADS, such kind of experiments was initiated in the FP5 programme MUSE where the first coupling of an accelerator to a fast subcritical assembly was studied. These experiments used a pulsed neutron generator GENEPI-1 that was coupled to a fast sodium assembly representative of sodium-cooled reactors. At the end of the MUSE programme, a first proposal for a methodology for the online reactivity monitoring of an ADS was made based on the outcome of the experimental investigations. For the future experiments, an accelerator working in continuous and pulsed mode would be needed and the subcritical core should be made of lead instead of sodium, being in this way much more representative for lead-cooled ADS.

Within the framework of the ECATS (Experimental activities on the coupling of an accelerator, a spallation target and a subcritical blanket) research domain of the FP6 IP-EUROTRANS programme, the GUINEVERE (Generation of Uninterrupted Intense Neutron pulses at the lead VENus REactor) project was launched in 2006 in line with the conclusions of MUSE. The GUINEVERE project was devoted to the coupling of deuterium accelerator GENEPI-3C, working in current mode with and without beam interruptions and in pulsed mode, with a subcritical fast neutron VENUS-F core with lead coolant simulation. The GUINEVERE project within the FP6 Programme was devoted to the construction of the installations and the execution of a limited experimental program.

The FP7 FREYA (Fast Reactor Experiments for hYbrid Applications) Project was launched in 2011 at the VENUS-F – GENEPI-3C facility, as a prolongation of the GUINEVERE project. The FREYA Project addresses several questions regarding the design of the MYRRHA facility. It consists of an extended
experimental program that especially investigates the issue of reactivity monitoring of a subcritical system, with different subcritical level configurations. These are configured by changing the amount of fuel assemblies in the VENUS-F core, the source position or the reflector material. Experiments in support of the design and licensing of both subcritical and critical modes are being performed, bringing important data also for lead fast reactors in general.

Part of the experiments were achieved with a core as representative as possible of the MYRRHA core design as it was in the beginning of the FREYA Project. But due to time constraints and material availability in the time framework of this project, limited configurations were studied. The impact of several in-pile sections (IPS) (foreseen for irradiations in the MYRRHA core) was also studied but with preliminary designs.

Since the MYRRHA core design in the beginning of the FREYA Project was not the final design, there were several changes such as positions and composition for the IPS, Si-doping rings and sections for Mo production during the FREYA Project execution. Therefore it was not possible within FREYA to work with the final core configuration.

That is why it is foreseen in the MYRTE Project to perform a limited set of additional experiments in complement to the ones achieved during the FREYA Project. It is foreseen, if necessary, to carry out additional integral measurements to study the effect of fuel, reflector and structure material changes in the specific MYRRHA core design, and to investigate the impact of dummy IPS on the reactivity for code validation. As specific questions from the safety authorities during the (pre-) licensing of the MYRRHA are expected, dedicated experiments have to be foreseen. Finally, new questions may emerge from the analyses of the FREYA experiments that will be addressed in MYRTE.

All these experiments aim at validating calculations performed for the reactor design and providing reliable data with regard to the safety studies of the facility.

**Tasks of the work package**

This work package of the project is divided into five separate tasks:

- **Task 5.1 VENUS-F cores definition**

  Within the constraints of available materials at the VENUS facility, the set and the order of the VENUS configurations simulating the current specific MYRRHA core design and accidental situations will be selected based on calculations. Mainly the MCNP code will be applied for these simulations. In addition the deterministic code ERANOS will be used to calculate the delayed neutron parameters. The features of the current specific MYRRHA core design will be taken as the basis for these simulations. This will be done in close collaboration with the MYRRHA design team.

- **Task 5.2 VENUS-F: MYRRHA mock-up reference core characterisation measurements**

  The standard core characterisation measurements will be performed for the reference critical core. These are axial (with fission chambers with fissile and fertile deposits) and radial (with U-235 and U-238 foils) flux distributions, calibration of the power, control rod worth measurements, rod drop experiments and spectral index measurements and minor-actinide responses with calibrated fission chambers.

- **Task 5.3 Detectors deposit and position**

  Preliminary FREYA results seem to show that the choice of the detector deposit and position in the core or reflector chosen for the reactor instrumentation can make reactivity monitoring easier or more difficult. A complete study should be carried out in this new project since the data provided by the FREYA...
Project on this specific issue are expected to be scarce. The position of the detectors should simulate as much as possible the positions of the detectors in the real MYRRHA design, considering that in MYRRHA the positions may not be optimal due to the design constraints. These details will be discussed with the MYRRHA design team. In the FREYA experiments almost all detectors have U-235 deposits except one fission chamber (U-238). This U-238 fission chamber being sensitive mainly to fast neutrons presents promising results, however with low statistics. Therefore the possibility to use fission chambers with other threshold deposits such as Np-237 will be investigated.

- Task 5.4 MYRRHA Mock-up Reactivity Effects

All reactivity effects that are important from the point of view of the new core design will be investigated in different sub-tasks. The reactivity effects will be measured applying the well-known MSM method or applying methods that are proven to be robust and simple in the FREYA Project.

- Sub-task 5.4.1. Reflector

As the MYRRHA core will have a rather thick reflector, subsequent effects should be studied in the VENUS-F reactor by simulating this thickness with material surrounding the VENUS-F vessel.

- Sub-task 5.4.2. in-pile sections

The current progress in the design of the in-pile section items (for material irradiation, MA transmutation or other) will also bring new experimental configurations in order to quantify their effect on the reactivity or flux profiles (also with location in an external reflector).

- Task 5.5 Safety issues and accidental situations

Dedicated experiments have to be foreseen to answer questions from the safety authorities related to the licensing of the new MYRRHA facility. The exact content will be determined as a function of the specific questions from the safety authorities making use of existing material and fuel at VENUS-F.

**Current status of the work package**

The loading of new the VENUS-F reactor core simulating the current status of the MYRRHA design will start in January 2017. The tasks described above will be performed in the course of 2017.

In the meantime, the work related to the simulation of the MYRRHA design peculiarities using the available materials in the VENUS-F storage is ongoing. In particular the configuration of the fuel assembly (FA) has been agreed (see Figure 2). The FA will contain Pb and Bi to simulate the coolant of MYRRHA. The materials and the design of the IPS (both for Mo production and for material tests) are ready.
Conclusions

A prolongation of the experiments performed within the FREYA Project in support of MYRRHA design and licensing will start soon in the frame of the MYRTE Project.

The work related to the design of the fuel assembly and VENUS-F core configurations to simulate the current status of the MYRRHA design is ongoing in close collaboration with the MYRRHA design team.

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References

Kinetic measurements in BRAHMMA accelerator-driven subcritical system

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Abstract
Accelerator-driven subcritical systems are being studied worldwide for their potential in burning minor actinides and reducing long-term radiotoxicity of spent nuclear fuels. Monitoring of subcriticality of such systems is very important as the power produced is directly related to the subcriticality of the system. In order to pursue the physics studies of accelerator-driven subcritical systems, a subcritical assembly BRAHMMA (BeO Reflected And HDPe Moderated Multiplying Assembly) has been developed at Bhabha Atomic Research Centre (BARC), India. The paper reports various experimental measurements on BRAHMMA subcritical assembly. Kinetic measurements of various parameters such as reactivity ($\rho$), prompt decay constant ($\alpha$) and mean generation time ($\Lambda$) have been measured using pulsed neutron source (PNS) techniques. The methods used include area ratio, slope fit and source jerk. One of the difficulties in using area ratio method is that the measured reactivity is spatially dependent on the source location. This can be overcome by evaluating a spatial correction factor. In this paper, we have used Bell Glasstone method for evaluation of correction factors. The corrected reactivity gives a measure of the global reactivity and is independent of the detector location.

Key words: BRAHMMA, prompt decay constant, reactivity, Bell Glasstone spatial correction

Introduction
Accelerator-driven subcritical system is an innovative concept of employing a subcritical reactor coupled to a high-power proton beam accelerator, through a spallation neutron source [1]. This idea has gained much interest because of the advantages of ADS such as inherent safety, better fuel utilisation, burning of waste products particularly minor actinides, Thorium utilisation and generation of lesser amount of long-lived nuclear waste products. The whole process is proliferation resistant as the production of plutonium is minimised [2]. However when we are operating an ADS core loaded with minor actinides and plutonium the safety margin (beta effective) is less and we have to avoid the reactor going critical at any point of time. Therefore efficient and accurate reactivity measurement is a major concern. Several techniques have been studied in this regard which may be broadly classified into PNS techniques and noise techniques [3]. In this paper PNS techniques have been used to evaluate reactivity at different positions in a subcritical core. The spatial dependence of reactivity has been removed by using a simulated correction factor.
BRAHMMA facility

BRAHMMA is a modular subcritical system with thermal neutron spectrum [4]. It consists of metallic natural uranium as fuel and high density polyethylene (HDPe) as moderator in a 13 X 13 square lattice configuration followed by 200 mm of beryllium oxide (BeO) as reflector. The pitch of the fuel-moderator lattice is 48 mm. The whole assembly is finally surrounded by borated (1% by weight) polyethylene to isolate the system from scattered neutrons. The system has a $k_{eff}$ value of 0.89. Figure 1 shows the basic schematic of the BRAHMMA core. The axial experimental channels are shown in the figure.

![Figure 1: BRAHMMA subcritical core](image1)

Figure 2: BRAHMMA subcritical core coupled to DD/DT neutron generator

![Figure 2: BRAHMMA subcritical core coupled to DD/DT neutron generator](image2)
The subcritical system is coupled to D-D/D-T neutron generator [5]. The central 3 X 3 positions of the lattice (dimension 144 mm X 144mm) are empty and serve as the cavity for inserting neutron source. The target end of neutron source is inserted in this cavity such that the target is located at the centre of the core. Figure 2 shows the subcritical core coupled to the neutron generator. The rear half of the cavity is filled with a (Pb + BeO) plug. The assembly has 7 experimental channels – 3 axial and 4 radial – for measurements of various kinetic parameters of the Subcritical Assembly.

Reactivity measurement using pulsed neutron source (PNS) techniques

Area ratio method

In the area ratio method, pulses are repeatedly injected in the subcritical system at a constant frequency such that a constant delayed neutron background is achieved and then the detector response is recorded. The pulse width (ΔT) and pulse period (T) are chosen such that ΔT<<T. Also the pulse period should be small enough compared to the shortest half-life of delayed neutron precursors so that the delayed neutron background can be assumed constant during the pulse period. The reactivity of the subcritical system is then given by

\[
\frac{\rho}{\beta} (\$) = - \left[ \frac{A_p}{A_d} \right] \text{ ...................... (1)}
\]

\(A_t\) and \(A_d\) can be obtained by integrating the area under the curve as shown in Figure 3. \(A_p\) can be calculated from the difference between total and delayed areas.

Figure 3: Detector response after a large number of pulse insertions
Slope fit method

This method also known as the prompt neutron decay fit method can be applied when we neglect the delayed neutrons. The prompt neutron decay constant is given as

$$\alpha = \frac{\rho - \beta_{\text{eff}}}{\Lambda} \quad \text{(2)}$$

The same PNS histogram as for area ratio method can be used. $\beta_{\text{eff}}$ and $\Lambda$ can be obtained from simulations and reactivity $\rho$ hence $k_{\text{eff}}$ can be calculated.

Source-jerk method

In this method the subcritical system is operated at a certain fixed power and suddenly switched off. If the neutron flux level with the source was $n_0$ and after the source is switched off becomes $n_1$ (which consists only delayed part of neutron flux) then the reactivity is given as

$$\frac{\rho}{\beta} (\$$) = \frac{n_1 - n_0}{n_1} \quad \text{(3)}$$

Bell and Glasstone spatial correction factor

The reactivity measured using area ratio method is spatially dependent and gives a measure of local reactivity as it depends on relative position of detector and source. In order to infer the true (global) reactivity we need a correction factor which has to be evaluated for each detector position [6]. The spatial correction factor for area ratio method, as suggested by Bell and Glasstone, is given as

$$f = \left( \frac{\rho_{\text{cri}}}{\rho_{\text{src}}} \right) = \left( \frac{A_d}{A_p} \right) \left( \frac{1}{\beta_{\text{eff}}} \right) \left( \frac{1 - k_{\text{eff}}}{k_{\text{eff}}} \right) \quad \text{(4)}$$

In the above Eq., $A_p$ is the prompt area, $A_d$ is the delayed area, $\rho_{\text{cri}}$ is the reactivity calculated by computer codes in criticality mode and $\rho_{\text{src}}$ is the reactivity calculated by computer codes in source mode with the external neutron source. In area ratio method $\rho_{\text{src}}$ (in $$) is calculated as the ratio of prompt area to delayed area. For evaluating $\rho_{\text{src}}$ the Static approach (time independent method) is used in which two separate simulations are done with steady-state external neutron source, one with total neutrons and other with delayed neutrons suppressed. Detector reaction rates determine the total area and prompt area respectively. Now we know that prompt area has contributions both from source as well as fission neutrons while delayed area has only fission neutron contribution, hence moving closer to the source will result in increased prompt area and hence lower value of $f$.

Experimental results

Reactivity is measured at the centre of the axial channels (EC1, EC2 and EC3) using three different pulsed neutron source methods, namely – area ratio, source jerk and slope fit. For measurements, miniature $^3$He detector (Active length: 70 mm; Diameter: 6.2 mm; Sensitivity: 0.4 cps/\text{nV}) is used [7]. The experiments are carried out with D-T neutron source at neutron yield of $1 \times 10^{10}$ n/s. The neutron generator was used
in pulsed mode with time period (T) of 10 ms and pulse width (ΔT) of 10µs. The effective delayed neutron fraction \( \beta_{\text{eff}} \) has been calculated theoretically and its value obtained is 704±10pcm, which has been used in determination of the reactivity using area ratio method. The results have been tabulated in Table 1 and Table 2. We observe quite clearly that the measured reactivity value using area ratio method and source-jerk method are dependent on the relative detector and source locations. To remove this dependence and obtain the global reactivity we need a spatial correction factor which is obtained from simulations. We further conclude that after applying this correction factor the values are almost same for all positions in the core. Table 3 shows the results for slope fit method.

### Table 1: Reactivity results using area ratio method

<table>
<thead>
<tr>
<th>Detector location</th>
<th>( \rho^{\text{meas}} ) (( $ ))</th>
<th>Spatial correction factor ( f )</th>
<th>( \rho^{\text{corr}} ) (( $ ))</th>
<th>( k_{\text{eff}}^{\text{corr}} )</th>
</tr>
</thead>
<tbody>
<tr>
<td>EC1</td>
<td>-28.03</td>
<td>0.62 ± 0.02</td>
<td>-17.38 ± 0.56</td>
<td>0.888 ± 0.003</td>
</tr>
<tr>
<td>EC2</td>
<td>-19.04</td>
<td>0.92 ± 0.03</td>
<td>-17.52 ± 0.57</td>
<td>0.885 ± 0.003</td>
</tr>
<tr>
<td>EC3</td>
<td>-16.61</td>
<td>0.99 ± 0.04</td>
<td>-16.44 ± 0.66</td>
<td>0.891 ± 0.004</td>
</tr>
</tbody>
</table>

### Table 2: Reactivity results using source-jerk method

<table>
<thead>
<tr>
<th>Detector location</th>
<th>( \rho^{\text{meas}} ) (( $ ))</th>
<th>Spatial correction factor ( f )</th>
<th>( \rho^{\text{corr}} ) (( $ ))</th>
<th>( k_{\text{eff}}^{\text{corr}} )</th>
</tr>
</thead>
<tbody>
<tr>
<td>EC1</td>
<td>-25.87</td>
<td>0.62 ± 0.02</td>
<td>-16.04 ± 0.51</td>
<td>0.894 ± 0.003</td>
</tr>
<tr>
<td>EC2</td>
<td>-17.39</td>
<td>0.92 ± 0.03</td>
<td>-16.00 ± 0.52</td>
<td>0.894 ± 0.003</td>
</tr>
<tr>
<td>EC3</td>
<td>-15.69</td>
<td>0.99 ± 0.04</td>
<td>-15.53 ± 0.62</td>
<td>0.897 ± 0.004</td>
</tr>
</tbody>
</table>

### Table 3: Reactivity results using slope fit method

<table>
<thead>
<tr>
<th>Experimental channel</th>
<th>( \alpha ) (ms(^{-2}))</th>
<th>( k_{\text{eff}} )</th>
</tr>
</thead>
<tbody>
<tr>
<td>EC1</td>
<td>-2.246 ± 0.112</td>
<td>0.887 ± 0.005</td>
</tr>
<tr>
<td>EC2</td>
<td>-2.318 ± 0.116</td>
<td>0.884 ± 0.005</td>
</tr>
<tr>
<td>EC3</td>
<td>-2.346 ± 0.117</td>
<td>0.883 ± 0.005</td>
</tr>
</tbody>
</table>

### Evaluation of spatial correction factor

The values of reactivity obtained from area ratio method were spatially dependent so in order to estimate the global reactivity we need to evaluate Bell Glasstone spatial correction factor at different detector locations. Simulations were performed using Monte Carlo technique to evaluate this factor. Separate simulations were done to evaluate the total and delayed areas. Prompt area was obtained as the difference of the two areas. For evaluating spatial correction factor in experimental channels (EC1 – EC7) miniature \(^3\)He detectors (Active length: 70 mm; Diameter: 6.2 mm) were simulated [8]. It can be observed that correction factor is lowest in the EC1 channel as it is closest to the source, hence maximum contribution of source neutrons to prompt area. The importance of this correction factor comes from the fact that applying this factor to the measured reactivity makes it spatially independent.

### Mean generation time

The mean generation time \( \Lambda \) used in Eq. 2 can be computed theoretically from prompt neutron lifetime. However, we can also use the PNS experiments to measure mean generation time. Eq. 1 and Eq. 2 may be combined to form the following expression:
The experimentally measured value of $\Lambda$ is $5.971 \times 10^{-5}$ s whereas the theoretically computed value is $8.38 \times 10^{-5}$ s.

Conclusions

Reactivity estimation has been done in the axial experimental channels of the BRAHMMMA subcritical assembly using area ratio, source jerk and slope fit method. The reactivity values computed from slope fit method do not suffer spatial dependence and are almost same at all locations. However the values obtained by other two methods viz. Area ratio and source jerk are not consistent throughout the core. They show spatial dependence and that needs to be corrected. For this we evaluated the spatial correction factor using Monte Carlo simulations. It is clearly visible that reactivity obtained after correction is nearly same at all positions. The post correction reactivity values obtained from area ratio method give a $k_{eff}$ value close to theoretical value. The prompt neutron decay constant obtained from slope fit method is also close to theoretical value further the mean generation time estimated theoretically and experimentally are consistent, which further validates the PNS techniques used.

References


Session 6: ADS data and simulations

Chair: L. Yang
Institute of Modern Physics, Chinese Academy of Sciences, China
The outcome of the post-irradiation examination of the Megawatt pilot experiment (MEGAPIE) target

Yong Dai, Michael Wohlmuther, Dorothea Schumann
Paul Scherrer Institut, Switzerland

Abstract
Megawatt pilot experiment (MEGAPIE) was successfully performed in 2006 at the Paul Scherrer Institut, under collaboration with nine international institutions. One of the important goals of MEGAPIE is to understand the behaviour of structural materials of the target components exposed to high fluxes of high-energy protons and spallation neutrons in flowing LBE (liquid lead-bismuth eutectic) environment by conducting post-irradiation examination (PIE). The PIE includes four major parts: non-destructive test, radiochemical analysis of production and distribution of radionuclides produced by spallation reaction in LBE, analysis of LBE corrosion effects on structural materials, T91 and SS 316L steels, and mechanical testing of the T91 and SS 316L steels irradiated in the lower part of the target.

The non-destructive test (NDT) including visual inspection and ultrasonic measurement was performed in the proton beam window area of the T91 calotte of the LBE container, the most intensively irradiated part of the MEGAPIE target to inspect the possible obvious failure and corrosion effect. Radiochemical analysis of radionuclides produced by spallation reaction in LBE is to improve the understanding of the production and distribution of radionuclides in the target. Detailed gamma and alpha measurements and chemical analysis were conducted on the LBE samples extracted from different positions of the target. The surface change on components induced by LBE corrosion under intensive irradiation was investigated by metallography, scanning electron microscopy etc. surface inspection techniques. The mechanical testing including tensile and bend tests was carried out on the specimens extracted from the T91 and SS 316L components in the intensive irradiation region, in order to detect the change in mechanical properties of the T91 and SS 316L steels after irradiation.

The experimental details and the results will be presented. The implications of the outcome of the PIE to the ADS target R&D will be discussed.
Release of volatile radionuclides from ADS and their capture

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\textsuperscript{d}Department of Materials Science and Engineering, University of Gent, Belgium

Abstract

The release of volatile radionuclides from the liquid-metal coolant of lead-bismuth eutectic (LBE) cooled ADS constitutes a significant safety issue for such facilities. For safety analyses the release behaviour must be well understood, and furthermore methods for capturing the released radioactivity are required. However, reliable physicochemical data on both evaporation and deposition processes are lacking, especially concerning the highly radiotoxic $\alpha$-emitting volatile element polonium. In order to obtain a better understanding of the processes underlying both the evaporation of volatiles from LBE and their capture from the gas phase and to close the gap in fundamental physicochemical data, a work package (WP6) dedicated to these topics was included in the project SEARCH, which was launched in 2011 within the 7\textsuperscript{th} framework programme of the European Commission in support of the development of the MYRRHA reactor. The work programme of SEARCH WP6 consisted of experiments studying the evaporation of polonium and mercury from LBE and the deposition of polonium on various surfaces under variation of moisture and redox potential in the gas phase. In parallel, the interaction of polonium with the LBE matrix and noble metal surfaces as well as the stability of small polonium-containing molecules were studied by quantum mechanical calculations within the framework of density functional theory. In this presentation, we will give a summary of the outcome of WP6 of the SEARCH project.
Neutronic and thermal-hydraulics coupling analysis for fast reactors: A developed computational tool applied to MYRRHA concept

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1Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas
2Empresarios Agrupados, Spain
3SCK•CEN, Belgium

Abstract

The increased capacity of the computational resources during the last years allows a better and detailed simulation of complex systems whatever the field of application. Nuclear reactors design activities take advantage of this computational capability, making possible the implementation of Monte Carlo and/or deterministic techniques for the resolution of the physics involved in the required calculations, achieving a suitable accuracy and spatial resolution in the results. In most cases, this requires high demanding computational calculations.

The computational tool developed and presented here links the neutronic and thermal-hydraulic disciplines taking into account the feedback existing when one affects other by means of reactivity-power and temperature-density as main connection factors between them. The used methodology is based on the data exchange between specific codes for each stage of the coupling through a driver programme – far from a comprehensive multi-physics code. The codes selected so far are MCNP5 and COBRA-IV, references on each application field.

The applications of the tool cover both steady state and transient analysis at full core or single fuel assembly scale. Expansion effects due to temperature changes from nominal conditions are also considered.

Results for an unprotected loss of flow (ULOF) for MYRRHA concept will be presented to test the simulation capabilities of the developed tool.

Introduction

The capabilities of prediction of the computational tools are essential from the point of view of their development when taking into account all relevant phenomena for nuclear reactor design through different codes coupling. The traditional approach for code coupling in nuclear engineering is based on loose coupling where individual validated codes perform their respective calculation and exchange information at specified time points [1,2].

Neutronics and thermal hydraulics are very closely related disciplines in nuclear reactors since one affects each other with power and temperature as main connection factors between them. Unexpected
reactivity variations (regardless the starting event) lead to temperature changes that imply changes in the behaviour of key functional materials in the core, potentially leading to exceeding failure limits. This mentioned feedback plays a key role on the design of nuclear reactors from the safety point of view, motivating the development of a computational tool able to deal with these coupled phenomena.

The tool described in this paper is applicable to both steady state and transient calculations for full core and individual fuel assembly scale. The coupling scheme follows the traditional approach of loose coupling where the information exchange is performed via shared file or message passing. The implementation is sufficiently generic so any type of reactor which can be handled by the involved codes – with proper databases – can be simulated with the tool.

**Coupling tool description**

The methodology followed in the coupling process is described in this section together with its practical implementation. The tool is programmed in C++ language, serving as driver program between executions of the codes selected for each stage of the calculations and performing the data exchange.

**Coupling scheme**

The traditional approach for code coupling in nuclear engineering is based on loose coupling where individual validated codes perform their respective calculation and exchange information at specified time points. With this coupling scheme, the time approximation order of each isolated code is masked by the time approximation order of the coupling approach. Loss of accuracy is a main drawback. However this approach is the state-of-the-art coupling methodology to perform multi-physics studies in nuclear reactor design and safety analyses because it allows the use of verified and validated codes for each stage of the problem without major modifications of the source code. System codes RELAP5 [3] or TRACE [4] serve as good examples.

The use of different codes to solve each field of study is equivalent to solve a set of partial differential equations with the operator splitting method [5]. With this technique, two validated codes can be coupled in a straightforward way, each one resolving its corresponding stage of the calculation. Because the coupling is carried out with two separated codes, only an explicit or semi-implicit scheme for time discretisation can be used without modifying the original codes, therefore leading to restrictions in the selection of the time step in order to avoid accuracy issues.

The method described here [6] uses a Monte Carlo code for simulating the neutron transport through matter with a given temperature and density profile. The neutron flux obtained with this simulation is then used to calculate the system power map. On the other hand, the thermal-hydraulic code calculates the temperature distribution in each sub-channel with the power map given by the neutronic stage. The two codes are coupled via data exchange by means of the driver program.

**Codes**

For the neutronics stage, the MCNP5 v1.60 from 2010 [7] is selected, which includes features as adjoint weighed tallies for point kinetics parameters calculations as delayed neutron fraction and neutron generation time. MCNP is a general purpose, continuous energy, generalised geometry, time-dependent, coupled neutron/photon/electron Monte Carlo transport code giving the possibility to work in several transport modes, being the interesting ones for this application those involving neutron interactions. MCNP can also be used to compute the effective multiplication factor and the fundamental mode eigenfunction in a critical system. A highlighted capability is also the possibility to simulate complex heterogeneous geometries – although demanding significant computational time.
For the thermal-hydraulics stage COBRA-IV-I [8] code has been selected. It is an extended version of COBRA-IIIC developed at the Pacific Northwest Laboratory. It is a sub-channel code which computes the flow and enthalpy — among others — distributions in rod bundles nuclear fuel elements and cores for both steady state and transient conditions. It can be used for water, liquid metal and gas cooled reactors. In a sub-channel code, the fuel bundle or core of a reactor is divided into computational cells. The balance laws of mass, energy and momentum for the fluid are solved for each cell where the independent variables of enthalpy, pressure and velocity are appropriate averages. The use of the computational cell concept allows for sub-channel analysis, core analysis and general flow field analysis to be considered in a unified approach.

In addition, the used version of COBRA has been updated to include the Ushakov correlation for wall to fluid convective fluid transfer in vertical pin bundles with liquid metals by means of the Nusselt number formulation proposed by [9]. On the other hand, the pressure drop model for wired rod bundles has been also updated by including Rehme correlation for the corresponding friction factor calculation [9]. The fluid properties for LBE and Na can be included from [11] and [12], respectively.

**Practical implementation**

The present description of the coupling code is focused on transient simulations, starting from a reactor in steady-state condition [2].

The coupled set of partial differential equations is solved through the conventional operator splitting technique, using a mono-physics code without any modification to solve each stage of the simulation. The coupling approach uses a staggered time mesh scheme to move forward in time the solution, where one physics component is solved first, passes data to the other component and then it is solved. The order of the numerical method used by each code to solve the partial set of differential equations is masked by the order of the coupling scheme. To obtain a high-order solution of the coupled problem an adaptive time-step algorithm should be applied [1].

A key idea in the transient treatment is the flux factorisation approach into shape and amplitude functions, assuming that the flux shape has weaker time dependence than the amplitude function. This allows for the use of bigger time steps to solve the shape equations than for the point kinetics equations used to calculate the amplitude. This is a great save in terms of execution time because in the shape equation the spatial, angular and energetic dependence of the neutron flux has to be found and it is more difficult to solve than the point kinetics equations.

With the adiabatic approach (from neutronics point of view [6]), MCNP is used to calculate the flux shape and also the reactivity passed to the point kinetics equations, assuming that prompt and delayed neutron source are in equilibrium. Although in a transient the flux shape remains almost constant along a short period, the reactivity may change significantly in a fraction of this period. Therefore the selection of the time step is governed by the time step required for an accurate reactivity estimation but not by the flux shape estimation.
The temporal scheme for the coupled transient calculation is as follows (see flowchart in Figure 1). The simulation starts from a converged steady-state coupled solution [13,2] and with kinetics parameters calculated in a previous separate simulation. Using the thermal power at steady state, the thermal-hydraulic solution is advanced until half of the time step. The results are used to update the MCNP input and calculate the reactivity that takes into account the thermal-hydraulic feedback (fuel temperature, coolant density and thermomechanical expansion). Then the point kinetic equation solver (the numerical method RADAU-IIA has been chosen because it is fully implicit and stable, allowing the use of larger internal time steps than a simple implicit Euler method [13]) calculates the power at the end of the time step. The reactivity calculated by MCNP is included in the point kinetic equations and kept constant for the entire time step. This power is used to normalise the fission reaction rates and to obtain the 3D power distribution in the core. For the new time step, the power distribution calculated at the end of the time step (dt) is used in the thermal-hydraulic transient for (dt/2,3dt/2) interval. The restart option of COBRA is employed to start the new thermal-hydraulics transient with the results of the previous calculation. This staggered coupling algorithm preserves the energy in the thermal-hydraulics simulation, because the time integral of the power (energy) with a linear variation is the same than the integral with a constant power value at half of the time step. The time constant that governs the change in thermal-hydraulic parameters depends on the fuel conductivity and the gap conductance (~0.1 seconds).

For the cross section updating due to changes in temperature, the pseudo-materials approach is employed. It consists in a weighted combination of a nuclide cross section between two temperatures.
The advantage of this is that it is not necessary to create new cross section data sets for each temperature, depending the accuracy on the interpolation interval [2].

In addition, thermal expansion effects are considered in the geometry of the model used in MCNP through the temperature changes provided by the thermal-hydraulics feedback, applying the linear expansion coefficient for each material. The materials that can be modified are fuel, cladding and diagrid.

Application to MYRRHA reactor

The MYRRHA facility is an initiative of the Belgian SCK-CEN centre [14], also supported in the European Sustainable Nuclear Industrial Initiative (ESNII) organisation as a part of the Sustainable Nuclear European Technological Platform (SNE-TP).

Figure 2: MYRRHA reactor layout and main characteristics

<table>
<thead>
<tr>
<th>Main characteristics</th>
<th>MYRRHA</th>
<th>Unit</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core power</td>
<td>100</td>
<td>MWth</td>
</tr>
<tr>
<td>Active core avg. power density</td>
<td>250</td>
<td>W/cm²</td>
</tr>
<tr>
<td>Acc. energy</td>
<td>600</td>
<td>MeV</td>
</tr>
<tr>
<td>Inlet temp.</td>
<td>270</td>
<td>°C</td>
</tr>
<tr>
<td>Coolant delta T</td>
<td>~140</td>
<td>°C</td>
</tr>
<tr>
<td>Max. core velocity</td>
<td>1.9</td>
<td>m/sec</td>
</tr>
<tr>
<td>Max. linear power</td>
<td>370</td>
<td>W/cm</td>
</tr>
</tbody>
</table>

It is designed as a flexible fast-spectrum irradiation facility so that a fast neutron spectrum is present at every location in the reactor core and that in selected fuel assembly position a dedicated experimental fuel assembly can be loaded. It is conceived as a pool-type reactor coupled with an accelerator-driven system (ADS), being able to operate in both sub-critical and critical modes, containing a proton accelerator of 600 MeV impinging on a spallation target and a multiplying medium with MOX fuel cooled by liquid Lead-Bismuth Eutectic (LBE). A schematic view of the reactor is shown in Figure 2 together with its main characteristics [14].

Simulations for a 100 MW critical core

For the application of the developed tool, a 108 fuel assemblies critical core; has been modelled. The fuel is a MOX with 30% Pu content (including a little Am content generated by Pu decay while in storage) in each fuel assembly. An unprotected loss of flow (ULOF) is simulated from a steady-state condition assuming a linear decrease down to 14% of the nominal flow rate in 15 seconds.
The fuel assemblies have been grouped into rings – for computational reasons – and the active axial length divided in six identical parts.

Figure 3 and 4 show preliminary results for the power and reactivity evolution during the first minute of the transient. Up to the first 20 seconds, the power decreases down to ~55% of the nominal value and from then a slowly decreasing tendency to a new lower power level is envisaged. In the case of reactivity, it reaches a minimum value of ~ 88 pcm at 20 seconds and from then a quest for a new steady-state condition (zero reactivity) is envisaged, describing an asymmetric well. A broaden time scale simulation would be necessary to confirm the final tendency of both quantities. These results only include the thermal expansion of cladding (axial and radial) and fuel (axial).

**Figure 3: Power results for ULOF simulation**

![Power results for ULOF simulation](image)

**Figure 4: Reactivity results for ULOF simulation**

![Reactivity results for ULOF simulation](image)
The fluctuations observed in the plots come from the number of histories used in the MCNP simulations, motivating the use of two different values to check the influence of this parameter into the results (“low stat.” and “improved stat.” plotted with thin and dashed lines, respectively). A fitted curve for the lower statistic case has been also plotted in order to describe a mean behaviour, labelled as “low stat. fitted”. The same tendency in the results is reproduced in both cases, reducing the fluctuation in the case of greater number of histories. In addition, the reduction of the time step compared with that used here (1 second) is planned to check its influence in the fluctuations of the studied quantities.

Further refinement to these results includes sensibility calculations of gap thickness and gap conductivity, expansion effect in other core components – such the diagrid in order to capture the flowering effect- and different fuel composition in selected fuel assemblies.

Conclusions

A computational tool able to simulate transients in nuclear reactor cores applying the adiabatic approach has been developed. It couples MCNP and COBRA codes by mean of a C++ driver programme and can handle any geometry that can be analysed individually with each mentioned code.

Main characteristics of the tool are the data exchange between codes as procedure to link the involved phenomena and the flux factorisation for an efficient treatment of the problem. Minor updating to COBRA to deal with liquid-metal cooled reactors calculations and the cross section management for MCNP are also outstanding features of the tool.

The application has been performed for an ULOF in a critical configuration for MYRRHA reactor, giving the evolution of power and reactivity during the first minute of the transient. Main results shows that reactor stabilises to a lower power level while reactivity decreases until coolant flow rate reaches its minimum value and from then shows a continuous stepping to “zero reactivity” condition – that is, a new steady-state condition is reached.

The fluctuations observed in the curves has been identified to be caused by the number of histories used in the MCNP calculations, expecting that size of the time step employed in the calculation will also have influence – reduction of the time step is part of future calculations.

Calculations with alternative set of data are planned as future work, such as gap thickness and conductivity and the inclusion of expansion effects of the diagrid.

References


Nuclear design and safety evaluation systems SuperMC

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Abstract

The Monte Carlo (MC) methods have been broadly adopted in nuclear design and analysis of advanced nuclear energy systems due to the great complexity and high requirements on the precision of design and safety margin. However, there are still great challenges in the current MC methods including the calculation modelling of complex geometries, simulation of deep penetration problem in radiation shielding, slow convergence of complex calculation, lack of experimental validation for new physical features, etc.

SuperMC is a general, intelligent accurate and precise simulation software for the nuclear design and safety evaluation of nuclear systems. It is designed to perform the comprehensive neutronics calculation, taking the radiation transport as the core and including the depletion, radiation source term/dose/biohazard, material activation and transmutation, etc. It supports the multi-physics coupling calculation including thermo-hydraulics, structural mechanics, biology, chemistry, etc. The main technical features are hybrid MC-deterministic methods and the adoption of advanced information technologies. The main usability features are automatic modelling of geometry and physics, visualisation and virtual simulation and cloud computing services. The latest version of SuperMC can accomplish the transport calculation of \( n \), \( \gamma \) and can be applied for criticality and shielding design of reactors, medical physics analysis, etc.

SuperMC has been verified by more than 2000 benchmark models and experiments, such as the International Criticality Safety Benchmark Evaluation Project (ICSBEP), Shielding Integral Benchmark Archive Database (SINBAD) and the comprehensive applications from the reactors including the fusion reactor (ITER models, FDS-II), fast reactor (IAEA-BN600, IAEA-ADS), PWR (BEAVRS, HM, TCA) and cases from the International Reactor Physics handbook Evaluation Programme (IRPhEP) and International Reactor Physics Experiment Revaluation Project (IRPhEP), etc. The benchmarking results have been compared with MCNP, demonstrating higher accuracy and calculation efficiency of SuperMC, and also significant enhancement of work efficiency due to its functions of automatic modelling and visualised analysis. As the supplementary of validation experiments of MC software for advanced nuclear energy systems applications, experiment for deep penetration problem in radiation shielding and neutronics integral experiment of fusion blanket are being particularly conducted using high-intensity D-T fusion neutron generator (HINEG) which will produce 14.1MeV neutrons.

SuperMC has been widely used in more than 50 nations and 23 major nuclear engineering projects. It has been passed the international benchmarking activity organised by ITER IO and selected for
ITER neutronics model building. SuperMC has been applied in the nuclear design and analysis Strategy Priority Research Programme of Chinese Academy of Sciences named “Advanced Nuclear Fission Energy-ADS Transmutation System”.

**Keywords:** Monte Carlo, transport, modelling, visualisation
A new hybrid fast-slow ADS for research and applications

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Abstract

We report on the studies of an irradiation facility based on an accelerator-driven subcritical nuclear research reactor, which can simultaneously provide a fast flux in the core and a thermal flux in the reflector, that we will call a hybrid fast-slow ADS. The conceptual design presented here, inspired by [1], starts from a 432 kW ($k_{\text{eff}}=0.967$) ADS composed by 110 solid lead fuel assemblies each with size $9.7 \times 9.7 \times 87$ cm$^3$, filled with 81 MOX pins (16.5% Pu+Am) of 0.357 cm radius and surrounded by a 0.068 cm thick AISI steel cladding. Source neutrons are produced by a 70 MeV 1 mA proton beam impinging on a beryllium target ($\sim 8 \times 10^{14}$ n/sec) [2]. The core is cooled by helium flowing in very thin pipes, 0.25 cm in diameter and is surrounded by a 80 cm lead reflector. Core and reflector are contained within a 2 cm steel vessel. The hybrid version ($k_{\text{eff}} 0.972 = P=527$ kW) is instead composed by 59 fuel assemblies, each hosting 81 MOX pins (22% Pu+Am) where:

- The lead reflector has been replaced by three concentric layers, the first of 35 cm lead, followed by 50 cm graphite and finally 10 cm lead.
- In the cooling system water flows in wider pipes, 0.5 cm in diameter, which allows to increase $k_{\text{eff}}$ while maintaining the fast character of the spectrum.

We simulated the neutron flux in three core positions (internal, intermediate and external) and in two graphite reflector positions (internal, intermediate), finding that the flux is still mostly fast in the core, while it exhibits a strong thermal component in the reflector, as shown in the following table.

This work is partially supported by the 7\textsuperscript{th} Framework Programmes of the European Commission (Euratom) through the CHANDA contract FP7-Fission-2013-605203.
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<th>Location</th>
<th>$\Phi_{\text{core}}$ (tot) ($n \text{ cm}^{-2} \text{s}^{-1}$)</th>
<th>$\Phi_{\text{core}}$ (&gt;1keV) %</th>
<th>$\Phi_{\text{core}}$ (&gt;500keV) %</th>
<th>$\Phi_{\text{ref}}$ (tot) ($n \text{ cm}^{-2} \text{s}^{-1}$)</th>
<th>$\Phi_{\text{ref}}$ (&lt;1eV) %</th>
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**Figure 1:** Radial power distribution in the core

**Figure 2:** Neutron flux in different core positions as a function of energy
Figure 3: Neutron flux in the intermediate reflector position
### List of participants

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<tr>
<th>Name</th>
<th>Institution</th>
<th>Country</th>
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