ROMANIAN NATIONAL COMMITTEE
OF WORLD ENERGY COUNCIL

THE GENERAL ASSOCIATION
OF THE ENGINEERS IN ROMANIA

EMERG

Energy ● Environment ● Efficiency ● Resources ● Globalization

Quarterly publication of Romanian National Committee
of World Energy Council (WEC/RNC)
and
The General Association of Engineers in Romania (AGIR)

ISSN 2668-7003 Year V / 2019
ISSN-L 2457-5011

www.cnr-cme.ro/publicatii/emerg
www.emerg.ro

EMERG

Energy ● Environment ● Efficiency ● Resources ● Globalization

Quarterly publication of Romanian National Committee
of World Energy Council (WEC/RNC) and
The General Association of Engineers in Romania (AGIR)

EDITORIAL BOARD

ROMANIAN NATIONAL COMMITTEE
OF WORLD ENERGY COUNCIL

THE GENERAL ASSOCIATION
OF THE ENGINEERS IN ROMANIA

EMERG 12

Energy • Environment • Efficiency • Resources • Globalization

Quarterly publication of WEC/RNC and AGIR

ISSN 2668-7003 Year V / 2019
ISSN-L 2457-5011

www.emerg.ro
www.cnr-cme.ro/publicatii/emerg

Papers' authors:

Minodora APOSTOL
Veronica ANDREI
Dorel BENGA
Marin CONSTANTIN
Daniela DIACONU
Raluca FAKO
Raluca GHERASIM
Daniela GUGIU
Alexandra ION
Virgil IONESCU
Viorel IONESCU
Dragos IONESCU
Andrea IVAN
Adrian JELEV
Sorin MEGLEA
Iulian NIȚĂ
Mirela NITOI
Alexandru NITU
Valentin OLARU
Adriana OLTEANU
Rodica PANCEF
Irina PARASCHIV
Constantin PAUNOIU
Daniel PETRESCU
Alexandru TOMA
Ilie TURCU
Lucian VELCIU
Diana VILCU
Florea VOICU

AGIR Publishing House
Bucharest, 2019

Romanian National Committee of World Energy Council (WEC/RNC)
and
The General Association of Engineers in Romania (AGIR)

WEC/RNC
B-dul Lacul Tei, nr. 1-3, Bucureşti, Sector 2, 020371, +40372-821-475
+40372-821-476, secretariat@cnr-cme.ro; www.cnr-cme.ro

AGIR
Calea Victoriei nr. 118, etaj 1, sector 1 Bucuresti, tel. 021 3168993, 021 3168994, fax. 021 3125531, e-mail: office@agir.ro; www.agir.ro

COMITETUL NAȚIONAL ROMÂN
AL CONSILIULUI MONDIAL AL ENERGIEI

ASOCIAȚIA GENERALĂ
A INGINERILOR DIN ROMÂNIA

CUPRINS

Situaţia mondială actuală a combustibilului nuclear pe bază de uraniu natural (*Raluca GHERASIM , Virgil IONESCU , Sorin MEGLEA*) 7

Producţia de apă grea în România: Declinul unei industrii & debutul unor noi orizonturi (*Raluca FAKO , Virgil IONESCU , Adriana OLTEANU , Sorin MEGLEA*) 19

Cerinte de mediu ce trebuie considerate la planificarea programului depozitului geologic national (*Veronica ANDREI , Adrian JELEV , Diana VILCU*) 26

Teste de ciclaj pe tuburi tip SEU-43 din Zircaloy-4 la 350°C (*Valentin OLARU , Viorel IONESCU , Dragos IONESCU , Alexandru NITU , Alexandra ION , Florea VOICU*) 34

Dezvoltarea tehnologiei reactoarelor cu neutroni rapizi și a demonstratorului ALFRED în contextul viitoarelor piețe de energie (*Marin CONSTANTIN , Ilie TURCU , Constantin PAUNOIU , Alexandru TOMA , Daniela DIACONU , Minodora APOSTOL , Mirela NITOI , Daniela GUGIU*) 44

Dimensionarea principalelor componente din circuitul secundar al reactorului ALFRED (*Iulian NIŢĂ , Rodica PANCEF , Andrea IVAN*) 57

Curățarea chimică a plumbului solidificat de pe oțelul austenitic 316 candidat pentru reactorii rapizi răciți cu plumb (*Daniel PETRESCU , Lucian VELCIU*)..... 64

Controlul oxigenului în sistemele răcite cu plumb, prin metoda fazei solide – teste preliminarii pentru dezvoltarea schimbătorilor de masă din PbO (*Irina PARASCHIV , Dorel BENGA*)..... 76

ROMANIAN NATIONAL COMMITTEE
OF WORLD ENERGY COUNCIL

THE GENERAL ASSOCIATION
OF THE ENGINEERS IN ROMANIA

CONTENTS

Current international case for natural uranium-based nuclear fuel (*Raluca GHERASIM , Virgil IONESCU , Sorin MEGLEA*) 7

Heavy water production in Rromania: Twilight of an industry & dawn of new perspectives (*Raluca FAKO , Virgil IONESCU , Adriana OLTEANU , Sorin MEGLEA*)..... 19

Environmental requirements to be considered for planning the national geological disposal program (*Veronica ANDREI , Adrian JELEV , Diana VILCU*) 26

Fatigue tests on SEU-43 Zircaloy-4 tubes at 350°C (*Valentin OLARU , Viorel IONESCU , Dragos IONESCU , Alexandru NITU , Alexandra ION , Florea VOICU*)..... 34

Lead fast reactor technology and ALFRED demonstrator development in the context of future markets of energy (*Marin CONSTANTIN , Ilie TURCU , Constantin PAUNOIU , Alexandru TOMA , Daniela DIACONU , Minodora APOSTOL , Mirela NITOI , Daniela GUGIU*) 44

Designing of main components of the secondary circuit of ALFRED reactor (*Iulian NIŢĂ , Rodica PANCEF , Andrea IVAN*)..... 57

Chemical cleaning of solidified lead from 316 austenitic steel a structure candidate material for lead-cooled fast reactors (*Daniel PETRESCU , Lucian VELCIU*)..... 64

Oxygen control in lead cooled systems via solid phase – preliminary tests for developing PbO mass echangers (*Irina PARASCHIV , Dorel BENGA*)..... 76

CURRENT INTERNATIONAL CASE FOR NATURAL URANIUM-BASED NUCLEAR FUEL

SITUAŢIA MONDIALĂ ACTUALĂ A COMBUSTIBILULUI NUCLEAR PE BAZĂ DE URANIU NATURAL

Raluca GHERASIM[1], Virgil IONESCU[2], Sorin MEGLEA[3]

Abstract: *Worldwide, there is a permanent concern to maintain in operation all nuclear power plants that contribute to each state's nuclear power production independence and, at the same time, limiting the dependence on imported energetic resources.*

Thus, a distinctive importance should be given towards energetic security in the context of increased instability, reflecting ESA's best practice recommendations of the states of developing and maintaining uranium stocks of different types, according to the kind of the nuclear fuel used, sufficient for an approximately 1-5 years of reactor functioning.

The present paper aims a review of the technological evolution of Nuclear Reactors and their influence on the demand for nuclear fuel, an analysis of the major suppliers on the market for the production and supply of natural uranium-based nuclear fuel, and also an overview of the supply price evolution for it.

Keywords: Nuclear Power Plants, nuclear fuel, natural uranium, security of supply.

Rezumat: *La nivel mondial, există o preocupare permanentă pentru menţinerea în funcţiune a tuturor centralelor nucleare care contribuie la independenţa producţiei de energie electrică a fiecărui stat şi, în acelaşi timp, la limitarea dependenţei de resursele energetice importate.*

Astfel, trebuie acordata o importanta deosebita asigurarii securitatii energetice în contextul instabilităţii crescute, reflectând recomandările ESA privind bunele practici ale statelor membre de a constitui şi menţine stocuri de uraniu sub diferite forme, în funcţie de tipul de combustibil nuclear utilizat, suficiente pentru o perioadă de consum pe reactor de aproximativ 1 ÷ 5 ani.

Lucrarea de faţă îşi propune o trecere în revistă a evoluţiei tehnologice a reactoarelor nucleare şi a influenţei lor asupra cererii de combustibil nuclear, o analiză a principalilor furnizori pe piaţa mondială pentru producerea şi furnizarea

[1] RATEN – CITON, e-mail: gherasimr@router.citon.ro
[2] RATEN – CITON, e-mail: ionescuv@router.citon.ro
[3] RATEN – CITON, e-mail: megleas@router.citon.ro

combustibilului nuclear pe bază de uraniu natural, dar, de asemenea, şi o prezentare generală a evoluţiei preţurilor de aprovizionare pentru combustibilul nuclear pe bază de uraniu natural.

Cuvinte cheie: centrale nucleare, combustibil nuclear, uraniu natural, securitate.

1. Introduction

At global level, among the governmental entities of the developed states, there is a permanent concern for maintaining the operational status of all the nuclear objectives which are strategic components of the nuclear fuel cycle and which contribute to the maintenance of the independence of each state in nuclear energy generation and, at the same time, to limit the dependence of imported energy resources According to the recommendations of the Euratom Supply Agency (Euratom), in order to choose the best alternative for nuclear fuel supply, each state must assess its own risk profile and, depending on the risk, use the appropriate ways which lead to risk mitigation and ensure the best energy security, each country taking the decision that it considers the best.

ESA's experience shows that some states, on the basis of their own assessments, have made the decision to set up safety reserve stocks for periods ranging from 1 ÷ 3 years to ensure the security of nuclear fuel supplying for nuclear plants.

The manufacture of nuclear fuel is carried out in specific installations, which make up the start-up part of the so-called nuclear fuel cycle – NFC (Nuclear Fuel Cycle), including nuclear reactors (research and nuclear power plants) and radioactive waste and used nuclear fuel storage facilities.

Nuclear fuel production facilities form those Front End Facilities that open the cycle and which are classified as non-reactor nuclear installations /. From the point of view of the importance of these objectives, considering them as a priority, concerns a series of economic and energy policy justifications and is also related to the anticipation, conception, organization and management of processes characterized by a significant quantity and quality impact , on economy, technology, education, as well as on the population in the adjacent areas to the site and on the environmental factors.

The main characteristics of a nuclear fuel cycle, starting with the identification of the quality and physico-chemical nature of uranium, and ending with the most important technological processes that are used for the production of triuranium octoxide, U_3O_8, or yellowcake and uranium dioxide, UO_2, or the nuclear quality sintered powder from which are manufactured the nuclear fuel bundles for Nuclear Power Plants.

World-class mining technologies (eg ISL) are used for extraction and separation of uranium from fields with enrichment levels below 0.1%. [1].

2. General Aspects of the Current Situation of the Nuclear Power Plants in the World.

An overview of the nuclear energy sector, both now and in its history since 1954, shows us that there are 667 nuclear reactors built or under construction worldwide.

Due to the problems arising from nuclear safety and particularly due to the Fukushima accident, many states that own Nuclear Power Plants have decided to discontinue nuclear development activities, even to shut down nuclear power plants by using renewable energy sources. According to data provided by the World Nuclear Association, in December 2016, there were 447 nuclear reactors in operation in 31 countries, with a total capacity of 391.3 GW covering about 11% of the electricity needs at world level. The 1.9% increase in energy production capacity in 2016 was largely due to investments in the Asia-Pacific area.

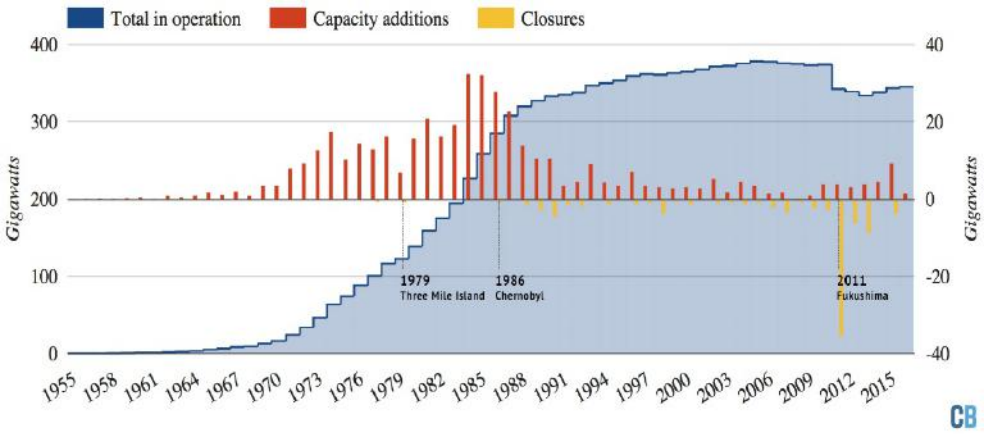


Figure 1. Total global nuclear power generating capacity in operation between 1955 and 2016 (blue area, left axis). Annual additions (red bars, right axis) and the capacity of reactors going offline or being shut down (yellow bars, right axis). Source: International Atomic Energy Agency (IAEA) PRIS database and Carbon Brief analysis. Chart by Carbon Brief.

2.1. Types of nuclear reactors

Nuclear reactors can be classified according to several criteria, namely: technological development, type of fuel used, moderator and cooling water and others, but this study will briefly present the four generations of reactors, the

resulting classification following the technological development and types of reactors classified according to the fuel used.

Thus, nuclear reactor technology has been developed and improved over the past five decades, with four distinct generations of nuclear reactors being developed.

Generation I: Reactors were developed in the 1950s and 1960s, of which very few are currently operational. These were called "early prototype reactors".

Generation II: Reactors belonging to this generation were developed in the mid-1960s, distinguishing themselves from generation I, through "active safety systems". Safety systems involve customized electrical or mechanical operation, which means that they are activated by operators and cannot work if the power supply systems are switched off. Approximately 90% of the nuclear power plants operating today use second-generation technology. Many have incorporated some passive or inherent safety features that do not require active controls or operational interventions to avoid accidents in the event of malfunction and can be based on gravity, natural convection or high temperature resistance.

Generation III: Advanced Reactors Developed in the Mid-1990s, their project incorporates other passive safety systems that improve reactor safety by operating without operator intervention.

Generation IV: Reactors of this generation are still at the project stage and will not be operational earlier than 2035. Many of the projects will be reactors using fast neutrons.

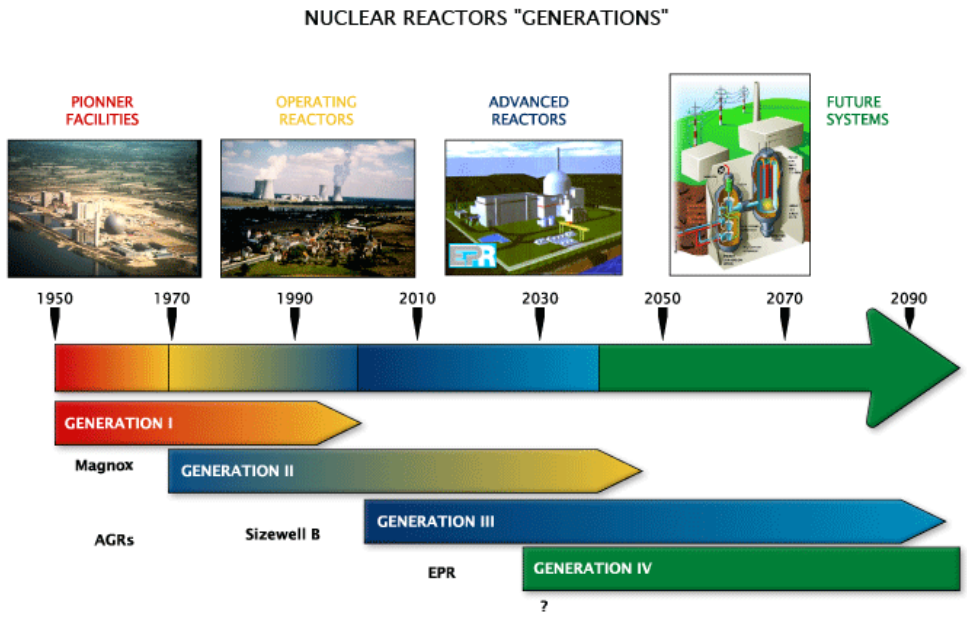


Figure 2. The Nuclear Reactors "Generations", source: http://www.bertrandbarre.com

If we take as a classification criterion the type of fuel used, we distinguish two categories of nuclear reactors: nuclear reactors using natural uranium fuel and nuclear reactors using uranium-enriched fuel.

2.2. Uranium nuclear fuel

In order to manufacture nuclear fuel there is a so-called "nuclear fuel cycle", NFC, which is a generic name designating a series of industrial processes involving the production of electricity from the uranium fission reaction in the nuclear power reactors.

The main production activities are carried out in two technological streams, where two distinct processes take place: obtaining the sodium diuranate (uranium technical concentrate) by the metallurgical processing of the uranium ore and obtaining the sinterable uranium dioxide powder (finished product) by refining uranium technical concentrates.

The processes are carried out on the basis of chemical reactions (wet and / or dried) to the production of triuranium octoxide U_3O_8 (stable intermediate product) and uranium dioxide sintering powder UO_2.

The technological process of nuclear fuel production starts with the processing of raw uranium ore. With this process, the nuclear fuel cycle is practically initiated

Then follows the process of chemical purification of the uranium technical concentrate and the production of uranium dioxide of nuclear purity. The process takes place at the interface between uranium aqueous solutions and the organic medium formed by tributyl phosphate and an organic solvent. The raw material used is sodium diuranate as a finished product of the hydrometallurgical flow [2].

The uranium dioxide thus obtained is processed for the purpose of producing the nuclear fuel pills with which the nuclear fuel elements and beams that will feed the nuclear reactors are manufactured.

The manufacturing of structural mechanical components, components of a nuclear fuel assembly (plugs, grids, skates, spacers, ties), is made of zircaloy and other components by means of turning, stamping, beryllium deposition, clamping welding , brazing, graphing, thermal treatment grafting, chamfering. [3].

2.2.1. Uranium ore global resources

Earth bark and oceans contain a uranium concentration of about 2.7 ppm. and 0.003 ppm,, respectively. The average U_3O_8 content of ores discovered so far is between 0.03% and 20%.

Uranium resources are generally classified as conventional or unconventional. Conventional resources are those that have a history of production where uranium is either a primary product, co-product or an important by-product.

Conventional resources are further divided into four categories, depending on the level of confidence in their occurrence: Rarely Guaranteed Resources (RAR), deducted resources, forecasting resources and speculative resources (SR). The first two are collectively known as identified resources.

The reasonably guaranteed resource category includes uranium in warehouses of size, grade and confined configurations, for which it is possible to specify the quantities that could be recovered with current proven technologies.

The deduced resource category includes uranium where occurrence is deduced from direct geological evidence, well-explored warehouse expansions, or deposits with established geological continuity.

The forecasting resources are uranium resources that are expected to occur in well-defined geological trends of known deposits or mineralized areas.

The speculative resources refer to uranium considered to exist in favorable geological, but unexplored geological areas.

The categories are subdivided according to the costs of the ore recovered in the processing plant: <40 USD / kg U, <80 USD / kg U and <130 USD / kg U.

Figure 3 presents the IAEA / Nuclear Energy Agency (IEA) Schedule for Classification of Uranium Resources, together with the estimated quantities in each category [4].

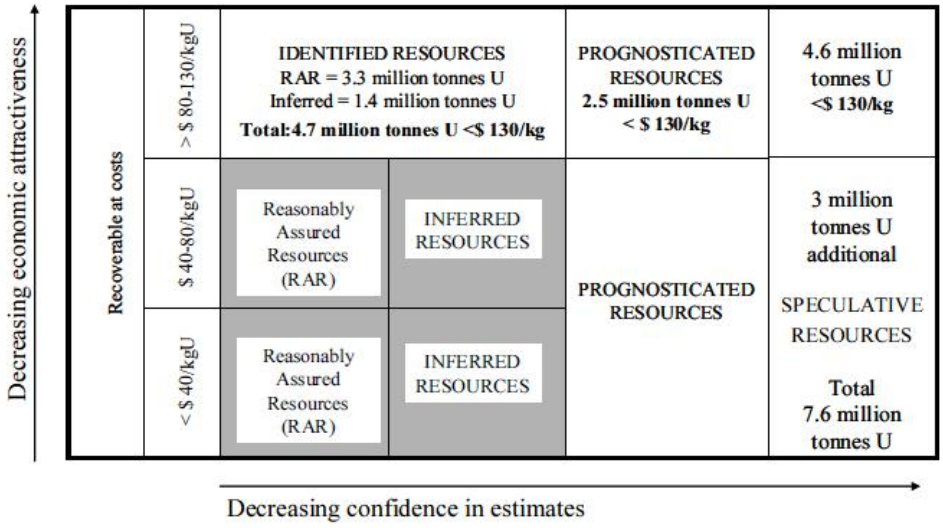


Figure 3. Classification scheme for conventional uranium resources

Uranium resources classified as unconventional, in which uranium exists at very low levels or can only be recovered as a minor by-product, include some 22 million tons occurring in phosphate deposits and up to 4 000 million tons in water great.

Uranium recovery technology from phosphates is developed, but costs are high (~ 60-100 USD / kg U). From sea water uranium was extracted only in small quantities, and costs are estimated to be close to $ 300 / kg U.

The total quantity of uranium resources identified worldwide in 2015 increased by 0.1% compared to 2013. This small percentage is due to low levels of investment and implicitly due to exploratory efforts reflecting current conditions and less favorable situation of the uranium market.

Total identified resources (reasonably assured and inferred) as of 1 January 2015 amounted to 5 718 400 tons of uranium metal (t U) in the <USD 130/kg U (<USD 50/lb U3O8) category, a decrease of 3.1% compared to 1 January 2013. In the highest cost category (<USD 260/kg U or <USD 100/lb U_3O_8), total identified resources amount to 7 641 600 t U, an increase of only 0.1% compared to the total reported for 2013.

The most significant change is reported in the <USD 80/kg U category, with an increase of 20.9% in inferred resources, compared to values reported in 2013. This can be primarily attributed to the addition of 208 400 t U of inferred resources from China and Kazakhstan. At the 2014 level of uranium requirements, identified resources are sufficient for over 135 years of supply for the global nuclear power fleet. [5].

2.2.2. Uranium ore production globally

Uranium production trough mining has fallen by 4% worldwide compared to 2013, but remains above the 2011 level, and Kazakhstan, now the world's largest producer, continues to increase production, but at a slower pace.

Overall, global uranium production declined from 58 411 t U in 2012 to 55 975 t U in January 2015. Changes are mainly the result of the decreasing production in Australia and of exploited uranium production from mines in Brazil, the Czech Republic, Malawi, Namibia and Niger.

In OECD countries, production declined from 17 956 t U in 2012 to 16 185 t U in 2014, mainly as a result of falling production in Australia and, to a lesser extent, of the production in the Czech Republic.

According to recent studies, uranium is produced in 21 countries, with the largest producer being Kazakhstan even though it has been slower lately, with output of 22 781 tons in 2014 and 23 800 tons in 2015.

Kazakhstan produced in 2014 more than Canada and Australia combined, countries which are the second and respectively the third largest uranium producers.

2.2.3. Uranium global demand

On the 1st of January 2015, a total of 437 commercial nuclear reactors with a net generation capacity of 377 G We were connected to the grid, requiring about 56 600 t U per year.

Taking into account the changes to nuclear development plans in several countries, it is estimated that the world's nuclear capacity by 2035 will increase to 418 G We net in the case of low demand and up to 683 G We net in the case of high demand which in percent means 11% and 81% respectively. Accordingly, it is estimated that global requirements for uranium fuel will increase to between 66 995 t U and 104 740 t U by 2035.

Projects to increase nuclear power electricity generation capacity differs considerably from one region to another. It is estimated that the East Asian region will see the largest increase, which by 2035 could result in the installation of a new capacity of 48 G We to 166 G We, representing increases of over 54% and respectively 188% from the capacity in 2014.

Nuclear capacity installed in the countries of Europe is also planned to increase significantly until 2035 with a value between 21 G We and 45 G We (an increase of about 49% and 105%, respectively).

Other regions planned to significantly increase nuclear capacity are: Middle East, Central and South Asia and Southeast Asia, with a more modest increase estimated in Africa and the Central and South American regions.

In the case of North America, where nuclear power projects have a low level, nuclear power generation capacity will remain roughly the same by 2035 and will increase by 11% to a large extent, depending on future electricity demand, extending the life of existing reactors and government policies regarding greenhouse gas emissions.

In the European Union, nuclear capacity in 2035 is estimated to fall by 48% in the case of a low demand scenario or to increase by 2% if the demand is high.

These predictions are subject to even greater uncertainty than usual after the Fukushima Daiichi accident, as the role that nuclear power will play in the future mix of generations in some countries that has not yet been established and China has not reported official targets for nuclear power capacity after 2020.

Key factors influencing the future capacity of nuclear energy include projected electricity demand, the economic competitiveness of nuclear power plants as well as funding arrangements for such intensive projects, the cost of fuel for

other power generation technologies, concerns about non-proliferation, and public acceptance of nuclear energy, which is a particularly important factor in some countries after the Fukushima Daiichi accident.

Concerns over the long-term security of fuel supply for nuclear energy considered to be beneficial for fulfilling the objectives of reducing greenhouse gas emissions, could contribute to the expected increase in demand of uranium [5].

3. Estimation and trends in uranium purchase prices

After extraction and initial processing at the mines, uranium is sent in the form of a yellowcake concentrate to a conversion facility. Here uranium is first separated from impurities through a series of chemical processes. This purification results in high purity uranium trioxide (UO_3).

This is the raw material for the next phase in the process of obtaining the fuel, namely conversion. There are converting facility in the world operating commercially in the US, Canada, France, Russia and China. The largest new facility is Areva's Comurhex II, which operates in two locations in France. It is expected that by 2025 and beyond, China will considerably increase its conversion capacity to meet the internal requirements [6].

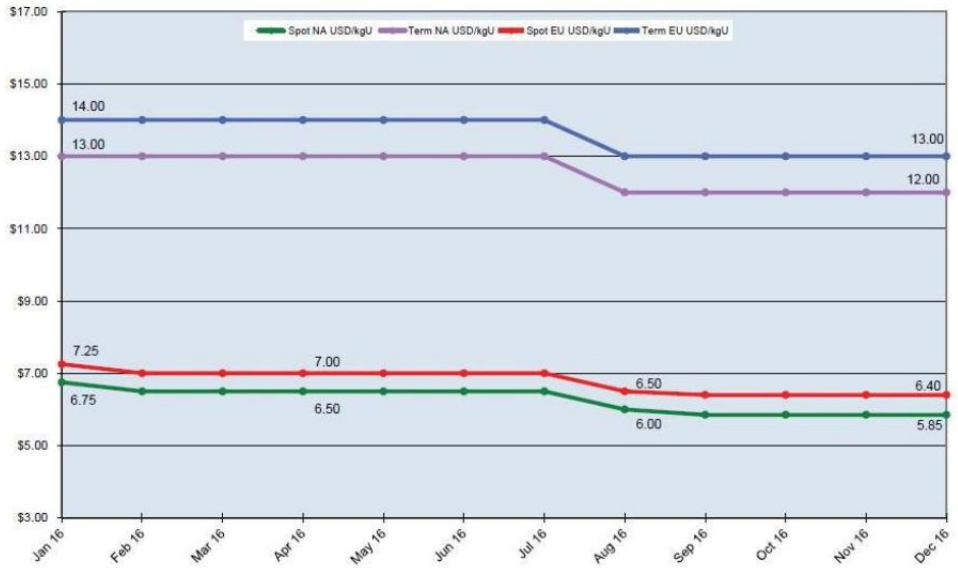


Figure 4. Evolution of conversion prices for natural and enriched uranium in the year 2016, source: https://www.uxc.com, 2018

Conversion spot purchase prices in Europe and North America, calculated by UxC, fell from $ 7.25 / kg U and $ 6.75 / kg U to $ 7.00 / kg U and $ 6.50 / kg U and remained stable until the end of July 2016. In August prices fell again and ended the year at $ 6.40 / kg U in the EU and $ 5.85 / kg U in the US.

The long-term UxC purchase prices for conversion in Europe and North America were stable from January to the end of July and amounted to $ 14.00 / kg and $ 13.00 / kg respectively. Since August, prices have started to decline and ended the year at $ 13.00 / kg in the EU and $ 12.00 / kg in the US [7].

4. Conclusions

The main purpose of the study is to analyze the main suppliers on the world market for the production and supply of nuclear fuel based on natural uranium, but also an overview of the evolution of supply prices for nuclear fuel based on natural uranium.

Starting from these general aspects of the present study, an estimate of the global nuclear energy situation can be made taking into account some of the main factors: population growth and, implicitly, the demand for electricity, approximation of a more accurate quantity of the uranium ore that can be extracted and processed, the amount of natural uranium fuel required for the operation of all operational nuclear power plants, those under construction and those planned to be built and commissioned over the next 30 years.

The increase of installed power of nuclear reactors is recognized as a very economical source of additional generation capacity. Turbine generator rehabilitation, combined with the use of initial margin benefits in reactor design, digital technologies and control tools, and investment in other generating capabilities, can increase production by up to 15-20%. There are many examples of this kind all over the world, but this has been a special concern in Sweden, the United States and Eastern European countries.

The overall picture of today's nuclear power plants is that they operate more efficiently than in the past, and unit operating costs are low compared to alternative power generation. These improvements have now become commonplace and will be integrated into the construction of new nuclear power plants.

World production and consumption of electricity grew to around 2.6% per year between 1990 and 2013, but many forecast factors believe that this growth rate will decrease in the future. For example, the International Energy Agency (IEA) scenario estimates growth in global electricity demand by 1.9% per annum over the 2014-2040 period.

It is estimated that by 2040 an important investment in new generation capacity will be needed to meet both the 10% anticipated demand growth and the need to replace a large number of nuclear power plants that will be closed during this period. As can be deduced from the study, there are sufficient uranium resources to support the continued use of nuclear energy and the significant increase in nuclear capacity for electricity generation and other long-term uses.

The identified resources, including the resources reasonably warranted and the resources deducted, are sufficient for more than 135 years, given the uranium requirements of about 56 600 t U (as of 1 January 2015). If estimates of current uranium consumption rates are used in power reactors, the resource base identified would be sufficient for over 160 years of nuclear reactor fueling.

The uranium resource described in this document is more than adequate meeting planned growth requirements by 2035. Complying with low expected estimates by 2035 would consume approximately 25% of the available identified resources at a cost of <130 USD / kg U and less than 20% of the identified identified resources at a cost of <260 USD / kg U.

Given the limited maturity and geographic coverage of uranium exploration worldwide, there is considerable potential for discovering new resources of economic interest. As has been clearly demonstrated in recent years, with adequate market signals, new uranium resources can be easily identified and exploited.

As mentioned in this paper, there are also considerable unconventional resources, including phosphate deposits and black shale / shale that could be used to significantly prolong the time needed for nuclear power to provide the demanded energy using current technologies.

Encouraging the advanced technology of the reactor and fuel cycle could also contribute significantly to the world energy supplies in the long term.

The transition to the advanced technology reactors and reused fuel is likely to increase the availability on the long-term viability of nuclear energy up to hundreds of thousands of years.

In addition, Thorium, which is more abundant than uranium is also a potential source of nuclear fuel, in case alternative fuel cycles are developed and successfully introduced in a cost-effective way.

The fuel reactors based on thorium have been proven and operated in the past. There are sufficient nuclear fuel resources in order to meet the requirements of the current energy demand and to increase in the future.

However, in order to achieve its full potential, it will be necessary to explore, research and to significant invest to develop in time new mining projects in order to facilitate the implementation of promising technologies.

R E F E R E N C E S

[1] *NEA/IAEA*, Report, „Environmental Activities in Uranium Mining and Milling", OECD, 1999;
[2] *Mihai POPESCU*, „Raport la Bilanţ de Mediu Nivel II, pentru CNU SA – Sucursala Feldioara", 2008 ;
[3] FCN-14712-DAU/SN-DSN, „Raport privind Analiza Siguranţei (RAS) şi Analiza Accidentelor pentru S.N. Nuclearelectrica S.A. – Sucursala Fabrica de Combustibil Nuclear Piteşti", RATEN CITON, 2013;
[4] NEA/IAEA (2006), Uranium 2005: Resources, Production and Demand, OECD Nuclear Energy Agency, NEA No.6098, Paris.
[5] Uranium2016: Resources, Production and Demand Nuclear Energy Agency, Organisation for Economic Co-Operation And Development;
[6] Ux Consulting Company, LLC (UxC), https://www.uxc.com, 2018
[7] EC, EURATOM Supply Agency, „Annual Report 2016", 2017

HEAVY WATER PRODUCTION IN ROMANIA: TWILIGHT OF AN INDUSTRY & DAWN OF NEW PERSPECTIVES

PRODUCŢIA DE APĂ GREA ÎN ROMÂNIA: DECLINUL UNEI INDUSTRII & DEBUTUL UNOR NOI ORIZONTURI

Raluca FAKO[1], Virgil IONESCU[2], Adriana OLTEANU[3], Sorin MEGLEA[4]

Abstract: *In November 1979 it was issued the Decree for the establishment of the Heavy Water Plant at Drobeta Turnu Severin. Growing as part of an ambitious nuclear program, this chemical production unit delivered the first heavy water batch in 1988 and it was recognized as the only heavy water producer in Europe and the largest in the world. In 2013, based on Government Decision no. 85/2013, the Heavy Water Plant entered in insolvency, and 3 years later was declared bankruptcy.*

A historical review of the manufacturing of the inventory for the first filling and subsequent make-ups for NPP Cernavodă Units 1&2 and, also, of the estimated inventory for future potential NPP Cernavodă Units 3&4 is delivered with highlights of achievements, drawbacks and challenges encountered within this industrial saga.

This paper aims to filter technical, economical and political aspects that could act as decision fundaments for future plans for heavy water stocks management (including the tritiated ones) or actions for redrawing the heavy water and tritium R&D program in Romania and for the prospection of new industrial developments.

Keywords: heavy water, tritium, management, resource

Rezumat: *În noiembrie 1979 a fost publicat Decretul de înființare al Combinatului Chimic pentru producerea apei grele Drobeta Turnu Severin.*

Dezvoltându-se ca parte a unui program nuclear ambițios, această unitate de producție chimică a livrat prima șarjă de apă grea în 1988 și a fost recunoscută drept singurul producător de apă grea din Europa și cel mai mare din lume. În 2013, în

[1] RATEN – CITON, e-mail: fakor@router.citon.ro
[2] RATEN – CITON, e-mail: ionescuv@router.citon.ro
[3] RATEN – CITON, e-mail: olteanua@router.citon.ro
[4] RATEN – CITON, e-mail: megleas@router.citon.ro

baza Hotărârii Guvernului nr. 85/2013, uzina de apă grea intră în insolvență, iar 3 ani mai târziu este declarat falimentul.

Este prezentat un istoric al fabricației inventarului de apă grea necesar pentru primă umplere și completare ulterioară pentru unitățile 1 și 2 de la CNE Cernavodă și, de asemenea, un istoric pentru inventarul estimat ca necesar pentru viitoarele unități 3 și 4 ale CNE Cernavodă; sunt subliniate aspecte importante referitoare la realizările, dar și privind dificultățile și provocările care au trebuit să fie înfruntate în această adevărată saga industrială.

Această lucrare își propune să scoată în evidență aspecte tehnice, economice și politice care ar putea să fundamenteze deciziile referitoare la planurile viitoare de gestionare a stocurilor de apă grea (inclusiv cele tritiate) sau acțiunile în vederea revizuirii programului de cercetare - dezvoltare în domeniul apei grele și tritiului în România și pentru prospectarea unor noi direcții de dezvoltare industrială.

Cuvinte cheie: apă grea, tritiu, management, resursă

1. Introduction

Heavy Water Plant (RAAN ROMAG PROD) was founded in 1979 and formerly was known as Drobeta Turnu Severin Chemical Plant. Its development was an exclusively Romanian achievement (research, design, execution, commissioning and operation) and this unit was an unique top technology facility of the Romanian industry.

In the `80s this unit was part of an ambitious nuclear program designated to sustain the economic development of the country and its energy independence. This goal was possible because ***the human & financial resources needed*** for this ***were available and allocated*** and, also, ***at that time the project had a strong political support and involvement***.

The first production line was tested starting with September 1987, and the first drop of heavy water was produced in July 1988. On July 17, 1988, heavy industrial water was obtained at ROMAG PROD - Drobeta Turnu Severin, at parameters set for nuclear reactors type CANDU 6 (table 1). At the beginning, in the first years after commissioning, the unit was shutdown for about three years (between 1990 – 1992) to upgrade technological equipment and environmental surveillance and protection systems. The modernization process had as result the improvement of the plant feasibility and had shortened the manufacturing time for a new supply of heavy water. During years this unit was recognized as the only heavy water manufacturer in Europe and the largest in the world.

2. Case study

Table 1. RAAN ROMAG PROD – heavy water parameters (source: RAAN ROMAG PROD website)

Heavy Water Quality As per Technical Sheet (that met AECL requirement for CANDU type reactors – Technical Specification TS-XX-38000-001, AECL, 01.02.1999)		
Parameter	**Value**	**Unit**
Isotopic concentration	Min. 99.78	% wt. D_2O
Conductivity	Max. 5	µS/cm
Turbidity	Max. 1	NTU (ppm SiO_2)
Organics ($KMnO_4$ demand)	Max. 10	mg/kg
Chloride	Max. 0.5	ppm
Tritium	none	µCi/kg
NOTE: 1. Also, could be delivered one grade of heavy water with concentration greater than 99.96 D_2O and four grades of depleted low-weight water with concentration ranging between 0 and 80 ppm D_2O. These grades are patented and homologated. 2. Considering high heavy water quality, ROMAG PROD is recognized on European and Asian markets, being exporter of nuclear grade heavy water supplies in South Korea, China, Germany, etc.		

The plant is situated on the territory of Rascovesti village, Izvoru Bârzii parish, at a distance of about 7km from Drobeta-Turnu Severin (Figure 1), and the area occupied by the plant is estimated at approximately 82 ha.

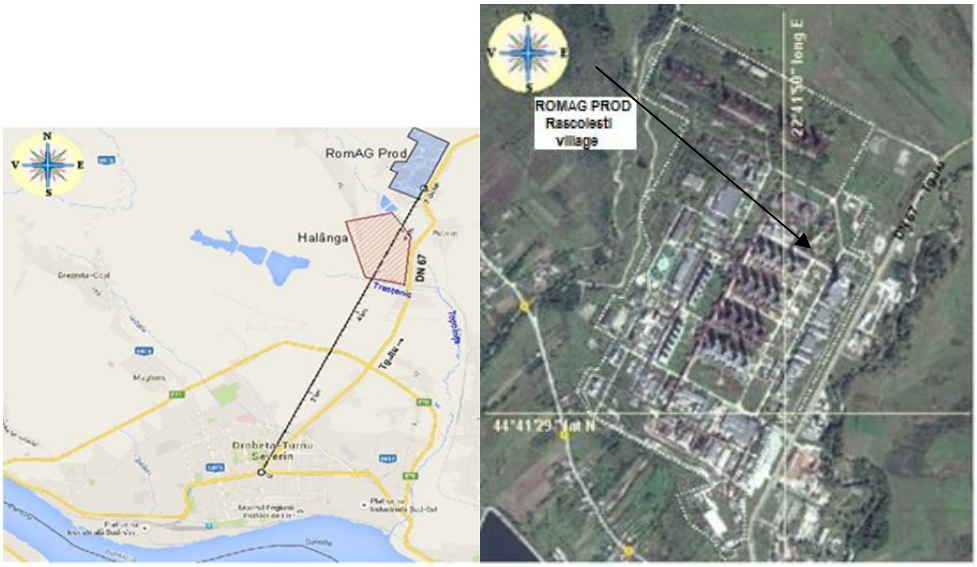


Figure 1. Heavy Water Plant location

Of the four production lines related to the first stage of development of the Water Factory, only three (GS1, GS3 and GS4) were put into operation. The process equipment from ROMAG PROD is made from rolled and forged G52/28 steel grade material developed in the eighties at Steel Plant Galati based on National Technical Normative NTR 440/83. It was clearly a particular attention to be paid to the materials from which the biterm separator (columns and pipes) were built. Wet sulfide hydrogen and water solutions with hydrogen sulfide (pH ~ 3) are very corrosive agents. For competitiveness, this carbon steel grade, developed in Romania was characterized by a smooth, homogeneous microstructure, it was subjected to a stress relief treatment and 100 % inspected in fabrication. The corrosion resistance was provided by a thermo-chemical treatment of the exposed surfaces, called "pyrite layer controlled formation".

This material was fully characterized in order to establish it fitness for use in hydrogen bearing environments as structural steel for isotopic exchange columns walls. Due to long term action of mechanical and thermal stresses in the presence of an extremely active and aggressive service environment (e.g. H_2S aqueous solutions) during service life of this plant an extensive RD program was developed to guarantee industrial safety. This preoccupation resulted in methodologies of action for long term serviceability of the equipment and other technological components at RAAN ROMAG PROD, in-situ experimental programs for the evaluation of long term behavior of steels that are serviced in hydrogen bearing environments, particularly for fine grained C-Mn steels serviced in H_2S environments etc.; methodologies to evaluate the technical state of pressure vessels for & after the extension of service life and the repair program for the isotopic exchange columns were developed and agreed with the Romanian regulatory body for pressure vessels (ISCIR).

This exceptionally complex program (and unique technical solution) was settled with the support of Romanian specialists and technicians considering that the following items are important for the development of repairs and monitoring schedule:

- The type of equipment to be evaluated;
- Technical expertise;
- Available resources.

As part of the activities it was demonstrated that it is possible to recover the capability of ROMAG equipments by repairs performed on metallic structure (deposition of metallic layers by welding in those areas reported as non-conformity areas (erosion – corrosion cavities)).

The heavy water process is based on the isotopic exchange of process water and hydrogen sulfide in a biterm system in GS (Girdler-Sulphide) plants – figure 2, where a primary deuterium oxide concentration of about 5-7 %.

The reaction occurs in the liquid phase, between process water and dissolved hydrogen sulfide, on special sieve-type plates:

$$D_2O + H_2S \underset{30^0C}{\overset{130^0C}{\rightleftarrows}} H_20 + D_2S \quad (1)$$

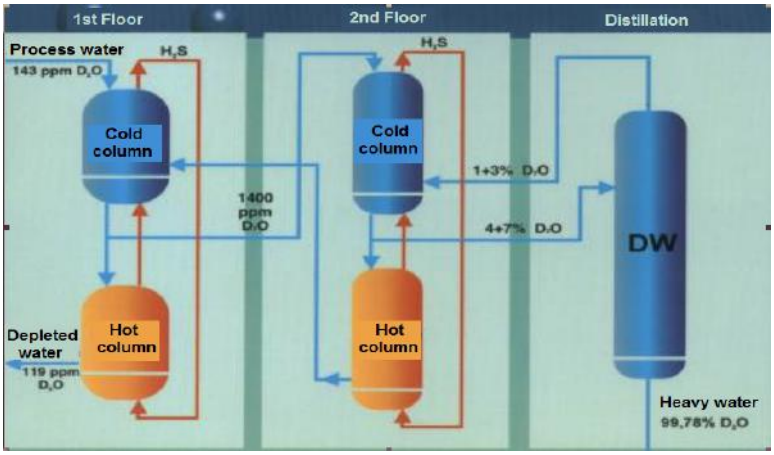


Figure 2. Heavy Water manufacturing process (source: RAAN ROMAG PROD website)

The basic element of the Girdler-Sulphide process is the pair of liquid-gas contact columns, of which the cold column works at temperatures of 30-35 ^oC and the hot column at 110-130 ^oC. Process water, with a deuterium content of about 143 ppm, flows from top to bottom through the cold and then the hot column, in counterflow with the hydrogen sulfide conveyed by a compressor on the base of the hot column. Water is progressively enriched in deuterium as it flows through the cold column and it is progressively depleted in deuterium while flowing through the hot column. The effluent leaving the hot column contains about 120 ppm deuterium. Thus, by deuterium enriching on each tray the heavy water concentration reaches about 1400 ppm. The production of the first floor is even greater as the deuterium concentration of the effluent is lower. On the second floor, the water resulting from the A, B and C biterms of the first floor is concentrated from 1400 ppm to 5-7% D_2O. The enriched water from the second floor feeds the vacuum distillation plant, DW.

The main dynamic equipment was the hydrogen sulfide recirculation compressor, virtually the vital element of the process, of high design complexity and therefore very expensive, so the plant was equipped with only active equipment. Due to the corrosive action of wet hydrogen sulfide, the compressor was made from special stainless steel.

The final concentration, up to a minimum of 99.85%, was achieved in the vacuum distillation plant, where it is obtained the finished product - heavy water

with parameters required for a CANDU reactor, according to the quality grade imposed by Cernavoda NPP.

For heavy water quantities produced by ROMAG PROD installations were taken a number of measures to ensure the long term preservation (qualitative and quantitative). The heavy water inventory was stored under inert gas in 220 liter steel cylinders and in tanks of 50 m^3, in safe warehouses.

During 1994 – 1998, this plant produced heavy water inventory with quality parameters requested for Unit 1 at Cernavodă NPP (556 t). During 1999-2004 was produced necessary heavy water inventory for first load for Unit 2 from CNE Cernavodă.

Several quantities were exported, in compliance with the Romanian legislation in force, to other countries such as Korea, China, Germany, USA.

Currently, it was produced the whole quantity set by law for the operation and extension of service life in case of four CANDU units at Cernavodă NPP (first fill & make up heavy water).

And, therefore, heavy water production has been stopped since 2015; previously in 2013, Heavy Water Plant entered in insolvency and 3 years later was declared bankruptcy.

For this reason, installations are in different stages of conservation, and the only activities aimed for the management of the existing heavy water inventory are carried out based on former RAAN ROMAG PROD procedures.

The situation is frozen in accordance with the initial design. The unit was designed with two development stages:

- Stage I - consisting in four isotopic exchange modules GS1, GS2, GS3 and GS4; this stage has been partially commissioned and produced the inventory forecasted for the development of the Romanian nuclear program;
- Stage II - consisting in three isotopic exchange modules GS5, GS6 and GS7; this was never commissioned and the construction was stopped in 2004.

3. Conclusions

Due to insolvency, and lately to bankruptcy, the resources for the management of heavy water inventories were a great challenge for the entity and for technicians. For this reason, the Government initiated discussions with all involved parties in order to find the best solution for heavy water inventories preservation.

Starting with 2017 a new entity was created in order to preserve the national reserve of heavy water. But this is not the end of the epic. Due to the evolution in the years before insolvency and further, the separation of another entity from the dying Heavy Water Plant is a delicate and hard topic.

In fact this new entity will have to re-design its structure and skills and capabilities, on the remains of the Heavy Water Plant. A lot of modifications and reallocations are to be imagined in order to guarantee de serviceability of this new entity and, also, to meet the requirements stated by law for the preservation of the national heavy water inventory.

In the beginning of the paper we mentioned that for the development of this unique heavy water plant in Europe both legal, economical and technical aspects were correlated. Now the same approach should be considered.

The social problem, the human resource (old staff, retirements, no new skilled young personnel trained for operation of the installation), lack of financial resources, or insufficient & delayed budgets, hesitations in the decisional process could impair the planning of the future steps for the transition to this new management entity for the existent virgin heavy water inventory. For this reason a strong governmental support is needed, sustained by the voices of recognized specialists that were the ones that in the past dedicated all their knowledge and efforts for the Drobeta Turnu Severin Chemical Plant.

R E F E R E N C E S

[1] Tehnical study and tehnical – economic documentation regarding actual heavy waters management after controlled closing of its production, required equipment, works and investigations on RAAN platform ROMAG PROD branch, code AG – MNG – DTH – 10216, by RATEN-CITON, for RAAN branch ROMAG PROD Drobeta Turnu Severin, 2016;

[2] Study on the foundation, planning the process of obtaining the necessary authorizations for carrying out the activities related to heavy water management at CNMAG and for providing logistical support, code AG-CNMAG181-DAU by RATEN CITON / ICSI Ramnicu Vâlcea for National Center for Heavy Water Management, 2018.

ENVIRONMENTAL REQUIREMENTS TO BE CONSIDERED FOR PLANNING THE NATIONAL GEOLOGICAL DISPOSAL PROGRAM

CERINTE DE MEDIU CE TREBUIE CONSIDERATE LA PLANIFICAREA PROGRAMULUI DEPOZITULUI GEOLOGIC NATIONAL

Veronica ANDREI[1], Adrian JELEV[2], Diana VILCU[3]

Abstract: *The elaboration of a comprehensive national program of the National Geological Repository destined for the disposal of the spent nuclear fuel in accordance with the European Directive on radioactive waste 2011/70/Euratom and at the level of the actual national geological disposal programs should consider the risks of the national context.*

The paper presents some information related to the environmental aspects that can be significant risk factors induced by the national context in the planning of the National Geological Repository. Based on the experience of over 20 years of direct participation in environmental authorization procedures of major nuclear projects in Romania, the authors confirm the results of a national risk management study on the current planning of the National Geological Repository and recommend integrated planning of major environment actions in the work breakdown structure of the stages for Siting and Licensing the Repository Site in order to minimize the negative impact of these risks on the development of the repository.

Keywords: National Geological Repository (NGR), national context risks, NGR program, risk planning

Rezumat: *Elaborarea unui program cuprinzator de sine-statator al Depozitului Geologic National destinat depozitarii finale a combustibilului nuclear ars, in acord cu Directiva europeana de deseuri radioactive 2011/70/Euratom si la nivelul unor programe nationale efective de depozitare geologica trebuie sa considere riscurile contextului national.*

Lucrarea de fata prezinta unele informatii cu privire la aspectele de mediu care pot constitui factori de risc semnificativ indusi de contextul national la planificarea Depozitului Geologic National. Pe baza experientei de peste 20 de ani de participare directa in procedurile de autorizare de mediu a unor

[1] Societatea Nationala „Nuclearelectrica” SA, e-mail: vandrei@nuclearelectrica.ro
[2] Societatea Nationala „Nuclearelectrica” SA, e-mail: ajelev@nuclearelectrica.ro
[3] Societatea Nationala „Nuclearelectrica” SA, e-mail: dvilcu@nuclearelectrica.ro

proiecte majore nucleare in Romania, autorii confirma rezultatele unui studiu al managementului riscului contextului national asupra planificarii curente a Depozitului Geologic National si recomanda planificarea integrata de actiuni majore de mediu in structura activitatilor din etapele de selectare si autorizare a amplasamentului depozitului in scopul minimizarii impactului negativ al acestor riscuri asupra dezvoltarii depozitului.

Cuvinte cheie: depozit geologic national (DGN), managementul riscurilor contextului national, , integrare planificare raspunsuri la riscuri in planificarea DGN

1. Introduction

In the last decades, organizations responsible for the disposal of radioactive waste, scientists and regulatory bodies have developed comprehensive knowledge demonstrating that deep geological disposal (hereafter referred to as geological disposal) is a feasible and safe technology [1]. The safety, the emplacement and feasibility of the engineering solutions for the final disposal of spent nuclear fuel, high-level radioactive waste and long-lived radioactive waste have been demonstrated by waste management organizations, evaluated by nuclear regulators and approved by governments as the basis for the stage of selecting the site of a geological repository.

Research, development and demonstration issues to be pursued in the future, including uncertainties associated with them, will no longer question the feasibility of geological disposal. Finland and Sweden are the first countries in the world that have already demonstrated that it is possible to establish the site of a geological repository following a process involving the broad participation of society. At the time of writing this paper, the French Nuclear Safety Authority, IRSN, gave its final agreement on the safety case sent by the French radioactive waste management agency, ANDRA, for the construction of the Cigéo geological repository [2].

However, international experience has shown that, and under the conditions of a strengthened national nuclear system, from the early stage of the development of a geological disposal program, a worrying question arises about the sustainability of the program for societal reasons [3].

From the study of the international standards and reports issued by IAEA, OECD/NEA, JRC-CE, etc., certain observations support the need to study the national context when designing and planning the geological disposal program [4].

The lessons learned from the reconsideration of some national geological disposal programs (e.g. programs in the UK, Canada, or the Czech Republic) included those that:

- In the discussions with the community, any "benefit package" must be an early and transparent part of the placement process;
- Communities have not encouraged the formation of a group to support the siting process, until the impetus of those against has been manifested;
- An adapted management is recommended, in which the plans of the program are divided into manageable steps or phases, each phase being

characterized by a decision that is taken in consultation with all stakeholders [5];
- Program managers should act to identify and implement solutions that will enable communities to accept how decision-making is being approached in site selection.

This paper further provide some information on the environmental aspects that can be a significant risk factor for NGR planning and the authors' recommendation for an integrated approach to action planning that minimizes the negative impact of identified risks on NGR planning.

2. Challenges of the national context to the application of environmental regulatory procedures for nuclear projects

Based on the international experience demonstrated in Finland, Sweden and recently in France, an average time scale for siting phases of a NGR program was about 25 years. This duration makes it difficult to predict the national context, as it includes many non-technical aspects, and its complexity and diversity bring about a degree of uncertainty for these forecasts. Therefore, the analysis of the national context may need to be resumed by the management of the responsible factors, if the uncertainties increase and the structure of the sources of impact and their actual impact impose. Much more, the analysis of the national context, beyond the authorization of the site of a geological repository, seems no longer necessary be treated by itself, because the international experience and the evolution of the Romanian society and economy will offer viable options for solving the risk sources in program development.

SNN specialists earlier have presented the experience and lessons learned in the authorization procedures for major nuclear capital projects on the CNE Cernavoda platform [6]. Thus, in a paper presented at Foren 2016 [7], the authors presented a topical issue encountered in the development of major nuclear projects, which are still topical and require effective solutions to solve. It is about the need to correlate the level of detail of the Environmental Impact Report (RIM) with the level of technical and safety information that is objectively available at the time of its collection for RIM. Furthermore, in order to produce RIM, the environmental requirements for RIM"s detailed documentation should be applied in a manner that is integrated with design and safety requirements. It is obviously about the design and safety requirements imposed by the Nuclear Regulatory Body for the development phase of the project that corresponds to the environmental impact assessment phase.

The authors support the view that the issue of complying with environmental requirements is more complex when preparing a geological disposal program.

A PESTEL analysis on the geological disposal of radioactive waste in Romania made in 2013 [8] showed that some aspects of the national context

associated with the Legal and Environmental Factors (from the PESTEL analysis) should be changed in order to have a sustainable program and to make the probability of errors in the program as small as possible.

Currently, there is no experience in Romania regarding the development of an appropriate environmental assessment or environmental impact study for a repository for radioactive waste.

It is necessary to correlate and integrate the legal environmental requirements [7, 9] with the legal requirements in the nuclear field and to establish the degree of detail / development of the disposal concept and all necessary data allowing a proper environmental impact assessment.

Ensuring the availability of data necessary for the environmental impact assessment would be in correlation with the development stage of the geological disposal program. To make clear this correlation implies a collaboration between the central environmental authority and the regulatory authority in the nuclear field, from the initiation of the NGR program, given the fact that the experience and knowledge in regulating geological disposal belongs to the nuclear authority

The PESTEL analysis mentioned above has shown that more in-depth analysis should clearly identify which are the risks specific to these risk factors, response actions to risk factors and how to integrate the actions into the NGR program planning. Such analyzes were conducted in a risk-management study of the national context that took place between 2013 and 2016 [10].

The environmental risks specific to the national context identified in the risk management study are presented in Table 1.

Tabel 1. The risks of the national context due to environmental and legal environmental factors

Risk Identifier*	Risk title
16E1	Lack of expertise/ technical support in the field, to authorities issuing environmental
17E2	The tradition of presenting technical information at technical design level in the EIA phase
20L3	The absence of the first authorization step in environmental legislation (SEA procedure)
21L4	The need to obtain the urbanism certificate for construction as a prerequisite for obtaining the environmental agreement

***** *The risk identifier is consistent with that in the Risk Register.*

For performing the risk analysis and establishing how to integrate the planning of risk response actions into updating the current NGR Strategy (the current NGR Strategy means the strategy developed at the level of 2009; there is no

public update of this strategy until the elaboration of the present paper) contributed the following:

• An in-depth analysis of the requirements of the current European and national environmental legislation;

• Experience in the development of similar approval, authorization and environmental approval processes in other similar national nuclear projects. It is about the activities carried out at Cernavoda NPP during the period 1996-2013 regarding: elaboration of the environmental balance and the procedure for obtaining the environmental permit at Unit 1 CNE Cernavoda; the procedure for obtaining the environmental agreement at the Intermediate Dry Spent Fuel Storage Facility at Cernavoda NPP; the procedure for obtaining the environmental agreement for Units 3 and 4 Cernavoda NPP [6]);

• Consultation of an environmental expert with large expertise at national level; The expert had direct responsibilities in the environmental authorization procedures of the major nuclear installations in Romania.

The National Risk Risk Management study proposed a revision of the current NGR Strategy planning in order to propose an optimal solution to integrate the risk response due to the environmental factor and legal environment factor. The results relevant to this paper are briefly discussed in the next section.

3. Integrating the response to national environmental risks due to the environmental factor into DGN planning

The risks of the national context were considered to be the risks associated with entrances, actions and inactions outside the NGR program developer organization over which the developer organization's management has no control but could have a significant impact on the program. This impact can be manifested itself in technical terms, cost, planning and / or acceptance of the program.

The national context for initiating the development of the NGR program was considered as described by the PESTEL analysis conducted in 2013. In the opinion of the authors of this paper, the results of the PESTEL analysis maintain its validity for the Legal and Environmental factors. We make this observation in the context in which the activities of the current NGR Strategy that was developed in 2009 have not been carried out as planned, and an effective geological disposal program is to be developed in the coming years by the Nuclear Agency and for Radioactive Waste.

The study on the management of the national context risks[10] has shown that effective integration of responses to these risks in NGR program planning can be done through a step-by-step approach to a set of actions that need to be deployed in the planning of the NGR program, in an integrated or iterative manner, because they are interconditioning. The optimal integrated risk response solution is to identify and establish the processes contributing to the planning of the NGR program and which should ensure such integration.

The authors of this paper confirm that, among all the processes that ensure the integration of the responses to the national context risks identified in the mentioned risk management study and which are presented in Figure 1, the "Environmental Agreement, Authorization and Approval (Based on the Environmental Impact Assessment and population health) "is the process that makes a significant contribution to the planning of the NGR program, as it relates significantly to the strategic planning of the program, particularly the planning of the program's technical activities. As a result of the schedule risk analysis on the current NGR Strategy, it has resulted, among other things, that necessary actions in this process are:

• A more realistic planning of the NGR program due to the need to reconsider the structure of those major activities within the program in which the risk response actions can be integrated

• The planning of major environmental activities whose work breakdown structure falls wholly or predominantly within the structure of activities on the critical route of estimating the NGR graphic.

More realistic planning of the selection and authorization schedule of the geological repository site has led to a 32 year duration estimated by expert judgment in the most likely scenario by difference with the 20.5 year duration of the current NGR Strategy. As a result of integrating the response to the risks induced by the environmental factor and the legal environmental factor, the new duration of the stages of selection and authorization of the NGR site was estimated at about 25 years, about 7 years less than the duration in the case of the most likely scenario.

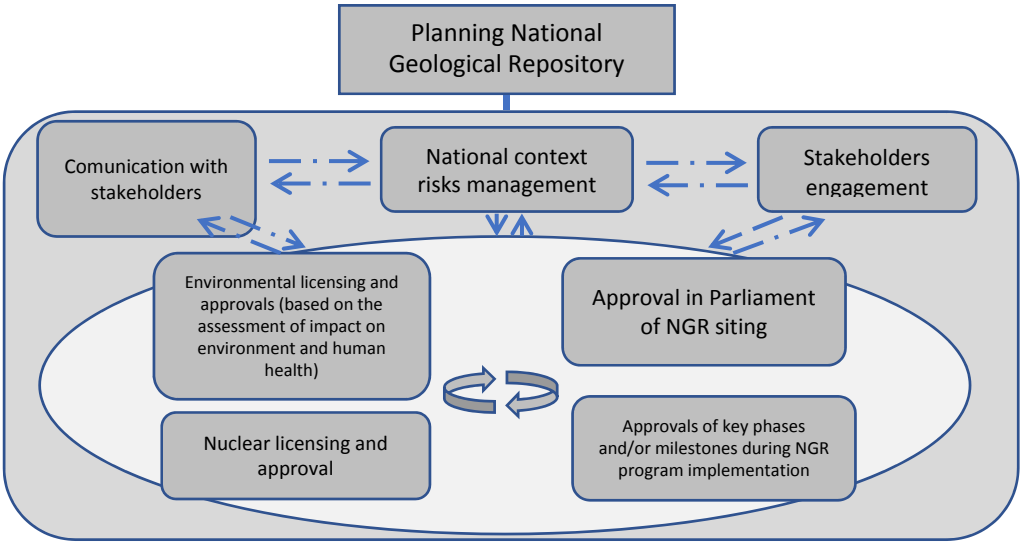


Figure 1. Processes that ensure the integration of the response to the national context risks in the planning of the current NGR Strategy

More realistic planning has led to a new structure for major activities on "Environmental Endorsement, Authorization and Approval", as identified in Figure 2.

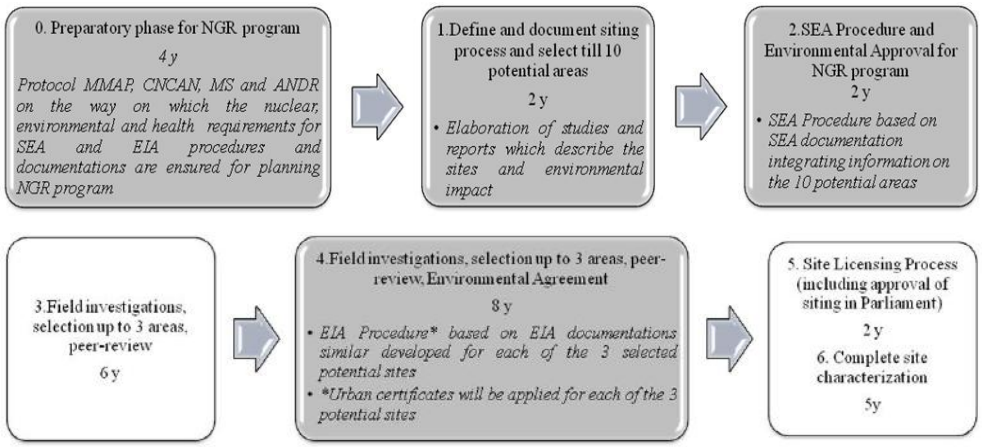


Figure 2. General planning of the „Environmental Permit, Authorization and Approval with reconsideration of the current NGR Strategy

Based on their own experience, the authors confirm that the structure of the major activities proposed by the risk management study ensures minimization of risks to the national context and beyond. However, where these risks combines with other risks of the national context, they can still have a significant impact, so it is also the study. The authors appreciate that the results of the study are useful to NGR program managers/ planners in the context of making the decision to develop a sustainable NGR program, for at least the following reasons:

• Effective recommendations based on previous national experience judgment is welcome given that there is very little information on how environmental permitting activities should be included in the project's graph in a practical way for managers and planners and/or executers of the project;

• Risk management cannot be approached properly or not at all if managers/planners are not aware of activities in the NGR program, within which risk response actions can be integrated; the study helps with the description of the actions that could prevent risk causes.

4. Conclusions

In this paper, the authors contribute by analyzing and interpreting, based on their experience of over 20 years of direct participation in the environmental authorization procedures of major nuclear projects in Romania, to increase the

transparency of the results of a risk management study of the national context on the planning of the future National Geological Repository destined for disposal of spent nuclear fuel.

The authors recommend to managers/planners of the National Geological Repository program to approach the program's preparation stage, integrated planning of major environmental activities in the structure of the activities of the stages of selection and authorization of the site of the repository, which according to the international experience lasts over 20 years, namely: the strategic environmental assessment of the program and the procedure for obtaining the environmental agreement, in order to minimize the negative impact of the environmental risks on the development of the repository.

REFERENCES

[1] OECD NUCLEAR ENERGY AGENCY, Engineered Barrier Systems (EBS) in the Safety Case: Design Confirmation and Demonstration (Proc. Workshop, Tokyo, 2006), OECD, Paris (2007).
[2] SFEN Newsletter, 18.01.2018, Cigéo est « techniquement mature » estime l'ASN]
[3] *Andrei V., Prisecaru I.,* "Risk Management Process for National Geological Repository Program," Proc. WEC Central & Eastern Europe Regional Energy Forum – 12th Edition, FOREN 2014, 22-26 June, Romania Paper s3-45-en, 2014, 1-8, Costinesti, Romani, 2014.
[4] IAEA. Safety Standards Series No. SSG-14, "Geological disposal facilities for radioactive waste, Specific Safety Guide," IAEA Vienna, 2011
[5] EPRI, "EPRI Review of Geological Disposal for Used Fuel and High Level Radioactive Waste," Vol. IV Lessons Learned Final report, September 2010, 7-2.
[6] *Jelev A., Marin F., Gheorghe M., Andrei V., Balas N., Transparenta procedurii AcM pentru proiectul Unitatilor 3&4 CNE Cernavoda*, Proc. WEC Central & Eastern Europe Regional Energy Forum – 12th Edition, FOREN 2014, 22-26 June, Paper s3-17-ro, 2014.
[7] *V. Andrei, R.Urjan, A. Jelev, E. Macovei, D.Vilcu,* "Integrating safety, technical and environmental aspects to obtain environmental agreement for nuclear capital projects"
[8] *Andrei V., Prisecaru I.,* "The Use of Pestel Analysis in Development of the Romanian Geological Repository," Scientific Bulletin of University Politehnica of Bucharest, Series C, Vol. 76, 2014, 247-254
[9] Petrisor S., "Participarea/implicarea publicului în cadrul procedurilor de reglementare din punct de vedere al protecţiei mediului," Workshop IPPA EC FP 7 Project, Simpozionul International SIEN 2011, Bucuresti, Romania, 2011.
[10] Andrei V., Teza de doctorat, "Contributii la fundamentarea programului Depozitului National Geologic de deseuri radioactive," 2016 WEC Central & Eastern Europe Regional Energy Forum – 12th Edition, FOREN 2016, 12-16 June, Costinetsi, Romania, 2016.

FATIGUE TESTS ON SEU-43 ZIRCALOY-4 TUBES AT 350°C

TESTE DE CICLAJ PE TUBURI TIP SEU-43 DIN ZIRCALOY-4 LA 350°C

Valentin OLARU[1], Viorel IONESCU[2], Dragos IONESCU[3], Alexandru NITU[4], Alexandra ION[5], Florea VOICU[6]

Abstract: *Presently, the CANDU type reactors use a fuel bundle made up of 37 rods filled with natural uranium pellets. The functionality of the current fuel at extended burn degrees doesn't offer the desired performance. Because of this fact, a new fuel bundle concept, named SEU-43, was proposed. This new concept uses low enriched uranium, obtained from the reprocessing of burnt fuel in LWRs. The SEU-43 fuel bundle represents an evolution of the standard CANDU fuel bundle with 37 elements.*

The paper presents the testing of samples worked from Zicaloy-4 tubes (''as-received'' metallurgical state), utilized in the composition of the CANDU SEU-43 fuel bundle. These tests are intended to simulate their behaviour in a power cycling process inside the reactor. The testing process is of low cycle fatigue type, done outside of the reactor, on "C-ring" samples, cut along the transversal direction. These samples are tested at 1%, 2% and 3% amplitude deformation, at a temperature of 350°C. To determine the amplitude deformation for both types of tube (small and big diameter), a numerical simulation must be done, using the finite element analysis in the ANSYS commercial computer code, resulting in a set of calibration curves. The curves are dependent on the metallurgical state, as well as on the mechanical and microstructural characteristics of the material. For determining mechanical properties, tensile tests were done on Zircaloy-4 samples ("ring tensile test" samples) at a temperature of 350°C.

The fatigue test results are in the form of a fatigue life curve (N-ε) for Zircaloy-4 used in the SEU-43 fuel bundle. The curve is determined by the experimental dependency between the number of cycles to fracture and the deformation amplitude. The low cycle fatigue mechanical tests, done at 350°C, together with electronic microscopy analyses and microscopic grain analyses have reflected the characteristic behaviour of the zircaloy-4 metal in the given environmental conditions.

Keywords: low cycle fatigue, SEU-43, 350°C, ANSYS computer code

[1] RATEN-ICN, e-mail: valentin.olaru@nuclear.ro
[2] RATEN-ICN, e-mail: viorel.ionescu@nuclear.ro
[3] RATEN-ICN, e-mail: dragos.ionescu@nuclear.ro
[4] RATEN-ICN, e-mail: alexandru.nitu@nuclear.ro
[5] RATEN-ICN, e-mail: alexandra.ion@nuclear.ro
[6] RATEN-ICN, e-mail: florea.voicu@nuclear.ro

Rezumat: *În prezent, reactoarele de tip CANDU utilizează un fascicul de combustibil alcătuit din 37 de teci umplute cu pastile de uraniu natural. Funcționalitatea combustibilului actual la gradele de ardere extinse nu oferă performanța dorită. Din acest motiv, a fost propus un nou concept de fascicul de combustibil, numit SEU-43. Acest nou concept utilizează uraniu cu un grad scăzut de îmbogățire, obținut prin reprocesarea combustibilului ars în LWR-uri. Fasciculul de combustibil SEU-43 reprezintă o evoluție a fasciculului standard de combustibil CANDU cu 37 de elemente.*

Lucrarea prezintă testarea probelor prelucrate din tuburile de Zicaloy-4 (stare metalurgica "as-received"), utilizate în compoziția fasciculului de combustibil CANDU SEU-43. Aceste teste sunt destinate să simuleze comportamentul lor într-un proces de ciclaj a energiei în interiorul reactorului. Procesul de testare este de tip oligociclic, efectuat în afara reactorului, pe eșantioane "C-ring", prelevate pe direcția transversală. Aceste probe sunt testate la o amplitudine a deformarii totale de 1%, 2% și 3%, la o temperatură de 350°C. Pentru a determina amplitudinea deformarii pentru ambele tipuri de tuburi (diametru mic și mare), trebuie făcută o simulare numerică, folosind codul numeric comercial cu elemente finite ANSYS, rezultând un set de curbe de calibrare. Curbele sunt dependente de starea metalurgică, precum și de caracteristicile mecanice și microstructurale ale materialului. Pentru determinarea proprietăților mecanice, testele de tracțiune au fost efectuate pe eșantioane din Zircaloy-4 (probe de tip "ring-test") la o temperatură de 350°C.

Rezultatele testelor de oboseală sunt sub forma unei curbe de durabilitate (N-ε) pentru teci de Zircaloy-4 utilizate în fasciculul de combustibil SEU-43. Curba este determinată de dependența experimentală dintre numărul de cicluri până la fisură și amplitudinea totala a deformării. Testele mecanice de oboseală oligociclice, efectuate la 350°C, împreună cu analizele microscopice electronice și analizele microscopice de graunte au reflectat comportamentul caracteristic al aliajului Zircaloy-4 în condițiile de mediu date.

Cuvinte cheie: oboseala oligociclică, SEU-43, 350°C, codul numeric ANSYS

1. Introduction

Presently, the CANDU type reactors use a fuel bundle made up of 37 rods filled with natural uranium pellets. The functionality of the current fuel at extended burn degrees doesn't offer the desired performance. Because of this fact, a new fuel bundle concept, named SEU-43, was proposed. This new concept uses low enriched uranium, obtained from the reprocessing of burnt fuel in LWR type reactors. The SEU-43 fuel bundle represents an evolution of the standard CANDU fuel bundle with 37 elements.

Based on a technology developed at ICN Pitesti and on material specifications and fabrication tolerance imposed through the SEU-43 fuel bundle project, attempts were made to formulate a series of demands which will constitute the base for a quality control plan focused on ICN's own experimental research.
In this paper, a mechanical testing of Zircaloy-4 is presented. This material is used in the forging of two different diameters (small and big) tubes used to encapsulate the SEU-43 fuel.

The fatigue tests done outside the reactor for non-irradiated Zircaloy-4 tubes were thought up and realised at ICN. These fatigue tests will help in

estimating the durability curves and other parameters that influence fatigue behaviour.

The testing processes of low cycle fatigue type, done outside the reactor, on "C-ring" samples, cut along the transversal direction. These samples are tested at 1%, 2% and 3% amplitude deformation, at a temperature of 350°C.

The testing equipment for "C-ring" samples was built at ICN and is capable of performing fatigue tests, being able to control displacement in the horizontal plane with the help of the proper transducer. For every sample deformation (1%, 2%, 3%), there is a corresponding displacement of the piston, measured by the transducer in millimetres.

The durability curve represents the fatigue behaviour of the cladding material. In this case it is obtained for a temperature of 350°C. In the current paper, having in mind the absence of high temperature resistant strain gages, the deformation calibration of "C-ring" samples, was done using finite element analysis in the ANSYS computer code [1].

Using the experimental data resulted from the tests, the durability curves are graphically traced for each type of sample (big and small diameter) and for metallurgical and microstructural state mentioned in the testing matrix.

2. Experimental set-up

2.1. Installation presentation

The fatigue testing equipment (Figure 1) was designed and built in 2002 at ICN and is used to test multiple types of samples.

Figure 1. The equipment used for simulated power cycling fatigue

The sample holders (Figure 2) are two "yoke"-like pieces, one of which is fixed to the support beam of the testing chamber, the other being coupled with a crank gear system, necessary for cyclic movement.

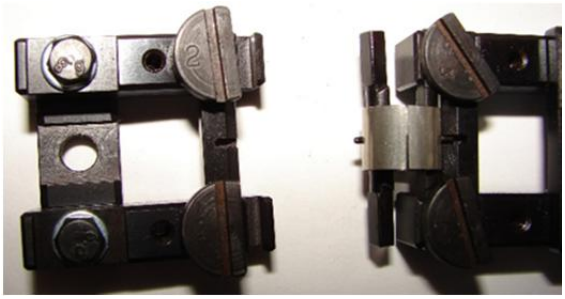


Figure 2. Sample holders

The command and data acquisition operations are handled by the Split Monitor Software. This application is developed and adapted for the Windows XP Operating System.

2.2. Sampling Process

The ring shaped samples (Figure 3) were cut from Zircaloy-4 tubes.

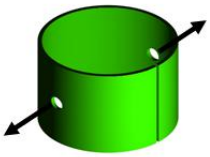


Figure 3. C-ring sample

3. Fatigue Test

3.1. Establishing the Calibration Curves

The calibration curves are dependent on the metallurgical state, as well as on the mechanical and microstructural characteristics of the material. These curves represent the dependency between the central deformation of the sample and the displacement of the piston, variables which can be measured with a strain gage and a displacement transducer. Every deformation (1%, 2% and 3%) is equivalent to a certain displacement of the piston which is measured by the transducer in millimetres.

Having in mind that the measuring of the deformation poses the most difficulties because of the small dimensions of the sample, the elaborate techniques utilized for attaching the strain gages and the lack of high temperature resistant strain gages, an alternative was chosen to obtain the calibration curves: the finite element analyses in the ANSYS computer code [1].

Using this method, the calibration curves for the 1% ÷ 3% deformation domain were traced for "C-ring" samples, at a temperature of 350°C. For this, a transversal cross-section of the specimen was modelled (Figure 4).

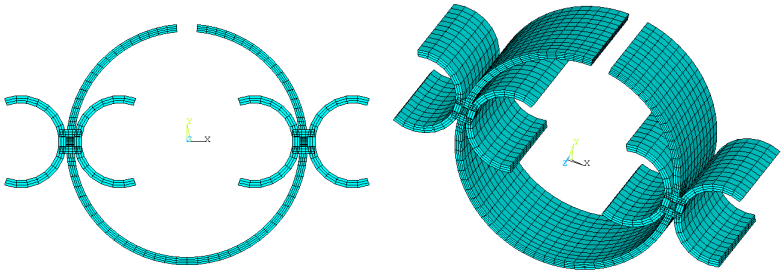


Figure 4. Modelling of "C-ring" samples using ANSYS

The code utilizes the material parameters (flow resistance, break resistance and elastic deformation) determined with the following formulas:

$$\varepsilon_{real} = \ln\left(1+\varepsilon_{\exp}\right) \qquad \sigma_{real} = \sigma_{\exp}\cdot\left(1+\varepsilon_{\exp}\right)$$

Where: ε_{real} is the calculated deformation; σ_{real} is the calculated flow resistance;
For determining mechanical properties, "ring tensile test" samples (Figure 5) were used.

Figure 5. „Ring tensile test" samples

The experimental values were determined from the curves (Figure 6 and Figure 7) obtained from tensile tests at a temperature of 350°C.

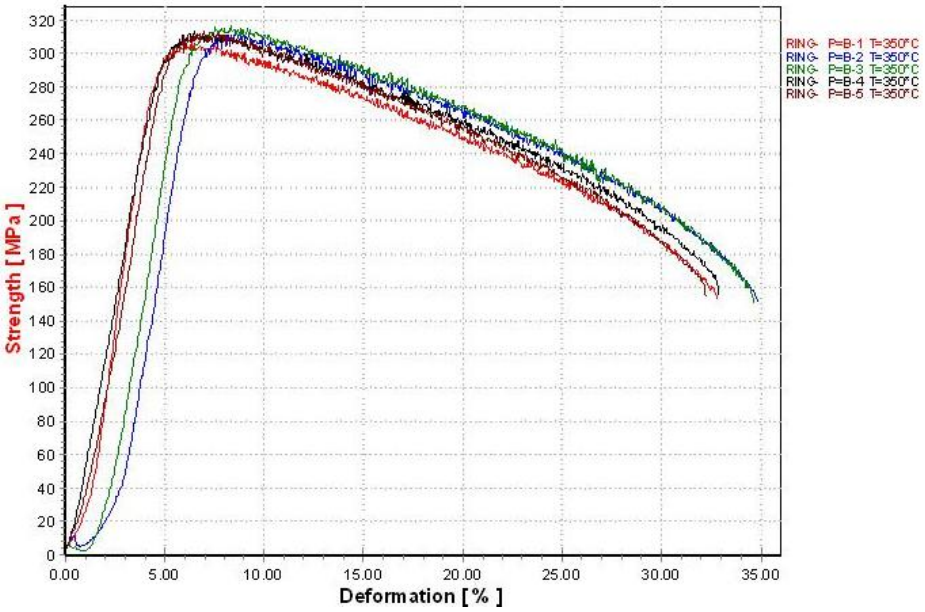


Figure 6. The experimental tensile curves for SEU-43 samples, big diameter (D), at 350°C

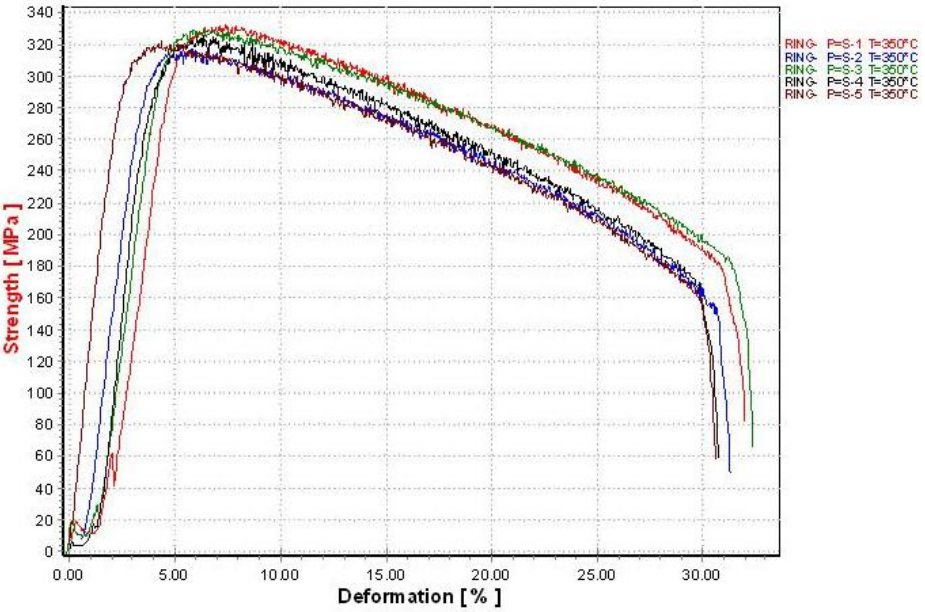


Figure 7. The experimental tensile curves for SEU-43 samples, small diameter (d), at 350°C

The mechanical parameters resulted from the calculated "real" curves are used for the ANSYS modelling. After the simulation, the results were compared with the results of calibrations done using strain gages and were found to be similar.

3.2. The Fatigue Tests

The fatigue tests which simulate power cycling conditions inside the reactor were executed in conformity with references [2], [3], [4], [5] and [6]. A series of five samples per deformation (1%, 2%, 3%), per diameter (small and big) were tested.

4. Results

4.1. Calibration Curves

The calibration curve is the graphical representation of the dependency between the central deformation of the sample and the displacement of the piston. The calibration curves obtained using the ANSYS code are represented in Figure 8 (SEU type I – big diameter) and Figure 9 (SEU type II – small diameter). The different curves are for calculations done in the central node and for mediation of multiple nodes starting from the central one. This mediation represents the length of the contact surface of the strain gage with the sample.

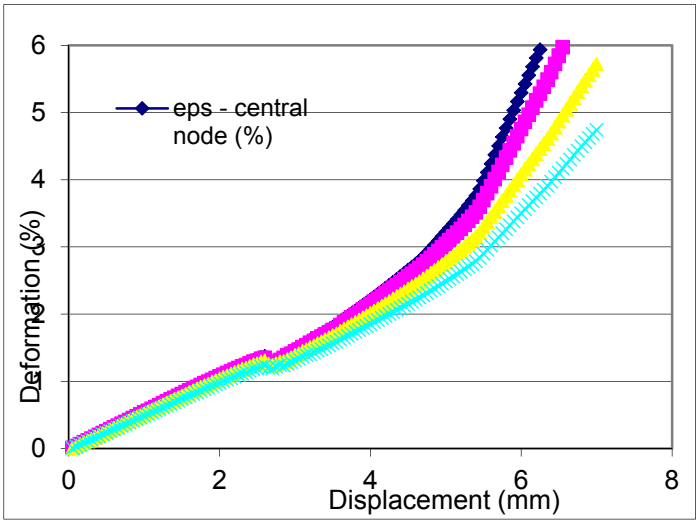


Figure 8. Piston displacement - deformation dependency at 350°C, when the fixing holes are drilled on the diameter axis of the big diameter (D) SEU-43 sample.

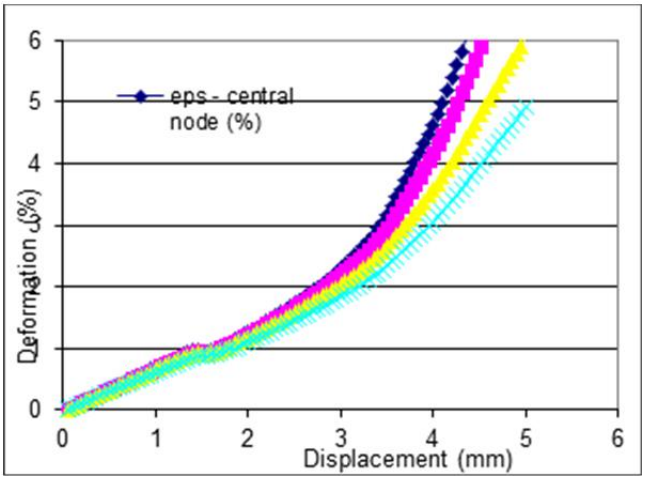


Figure 9. Piston displacement - deformation dependency at 350°C, when the fixing holes are drilled on the diameter axis of the small diameter (d) SEU-43 sample.

It was established for both diameters that the best representation of the dependency between the deformation (1%, 2%, 3%) and the piston displacement is correlated by the curve for the central node, coresponding to the smallest possible area of contact with a strain gage.

4.2. Determinating the Durability Curve

The experimental data are presented in Figure 10 and Figure 11 in the form of a power curve dependent on the equation . A comparison was made with data from references [6] and [7].

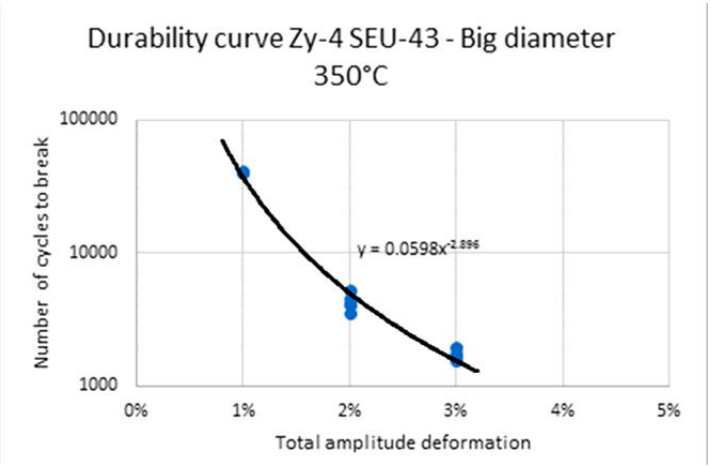


Figure 10. Durability Curve for Zircaloy-4, SEU-43, big diameter(D) samples, at 350°C

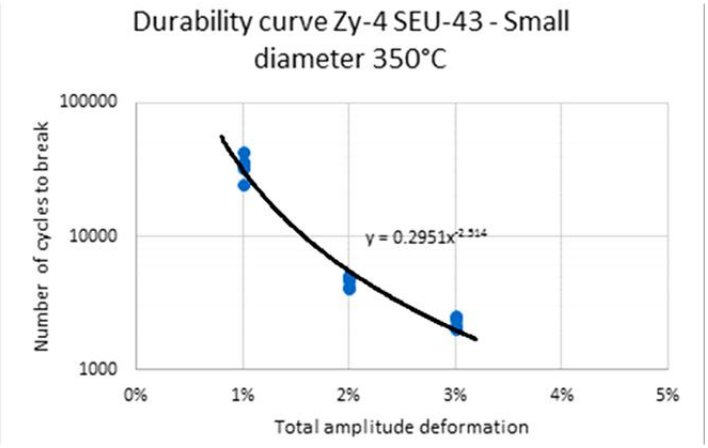


Figure 11. Durability Curve for Zircaloy-4, SEU-43, small diameter(d) samples, at 350°C

The curve functions respect the Coffin – Manson law: $\frac{\Delta\varepsilon}{2} \cdot N^{\alpha} = C$.

4.3. Microstructural analysis of break surface

A SEM analysis was done on the fatigue test samples. The different aspencts of the break surface are shown in Figure 12. The SEM analysis revealed the ductile-brittle break mode, done in phases along the fatigue striations, phenomenon present in all Zircaloy-4 fatigue tests.

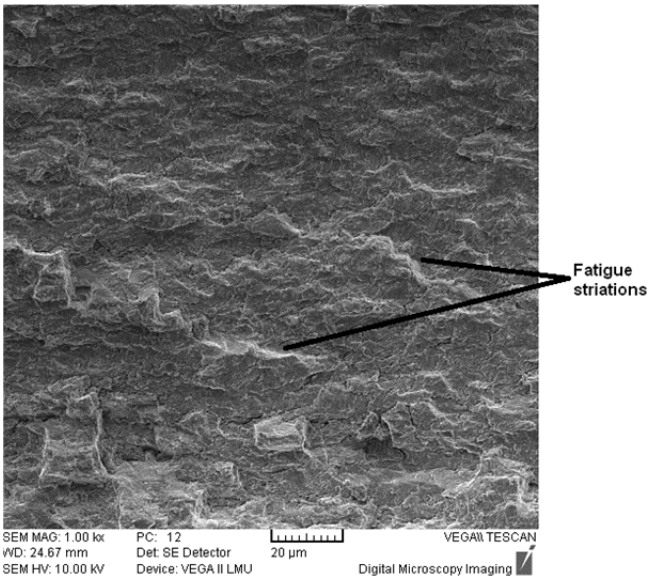


Figure 12. Zircaloy-4 break surface of fatigue sample at 350°C (x1000)

The fatigue tests described in this paper simulate power cycling conditions inside the reactor. The "C-ring" samples worked from Zircaloy-4 tubes ("as received" state) were tested using an installation designed and built at ICN. The tests were done at a temperature of 350°C for 1%, 2% and 3% amplitude deformation in accordance with ref. [4].

For determining the calibration curves, the finite element analyses in the ANSYS computer code was used. The prediction capacity of the ANSYS simulation was verified in a previous project, comparing the results of the calibration with the results from using strain gages, which were found to be acceptably close.

The results include the durability curve describing the behaviour of "C-ring" samples worked from Zircaloy-4 tubes (as received state) subjected to low cycle fatigue tests done at 1%, 2% and 3% deformation amplitude.

The SEM analysis revealed the ductile-brittle break mode of the cycling fatigue tested samples.

R E F E R E N C E S

[1] ANSYS 7.1 Documentation
[2] M. Roth, C. Gheorghiu, V. Radu, V. Pitigoi, C. Ducu, V. Malinovschi, M. Abrudeanu – "Zircaloy-4 CANDU Reactor Sheaths Fatigue Lifetime", Progress of Cryogenics and Isotopes Separation Conference, Rm. Valcea, 2004.
[3] M. Tayal, A. M. Manzer, R. Sejnoha – The integrity of CANDU Fuel During Load Following, AECL, 1991.
[4] Standard Practice of Linearized Stress – Life and Strain – Life Fatigue Data, ASTM E 739 – 91.
[5] Testing Specification, "Fatigue tests done outside the reactor, which simulate power variations of CANDU fuel", INR-372000-400-002, Rev.0
[6] Toma Pana, Stefan Dan Pastrama – Integrity of Metallic Structures, Ed. Fair Partners, Bucuresti, 2000.
[7] Palleck Steve, Waheed Asif, Sim Ki-Seob, AECL Document, 2009.

LEAD FAST REACTOR TECHNOLOGY AND ALFRED DEMONSTRATOR DEVELOPMENT IN THE CONTEXT OF FUTURE MARKETS OF ENERGY

DEZVOLTAREA TEHNOLOGIEI REACTOARELOR CU NEUTRONI RAPIZI ȘI A DEMONSTRATORULUI ALFRED ÎN CONTEXTUL VIITOARELOR PIEȚE DE ENERGIE

Marin CONSTANTIN[1], Ilie TURCU[2], Constantin PAUNOIU[3], Alexandru TOMA[4], Daniela DIACONU[5], Minodora APOSTOL[6], Mirela NITOI[7], Daniela GUGIU[8]

Abstract: Lead Fast Reactor is one of Generation IV technologies recommended by GIF and European Strategies (SNETP Strategic Agenda, SET Plan) to be developed in the future in order to achieve a better sustainability of nuclear power. An important step of this effort consists of the implementation of ALFRED (Advanced Lead Fast Reactor European Demonstrator). It is a 125MWe lead cooled fast reactor demonstrator, connected to the grid, with a target date for operation start-up in 2028. The reference site was nominated as Mioveni nuclear platform, the location of RATEN-Institute for Nuclear Research. The paper discuss the potential of LFR in the future energy market. Some important characteristics are important: load following capabilities in a market with large penetration of variable renewables, drastical decrease of the amount of radioactive wastes and their radiotoxicity, improved safety, reduction of emergency preparedness zone, enhancing the proliferation resistance, heat market, non-electric applications of nuclear, including H2, transport, cost reductions by standardization and modularization, regulations, safety and environmental. The paper proofs that in some market conditions introduced by large share of variable renewable the nuclear hydrogen production offers good performances for load following. The use of wasted heat is a great reserve and can be capitalised in the conditions of the New Member states of Europe Union.

Keywords: Nuclear power, Generation IV, lead fast reactors, energy markets

[1] RATEN-ICN, e-mail: marin.constantin@nuclear.ro
[2] RATEN-ICN, e-mail: ilie.turcu@nuclear.ro
[3] RATEN-ICN, e-mail: constantin.paunoiu@nuclear.ro
[4] RATEN-ICN, e-mail: alexandru.toma@nuclear.ro
[5] RATEN-ICN, e-mail: daniela.diaconu@nuclear.ro
[6] RATEN-ICN, e-mail: minodora.apostol@nuclear.ro
[7] RATEN-ICN, e-mail: mirela.nitoi@nuclear.ro
[8] RATEN-ICN, e-mail: daniela.gugiu@nuclear.ro

Rezumat: *Tehnologia reactoarelor cu neutroni rapizi răciți cu plumb (LFR) este una dintre tehnologiile viitorului, recomadată la nivel global de GIF, iar la nivel european de către agenda strategică a SNETP și de către SET Plan. Dezvoltarea este orientată către o sustenabilitate îmbunatațită a energiei nucleare. Un pas important în dezvoltarea LFR constă în realizarea reactorului de demonstrație ALFRED. Acesta va avea o putere electrică de 125 MWe și va fi conectat la rețeaua națională de electricitate. Intrarea în operare este prevazută pentru anul 2028. Amplasamentul de referință este platforma nucleară Mioveni, pe aceași locație cu Institutul de Cercetări Nucleare, RATEN ICN.*

Lucrarea analizează potențialul tehnologiei LFR în contextul viitoarelor piețe de energie. Câteva caracteristici ale dezvoltării LFR sunt cruciale pentru viitor: flexibilitatea în operare într-o piață cu pondere ridicată a regenerabilelor cu producție intermitentă, scăderea drastică a cantității de deșeuri radioactive generate și a radiotoxicității asociate, securitate operațională îmbunatățită, reducerea drastică a zonei de intervenție la urgență, întărirea rezistenței la proliferare, aplicații pe piața de energie termică, potențialul de operare în cogenerare, producerea de hidrogen pentru decarbonarea transportului, reducerea de costuri prin standardizare și modularizare, reducerea impactului asupra mediului în operare normală, precum și în situații accidentale.

Lucrarea arată că, în anumite condiții ale pieței caracterizate de o pondere mare a regenerabilelor cu producție intermitentă, o abordare coordonată a dezvoltării sinergice nuclear-regenerabile-hidrogen oferă performanțe de flexibilitate și eficiență economică crescute. Pe de altă parte utilizarea căldurii uzual evacuate în mediu reprezintă o rezervă apreciabilă de creștere a performanțelor economice, în special pentru noile state membre din UE unde infrastructura de încălzire urbană este construită și, în multe orase, încă funcțională.

Cuvinte cheie: Energetica nucleară, Generație IV, reactori rapizi răciți cu plumb, piața energetică

1. Introduction

Among the promising fast reactor technologies Lead Fast Reactor (LFR) has been identified as a route with great potential to meet the goals of increased safety, improved economics for electricity production, reduced nuclear wastes for disposal, and increased proliferation resistance.

In Europe, the European Commission organized the Sustainable Nuclear Energy Technology Platform (SNETP) that, through its Strategic Research and Innovation Agenda [1] promoted the development of fast reactors with closed fuel cycle. The Roadmap proposed by the European Sustainable Nuclear Industrial Initiative (ESNII), includes the lead-cooled fast reactor as an alternative technology to be developed in parallel with the sodium-cooled fast reactor.

The LFR is based on a closed fuel cycle for efficient conversion of fertile uranium and management of actinides (enhanced sustainability), the inert nature of

the coolant provides important design simplification (improved economics) and allows for designing decay heat removal systems based on well-known light water technology and passive features (increased safety). Moreover, the reference LFR fuel (MOX) constitutes a very unattractive route for diversion or theft of weapons-usable materials and provides increased physical protection against acts of terrorism (Non-proliferation and Physical Protection).

The LEADER project [2] incorporates the vision emerged from SNETP, as well as the main goals of ESNII, and succeed at the end of the project (2013) to develop the conceptual design for ELFR (European Lead Fast Reactor) industrial size plant and, also, for a scaled demonstrator of the LFR technology – ALFRED (Advanced Lead Fast Reactor European Demonstrator). The project involved 17 partners from Industry, research organizations and universities. The total effort was 502 person-months over a period of 36 months.

The main features of the core design and the effectiveness of the European LFR demonstrator are presented in [3, 4].

According with the European vision the experimental infrastructure in the New Member States should be enlarged in order to balance the differences between different countries. For this reason the construction of LFR and GFR demonstrators is foreseen to be in Central and East-European countries [5].

In the current conditions of the constraints imposed by the 20C scenario nuclear power is a powerful pillar for climate change prevention/mitigation. Nuclear and renewables (RES) are the alternatives with zero CO2 emissions. The paper discuss the synergies between nuclear and RES in the context of future energy markets. One of the difficulty is induced by the tendency to increase as much as possible the RES (especially the variable RES, represented by photovoltaic (PV) and wind (W)) and the difficulties of nuclear to be operate in load following conditions. The paper propose some possible solutions in the context of LFR technology development. On the other hand the economic impact of possible use of heat, non-electric applications of nuclear (including H2), and cost reductions by standardization and modularization are introduced. Also the technical aspects of LFR such as the decrease of the amount of radioactive wastes and their radiotoxicity, improved safety, reduction of emergency preparedness zone, enhancing the proliferation resistance are discussed.

2. New energy markets

In the present market the nuclear energy is on the first place (18,3%) among the alternatives with zero carbon emissions (the second is the hydro with 13.5%). At global level, the most important competition is with the natural gas, especially due to the new capacities based on shale gas, but the cost of the de-

carbonization introduces some advantages for nuclear. On the other hand, the nuclear power has an important advantage in terms of security of supply.

The difficulties introduced by the continuous increase of the world average temperature have introduced urgent actions to limit the variation at 20C until 2050. In this action the renewable energies and the nuclear power play an important role by their almost zero CO2 emissions. In the last decades in many countries, especially in Europe Union member states, an amazing development of the renewable (RES) occurred, boosted by the variable RES (vRES) represented by photovoltaic (PV) and wind (W) systems. A large share of the RES in the energy mix creates some difficulties in the market especially in load following characteristics, imposing a greater number of peaking units and a greater capacity for load following.

Wind energy is extremely variable and the present prediction capacity in wind production is very limited. The PV are better predicted and shows a daily and seasonable variation that can be considered in energy planning.

On the other hand nuclear has limited capabilities for load following. The reduction of the power to a certain demand may be achieved by appropriate adjustment of the neutron flux (for example by grey rods) or by variation of the coolant velocity, or by variation of the moderator's level. The usual reduction is at a level of 50% of nominal power, in some special cases to 20%. The increase rate of the power is limited in order to follow the dynamics constraints and to protect the materials to different stresses. There are some requirements for load following (EUR=European Utilities' Requirements) applicable to the existing LWR design; (1) continuous operation between 50% and 100%, (2) compulsory design to ensure a low power operation ~20%, (3) rate of change of 3% of Pr/min, (4) maximum no of changes: 2/day, 5/week, 200/year, (5) special units (participating in emergency load variations) has a rate 20% Pr/min.

A high share of nuclear in the energy mix (like in France, Belgium, Slovakia) imposes the need for load following. In other case, for example Germany, the large share of RES (introducing large variability of the energy productions) introduces the need for nuclear flexible operation. The most important example for load following is the case of France with a daily total variation of 5-10%. It should be noted for some periods the variation can exceed 20%. The ramps used are 1-5% P/min, starting from the 50% level of power. Some plants are operated in special modes (18 h at rated power, 6h at low power).

In case of new energy markets a deep penetration of RES will produce effects in economics for the nuclear power plants (NPP). Low or negative electricity spot market prices induced by the vRES may lead to financial losses for nuclear power. At the same time the intermittency will create pressure on load following and consequently a stringent need of more peaking units.

On the other hand, these penalties will impact the performances of the nuclear power and introduce in the debate the need for a reduction of the costs (investment, operation and maintenance, fuel) in order to successfully compete with the other energy alternatives. In the same direction the increase of the economic efficiency can be based on co-generation and the use of nuclear for non-electrical applications including desalination and hydrogen production in order to decarbonize the transport and industrial chemistry based on hydrogen.

In Romania the nuclear energy is a stable component of the energy mix and a significant contributor to the de-carbonization of the economy. Nuclear power is used in base load mode, at a high capacity factor (due to the quasi continuous refueling and reduced outage periods). At the same time a great development of the RES occurred based on investment incentives, green certificates for production, priority to the consumption. This development has introduced some difficulties created by the intermittency of around 12% of the electricity production, including the impact of a high cost of transport due to the lack of modernization of the grid and the localization of the important resources (nuclear, hydro, RES) in the south part of the country. A new investments in grids and networks would be needed to support such increase in the intermittent capacity, with new peaking generation added to existing infrastructure.

3. LFR technology

LFR technology was started, in early 60s, with lead-bismuth cooled fast reactors in Russia, especially for submarine propulsion. The fast spectrum introduces the advantage of converting the fertile isotope U238 into fissile Plutonium with an important effect on the optimal use of the Uranium resources. After 2000 based on the recommendation of GIF and current progress in fast reactor technologies, the strategic documents of the EU shows a clear orientation to the development of three Generation IV systems: (1) Sodium Fast Reactors (SFR), (2) Lead Fast Reactors (LFR), (3) Gas Fast Reactors (GFR).

The LFR development is based on gradual and progressive steps to reach the maturity [6]: (1) identification of main issues related to the technology, (2) the construction and exploitation of small scale to large scale experimental facilities to offer optimal solutions for the open issues, (3) perform irradiation tests, develop the fuel and the materials, (4) implementation of the demonstrator, (5) develop the prototype, (6) development of the FOAK, (7) commercial fleet development.

Based on a large collaboration of RDI organization in the frame of ELSY and LEADER projects a conceptual design for a 600 MWe LFR system and for its demonstrator (ALFRED) was developed. More than other 10 projects (FP6, FP7,

and H2020) supported RDI and technological development for LFR. Romania, by RATEN ICN, contributed to these efforts as a valuable partner in the majority of these projects. ALFRED is a 300 MWth reactor addressing the concerns on safety, economics and sustainability of LFR technology. At the same time ALFRED is also a crucial component of a pan-European Distributed Research Infrastructure based on 4 nodes (one in Brassimone-Italy, and Rez-Czech republic, and two nodes in Mioveni-Romania consisting in the licensing supporting infrastructure and ALFRED itself).

On this basis, and after the consultation of the main organization acting in the nuclear field, in February 2011, Romanian Government approved the Memorandum 2025/2011, "Romanian option to host ALFRED demonstrator", initiated by Ministry of Economy, Trade and Business Environment. The document declares the availability of Romania to host ALFRED demonstrator and nominates Institute for Nuclear Research (INR) to initiate the preparatory actions at international level for ALFRED consortium construction.

In February 2012 a MoU (Memorandum of Understanding) was signed by Ansaldo Nucleare, ENEA, and INR in order to prepare the future ALFRED consortium. MoU considers Romania as the reference option for the siting and acted to the construction of an international consortium for ALFRED implementation in Romania.

In December 2013 the consortium agreement for co-operation on the development of the Lead-Cooled Fast Reactor Demonstrator, named FALCON was signed by the three entities that signed MoU before. In 2014 CVR (Czech Republic) has joined the consortium.

Related to the location of ALFRED a reference option for the siting is considered consisting of the nuclear platform Mioveni, near Pitesti. The site benefits of the: electrical supply, including 110 kV electrical station; water supply; demineralized water; waste water treatment plant; radioactive waste facilities; fire brigade; heating plant; natural gas supply; physical protection; access road to transport large equipment; already investigated field (e.g. earthquake characterization). The site may benefit for the existing infrastructure, proximity of the INR departments, availability of the existing personnel for different activities. The choice will avoid some expenses with services and construction of buildings for these services.

The Roadmap of ALFRED demonstrator implementation is structured into 4 phases: (1) Viability (to investigate the feasibility, until 2018), (2) Preparatory (to perform siting, licensing, technical design, etc.) (2019-2023), (3) Construction (2023-2028), (4) Operation (after 2028). The Viability phase will be finalized by the release of Feasibility Study.

The main parameters of ALFRED are presented in Table 1.

Table 1. ALFRED – Main parameters

Items	**ALFRED Options**
Thermal Power (MWth)	300
Electrical Power (MW_e)	125
Primary Coolant	Pure Lead
Primary System	Pool type, Compact
Primary Coolant Circulation: Normal operation Emergency conditions	Forced Natural
Allowed maximum Lead velocity (m/s)	2
Core Inlet Temperature (°C)	400
Steam Generator Inlet Temperature (°C)	480
Secondary Coolant Cycle	Water-Superheated Steam
Feed-water Temperature (°C)	335
Steam Pressure (MPa)	18
Secondary system efficiency (%)	40
Maximum Structural material neutron Damage (dpa)	2
Fuel type	MOX (max Pu enr. 30%)
Maximum discharged burn-up (MWd/kg-HM)	90÷100
Maximum Clad Neutron Damage (dpa)	100
Maximum Clad Temperature in Normal Operation (°C)	550

The configuration of the ALFRED primary system is pool-type. It eliminates all problems related to out-of vessel circulation of the primary coolant. A simple flow path of the primary coolant with a Riser, Pump, Steam Generator, and a Downcomer is present allowing also an efficient natural circulation of the coolant. The Reactor Vessel is cylindrical with a toro-spherical bottom head. It is anchored to the reactor cavity from the top, by means of a vessel support (Figure 1).

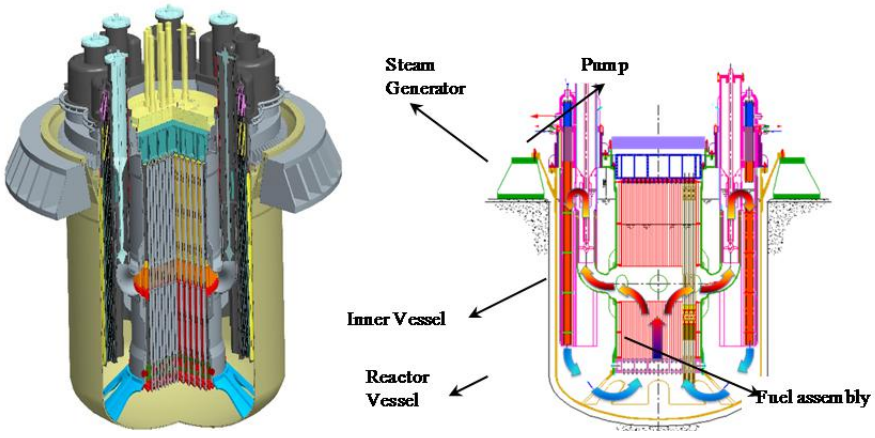


Figure 1. ALFRED 3-D Sketch and Reactor Block Vertical Sections

A steel layer covering the reactor pit, constitutes the Safety Vessel. The dimensions of gap between the safety vessel and the reactor vessel are sufficient for the In-service Inspection tools. The safety vessel is cooled by the same system that cools the concrete of cavity walls. This system is inserted inside the concrete and is independent from the reactor cooling systems. This design solution mitigates the consequences of through-wall cracks with leakage of lead: any reactor vessel leakage is discharged into the safety vessel.

The adopted core configuration of ALFRED [4] is constituted by wrapped Hexagonal Fuel Assemblies. The fuel is MOX type with hollow pellets and a low active height in order to improve the natural circulation. The core (Fig. 2) consist of 171 Fuel Assemblies (FAs), 12 CR (Control Rods) and 4 SR (Safety Rods), surrounded by 108 Dummy Elements (ZrO2-Y2O3) shielding the Inner Vessel [7].

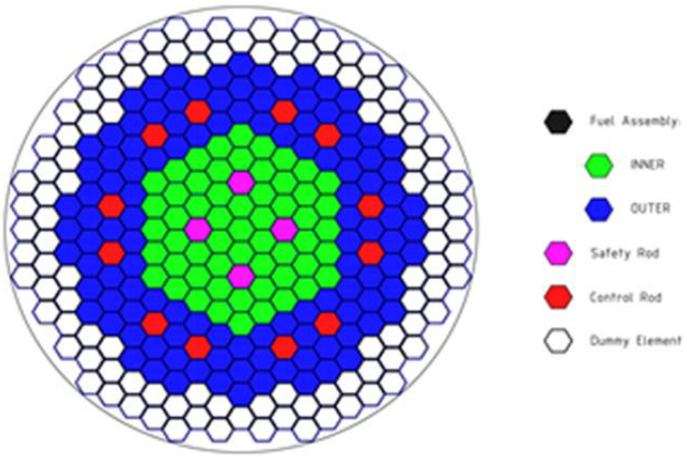


Figure 2. ALFRED core configuration

ALFRED is equipped with two diverse, redundant and separate shutdown systems: (1) Control Rod (CR) system, used for both normal control of the reactor (start-up, reactivity control during the fuel cycle and shutdown) and for SCRAM in case of emergency; (2) Safety Rod (SR) system, is the redundant and diversified complement to the control rods for SCRAM only. For both systems the materials considered are B4C enriched in 10B at 90% as absorber, T91 for the guide tube, 15-15 Ti for the clad and ZrO2 (95%) - Y2O3 (5%) for the insulator and reflector [7]. The CRs are extracted downward and rise up by buoyancy in case of SCRAM. The structure of CR consists of a 19 pins absorber bundle, cooled by the primary coolant flow. These pins are fitted with a gas plenum collecting the Helium and Tritium, produced by nuclear reaction of B10.

The absorber bundle for SR stays in the primary coolant. The rod is extracted upward and inserted downward against the buoyancy force. The absorber

gets inserted by the actuation of a pneumatic system. In case of loss of this system, a tungsten ballast will force the absorber down by gravity in a slow insertion.

The steam generator (SG) and primary pump (PP) are integrated into a single vertical unit. Each SG consists of a bundle of 542 bayonet tubes immersed in the lead vessel pool for six meters of their length. Eight SG/PP units are located in the annular space between the cylindrical inner vessel and the reactor vessel wall. The primary pump is placed in the hot side of the steam generator, having its mechanical suction in the hot pool inside the inner vessel. The primary coolant moves upward through the pump impeller to the vertical shaft, then enters the SG through the lead inlet holes, flows downwards on the shell and exits the steam generator. The pump motor is located above the reactor roof.

The Decay Heat Removal system (DHR) consists of two passive, redundant and independent systems, DHR1 and DHR 2, both composed of four Isolation Condenser systems (ICs) connected to four Steam Generators (SGs) secondary side (i.e. one IC for each SG).

The Secondary system proposed for ALFRED is based on a dual turbine configuration with three extractions in the HP turbine and three more in the LP turbine, with an axial outlet.

An auxiliary lead heating system is added. This system would work when the power cycle is not in operation, in order to ensure the minimum temperature of the lead by transmitting heat from the secondary system if it is needed.

4. LFR estimated performances in the new energy markets

The LFR technology development will definitely contribute to the improvement of sustainability of the nuclear development. A reduction of the total amount of radioactive wastes and their radio-toxicity at 1/10 of the present NPP is expected. This feature offers important reduction in the costs of the investment and operation of the disposals.

At the same time two important aspects form the point of view of the optimal use of resources can be discussed: (1) the possibility to burn the existing stocks of Plutonium (produced in the past for military purpose), (2) the use of U238 isotope to produce fissile isotopes. The second option will extend the actual Uranium reserves with more than 1000 years. The first option is a valuable path to reduce the costs for storing a critical material.

Due to the properties of the lead (it does not react with water or air, it has a high boiling point, it has a higher density than the oxide fuel, it is compatible with existing cladding materials such as 15-15/Ti and T91) some intrinsic safety features are present. The phenomenology does not involve paths to the severe accident initiations. The presence of passive systems to intervene in case of some transients is very important and enough large grace times are available for the action of the operators. There is no need for core catcher due to the fact the core melt is eliminated

by design. Also there is no risk of re-criticality in case of core damage due to the floating of the fuel elements in the lead coolant. There are large margins in terms of core voiding by a very improbable vaporization of the lead. On the other hand, lead is a low moderating medium and has low absorption cross-section leading to a hard neutron fast spectrum.

However some issues are introduced by LFR technology in terms of the resistance of the materials and coatings to the corrosion and erosion produced by a high density coolant, control of the oxygen content to minimize the effect of corrosion, seismic risks, lead freezing in case of accidental extraction of additional heat from the coolant, possible flow blockage due to deposition of corrosion/erosion products. All these aspects are treated as open issues by RDI activities and solutions are expected in the next years, including at industrial scale level.

Some important questions are raised in the context of new energy markets: (1) are the innovative systems able to compete with the most performing alternatives of energy production? (2) are the new systems able to drastically reduce the present difficulties of nuclear power? (3) are the new systems more adaptable to the exigencies of the new grids and future societies?

Some elements are approached in the current section. In terms of the expected improved performances, the reduction of the amount and radiotoxicity of radioactive wastes, reduction of the emergency planning zone, elimination by design of severe accidents, more efficient use fuel resources, and enhanced proliferation resistance are important aspects.

In term of economics it is expected a reduction of the costs based on standardization, simplification in design, and the general features of Small Modular Reactor (SMR). A secondary objective of ALFRED implementation is to demonstrate LFR capability to produce a SMR. Important efforts were devoted to the conceptual development of LFR SMR.

From the point of view of energy mix the new systems should be able to perform load following, rather than work predominantly in the base load regime. ALFRED is a fast system that is no restricted to re-start by the poisoning effect. On the other hand the materials and developed coatings are intended to cover a large temperature range as a basis for a good behavior at variations of the power level.

The route of the development of ALFRED and LFR technology towards the SMR deployment offers important advantages in terms of the flexibility due to the number of units and improved load following capabilities. This paper will not approach this obvious feature and it will be focused on load following based on H2 production.

One of the most important features is derived from the coolant temperature (around 5000C) and possibilities to use the usual lost heat (around 60% of the total power) as a resource for hydrogen production. Usually the High Temperature Splitting Electrolysis (HTSE) of water are optimal at 800-10000C, but some important progress was announced by AREVA, with HTSE at 5000C.

Two options are possible for load following by using H2 production: (1) produce H2, inject it into gas grid (power to gas) and use combustion turbines to reconvert into electricity, (2) produce H2 and sell it on the market for industry and transport.

In the first option the intention is to produce hydrogen in the period imposed by the grid to reduce the nuclear power level, and later to use the period of peaks to produce electricity and sell it at a greater price [8]. Taking into consideration the global efficiency of this process ($\varepsilon = 40\% - 45\%$) a very simple primary conditions between averaged low and high spot prices can be derived:

$$\varepsilon p_{high}^{ave} \geq p_{low}^{ave} \quad \text{(C1)}$$

On the other hand if the investment, operation and maintenance costs are considered the condition should be reformulated as:

$$\varepsilon p_{high}^{ave} - p_{low}^{ave} \geq \frac{c_{add}}{p_{nom} N_h^-} \quad \text{(C2)}$$

Where $\varepsilon p_{high}^{ave} - p_{low}^{ave} \geq \frac{c_{add}}{p_{nom} N_h^-}$ includes construction and investment, operation and maintenance and other additional costs, N_h^- is the total (equivalent at nominal power) number of hours of use of the electrolyser during its whole lifetime.

The condition (C1) can be rewritten as: $p_{high}^{ave} \geq 2.2 * p_{low}^{ave}$ if we consider the actual global conversion efficiency. On the other hand considering an investment price for electrolyser of 0.8-1.0 mil Euro/MW, a lifetime of 20 years, a discount rate of 7%, running costs at 5Euro/MWh, and a number of working hours for the electrolyser of 2000 h/year the condition (C2) may be written as: $p_{high}^{ave} \geq 2.2 * p_{low}^{ave} + 25$ (euro/MWh). The condition is more drastically if the number of annual working hours of the eletrolyser is reduced to 1000 h, or to 500 h ($p_{high}^{ave} \geq 2.2 * p_{low}^{ave} + 85$). This parameter is a critical one and the normal way to increase the hours for the electrolyser is to couple it with vRES in order to smooth the produced peaks.

Considering the large variability of price introduced by a large share of vRES in the grid it is possible to reach periods with very low or even negative spot prices (quite frequent in case of Germany). In Romania at 30th November 2017 (the windiest day) the average spot prices was reduced at around 10 Euro/MWh, compared with higher usual spot prices around 60 Euro/MWh. In such conditions there is enough room to compete on the load following market with the classical gas units used for peaking in the most countries with high share of vRES. For Romania this case may be a realistic approach if the RES targets for 2030 will be achieved.

Considering the second option (sell the H2 on existing market) it should be noted the price of 1 kg H2 (US market) is between 3 and 10 USD. The global efficiency of the process may be converted in the production of H2 per MWh that is estimated, based on the literature parameters, at 17.96 Kg/MWh. By using the same

values of C&I, O&M costs, lifetime, discount rate etc. for number of working hours for the electrolyser of 2000 h/year the price of produced H2 is estimated at 3.5 Euro/Kg. For a reduced functioning at 1000 h/y the price will be 4.9 Euro/Kg, and for 500 h/y the price becomes 7.5 Euro/Kg. Even in the last cases the price is competitive on the existing markets. It should be noted that all the estimations did not considered the value of de-carbonization by H2 production. In such case the performances of nuclear load following by H2 production are better.

However the cost of investment is high and on short term it is not expected a high appetite of the investors for electrolysis, except in case of implementation of policies with high penalties for CO2 emissions. Due to this fact, a public funds support for the development of the H2 infrastructure will be beneficial. After that, an efficient business for the de-carbonization of transport and de-carbonization of H2 production (used massively for ammonia industry by classical approach of steam reforming of the methane) may be developed.

ALFRED and LFR has good characteristics to go further in a modern era of nuclear power. At the same time its flexibility is great based on modular approach and range of temperature able to implement an efficient H2 production.

Finally it should be noted a great reserve for economic performance for LFR modular reactors consists of the use the wasted heat. Around 30% of the 60% wasted energy can be easily used for district heating or for industrial heat processes.

5. Conclusions

(C1) Romania expressed the availability to host ALFRED demonstrator. An international consortium was built in December 2013 by RATEN ICN, Ansaldo and ENEA considering Romania as the reference option for siting. CVR (Czech Republic) has joined the FALCON consortium in 2014. The Consortium acted to analyze the feasibility of the implementation, to pave the way for the preparatory activities, and to identify and capture the implementation resources.

(C2) The central objective of ALFRED implementation is to demonstrate the technical and economic viability of LFR technology towards the FOAK stage and commercial deployment, after 2050. A second objective is important from the point of view of the energy market reconfiguration: to demonstrate ALFRED capabilities to develop a SMR design.

(C3) In the context of the future energy markets due to the de-carbonization policies it is expected a more robust role of nuclear power. At the same time, nuclear will compete with a large development of RES and should offer important advantages (beyond the security of supply) from the point of view of economics.

(C4) New nuclear systems may be developed to offer more flexibility in the grids where some difficulties introduced by large share of renewables, especially more needs for peaking units, are present. Nuclear can contribute to the development of a sustainable hydrogen economy (clean, competitive, and based on water) and it can reduce the emissions due to peaking units based on coal/gas/oil.

(C5) On the other hand, the LFR offer more versatility by heat and electricity to drive different hydrogen production processes. It is clear than more deeply economic analysis are necessary to take into consideration the diverse aspects of the markets and to proof the competitiveness. However, based on the simplified analysis in certain conditions of high penetration of RES, a profitable nuclear H2 production may appear. At the same time, at least in the current predicted conditions, a need for global/national support for H2 infrastructure development is obvious.

(C6) Some synergies are identified between nuclear power and vRES development: (1) common use of H2 infrastructure, (2) RDI for optimal use of the wasted heat in nuclear and concentrated PV, (3) digitalization to an optimal use of local grids and resources.

(C7) District heating in EU New Member States remain an important reserve to improve the economic performances of the nuclear at may be revived based on the existing networks and the concentration of households in blocks of flats/quartiers. Such decisions will be based on a deeper analysis of the future energy market conditions.

REFERENCES

[1] SNETP Strategic Research and Innovation Agenda, February 2013;
[2] LEADER project, www.leader-FP7.eu;
[3] M. Frogheri, A. Alemberti, L. Mansani, "The Lead Fast Reactor: Demonstrator (ALFRED) and ELFR Design," International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, France (2013);
[4] G. Grasso, C. Petrovich, K. Mikityuk, D. Mattioli, F. Manni, D. Gugiu, "Demonstrating the effectiveness of the European LFR concept: the ALFRED core design," International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, France (2013);
[5] Deloitte study – "Funding Opportunities and Legal Status Options for the Future European Sustainable Nuclear Fission Industrial Initiative of the Strategic Energy Plan," EC, February, 2010;
[6] C. Artioli, G. Grasso, and C. Petrovich, "A new paradigm for core design aimed at the sustainability of nuclear energy: The solution of the extended equilibrium state," Ann. Nucl. En. 37(7):915-922 (2010);
[7] A. Alemberti, L. Mansani, G. Grasso, D. Mattioli, F. Roelofs, D. De Bruyn, "The European Lead Fast Reactor Strategy and the Roadmap for the Demonstrator ALFRED," International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, France (2013);
[8] M.Constantin, "Load Following based on Hydrogen and Synergies with vRES," The 2nd meeting of NEA/OECD expert group on Advanced Reactors in Future Energy Markets, Paris, 24-25 January 2018.

DESIGNING OF MAIN COMPONENTS OF THE SECONDARY CIRCUIT OF ALFRED REACTOR

DIMENSIONAREA PRINCIPALELOR COMPONENTE DIN CIRCUITUL SECUNDAR AL REACTORULUI ALFRED

Iulian NIŢĂ[1], Rodica PANCEF[2], Andrea IVAN[3]

Abstract: *The Advanced Lead Fast Reactor European Demonstrator is one of the main project to be built in Romania next to ELI-NP laser project. In order to be built, there are a great effort from Romanian and Italian researchers in order to design and built a new technology for Europe. In this paper we started from know data of steam generators feed water flowrate and main steam parameters and we obtained by iterative calculation all flowrates, pressure, thermal loads of all main equipment from secondary regenerative circuit of ALFRED. The main focus till this paper, research and design effort was focused on primary circuit, due to new provocations of using lead as cooling agent. Based on current research we can make big steps in procuring main components of secondary circuit of this important project who will be built in Romania at Mioveni.*

Keywords: ALFRED, Secondary Circuit, Heat balance

Rezumat: *Reactorul European de Demonstraţie Rapid Răcit cu Plumb va fi construit în România după construirea proiectului celui mai mare laser din lume proiectul ELI-NP. Pentru a putea fi construit, se face un efort considerabil din partea cercetătorilor români şi italieni pentru a proiecta şi constrai o nouă tehnologie în Europa. In această lucrare am pornit de la datele cunoscute despre generatoarele de abur şi anume debitul şi temperatura apei de intrare şi temperatura şi presiunea aburului viu şi am obţinut, prin calcul iterativ, toate debitele presiunile, sarcinile termica ale principalelor echipamente din circuitul secundar al reactorului demonstrativ ALFRED. Principala ţintă până la această lucrare au fost cercetarea si proiectarea circuitului primar de răcire a reactorului, datorită noilor provocări datorate de folosirea plumbului ca agent de răcire. Pe baza cercetării curente putem face mari progrese în procurarea componentelor principale din circuitului secundar al acestui proiect important ce va fi construit în Romania at Mioveni.*

Cuvinte cheie: ALFERD, Circuit secundar, bilanţ termic

[1] RATEN-CITON, e-mail: nitai@router.citon.ro
[2] RATEN-CITON, e-mail: pancefr@router.citon.ro
[3] RATEN-CITON, e-mail: ivana@router.citon.ro

1. Introduction

This paper has as main objectives indentifying and analysing of technical parameters in order to define a solution for secondary circuit diagram of ALFRED demonstrator reactor (Advanced Lead Fast Reactor European Demonstrator).

Nuclear energy based on fission reaction represent, in present, the main option of sustainable energy supply, clean and competitive to global scale.

Aware of the advantages and disadvantages of using nuclear energy, government and technical-scientific community problems related to nuclear safety, treatment and conditioning of radioactive waste, costs of investments in nuclear installations, physical protection of radioactive materials and nuclear facilities, resistance to proliferation.

Fast generation IV reactors are an acceptable and feasible solution with cleaner nuclear safety features that promise to prevent Fukushima-like consequences.

European Union and the major economic powers have set up a working group for the development of Generation IV reactors. Basic research on Generation IV reactors was conducted through a global collaboration effort coordinated by the Generation IV International Forum (GIF-Generation IV International Forum).

In order for Romania to adopt a modern generation of Generation IV nuclear reactors, it is imperative that we enter the club of countries that promote these branches in the research and experimental demonstration phase.

The design of the ALFRED demonstrator has so far been carried out under the 7th EURATOM Research Framework Program (European Atomic Energy Community) and Ansaldo Nucleare has been the leader of the ENEA integration in technological development.

The Nuclear Research Institute - I.C.N. Pitesti, participated in the design of ALFRED installation, starting with its first stages, contributing to the design of the core of the reactor and promoting the preliminary activities necessary for the approval of the project in Romania.

Reactors of IV generation are truly innovative technologies that combine maximum security with economic competitiveness. The technology is based on passive safety aspects that provide a "zero impact" outside the directly affected area, even for the worst accident scenarios.

Liquid-cooled reactors (sodium, lead and bismuth lead) are fast reactors, currently in the conceptual design stage in Europe.

The ALFRED (Advanced Lead Fast Reactor European Demonstrator) project is a reactor that is part of the Lead cooled Fast Reactor (LFR) line of the

fourth generation nuclear reactor, a potential characterized pipeline to deliver efficient, simple and robust concepts. The LFR system is considered the first place in sustainable development because it uses a long-lived active area, has a closed fuel cycle and has a high resistance to proliferation and physical protection.

The ALFRED reactor uses a spectrum of fast neutrons and a closed fuel cycle for efficient uranium conversion and good management of minor actinides.

Data related to the ALFRED demonstration reactor were extracted from the existing specialized documents on the internet and used as input data for the calculation of the secondary circuit.

2. Secondary Circuit Description

ALFRED Secondary Circuit Operation is based on the second principle of thermodynamics, according to which a cyclic thermal machine can only produce mechanical work if it is in contact with two heat sources: hot and cold.

With help of all the parameters, a heat balance was made on the regenerative preheating line, determining the flows extracted from the turbine and the flows flowing through the regenerative preheaters.

For the ALFRED secondary circuit scheme design, the structure of the regenerative preheater circuit is the following:

- 3 low pressure heaters (LPH);
- 1 deaerator;
- 3 high pressure heaters (HPH);
- 1 Feed Water Temperature Control Heater (FWTCH).

Starting from the general scheme of an intermediate overheating condensation group (see figure 1), by calculating the decomposition in the steam turbine, the thermodynamic parameters of the steam were determined at the characteristic points of the thermal cycle, which are on the decomposition path between the exit from the steam generator and the entry into the steam condenser. Then the calculation for the main condensate for the supply water and condenser cooling circuit continued in three distinct situations: nominal mode (ambient temperature 18 °C); Severe winter (ambient temperature -25 °C) and severe summer (ambient temperature 37 °C).

To achieve the purpose of the paper, the input data characteristic of ALFRED project was set, establishing the steam generator parameters (≈300 MWt) of steam in the secondary circuit. A general scheme of an intermediate overheating

condensation group was designed, calculating the power at the turbine generator terminals and the thermal cycle yield.

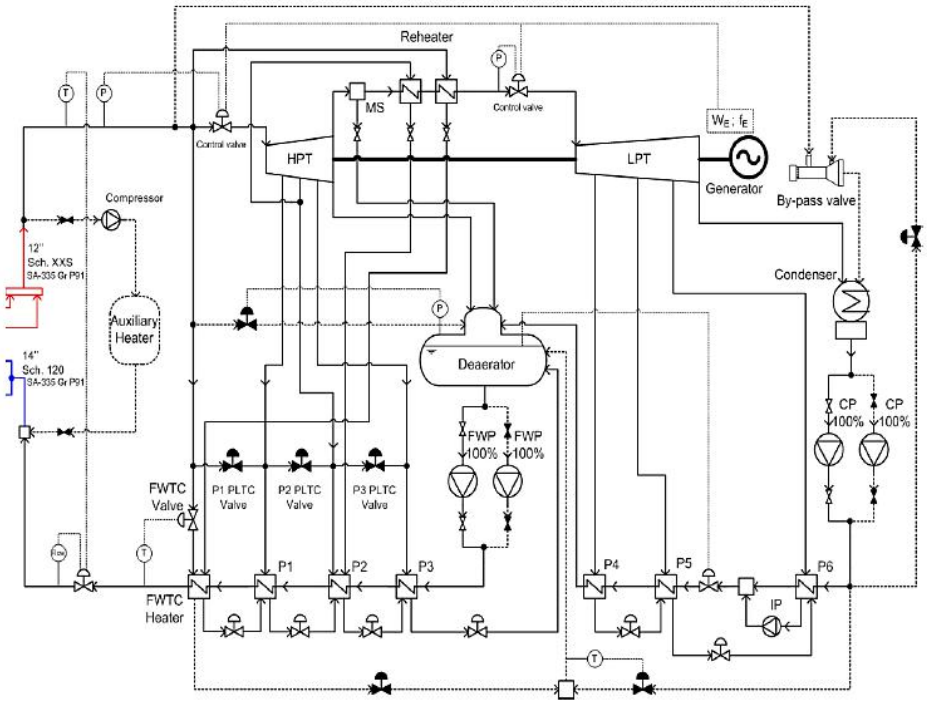


Figure 1. Secondary circuit diagram [ref. 1].

3. Obtained results

In the paper, the conceptual diagram of the secondary circuit for the ALFRED demonstration reactor [ref. 2 and 3] was drawn up and important parameters were calculated: pressure, flow, temperature, enthalpy at all characteristic points of the scheme. Interface with the requirements of the steam generator has been provided.

The secondary circuit diagram was calculated for three operating modes. Thus, in addition to the nominal exploitation regime (100% of the nominal load), the severe summer and severe winter regimes were also calculated. These calculations were made to determine the performance of the projected secondary circuit throughout the calendar year [ref. 4].

Following the calculation, the following results were obtained:

- For normal operation, the turbine generator output is 130.78 MW at a secondary circuit output of 43.92%;

- For the "extreme summer" operating mode, the turbine generator output is 126.6 MW at a secondary circuit efficiency of 42.52%- the computational diagram is showed in figure 2;

- For the "severe winter" operation, the turbine generator output is 132.1 MW at a secondary circuit efficiency of 44.38%- the computational diagram is showed in figure 3.

The internal report presents the results of the secondary circuit diagram parameters for all ALFRED thermal circuit components.

The results of steam expansion in turbine and the distribution of thermal loads for the regenerative circuit allow the dimensioning of these equipment in the next stage of the technical project.

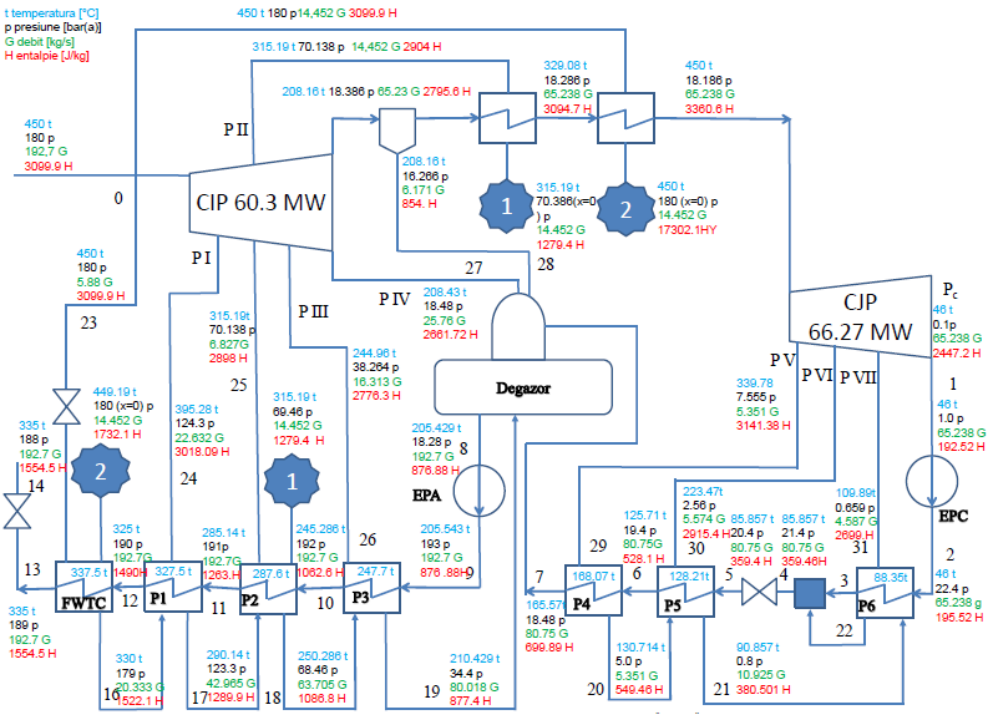


Figure 2. Heat Balance Diagram for one severe summer conditions

Because the ALFRED site does not allow cooling of the condenser in open circuit, closed-loop cooling solution is selected by means of a cooling towers battery that will discharge a thermal load of 147.1 MW in severe summer mode.

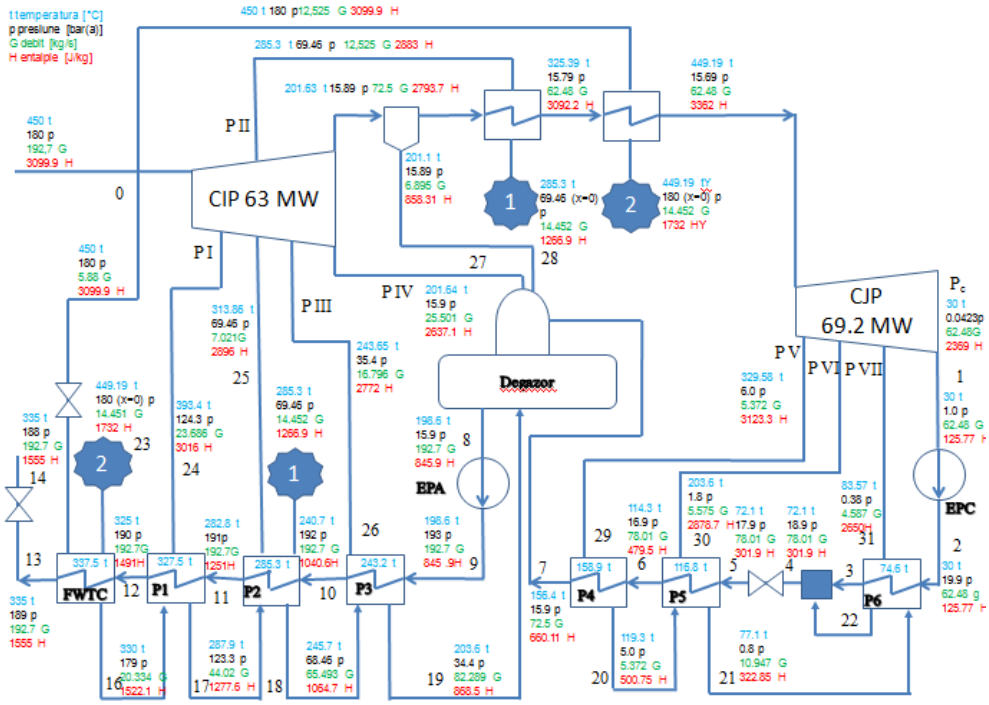


Figure 3. Heat Balance Diagram for one severe winter conditions

4. Conclusions

Nuclear power is the only energy source capable of meeting decarbonisation targets at the same time as supply security, being stable and not dependent on price fluctuations in fossil fuels. Nuclear power ensures the energy needs of mankind, it is a source of energy security because it brings a diversification of the energy mix; Through its carbon-free character, is part of the current decarbonisation trend in electricity production, fully responding to climate change prevention concerns.

The recent Communication on a new European Strategic Energy Technology Plan also states that the priority for nuclear energy is to support the development of the most advanced technologies in order to maintain the highest level of nuclear reactor security and to improve the efficiency of the operation, Processes from the final stage of the fuel cycle and decommissioning.

In this respect, the current EURATOM research initiative is to implement the European Industrial Initiative for Sustainable Nuclear Energy, which aims to prepare for the future deployment of the fourth generation nuclear power system, based on fast neutron fuel cycle technology . A number of reactors are in the research stage. These are: ALFRED; ALLEGRO; MYRRHA and ASTRID.

Fast generation IV reactors are an acceptable and feasible solution with clearly superior nuclear safety features that promise to prevent Fukushima-like consequences.

Generation IV reactors were designed to achieve a high level of security, proliferation resistance, low fuel consumption, high technical and economic performance, and to reduce the amount of radioactive waste generated.

The Russian experience with the deployment of LBE cooling systems for propulsion of submarines has provided excellent evidence that LFR reactors can be produced and operated on an industrial scale.

Fast lead-cooled reactors offer a great promise in terms of simplification of the boiler scheme, performance and safety response, while offering advantages over other fast reactors.

As a result of the calculation, good yields were obtained for the three studied operating regimes (nominal, severe summer, severe winter), which justifies the sizing of the secondary circuit of the ALFRED demonstration reactor.

The results obtained will be used to sizing all the main components of the ALFRED secondary circuit diagram, a big step forward in order to put in implementation a great project of generation IV reactor in Romania.

This paper is in the strategic direction of support activities for the development of its capability, competence and expertise in the field of LFR in order to strengthen the contribution to the ALFRED demonstration reactor.

In conclusion, Romania is the reference option for hosting the ALFRED demonstration plant of the fast-cooled lead reactor (LFR), based on the benefits of accessing structural funds, existing technical capabilities and nuclear expertise. At European level, the idea of building the ALFRED demonstrator (LFR) in the new Member States is agreed to reduce the imbalances in the development of the technological infrastructure.

REFERENCES

[1] *S. Lorenzi, A. Cammi, L. Luzzi, R. Ponciroli,* Politehnico di Milano, Department of Energy, Vienna, Austria, May 19-22, 2014, p. 35

[2] ALFRED – SECONDARY SYSTEM 3th LEADER international Workshop 2012 Pisa, 4th-5thSeptember 2012 ; Bologna, 6th – 7th September 2012.

[3] *A. Alemberti*, "The Lead fast reactor – Demonstrator (ALFRED) and ELFR design," International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, France (2013).

[4] Oak Ridge National Laboratories, "Thermophysical properties of saturated light and heavy water of advanced neutron source applications," 1993

CHEMICAL CLEANING OF SOLIDIFIED LEAD FROM 316 AUSTENITIC STEEL A STRUCTURE CANDIDATE MATERIAL FOR LEAD-COOLED FAST REACTORS

CURATAREA CHIMICĂ A PLUMBULUI SOLIDIFICAT DE PE OȚELUL AUSTENITIC 316 CANDIDAT PENTRU REACTORII RAPIZI RĂCIȚI CU PLUMB

Daniel PETRESCU[1], Lucian VELCIU[2]

Abstract: *The aim of this study is to find a specific cleaning solution that removes solidified lead from a candidate material for Lead-cooled Fast Reactors (LFR) without damaging it.*

Among the promising reactor technologies being considered by the Generation IV International Forum (GIF), the Lead-cooled Fast Reactor has been identified as a technology with great potential, because this system can fulfill simultaneous several key objectives like sustainability, non-proliferation, economics and safety. Some of these goals are achieved thanks to the favourable inherent properties of lead. With his very high boiling point, low vapour pressure, high shielding capability, high heat transfer, high density, small neutron absorption cross-section, good compatibility with air and water, lead permits to extend the viability domain for reactor design, opening new possibilities for a greatly simplified system configuration.

In service inspection and repair (ISIR) is an important issue in the development of the next generation nuclear systems as it contributes greatly to the safety of the system. For the inspection of components removed from the reactor vessel, it is very important to be able to clean the structural material from residual lead without damaging it. Therefore, in this paper, several chemical cleaning solutions were studied. In order to remove residual lead from the surface of candidate materials for LFR, first were tested 316 stainless steel specimens in liquid lead at 450 ^{0}C for 48 hours.

Among the tested solutions that proved their efficiency for lead removal were several mixtures based on acetic acid, hydrogen peroxide, and alcohol (ethanol and propanol). All of these were tested under two temperature conditions: 20 ^{0}C and 75 ^{0}C. The cleaning speed was significantly faster for the mixtures that had a higher temperature. Other solutions that were totally ineffective or had affected the austenitic steel and the protective oxide layer are nitric acid, aqua regia and the electrolytic solution of fluoboric acid.

A series of analyses were performed on all austenitic steel specimens after each immersion period in the cleaning solutions: microscopic, gravimetric and Vickers microhardness. These analyses showed that the solutions based on organic acids were more appropriate for removing residual lead from the 316 austenitic steels.

Keywords: lead-cooled fast reactor, chemical cleaning, austenitic steel

[1] RATEN Institute for Nuclear Research Pitesti, Romania, e-mail: daniel.petrescu@nuclear.ro
[2] RATEN Institute for Nuclear Research Pitesti, Romania, e-mail: lucian.velciu@nuclear.ro

Rezumat: *Scopul lucrari este acela de a identifica o solutie de curatare specifica pentru inlaturarea plumbului solidificat de pe un material candidat pentru Reactorii Rapizi raciti cu Plumb (LFR) fara deteriorarea acestuia.*

Printre tehnologiile promitatoare de reactoare care au fost luate in considerare de Forumul International de Generatie IV (GIF), LFR a fost identificat ca o tehnologie cu un mare potential, deoarece acest sistem poate indeplini simultan mai multe obiective cheie, cum ar fi durabilitatea, neproliferarea, economia si siguranta. Unele dintre aceste obiective sunt realizate datorita proprietatilor inerente favorabile ale plumbului. Cu punctul de fierbere foarte ridicat, presiunea scazuta a vaporilor, capacitatea de ecranare ridicata, transferul de caldura ridicat, densitatea mare, sectiune scazuta de absorbtie a neutronilor si compatibilitatea buna cu aerul si apa, face ca plumbul sa extinda domeniul viabilitatii pentru proiectarea reactorului, pentru o configurație a sistemului foarte simplificata.

Inspectia si reparatia sunt niste probleme importante pentru dezvoltarea sistemelor nucleare de generatie noua, deoarece contribuie in mare masura la siguranta intregului sistem. Pentru inspectia componentelor scoase din vasul reactoarului este foarte important sa se poata curata materialul structural de plumbul rezidual fara ca acesta sa fie afectat. Prin urmare, in aceasta lucrare au fost studiate cateva solutii chimice de curatare. Pentru a indeparta plumbul rezidual de pe suprafata materialelor candidate pentru reactoarele LFR, s-au testat mai intai probe de otel inoxidabil 316 in plumb lichid la 450 °C timp de 48 de ore.

Dintre solutiile testate care si-au dovedit eficienta pentru inlaturarea plumbului au fost cateva amestecuri pe baza de acid acetic, apa oxigenata si alcool (etanol si propanol). Toate acestea au fost testate in doua conditii de temperatura: 20 ^{0}C si 75 ^{0}C. Viteza de curatare a fost semnificativ mai rapida pentru amestecurile care au avut temperatura mai mare. Alte solutii care au fost testate, dar s-au dovedit a fi total ineficiente sau au afectat otelul austenitic si stratul oxidic protector sunt acidul azotic, apa regala si solutia electrolitica de acid tetrafluoroboric.

S-au efectuat o serie de analize pe toate probele de otel austenitic dupa fiecare perioada de imersie in solutiile de curatare chimica: microscopice, gravimetrice si microduritate Vickers. Analizele au aratat ca amestecurile de solutii cu acizi organici au fost mai potrivite pentru indepartarea plumbului de pe otelurile austenitice 316.

Cuvinte cheie: reactor rapid racit cu plumb, curatare chimica, otel austenitic

1. Introduction

High-level radioactive waste disposal is an issue of great importance in the discussion of the sustainability of nuclear power generation. The main contributors to the high radioactivity are the fission products and the minor actinides. The long-lived fission products and minor actinides set severe demands on the arrangements for safe waste disposal [1].

Lead-cooled Fast Reactors (LFRs) and Accelerator Driven Systems (ADS) are studied in Member States to reduce the long-term hazard of spent fuel and

radioactive waste, taking advantage of their incineration and transmutation capability. The conceptual design of lead/lead-bismuth cooled fast reactors have been developed to meet enhanced safety and non-proliferation requirements, aiming at both energy production and transmutation of nuclear waste. Some R&D studies indicate that the use of lead and lead-bismuth coolant has some advantages in comparison with existing sodium-cooled fast reactor systems, e.g.: simplified design of fast reactor core, enhanced inherent safety, and easier radwaste management in related fuel cycles [2].

In Figure 1 are presented the conceptual designs for MYRRHA (Multi-purpose hybrid Research Reactor for High-tech Applications), an accelerator driven system developed by SCK-CEN, Belgium, which uses lead-bismuth eutectic (LBE) as coolant, and ALFRED (Advanced Lead-cooled Fast Reactor European Demonstrator) designed by FALCON consortium and planned to be built in Mioveni, Romania.

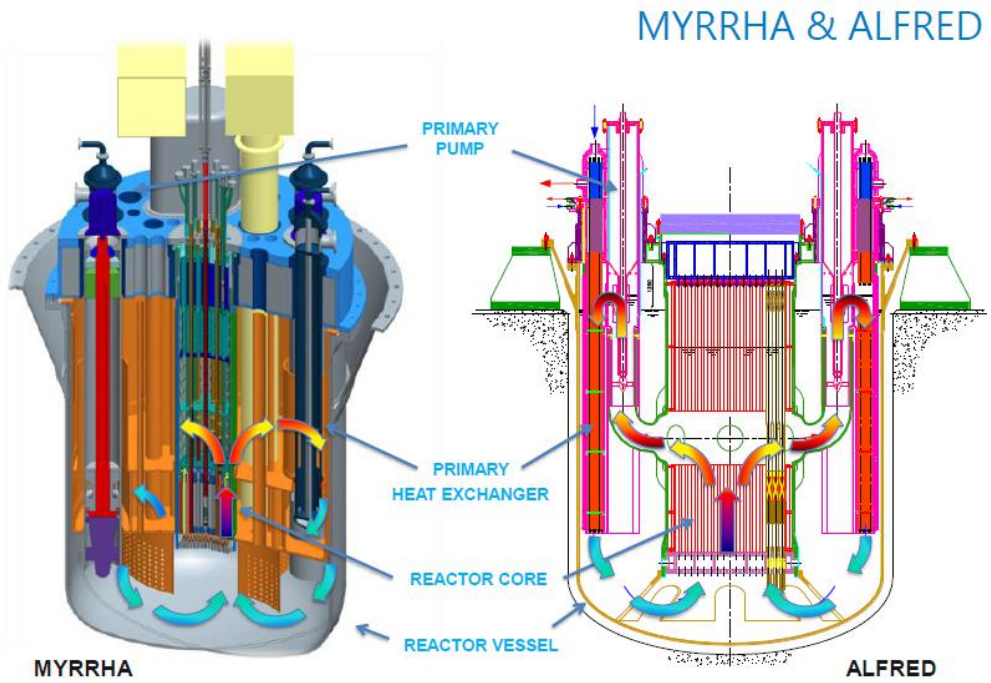


Figure 1. MYRRHA and ALFRED system designs [3]

The main features of these two reactor concepts, which are cooled with heavy metals, are presented in table 1.

Table 1. Main characteristics for MYRRHA and ALFRED reactors [3]

	MYRRHA v1.6	ALFRED
Thermal power	100 MWth	300 MWth
Primary coolant	Lead Bismuth Eutectic	Lead
Primary circulation		
Normal operation	Forced (2 mechanical pumps)	Forced (8 mechanical pumps)
Accident conditions	Natural	Natural
Primary system temperature	270 ÷ 325 °C	400 ÷ 480 °C
Primary system flow rate	13800 kg/s	26000 kg/s
Secondary system coolant	Water/Steam	Water/Superheated-Steam
Secondary system pressure	1,6 MPa	18 MPa
Secondary system temperature	200 °C	450 °C
Residual heat removal systems	DHR system, 4 independant loops	2 DHR systems, 4 loops each
	Passive	Passive
Vessel diameter	10,4 m	8 m
Vessel height	12,56 m	10,13 m

2. Experimental

For the inspection of components removed from the reactor vessel, it is very important to be able to clean the structural material from residual lead without damaging it. Therefore, in this paper, several chemical cleaning solutions were studied. In order to remove residual lead from the surface of candidate materials for Lead-cooled Fast Reactors, first were tested 316 stainless steel specimens in liquid lead at 450 ^{0}C for 48 hours. The visual aspect of specimens, before and after testing in liquid lead, it is presented in Figure 2.

a)

b)

Figure 2. 316 stainless steel specimens a) before testing in liquid lead b) after testing in liquid lead

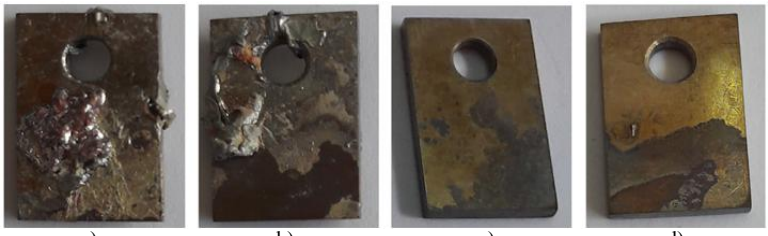
After 48 hours of immersion in liquid lead, all the austenitic steel samples were cooled in air and then sent to analytical balance to see the weight gain. Forwards, several organic and inorganic mixtures were prepared in order to remove solidified lead from the specimens. Table 2 presents the tested solutions for each specimen in different temperature conditions.

Table 2. Solutions tested in different condition for lead removing from the specimens

Solution	Testing condition	Specimen number
Hydrogen peroxide + acetic acid	20 ^{0}C	4
Hydrogen peroxide + acetic acid + propanol	20 ^{0}C	3
Hydrogen peroxide + acetic acid + ethanol	20 ^{0}C	5
Hydrogen peroxide + acetic acid	75 ^{0}C	2
Hydrogen peroxide + acetic acid + propanol	75 ^{0}C	7
Hydrogen peroxide + acetic acid + ethanol	75 ^{0}C	9
Azotic acid	75 ^{0}C	1
Aqua regia	75 ^{0}C	8
Fluoboric acid	20 ^{0}C, 10 V	11

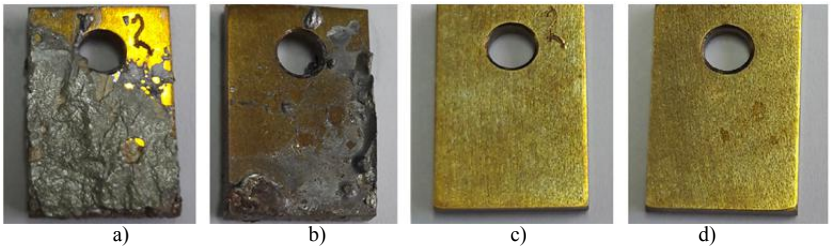
3. Results and discussion

The mixture of hydrogen peroxide and acetic acid (Sol.1) was tested at 20 ^{0}C on specimen 4 and at 75 ^{0}C on specimen 2. Figures 3 and 4 present some visual aspects of specimens 4 and 2 before and after the cleaning test.



a) b) c) d)

Figure 3. Specimen 4 a), b) front-back before cleaning test c), d) front-back after cleaning test for 15 minutes in Sol.1 at 20 ^{0}C



a) b) c) d)

Figure 4. Specimen 2 a), b) front-back before cleaning test c), d) front-back after cleaning test for 3 minutes in Sol.1 at 75 ^{0}C

Almost 99,85% of residual lead was removed from the sample 4 after 15 minutes of immersion in Sol. 1 at 20 ^{0}C and more than 99,99 % of lead from the sample 2 after only 3 minutes of sinking in the same solution but at 75 ^{0}C.

A solution made with hydrogen peroxide, acetic acid, and propanol (Sol.2) was tested at 20^{0}C on specimen 3 and at 75 ^{0}C on specimen 7. The visual aspects of the samples, before and after cleaning, are presented in Figure 5 and Figure 6.

Figure 5. Specimen 3 a), b) front-back before cleaning test c), d) front-back after cleaning test for 15 minutes in Sol.2 at 20 ^{0}C

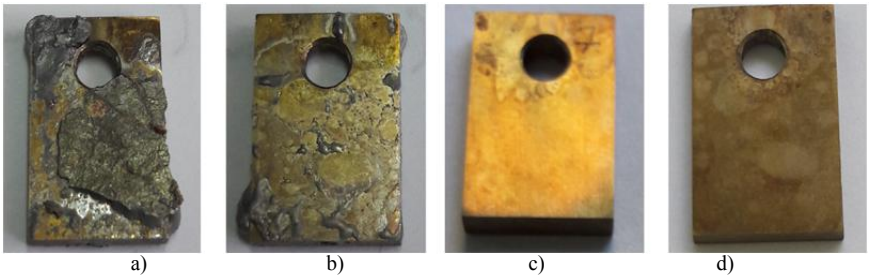


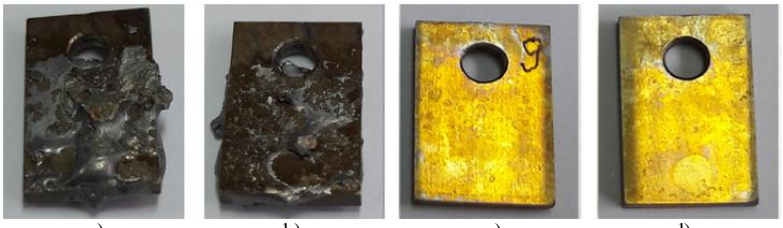
Figure 6. Specimen 7 a), b) front-back before cleaning test c), d) front-back after cleaning test for 1 minute in Sol.2 at 75 ^{0}C

Specimen 3 had lost ~83,78 % of the lead after about 5 minutes of immersion in Sol 2 at 20^{0}C and almost ~99,99 % after another 10 minutes of immersion. The percentage of residual lead removed in the case of specimen 7, tested in the same solution but at 75 ^{0}C, was about 99,95 % after only 1 minute of immersion. The bright orange color that can be seen in figures 5 c), d) and 6 c), d) is a thin protective oxide layer formed on the surface of the specimens after testing in liquid lead. This film was not affected by the cleaning solution.

Another solution that was tested at 20 ^{0}C on specimen 5 and at 75 ^{0}C on specimen 9 was obtained by mixing hydrogen peroxide, acetic acid, and ethanol (Sol. 3). In Figures 7 and 8 are presented the visual aspects of these specimens after the cleaning tests for lead removal with Sol. 3.

a) b) c) d)

Figure 7. Specimen 5 a), b) front-back before cleaning test c), d) front-back after cleaning test for 15 minutes in Sol.3 at 20 ^{0}C



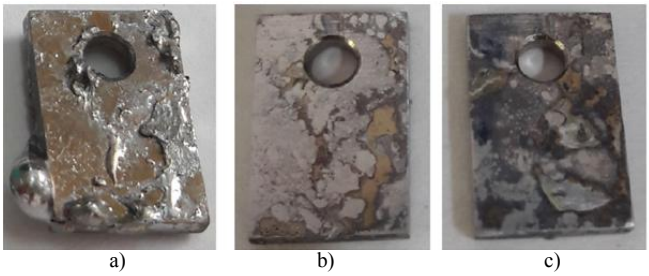
a) b) c) d)

Figure 8. Specimen 9 a), b) front-back before cleaning test c), d) front-back after cleaning test for 3 minutes in Sol.3 at 75 ^{0}C

Almost 85,81% of residual lead was removed from specimen 5 after 15 minutes of immersion in Sol. 3 at 20 ^{0}C and about 99,97 % of lead was dissolved from specimen 9 after only 90 seconds of sinking in the same solution but at 75 ^{0}C. After another 90 seconds of immersion at 75 ^{0}C, the percentage of lead removed from specimen 9 was about 99,99 %.

Specimens 5 and 9 presented large drops of solidified lead on their surface. In this case, the lead was not distributed uniformly and the contact surface area with the cleaning solution was reduced, which resulted in an increase of dissolution time.

A separate series of solutions that were tested for lead removal contained inorganic acids. Thus, sample 1 was kept in a nitric acid solution (Sol.4) and its visual aspects, before and after immersion, are shown in Figure 9.



a) b) c)

Figure 9. Specimen 1 a) before cleaning test b), c) front-back after cleaning test for 15 minutes in Sol.4 at 75 ^{0}C

Specimen 1 was immersed in nitric acid at 75 ^{0}C for about 15 minutes. After that, it had lost theoretically 96.07% of solidified lead. As the sample does not look good visually (see Figure 9 b), c)), the mass loss may have occurred from the base material. Also, the protective oxide layer formed on the material surface was affected.

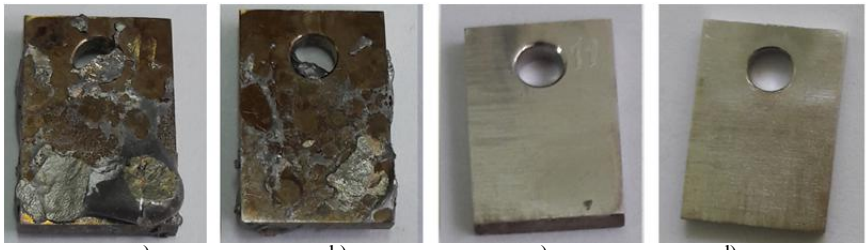
Aqua regia (Sol.5) was another cleaning solution which was tested at 75 ^{0}C for lead dissolving. The visual aspects of specimen 8 covered with lead and after the cleaning procedure in Sol.5 are shown in Figure 10.

a) b) c) d)

Figure 10. Specimen 8 a), b) front-back before cleaning test c), d) front-back after cleaning test for 5 minutes in Sol.5 at 75 ^{0}C

After 5 minutes of immersion in aqua regia at 75 ^{0}C, sample 8 underwent a significant degradation. Sol. 5 dissolved both lead and 316 austenitic stainless steel. The degradation of the base material was so strong that it lost 0.5669 grams from its initial weight of 2.0463 grams. The protective oxide film was also destroyed.

The last solution tested for lead removal was fluoboric acid (Sol. 6). The cleaning method involved the electrodeposition of solid lead from the anode on a copper cathode using an electrolytic solution (HBF_4) at a potential of 10 V. Specimen 11 was tested by this method and the visual aspects before and after cleaning are shown in Figure 11.



a) b) c) d)

Figure 11. Specimen 11 a), b) front-back before cleaning test c), d) front-back after cleaning test for 5 minutes in Sol.6 at 10 V

As can be seen in Figure 11 c) and d), the solidified lead from sample 11 was removed. However, according to gravimetric analysis, the austenitic steel

suffered a slight weight loss, and the protective oxide film was removed. It is possible that the 10 V potential has been a little too high and thus has affected the oxide layer.

Micrographs of specimens 4 and 2, before and after cleaning test in Sol. 1, are presented in Figures 12 (A, B) and 13 (A, B). These pictures show that solidified lead was successfully removed from specimen surfaces. The micro-indentations for Vickers hardness calculation, which were made in cross-sections of the samples, near the protective oxide layer, are shown in Figure 12 (C) for sample 4 and in Figure 13 (C) for sample 2. The mean microhardness value for specimen 4 was 183 Kgf/mm^2 and for specimen 2 was 189 Kgf/mm^2.

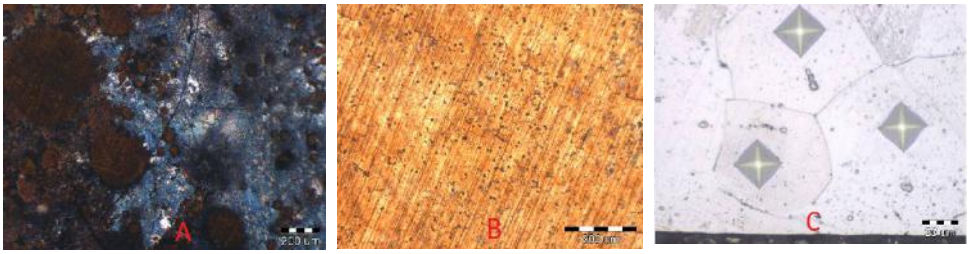


Figure 12. Specimen 4 A) covered with lead B) after cleaning test at 20 ^{0}C in Sol. 1 C) Vickers microhardness after cleaning test

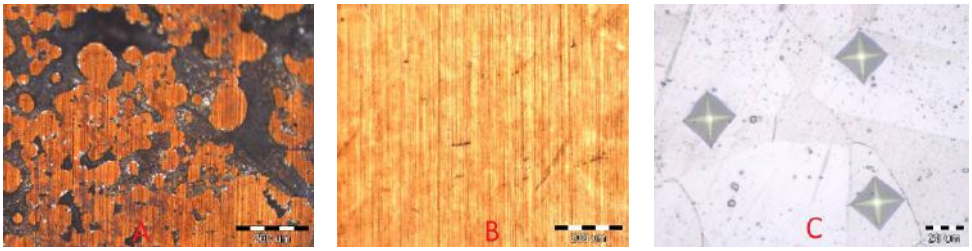


Figure 13. Specimen 2 A) covered with lead B) after cleaning test at 75 ^{0}C in Sol. 1 C) Vickers microhardness after cleaning test

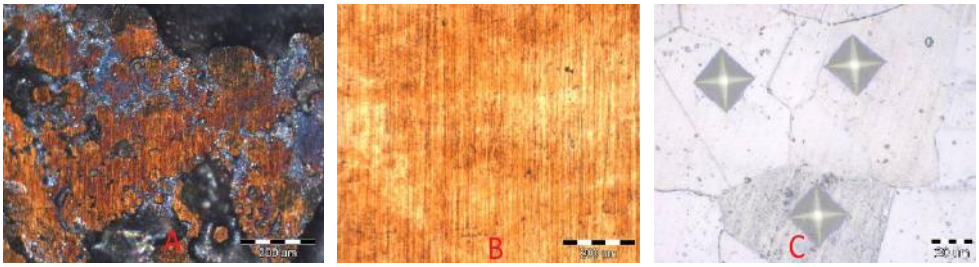


Figure 14. Specimen 3 A) covered with lead B) after cleaning test at 20 ^{0}C in Sol. 2 C) Vickers microhardness after cleaning test

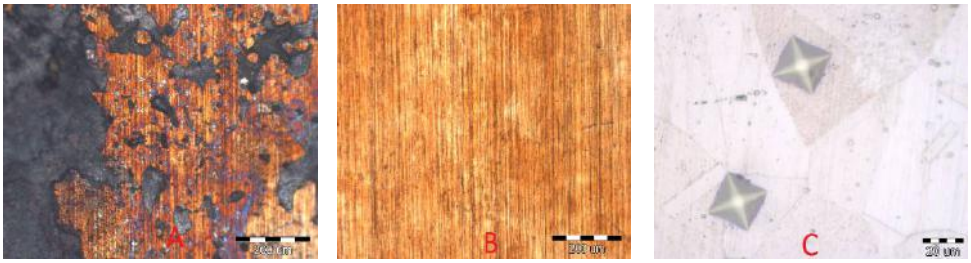


Figure 15. Specimen 7 A) covered with lead B) after cleaning test at 75 ^{0}C in Sol. 2 C) Vickers microhardness after cleaning test

Figures 14 (A, B) and 15 (A, B) presents some micrographs of specimens 3 and 7 before and after cleaning test in Sol. 2. These pictures show that Sol. 2 was also efficient for lead removal from the surface of both samples. Microhardness measurements showed values of 183 Kgf/mm^2 for sample 3 and 187 Kgf/mm^2 for sample 7.

As can be seen in Figures 16 (A, B) and 17 (A, B), Sol. 3 removed almost all the lead from the samples without affecting the protective oxide layer from the stainless steel. The Vickers microhardness for specimens 5 and 9 was 181 Kgf/mm^2 respectively 186 Kgf/mm^2.

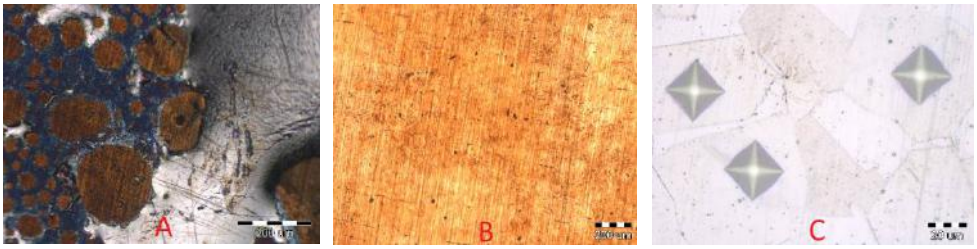


Figure 16. Specimen 5 A) covered with lead B) after cleaning test at 20 ^{0}C in Sol. 3 C) Vickers microhardness after cleaning test

Figure 17. Specimen 9 A) covered with lead B) after cleaning test at 75 ^{0}C in Sol. 3 C) Vickers microhardness after cleaning test

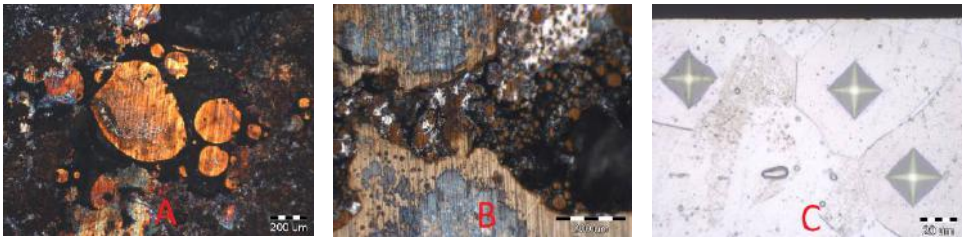


Figure 18. Specimen 1 A) covered with lead B) after cleaning test at 75 ^{0}C in Sol. 4 C) Vickers microhardness after cleaning test

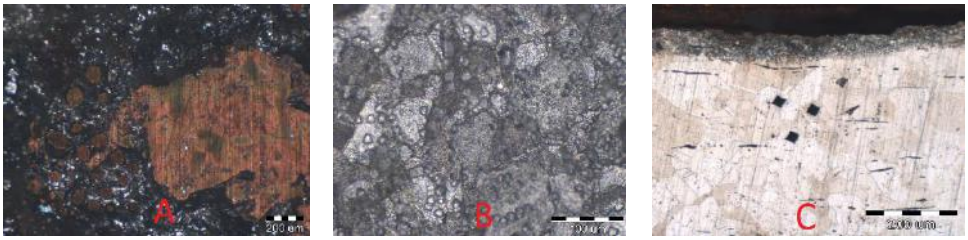


Figure 19. Specimen 8 A) covered with lead B) after cleaning test at 75 ^{0}C in Sol. 5 C) Vickers microhardness after cleaning test

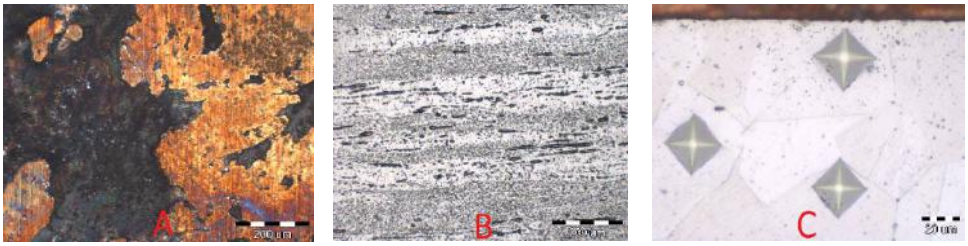


Figure 20. Specimen 11 A) covered with lead B) after cleaning test at 10 V in Sol. 6 C) Vickers microhardness after cleaning test

As seen in the micrographs of Figures 18, 19 and 20, solution 4 failed to remove the lead from sample 1, solution 5 dissolved both the lead and the base material of sample 8, and solution 6 removed the lead but also the protective oxide layer from sample 11. The calculated microhardness values were 186 Kgf/mm^2 for specimen 1, 189 Kgf/mm^2 for specimen 8 and 184 Kgf/mm^2 for specimen 11.

4. Conclusions

- After testing 316 austenitic steel in liquid lead for 48 hours, a protective oxide layer was formed on its surface;
- Chemical solutions based on hydrogen peroxide, acetic acid, and alcohol (ethanol/propanol) were able to remove almost completely the lead from the specimens without affecting the protective oxide layer or the steel structure;

- The speed for lead removal from the samples was significantly influenced by the temperature. Thus, the same solutions that cleaned the lead from the samples within 15 minutes at 20 °C were able to remove the lead in just 1-3 minutes at 75 °C;
- The volume of solidified lead droplets on the surface of the material influences the rate at which the solution manages to remove the lead. When the drops are too big, the contact surface of the lead with the cleaning solution is small and the cleaning speed decreases.
- Some chemical cleaning solutions like aqua regia, nitric acid and fluoboric acid were ineffective for lead removal or have damaged the protective oxide layer from the surface of the steel.

REFERENCES

[1] NEA/OECD, "Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermal-hydraulics and Technologies," 2007;
[2] TECDOC-1348, „Power reactors and sub-critical blanket systems with lead and lead–bismuth as coolant and/or target material," IAEA, 2003;
[3] *Tichelen K.,* „Thermal hydraulic phenomena in HLM pools," Heavy Metal Summer School, Belgium, June 2016.

OXYGEN CONTROL IN LEAD COOLED SYSTEMS VIA SOLID PHASE – PRELIMINARY TESTS FOR DEVELOPING PbO MASS ECHANGERS

CONTROLUL OXIGENULUI ÎN SISTEMELE RĂCITE CU PLUMB, PRIN METODA FAZEI SOLIDE – TESTE PRELIMINARII PENTRU DEZVOLTAREA SCHIMBĂTORILOR DE MASĂ DIN PbO

Irina PARASCHIV[1], Dorel BENGA[2]

Abstract: *Heavy liquid metals such as lead or its alloy lead-bismuth eutectic (LBE) are candidate materials for the coolant agent of Generation IV nuclear reactors. One of the most important issue in using these materials is mantaining an optimum oxigen concentration, in order to avoid the formation of undesired oxides, thus influencing the corrosion of structural materials.*

There are two methods used for controlling the concentration of dissolved oxygen: one based on the gas/liquid equilibrium between the cover gas and the liquid bulk when the liquid is below saturation and the other based on the dissolution of PbO, in solid form. The last method is known as "solid phase control with mass exchangers" and it consists in controlled dissolution of PbO compacts, placed in a special reaction vessel through which passes the coolant, compacts thus dissolving the oxygen, which is further transported in the cooling circuit, by the coolant agent.

In this paper, preliminary technological studies were carried out in order to develop and establish a technology for obtaining PbO compacts with different geometries, which can be used as mass exchangers for controlling the oxigen concentration. The method used for obtaining these compacts is based on the operations from ceramic powder technology. After different operations, PbO compacts with mechanical strength were obtained, in three different shapes: oval, parallelepiped and cylindrical, which were dimensionally characterized.

Keywords: mass exchangers, PbO, oxygen control, LBE

Rezumat: *Metalele grele în stare lichidă, precum plumbul sau aliajul său Pb-Bi (LBE) sunt materiale candidate pentru a fi folosite drept agent de răcire al reactorilor nucleari de Generație IV. Una dintre cele mai importante probleme în utilizarea acestor materiale este menținerea unei concentrații optime de oxigen, astfel încât să se evite formarea unor oxizi nedoriți, care să influențeze coroziunea materialelor structurale.*

Se folosesc două metode pentru controlul concentrației de oxigen dizolvat: prima este metoda bazată pe echilibrul gaz-lichid iar cea de-a doua metodă este bazată pe

[1] Institute for Nuclear Research, Arges, e-mail: irina.paraschiv@nuclear.ro
[2] Institute for Nuclear Research, Arges, e-mail: dorel.benga@nuclear.ro

disoluția PbO, în formă solidă. Cea din urmă metodă este cunoscută sub denumirea de "control în fază solidă cu schimbători de masă" și constă în disoluția controlată a unor compacte din PbO, plasate într-un vas special de reacție, prin care circulă agentul de răcire, compactele dizolvând astfel oxigenul, care este transportat mai departe în circuitul de răcire.

Pentru această lucrare, au fost derulate studii tehnologice preliminarii, pentru dezvoltarea și stabilirea unei tehnologii în vederea obținerii unor compacte din PbO cu diferite geometrii, care să poată fi folosite drept schimbători de masă pentru controlul concentrației de oxigen. Metoda folosită pentru obținerea acestor compacte este bazată pe operațiile din tehnologia pulberilor ceramice. După diferite operații, au fost obținute compacte din PbO cu rezistență mecanică, în trei forme diferite: ovale, paralele și cilindrice, care au fost apoi caracterizate dimensional.

Cuvinte cheie: schimbători de masă, PbO, controlul oxigenului, Pb-Bi

1. Introduction

The importance of oxygen control in Pb/LBE systems. Liquid lead and its alloy lead-bismuth eutectic (LBE) are considered at present as potential candidates for the coolant material of Generation IV fast reactors (critical and subcritical) and for liquid spallation neutron sources and accelerated driven systems (ADS). This choice is based on the advantages of these materials: good neutron transparency, high spallation neutron yield, good ɣ shieldieng, high thermal conductivity, low vapor pressure, low melting point and high boiling point, inert on water and air. However, there are some important aspects that needs to be taken into consideration. One of them is represented by the solubility of the structural materials in liquid lead/LBE, the corrosion of the steels and the coolant chemistry control.

The development of heavy liquid metal (HLM) chemistry control and monitoring is one of the critical issues for nuclear systems using lead alloys either as a spallation target or as a coolant in the reactor primary circuit of a critical or sub-critical system. Thus, for the safe operation of an HLM nuclear system, the chemistry control and monitoring is a critical issue for at least three distinct requirements:

- contamination, the assurance of a stable hydrodynamics and heat transfer during service lifetime requires avoiding of PbO formation. Lead oxide production may result in plugging due to the mass transfer in a non-isothermal system. Also, the appearance of some deposits of other contaminants may eventually reduce the overall heat transfer capacity, etc.
- corrosion and/or dissolution must be kept to a minimum rate in order to ensure a sufficient resistance of the structural materials during the service lifetime; this might require the use of an active oxygen control system for ensuring and maintain a protective film.

- activation due to corrosion, spallation and fission products might require liquid metal specific control in order to ensure the safe management of the operation and maintenance phases.

The accomplishment of these requirements makes chemistry control an essential element of nuclear system operation: control of oxygen and other relevant impurities including corrosion products, spallation and activation products. For this reason, control processes, in conjunction with monitoring systems, must be developed and/or qualified for application to an ADS system for both the coolant loop (refereeing to the primary circuit), for the spallation target loop and for the primary circuit in any critical system.

Oxygen is the most important chemical element for the management of Pb or LBE, due to its potential to contaminate the liquid by forming solid oxides and its influence on corroding structural materials of the reactor. Thus, the concentration of dissolved oxygen is the most important chemical aspect in lead or LBE systems and it must be adjusted to a specific value in order to control corrosion and to avoid the formation of Pb oxides (especially PbO), for the entire service lifetime of the reactor.

In this sense, two limits for oxygen concentration are defined:
- upper limit, for avoiding the contamination of the coolant agent with oxides;
- lower limit, to enhance the corrosion protection, by the protective layer of oxides (which depends on the structural materials). [1]

As it can be observed from Figure 1 [2], the superior limit corresponds to the solubility limit of oxygen in the liquid metal, to avoid the formation of PbO; this limit is giving the maximum allowable oxygen chemical activity in the liquid metal. The inferior limit corresponds to the dissolution limit below which the magnetite (Fe3O4) layer is dissolved and consequently doesn't protect the structural metallic materials against corrosion.

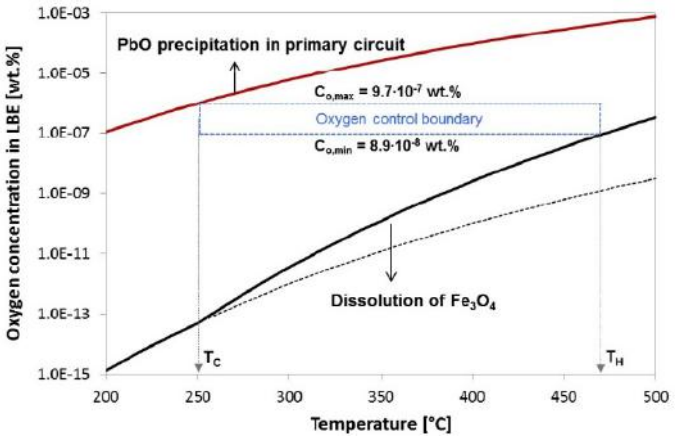


Figure 1. Oxygen specifications in LBE

When the oxygen concentration is too low for a stable oxide film, dissolution of metal occurs. At high oxygen concentrations rapid oxidation occurs which results in degradation of the structure and/or the formation of Pb oxides. Between these two extremes, there is a transition region where the kinetics transitions between dissolution and oxidation and the overall reaction rate is kept very small.

2. Methods for controlling the oxygen concentration in Pb/LBE systems

The oxygen control is a basic requirement for nuclear relevant system, for which long service lifetime is expected. A requirement for the oxygen control systems consists in an active oxygen control for promoting a protective oxide film formation on the structure by controlling the oxygen potential in the liquid metal.

For the control of oxygen concentration in Pb or LBE systems two different methods are used:

- oxygen control via the gas phase (based on gas/liquid equilibrium between the cover gas and the liquid bulk when the liquid is below saturation);
- oxygen control via solid phase (which consists in dissolving solid lead oxide in a device where thermal-hydraulics is controlled).

2.1. Oxygen control via solid phase

In this paper, the oxygen control via solid phase is treated. This method involve the use of lead oxides in solid phase (PbO compacts), placed and maintained in a part of the cooling circuit, with a limited volume (the reaction vessel of the mass exchanger), and thermo-hydraulic controlled. By passing the liquid metal through the mass exchanger vessel, the solid phase PbO oxides are dissolved, releasing the oxygen which is then transported through the circuit by the coolant agent flow (Figure 2).

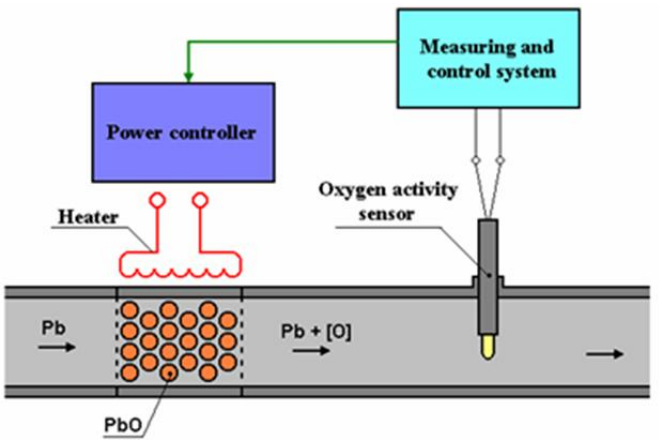


Figure 2. Automatic oxygen control system

The principle of oxygen control via solid phase, with mass exchangers, is the following: when liquid metal unsaturated with oxygen is pumped through a mass exchanger, the lead oxides are dissolved in the liquid and in this case the mass exchanger is a dissolved oxygen generator. On the coldest side of the cooling loop, the coolant agent will be slightly unsaturated with oxygen. As the coolant flow transports the dissolved oxygen to other heated sides, the coolant will become unsaturated in oxygen and an additional contamination of the liquid will be generated. The control of dissolved oxygen supplied for forming protective films, corrosion-resistant, from the steel's surface is possible by controlling the temperature of the coolant from the mass exchanger, the pumping speed and the pumping period. Thus, the formation of the protective films depends on the monitoring parameters from the oxygen sensors. [1]

For the design of PbO mass exchangers, in the specialty literature, different geometries for PbO compacts were used (pebbles, oval, cylindrical). In this context, the aim of this paper was to obtain PbO compacts with mechanical strength, under different geometries.

3. Steps in obtaining PbO mass exchangers

In order to develop and establish a fabrication technology for obtaining PbO mass exchangers, the starting point were some preliminary technological tests to see if we can obtain these mass exchangers. The method used for obtaining the PbO compacts is based on the ceramic powder technology.

1. Preparation of PbO powder

The PbO powder used for this paper was obtained by calcination of the PbO2 powder, in air, at a temperature of 600°C, for one hour (Figure 3). After this, the powder was weighted and prepared for the next operations.

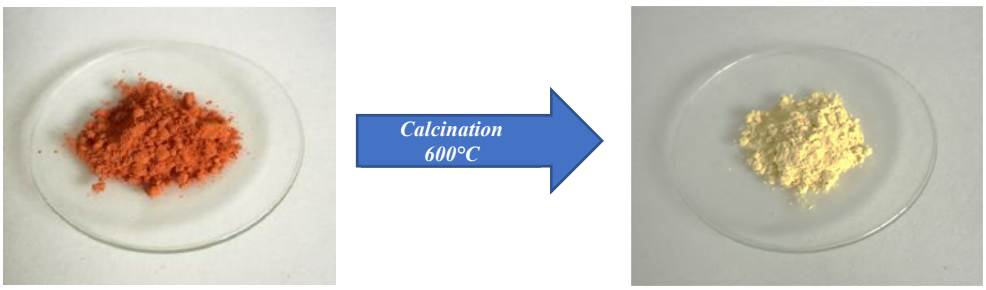


Figure 3. Obtaining PbO powder from PbO_2

The main physical and chemical characteristics of lead oxide are shown in Table 1.

Table 1. Physical and chemical characteristics of PbO

Characteristic	*Value*
Molar mass	223.2 g / mole
Theoretical density	9.53 g / cm³
Melting point	888 ^{0}C
Boiling point	1477 ^{0}C
Colour	red - yellow (powder)
Crystalline structure	tetragonal

2. *Preparation of the moulds and punches*

For developing a technology for obtaining PbO mass exchangers, in our preliminary tests, different moulds were used in order to make some correlations and also to see if compacts with different geometries can be attained.

The molds and the punches used were made from C120 material, having the following shapes:

- cylindrical mold with 15.5 mm interior diameter and straight punches (Figure 4 a);
- cylindrical mold with 15.5 mm interior diameter and punches with concavity (Figure 4 b);
- rectangular mold (28.5 x 9.25 mm) with rectangular punches (Figure 4 c).

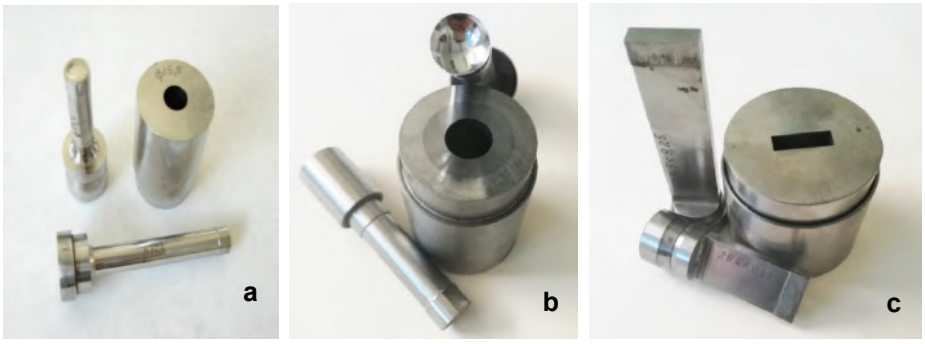


Figure 4. Molds and punches used for pressing operation

3. Pressing the PbO powder

The formation of compacts from ceramic powders by pressing is the most often used process in industry. The factors affecting the formation of compacts by pressing are:

- the pressing method;
- the pressing time and temperature;
- the pressing speed;
- the pressing atmosphere;
- the lubricant;
- the quality of the moulds.

Obtaining a stable compact, which can allow a safe manipulation, depends on choosing the right factors.

For obtaining the PbO compacts, the bilateral pressing method was used. In this method, the pressure acts through both punches (inferior and superior), which determines a density decrease in the pressure propagation direction only in half of the pressed compact high. Thus, in many cases, uniform density distributions can be obtained.

The density dependence on the pressing force in represented as a characteristic curve which increases abruptly thereupon it collapses moreover. By increasing the pressing force, the powder particles are being elastically or plastically deformed. There is a pressure limit after which the density of the compact does not increase. Increasing the compaction pressure is leading to an increase of the material plasticity, decreasing therewith the deformation at sintering.

Regarding the compression time (the time in which the final pressure acts on the compressed powder), researches have shown that maintaining constant the maximum pressing force for a short time, is leading to a considerable improvement of the density. [3]

In order to see if PbO compacts can be obtained and establish the optimum pressing parameters, several tests were performed, at different levels of compaction force, using the cylindrical moulds. PbO powder was added in the moulds and the powder was then pressed at different pressures, obtaining “green” PbO compacts, like the one from Figure 5.

Figure 5. “Green” PbO compact

The main parameters used in the pressing operations are shown in Table 2.

Table 2. Pressing parameters of PbO powder

Parameter	*Value used*
Compaction pressure	270-450 MPa
Compaction time	1 ÷ 7,5 s
Pressing type	bilateral
Matrix	ϕ = 15.5 mm

After this operation, the compacts were dimensionally controlled, studying the influence of the pressing force on the density. As it can be seen from Figure 6, the density of the obtained "green" compacts does not depend very much on the pressing force, meaning that we don't have to use higher compacting forces. [4]

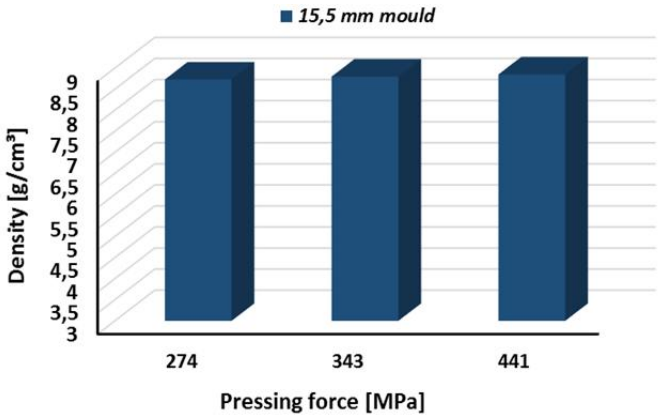


Figure 6. Density evolution of "green" PbO compacts

4. Sintering the "green" PbO compacts

In powder technology, sintering represents the welding, densification and recrystallization processes, by thermal activation of some powder agglomerates, in the presence or absence of a liquid phase. During the sintering operation, the powder agglomerate consolidates by forming continuous and stable bounds between the particles.

The sintering process practically begins at 0.5% of the melting temperature. At this temperature, all the transport mechanisms of the material act, but not all of them contribute with the same intensity at the sintering process. Some of these mechanisms can be dominant at certain temperatures or during certain sintering stages and other mechanisms act during all stages.

In order to establish the optimum parameters for sintering, different tests were performed, varying the sintering time and the temperature. The "green" compacts obtained previously, were subjected to sintering thermal treatments, at a temperature of ~ 2/3 of PbO melting temperature, for a period of time between 1.5 and 6 h, in air atmosphere. After this operation, sintered PbO compacts with mechanical strength were obtained.

For investigating the influence of temperature on the density of the compacts, two sintering thermal treatments were performed, at different temperatures (700°C and 720°C), maintaining constant the sintering time. After this test, as it can be observed from Table 3, the differences are insignificant, meaning that we don't have to increase the sintering temperature.

Regarding the influence of the sintering time (Figure 7), it can be observed that small variations are occurring thus increasing the sintering period. [4]

Table 3. Dimensional evolution of PbO compacts sintered at 700°C and 720°C

Temperature	*Pressing force [MPa]*	*Mass [g]*	*Dimeter [cm]*	*High [cm]*
700^0C	274	9.889	1.519	0.571
	343	9.947	1.513	0.581
	441	9.909	1.508	0.582
720^0C	274	9.940	1.519	0.575
	343	9.787	1.509	0.576
	441	9.932	1.606	0.587

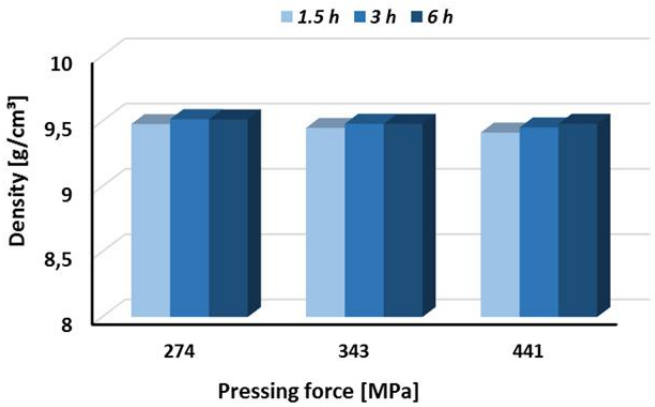


Figure 7. Density evolution of PbO compacts sintered for different periods of time

5. *Obtaining PbO mass exchangers with different geometries*

The remarks obtained after the first studies presented above, were then used for obtaining PbO mass exchangers with other geometries.

Thus, the PbO compacts were obtained through ceramic powder technology, by bilateral pressing at a compaction pressure of ~ 270÷450 MPa and a compaction time of ~5 seconds, using the cylindrical, oval and parallelepiped molds.

The "green" compacts were then subjected to different sintering thermal treatments, at 700°C and 720°C (~ 78% of PbO melting temperature), in a furnace for thermal treatments without reducing atmosphere, for a period of time between 1.5 hours and 2 hours. After this operation, PbO compacts with mechanical strength, in three different shapes: oval, parallelepiped and cylindrical were obtained (Figure 8). [5]

Figure 8. Sintered PbO compacts with different *geometries*

These compacts were then visually and dimensionally controlled, determining the density before and after sintering (geometrical and respectively by immersion), the results being presented graphically in Figure 9.

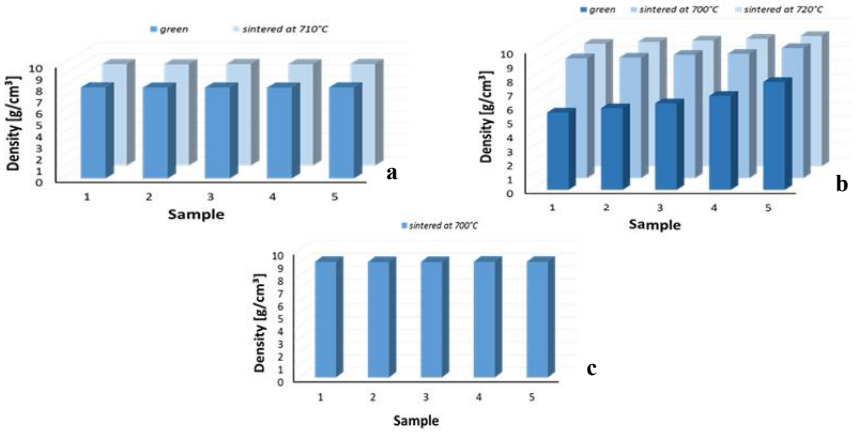


Figure 9. Density evolution of sintered PbO compacts with different geometries
a – cylindrical b- parallelepiped c - oval

4. Conclusions

The chemistry control in a nuclear system is a key element important for ensuring the nuclear safety of Generetion IV nuclear reactors. Besides, this aspect appears as a quite complex issue, since is critical to keep under control the corrosion in a wider operating temperature range, as well as to keep the coolant free of any contamination by oxides, which is basically the first requirement.

Developing the necessary systems and techniques for optimizing the oxigen concentration in the coolant agent is the subject of many studies worldwide.
The present paper presents the first technological studies initiated in RATEN-ICN Piteşti, in order to develop a technology for obtaining PbO mass exchangers which can be used for controlling the dissolved oxygen.

PbO compacts, with different geometries and mechanical strength, were obtained, trough ceramic powder technology (pressing and sintering operations). The relative contraction of the compacts, after sintering, was between 3.5% - 6%, depending on the shape and the dimensional orientation on which it was calculated (diameter, high, length, width). In regards with the density of the compacts, it was observed that after sintering, it is barely depending on the compaction pressure and is varying between 92% and 97% of the theoretical density of PbO (9.53 g/cm³).

In order to develop PbO mass exchangers, the paper will continue with more studies regarding the microstructural properties of the compacts and some thermal treatments for investigating the behaviour of the compacts. Regarding the behavior of the compacts in the HLMC, some tests in liquid lead, to see if the compacts maintain their integrity, are preconized.

REFERENCES

[1] OECD/NEA, "Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermal-hydraulics and Technologies," 2007 Edition, ISBN 978-92-64-99002-9;
[2] *J. Van den Bosch* (SCK•CEN), "Oxygen Control Systems for HLM Systems," International Workshop on Innovative Nuclear Reactors Cooled by Heavy Liquid Metals 2012, Pisa, Italy;
[3] *T. Surdeanu, M. Pernes,* "Piese sinterizate din pulberi metalice – Tehnologii, proprietăţi, utilizări," Ed. Tehnică, Bucureşti, 1984;
[4] *I. Paraschiv, D. Benga,* "Technological Studies for Obtaining Lead Oxide Compacts Used in Generation IV Nuclear Reactors," Nuclear 2016 - The 9th Annual International Conference on Sustainable Development through Nuclear Research and Education, 18-20 May 2016 Piteşti, România, ISSN 2066-2955;
[5] *I. Paraschiv, D. Benga,* "Technological Tests for Obtaining PbO Mass Exchangers with Different Geometries Designed for Oxygen Control in Pb or LBE Systems," Nuclear 2017 - The 10th Annual International Conference on Sustainable Development through Nuclear Research and Education, 24-26 May 2017, Piteşti, România, ISSN 2066-2955.