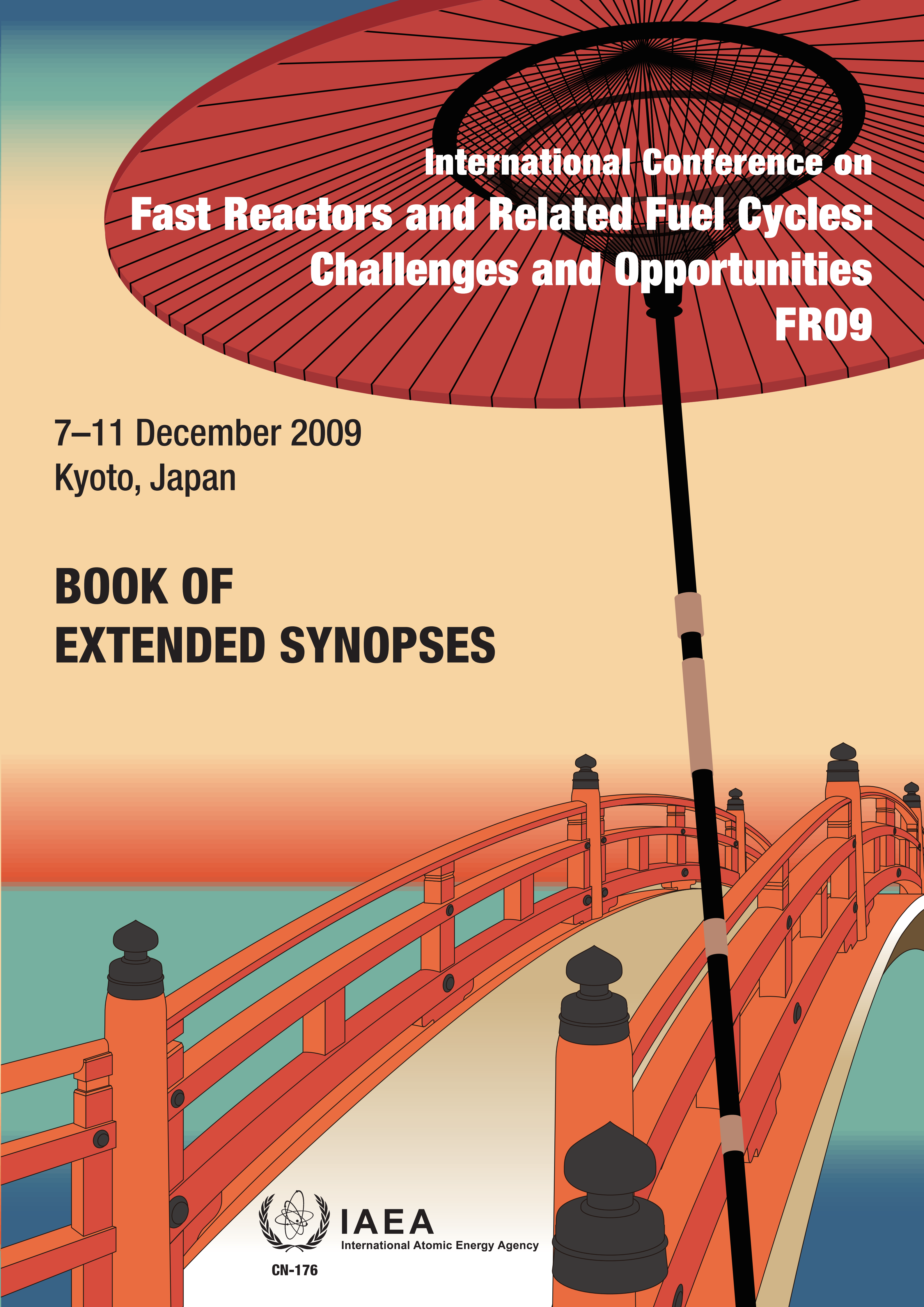

International Conference on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities FR09

7–11 December 2009
Kyoto, Japan

BOOK OF EXTENDED SYNOPSES

IAEA
International Atomic Energy Agency

CN-176

Organized by the

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Japan Atomic Energy Commission
Ministry of Economy, Trade and Industry (Japan)
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Atomic Energy Society of Japan
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Fast Reactor Development for Sustainable Nuclear Energy Supply in China

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China needs a very huge energy supply for national economy development and living standard improvement of 1.3 billion population. The nuclear energy is a new member of the energy supply family in China. A satisfied operation records of all 11 units of NPPs, especially with the total average load factor 85.8% of all NPPs in 67 reactor-years since commercial operation of each unit encourage the public to believe that the nuclear power is a safe, reliable, economically-acceptable, CO_2 avoidable one and could be used in large scale. The government has decided in 2006 to accelerate the nuclear power development with the target of 40GWe in operation and 18GWe in construction in 2020. Right now 13 units with total capacity 13.05GWe are under construction and other 11 units of total capacity 12.05GWe have been approved by the government and the preparation for construction is underway.

For the sustainable supply of nuclear energy, as the principle strategy, PWR-FBR matched with closed nuclear fuel cycle has been decided by the government for a long time. Three FBR development strategy targets suggested as following.

(1) to realize FBR commercial utilization in small batch in 2030;
(2) to increase nuclear capacity to 240GWe, sharing about 16%, mainly by FBRs in 2050, and

(3) to replace coal fired plants by nuclear power in large scale, in the period about 2050-2100.

For that, the suggested FBR development strategy is shown in Table 1.

Table 1 Suggested FBR Development Strategy

Reactor	Electric power/MWe	Design Beginning	Commissioning
CEFR	20	1990	2010
CDFR	800	2007	2018
CCFR	n×800	2015	2030
CDFBR	1000～1500	2015	2028
CCFBR	n×(1000～1500)	2018	2030～2032

China Experimental Fast Reactor (CEFR) with the power 65MWt is a pool type sodium-cooled fast reactor. Pre-conceptual design started in 1990 with the first pot of concrete in 2000, and architecture engineering launched in 2001. Now it is under commissioning tests stage. The experiences of design fabrication and construction for this type of reactor have been gained.

The possibility of large striding from 20MWe CEFR to 800MWe China Demonstration Fast Reactor (CDFR) has been studied. The favorableness is estimated mainly as following.

(1) The main technical scheme options of design for CEFR, CDFR and CDFBR have a basic consistency;
(2) A set of computer codes, data files and design criteria of different specialties applied to sodium-cooled fast reactor were developed and their validation and verification are underway following the CEFR commissioning and pre-operation testing;
(3) A set of fabrication enterprises and factory having fabricating license for nuclear safety grade components for fast reactor have been organized through CEFR construction;
(4) A good international cooperation environment has been established.

To support PWR-FBR matched strategy, the closed nuclear fuel cycle architecture is under development including 100t/a reprocessing pilot hot tested, an industrial reprocessing plant with the capacity about 1000t/a and MOX fuel manufacture plant of about 50t/a are under programming and carrying into execution.

French R&D program on SFR and the ASTRID prototype

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To be capable to offer a Gen IV SFR for future deployment (typically by 2040 and after) and although there exists a significant experience, the SFR technology must still progress. The CEA and its industrial partners, AREVA NP and EDF, finalized in spring 2007 an R&D program over the period 2007-2012. This program comprises research in four domains of innovations:

- The development of an attractive and safe core, taking into account the specificities of the fast neutrons and sodium, and also able to transmute minor actinides,
- A better resistance to severe accidents and external hazards,
- The search for an optimized energy conversion system reducing the sodium risks,
- The re-examination of the reactor and components design to improve the conditions of operation and the economic competitiveness

The R&D program aims at selecting, by 2012, the most promising innovations for a new generation of SFRs and to establish the specifications of the prototype which would demonstrate whole or part as of these promising technologies. One considers a prototype power of some hundreds MWe to be representative at the same time of the core physics and of the general architecture of the reactor. This prototype, named ASTRID (for Advanced Sodium Technology Reactor for Industrial Demonstration), is considered as the precursor of a first of a kind of the commercial reactor.

The ASTRID prototype will be used at the same time for the demonstrations required in the June 2006 French act on the sustainable management of wastes (irradiation of transmutation fuels at a scale varying from several pins to a whole assembly), which requires to consider it as an irradiator, and for the demonstration of the potential of SFRs as an innovative Gen IV system preserving uranium resource, sure and economic. The paper will present a first discussion of the trends concerning the power level of ASTRID.

The international cooperation on the SFR was particularly active, in 2007 and 2008, with Japan and the United States in particular, countries which also envisage (or were envisioning) prototypes by 2020-25. The stakes of the international co-operation are multiple: to share the vision of progress still necessary on the SFR, to share some of the effort of R&D, to seek for a convergence on the international standards (safety, non-proliferation…), and if possible to articulate in a constructive way objectives of demonstration of the prototypes (to make them complementary) and to optimize the associated necessary infrastructures. 2008 saw the preparation of a European coordinated project on the SFR, the ESFR project, which was

officially launched at the beginning of 2009. It contributes as a significant and original contribution to international R&D on SFR.

First contacts with French Nuclear Safety Authority took place in June 2008 and are continuing now : this features the probable start of an accompaniment process of French SFR project on the safety orientations, first on the visions of the future commercial reactor, and then by 2011/2012 on the ASTRID prototype.

First preliminary results have already been gained in the current R&D program on important issues:

- The definition of a large reference core using oxide fuel and characterized by a very low reactivity loss (self-sustainable core) and attractive safety parameters ,
- The realization of two irradiation tests in Phénix, concerning structural materials (ODS F/M steel) and oxide fuel elaborated with (U,Pu) co-precipitated powder,
- The characterisation of new solutions for the recycling of minor actinides in the heterogeneous mode,
- The first calculation of severe accident sequences with SAS4A and SIMMER multi-physic computational tools
- The selection for complementary evaluations of two alternate fluids to sodium for the secondary circuit
- The establishment of the development plan of the technological loops necessary for the R&D and the prototype development,
- The definition and launching of a comprehensive program on In Service Inspection and Repair (ISIR)
- The preliminary reactor designs for innovations assessment (pool, loop, fuel handling options, advanced energy conversion systems, ..)

2009 constitutes a year of particular importance for the French trilateral program on SFR : being :10 synthesis reports have been produced in the various fields covered by the program. They are the basis for the detailed definition of the next phase of the program (2010 to 2012). The topics covered in the synthesis reports of which some conclusions will be reported in the paper are listed hereafter:

- Loop type reactor concepts
- Pool type reactor concept
- Review of innovative options: advanced energy conversion system, very innovative reactor concepts…
- Incidence of power level and of modularity on safety and economics
- Fuel Handling
- Fuel, assembly and core design
- Safety and severe accidents
- Assessment of the T91 type steels for pipings and components
- Status and perspective of ODS steel development for cladding
- ISIR: US sensors, inspectability, repairability, robotics

Perspective on Development of Future SFRs in India

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In the Indian energy scenario projections for the future, the nuclear power through fast reactors is expected to play an important role of ~ 20% of total installed electrical capacity by 2052. Successful operation of 40 MWt/13 MWe capacity Fast Breeder Test Reactor over 23 years, strong R&D executed in multidisciplinary domain, and construction of 500 MWe Prototype Fast Breeder Reactor (PFBR) based on indigenous design in a smooth manner have provided high confidence on the success of fast breeder technology.

Beyond PFBR, it is planned to construct 4 more FBRs of 500 MWe capacity each by 2020. Towards this, a systematic roadmap has been drawn for improved economy and enhanced safety through a number of measures. The major features incorporated to achieve economy are twin unit concept, plant life increased to 60 years in comparison to 40 years for PFBR, reduced fuel cycle cost with higher burn-up, reduction in number of steam generators from 8 to 6, in-vessel primary sodium purification, minimizing the use of SS 316LN for NSSS components, reduction in special steel specific weight requirements, compact plant layout, improved load factor, reduction in construction time by atleast 2 years, and co-location of fuel cycle facility.

The major features towards enhancing safety features are improvements in reactor shutdown system to provide reliability a target of 10^{-7}/reactor-year, enhanced diversity in decay heat removal system, integrated primary sodium purification, reduction in number of tubes to tubesheet welds by increasing the seamless tube length of the steam generator, and enhanced in-service inspection.

Beyond 2020, a series of 1000 MWe capacity metallic fuel with high breeding potential will be constructed. R&D activities have been systematically planned for metallic fuel development.

The paper addresses the highlights of the conceptual design features of future sodium cooled fast reactors in India.

Research and Development Policy on FBR Cycle Technology in Japan

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The Fast Breeder Reactor (FBR) and its fuel cycle (hereinafter "FBR cycle") technology will provide harmonic solutions for global energy resource and environment issues. In Japan, the significance of FBR cycle technology development has been recognized for decades.

The Japan Atomic Energy Agency (JAEA) has been the principal agency for the FBR cycle development in Japan. The experimental fast reactor, Joyo, had been successfully operated for about 30 years, beginning in 1977. The prototype FBR, Monju, achieved initial criticality in 1994. Monju is designed on the basis of research results gleaned from Joyo. Monju has the role of confirming the technological data base for design and safety evaluation tools, and of accumulating operation experiences for sodium-cooled reactors, with an eye toward commercialization. Both reactors' operations have been suspended since 2007 and 1995, respectively, due to troubles; presently, JAEA is preparing for the re-launch of operations. Furthermore, the development of FBR spent fuel reprocessing technologies was initiated in 1975, and JAEA has successfully achieved MOX fuel fabrication at the Plutonium Fuel Center, as far back as 1972.

In 1999, the "Feasibility Study on Commercialized FBR Cycle Systems (FS)" was initiated to present an appropriate picture of FBR cycle technology commercialization by 2015, as well as its research and development (R&D) program. In this study, conceptual design features were evaluated in order to select promising FBR cycle systems that could meet the design requirements that embodied the five development targets: 1) safety; 2) economic competitiveness; 3) efficient utilization of nuclear fuel resources; 4) reduction of environmental burden; and 5) enhancement of nuclear non-proliferation. As a result, the combination of sodium-cooled FBR with oxide fuel, advanced aqueous reprocessing and simplified pelletizing fuel fabrication was selected as the most promising concept for the FBR cycle system. Figure 1 shows major features of the Japanese sodium-cooled FBR.

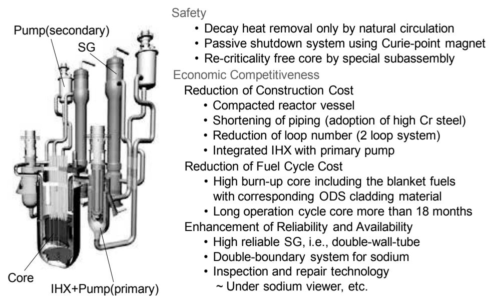

Fig.1 Major features of Japanese sodium-cooled FBR

The Atomic Energy Commission of Japan (AECJ) issued the "Framework of Nuclear Energy Policy" in October of 2005, which is the foundation of Japanese policy on research, development, and utilization of nuclear energy. In this framework, the target for development of FBR cycle technology is commercialization by approximately 2050. In March of 2006, FBR cycle technology was selected as one of the key technologies of national importance in the third-term "Science and Technology Basic Plan." Subsequently, the Ministry of Education, Culture, Sports, Science and Technology (MEXT) and the Ministry of Economy, Trade and Industry (METI) formulated the action plan for the development of nuclear technologies to materialize the framework reviewing the report of FS. Following the action plan, the council of five-parties, which consisted of MEXT, METI, the Federation of Electric Power Companies of Japan (FEPC), the Japan Electrical Manufacturer's Association (JEMA) and JAEA, was set up to discuss the smooth transition from the R&D stage to the demonstration and deployment stages, in the process developing a solid mutual understanding of the issues. The "Basic Energy Plan," issued in March of 2007, explains that the FBR cycle development should be promoted as one of the most important technologies, with the aim to commercialize the FBR cycle system by 2050.

Following the Japanese policy on FBR cycle technology development, MEXT, METI and JAEA, launched the Fast Reactor Cycle Technology Development (FaCT) project in 2006, in cooperation with the Japanese electrical utilities. Figure 2 shows an outline of the development plan toward commercialization of the FBR cycle technology in Japan. In the FaCT project, design and experimental study for the main concept will be implemented in order to present the conceptual designs of the commercial and demonstrative FBR cycle facilities by 2015, along with the development plan to realize them. R&D has progressed to the development stage, to establish the realization of innovative technologies that can meet development targets. Consequently, the development of innovative technologies should be completed by 2015. Thereafter, the FBR cycle development project will enter the introduction stage through a first system demonstration. The demonstration FBR will be launched sometime around 2025. By around 2050, the commercial FBR cycle system will be deployed, based on experiences with the demonstration FBR cycle system. In the FaCT project, Joyo and Monju will play important roles, through demonstration as a reliable power plant and the establishment of sodium handling technology, as well as MA burning and irradiation of materials.

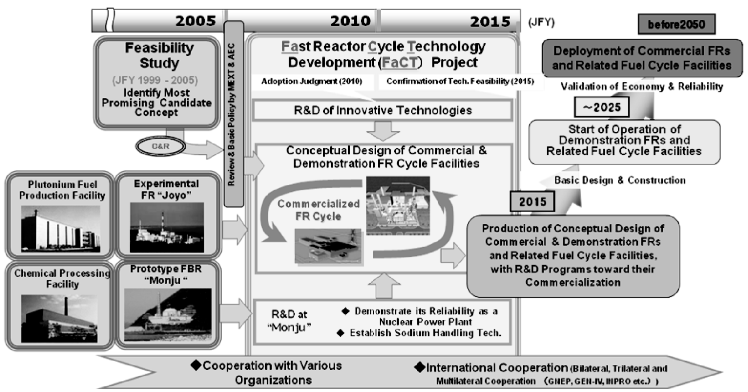

Fig.2 Outline of development plan toward commercialization of FBR cycle technology in Japan

Status of Fast Reactor and Pyroprocess Technology Development in Korea

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For the time being, PWRs will remain as the major source of nuclear power in Korea. However, the storage of the spent fuels produced from those PWRs is a big issue. The on-site spent fuel storage capacity will reach its limit by 2016. Therefore, a decision-making process for spent fuel management is under way.

It has been recognized that one of the most promising nuclear options for electricity generation is a fast reactor system which efficiently utilizes uranium resources and reduces radioactive wastes from nuclear power plants, thus contributing to sustainable development. In response to this recognition, the widespread concern about the management of spent fuels caused us to develop technologies for Sodium-cooled Fast Reactors (SFRs) as one of the most promising future types of reactors in Korea.

The pyroprocessing technology capitalizes on the recovery of actinide elements from spent fuel for the recycling and fissioning in SFRs for the purpose of power generation. The overriding goal of this R&D plan for pyroprocessing technology combined with SFRs is to develop a closed nuclear fuel cycle that is economically viable, resistant to diversion of nuclear materials, and minimizes generation of waste products, thereby efficiently increasing the capacity of a final spent fuel repository by approximately 100 times. In this fuel cycle, plutonium remains with other isotopes and impurities throughout the processes and cannot be chemically separated in pure form, which reduces the risk of nuclear proliferation. Cofining the final product in a hot cell also makes it far less open to misuse.

In order to provide a consistent direction to long-term R&D activities the Korea Atomic Energy Commission (KAEC) approved a long-term development plan for future nuclear reactor systems which include SFR, pyroprocess and VHTR on December 22, 2008. This long-term plan will be implemented through nuclear R&D programs of the National Research Foundation, with funds from the Ministry of Education Science and Technology (MEST). A detailed implementation plan is now being developed.

The long-term SFR development plan approved by the KAEC will be carried out with the long-term vision of the construction of an Advanced SFR demonstration plant by 2028 in association with the pyroprocess technology development in three phases: (1) First Phase (2007 – 2011)- Development of an Advanced SFR design concept, (2) Second Phase (2012 – 2017)- Standard design of an Advanced SFR plant, and (3) Third phase (2018 – 2028)- Construction of an Advanced SFR demonstration plant.

The KALIMER-600 design that evolved on the basis of the KALIMER-150 design, serves as a starting point for developing the new advanced design. This new design will adopt advanced design concepts and features with the potential to better meet the Generation IV (Gen IV) technology goals of sustainability, safety and reliability, economics, proliferation resistance and physical protection. A conceptual design of the Advanced SFR will be developed by 2011.

R&D efforts are being made for the conceptual design of the Advanced SFR with emphasis on the core and reactor systems, and on the development of the Advanced SFR technologies necessary for the enhancement of competitiveness. These R&D activities include the developments of a Passive Decay Heat Removal Circuit (PDRC) system, a supercritical carbon dioxide (S-CO_2) Brayton cycle system, an under-sodium viewing technique, and a metal fuel. In addition, R&D activities for development of basic technologies are being performed, mainly focusing on validating computational tools and developing sodium technologies.

The development of Metal fuel for the SFR started in 2007. The development of U-TRU-Zr metal fuel assembly in combination with pyroprocess is expected to be completed by 2021 according to the long-term plan. The mass production of U-TRU-Zr metal fuel assembly will be started in 2025, and will support the demonstration SFR.

KAERI is currently establishing a PyRoprocess Integrated inactive DEmonstration Facility (PRIDE), a mock-up facility for pyroprocessing, to produce the engineering data to be incorporated into the design of an Engineering-Scale Pyrochemical Process Facility (ESPF) which is scheduled to be constructed by 2016. In addition, the proliferation resistance characteristics and safeguardability of the pyrochemical process are being anlyzed so as to ensure that pyroprocessing is qualified as a proliferation resistant option for the spent fuel management in Korea. The demonstration of an engineering-scale pyrochemical process will be started in 2016, and the construction of a prototype(or commercial scale) pyroprocess facility, Korea Advanced Pyroprocess Facility (KAPF), will be completed by 2025, followed thereafter by the fabrication of metal fuel for the SFR.

The pyroprocessing technologies listed in the long-term development plan include an Electrolytic Reduction System of PWR spent fuel, a High-throughput Electrorefining System, an Electrowinning System for TRU recovery, Waste Salt Regeneration & Solidification, and System Engineering Technology Development.

The electrolytic reduction system converts PWR spent oxide fuel to an electrorefiner feed material (metal form) by an electrochemical reaction in a molten salt. A high speed electrolytic reduction system with a capacity of 20 kgU/batch is under development. The electrorefining system is composed of a pure uranium recovery system from the electrolytic reduction product, salt distillation system, melting furnace, and UCl_3 fabricator. A continuous electrorefining system with a capacity of 20 kgU/batch is under development. The electrowinning technology aims at recovering of actinides from LiCl-KCl eutectic salt after the electrorefining operation. A lab-scale electrowinning apparatus with a mesh-type LCC structure to prevent the formation of the uranium dendrite is being developed. An innovative process using LCC electrolysis and oxidative extraction for residual actinide recovery has been developed.

The goals of KAERI’s research on the waste salt treatment technology are to minimize an amount of waste generated and to develop a durable form for high level waste disposal. The salt recycle technologies such as melt crystallization and oxidative precipitation/distillation to separate salt from the waste are under development. The final waste is fabricated into durable ceramic waste forms to enhance the integrity during the interim storage or final disposal.

Plenary Session 2:

Invited papers

National and international fast reactor programmes

The Program of Fast Reactor Development in Russia

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The paper presents the main trends of development of fast reactors in Russia for the mid-term (up to 2020) and long-term (up to 2050) future.

The mid-term tasks are formulated under the Federal Target Program (FTP) “Nuclear power technologies of a new generation” that covers the years of 2010-2020. This program is currently under approval in the Government of the Russian Federation.

The purpose of this FTP is to develop and construct a new technological platform for nuclear power that is based on transfer to the closed uranium and plutonium nuclear fuel cycle with fast reactors of the fourth generation. The objective is to solve the system problems typical of the current large-scale nuclear power in Russia: a considerable, constantly growing amount of spent nuclear fuel and limited raw material base of nuclear power with thermal neutron reactors.

In order to implement the transfer to the new technological platform the FTP envisages the activities in the following directions:

- to develop advanced reactor technologies of the fourth generation;
- to develop and construct new test facilities, to fabricate new equipment, to upgrade and develop an experimental and bench-scale base in support and justification of the reactor technologies under development;
- to develop the technologies for production of advanced types of fuel for reactors of the next generation;
- to develop materials and technologies of the closed fuel cycle (CFC) for nuclear power systems with fast and thermal reactors of a new generation.

The basic approach considered in the FTP is the approach that envisages simultaneous development of several reactor technologies, i.e. a sodium-cooled fast reactor (SFR), a lead-cooled fast reactor (BREST) and a lead-bismuth fast reactor (SVBR), and the respective fuel cycles.

The FTP implementation is planned to be in two stages.

The first stage (2010-2014) assumes to cover the following main activities:

- to develop basic designs for all the above-mentioned fast reactors of the fourth generation, with a nuclear fuel breeding ratio not less than 1 and with a high level of inherent safety; as well as respective technologies of the closed nuclear fuel cycle;
- to complete designing and commissioning of uranium-plutonium oxide fuel production for fast reactors of a new generation;
- to develop a detailed design for construction of a multi-purpose research fast reactor (MBIR) purposed to conduct reactor studies, including testing of new types of fuel, various coolants, fuel and structural materials;
- to perform work on extension of BOR-60 research reactor lifetime;
- to develop and construct the facility to produce dispersion composite materials for the reactors of a new generation.

Among the most important activities envisaged to be implemented at the second stage (2015-2020) it is necessary to highlight the following:

- to construct the demonstration and prototype reactors of BREST and SVBR;
- to commission the technically reequipped complex of big test facilities (BFS);
- to set up a pilot production of compact fuel for nuclear reactors of a new generation;
- to construct a demonstration semi-commercial pyrochemical complex for fuelling nuclear reactors of the fourth generation;
- to construct, refurbish, technically upgrade and commission the required research base with the aim to justify the new technological platform of nuclear power, including the MBIR reactor construction.

The FTP under consideration is aimed at providing scientific and technical basis for innovative reactor technologies together with the CFC, first of all for those that require experimental justification of their implementation at the level of demonstration prototypes (BREST, SVBR). For SFRs, which have already proved their technical feasibility, with BOR-60, BN-600 reactors as good examples, and are being the most advanced in comparison with other reactor technologies, the objectives are to further improve technical and economical characteristics and safety up to the level that meets the requirements imposed to the reactors of the fourth generation. From this point of view the important stages in the SFR development will consists in construction of BN-800 reactor and demonstration of the closed nuclear fuel cycle, as well as development and construction of the advanced commercial SFR (BN-K).

The paper considers various scenarios for development of Russian fast reactors in the long-term future (up to 2050).

The US Advanced Fuel Cycle Program: Objectives and Accomplishments

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The US Department of Energy has been running for approximately a decade an advanced fuel cycle program (currently named the Fuel Cycle R&D Program) devoted to lessening both the environmental burden of nuclear energy and the proliferation risk of accumulating Used Nuclear Fuel. Until very recently, this program was technically focused on achieving an optimized symbiosis between fuel cycle options on one hand, and the US geological repository on the other, with a relatively short term deployment focus. Following detailed technical analyses, this focus led to the selection of a limited set of technologies that were expected to both meet specific geology related criteria, and would be based on limited extrapolations of existing technologies.

Recent developments in the US indicate that the Yucca Mountain repository might not be anymore the geology of reference; furthermore, the need for advanced fuel cycles has been postponed to the middle part of the century, with increased reliance on temporary storage of Used Nuclear Fuel in the interim.

Consequently, the Fuel Cycle R&D Program is being redirected towards a science based, goal oriented focus, driven by the following three considerations:

1. the program is currently examining a broad set of options, including different geologic media and transmutation technologies, in order to understand their relationships and provide information for later decisions.
2. the R&D component of the program is focused on acquiring the basic understanding of key phenomena, defining the relevant challenges, and acquiring the basic tools necessary to resolve them.
3. the timeline of the program allows for a deployment of the successfully demonstrated technology in the 2040-2050 timeframe; this allows us to consider technologies that are not yet mature, but that might provide significant improvements in performances.

The technical program is articulated along the following elements:

- a systems integration task that analyzes the relationships between technologies and defines requirements on each technology to achieve overall system objectives.
- a separations research programs that is aimed at understanding the fundamentals of actinide chemistry, in order to develop processes that achieve specific separation goals with very low losses
- a fuels research program that is also aimed at a better understanding of the fundamentals of fuel behavior, in order to design minor actinide containing fuels with high burnup capabilities.
- A fast reactor development program aimed at reducing the cost of fast reactors, with increased safety performance,
- Significant efforts are also being devoted to achieving a better understanding of geologic repository options, and to the development of better safeguards techniques.

The program also contains a significant Modeling and Simulation component.

Fast reactor research in Europe: the way towards sustainability

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The European Union has taken the lead in responding to climate change, announcing far-reaching initiatives from promoting energy efficient light bulbs and cars to new building codes, carbon trading schemes, the development of low carbon technologies and greater competition in energy markets.

Nuclear energy remains central to the energy debate in Europe. One third of EU electricity is produced via nuclear fission, and eight new reactors are under construction. Traditionally non-nuclear countries are manifesting an interest in building nuclear power plants while the clock is ticking down on Belgium, Germany and the UK's decision to renew or close existing nuclear infrastructures.

Sustainability in nuclear energy production is ensured in the medium term due to the large and diverse uranium resources available in politically stable countries around the world. The quantities available with high probability ensure more than hundred year of nuclear energy production. This extrapolation depends however on the forecast for the future nuclear energy production. The use of fast neutron breeder reactors would lead to a much more efficient utilisation of the uranium, extending the sustainable energy production to several thousands of years. The presentation will outline the fast reactors of the new generation currently being developed within the "Generation IV" initiative.

Broad conclusions of the presentation will be that:

- There is a growing *nuclear renaissance* in Europe for good reason;
- Nuclear energy is a *green and sustainable option* for Europe and indeed the world's energy needs;
- Nuclear energy is a *competitive* energy that makes *economic sense*;
- Nuclear fission reactors have a *safety* and *environmental* track record that is second to none, yet public misperceptions persist and must be tackled;
- *Waste management* solutions exist while new developments hold great promise;
- The evolution and *promise* of nuclear technologies must also be examined against the costs and risks in a balanced approach;
- Research on fast neutron reactors is being strengthened in Europe, under the umbrella of the Generation IV International Forum. European coordination is entrusted to the Joint Research Centre.

IAEA programme on fast reactor, related fuels, and structural materials technology

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For obvious sustainability reasons, spent fuel utilization and breeding are returning to centre stage, and with this the fast reactor as the necessary linchpin. The necessary condition for successful deployment in the near and mid-term of fast reactors and the associated fuel cycles is the understanding and assessment of technological and design options, based on both past knowledge and experience, as well as on research and technology development efforts. Achieving the full potential of fast neutron systems and closed fuel cycle technologies with regard to both efficient utilization of the fissile resources and waste management is conditional on continued advances in research and technology development to ensure improved economics and maintain high safety levels with increased simplification of fast reactors.

The IAEA's fast reactor technology development activities are pursued within the framework of the Technical Working Group on Fast Reactors (TWG-FR). Currently, the TWG-FR comprises 14 IAEA Member States, the European Commission (EC), the ISTC, and the OECD/NEA, as well as Belgium and Sweden as observers. The TWG-FR assists in the implementation of IAEA activities, and ensures that all technical activities performed within the framework of the IAEA project on Technology Advances in Fast Reactors and Accelerator Driven Systems are in line with expressed needs from Member States. The scope of the TWG-FR is broad, covering all technical aspects of fast reactors and sub-critical systems, including: research and development, design, deployment, operation, and decommissioning.

The TWG-FR has focused on experimental and theoretical aspects of fast reactor technology and safety. A benchmark test with experimental data was conducted to verify and improve the codes used for the seismic analysis of reactor cores. A coordinated research project (CRP) was conducted to apply acoustic signal processing for the detection of boiling or sodium/water reactions in liquid metal cooled fast reactors. Benchmark analyses addressed accident behaviour and design improvements of the Russian BN-800 reactor. In cooperation with the IAEA's Department of Nuclear Safety, assistance was provided to ensure safe operation during the remaining lifetime and the development of an effective decommissioning programme for the BN-350 reactor in Kazakhstan. A CRP is being conducted with the objective of reducing the calculational uncertainties of fast reactor reactivity effects. Another CRP is contributing to the IAEA Fast Reactor Knowledge Preservation (FRKP) initiative through bibliographic catalogues and synthesis (lessons learned) reports related to feedback from fast reactor operational experience in the areas of steam generators, fuel and blanket subassemblies, and structural materials. Advanced reactor technology options for effective utilization and transmutation of actinides from spent nuclear fuel is addressed in another CRP. Its focus is on the transient behaviour of advanced transmutation systems, both critical and sub-critical. An ongoing CRP is performing computational and experimental benchmarking of ADS and non-spallation neutron source driven sub-critical systems. Two new CRPs were initiated in 2009: the first one aiming at the validation of multi-dimensional fluid dynamics codes based on thermal stratification measurements performed during the 1995 Monju start-

up experiments; the second one performing blind benchmarking and post-experiment analyses for two Phénix end-of-life tests, viz. the Control Rod Withdrawal Test and the Sodium Natural Circulation Test. More CRPs are planned for 2010/2011 and beyond, e.g. on the estimation of the source term in a fast reactor for radioactivity release, and on thermal hydraulics code verification and validation of liquid metal and molten salt coolants.

The IAEA maintains a database to foster information exchange in the field of advanced fast reactor technology development. It is planned to establish a "living" (WWW-based) innovative fast reactor technology status report. Last but not least, topical technical meetings are held to ensure in-depth information exchange on various scientific and technology related issues (e.g. in-service inspection, advanced steam generators, seismic designs, etc), as well as education and training events (Workshops and Schools) in collaboration with the International Centre for Theoretical Physics in Trieste.

A key aspect of fast reactor technology involves the development of the associated fuels and fuel cycles. These include the use of depleted, natural and recycled uranium in conjunction with the multiple recycling of plutonium. Additional requirements which may be placed on the fuels include the ability to operate safely to very high burnup values (150 000-200 000 MWd/tHM), to breed efficiently, and to burn minor actinides effectively. The fuel assembly structural materials should be able to withstand high doses of fast neutron irradiation (up to 200 dpa) with minimum swelling, creep and embrittlement.

During the last few years, the IAEA's programme on fast reactor fuels and fuel cycle technology has been active in all of these areas. In collaboration with international experts, the IAEA has organized workshops, conducted CRPs and maintained relevant databases. Three upcoming publications will cover work in all of these areas.

"Status and Trends of Nuclear Fuels for Sodium Cooled Reactors" will provide a compilation of updated information on the manufacturing technology of oxide and non-oxide ceramic nuclear fuels by different powder metallurgy routes, the fabrication of U-Pu-Zr metallic fuel by vacuum melting followed by injection casting, and the out-of-pile thermophysical and thermomechanical properties of fast reactor fuels and their irradiation behaviour.

"Back End of the Fast Reactor Fuel Cycle: Status and Perspectives" will cover the different aqueous and pyro electrolytic processes for reprocessing of spent fast reactor fuels and recycling the uranium, plutonium and minor actinides.

"Structural Materials for Sodium Cooled Fast Reactor Fuel Assemblies – Fabrication, Properties and Irradiation Behaviour" will give an overview of the development of austenitic, ferritic-martensitic and oxide dispersion strengthened (ODS) steel for fast reactor fuel assembly covering the manufacturing technology of fuel cladding and duct tubes, their out-of-pile tensile and fracture properties and their irradiation behaviour under high fast neutron dose.

A workshop on Physics and Technology of Fast Reactor Systems was just held at the International Centre for Theoretical Physics in Trieste, Italy, which covered all areas of fast reactor and related fuel cycle technology. A CRP on Simulation and Modelling of Radiation Effects (SMoRE) was launched in December 2008. In this CRP, ion beam irradiation from accelerators will be utilized to simulate fast neutron irradiation to high values fluence. Another CRP is planned for 2010 to provide a major update to the Minor Actinide Database and work has begun on developing a database on the properties of fuels and structural materials for fast reactors. Technical meetings on advanced pyrochemical partitioning methods and minor actinide bearing fuels are envisaged for 2011.

OECD Nuclear Energy Agency activities related to fast reactor development

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The renewed interest in nuclear energy, to a large extent due to concerns about global climate change, high volatility of fossil fuel prices and security of energy supply, has also revived discussions on the attractiveness of fast neutron reactors with closed fuel cycles. The OECD Nuclear Energy Agency (NEA), whose role is to assist its Member countries to develop, through international cooperation, the scientific and technological bases required for the safe, environmentally friendly and economical use of nuclear energy, conducts work related to fast reactor systems along two areas of activity: one focussed on scientific research and technology development needs and one dedicated to strategic and policy issues.

Recent scientifically oriented fast reactor related activities in the NEA comprise:

- a series of workshops on “Advanced Reactors with Innovative Fuels (ARWIF)”,
- a series of information exchange meetings on Actinide and Fission Product Partitioning and Transmutation,
- a coordinated effort to evaluate basic nuclear data needed for the development of fast reactor systems [1],
- an ongoing study on “Homogeneous versus Heterogeneous Recycle of Transuranic Isotopes in Fast Reactors” and
- a recently started review of “Integral Experiments for Minor Actinide Management”.

The NEA has also conducted a review of the technical implications of a transition from thermal to fast neutron nuclear systems [2] on reactors, as well as fuel fabrication and reprocessing facilities. Country dependent scenarios for Belgium, Canada, France, Germany, Japan, Korea, Spain, the UK and the USA are presented, as well as a list of the key technologies that were identified as crucial for the implementation of advanced fuel cycles. In addition, the NEA launched three transition scenario benchmark studies devoted to:

1. the performance of scenario analysis codes,
2. a regional (European) scenario, and
3. a global transition scenario.

Among the more strategic and policy oriented NEA activities related to fast reactors could be mentioned a recent study on issues raised by the transition from thermal to fast nuclear systems [3]. The overall goal of the later study was to provide a comprehensive overview on issues raised by the transition from thermal to fast neutron reactors and fuel cycles with emphasis on topics of interest to policy makers. Key parameters affecting the cost/benefit analysis of transitioning include the size and age of the nuclear reactor fleet, the expected future reliance on nuclear energy, access to uranium resources, domestic nuclear infrastructure and technology development, and radioactive waste management policy in place. The study took advantage of previous work carried out by the NEA, including the scientific transition scenario study mentioned above.

The NEA is also an active player in many other international activities related to fast neutron systems, such as the Generation-IV International Forum (GIF) where the NEA provides the technical secretariat support for the project.

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Plenary Session 3:

Keynote papers

Advanced concepts and coolant technologies

Advanced and innovative reactor concept designs, associated objectives and driving forces

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Nuclear Energy appears more and more as an option which cannot be ignored, in the quest for solutions that meet the increasing world energy demand all while reducing the release of green house gases. Today's world installed nuclear capacity amounts to some 370 GWe, which contributes to about 15% of the world's electricity generation. In the next decades, nuclear electricity production will mainly originate from third generation light water reactors (LWRs), which are safe, reliable and efficient.

To date, LWRs consume less than 1% of natural uranium, and the issue of uranium resources will appear more acute as the size of the LWR fleet grows bigger. As for the spent fuel, some countries have recourse to an open cycle, while others have adopted a closed cycle with spent fuel treatment and partial recycling. Open cycle leads to storage then to geological disposal of the bulk of spent fuel, which is an option that does not seem compatible with a strong increase of the nuclear fleet in a large number of countries. Closed cycle allows for the sorting out of the different components of the spent fuel: uranium and plutonium can be recycled once in LWRs, which enables a 20 to 30% saving of natural uranium consumption, the ultimate wastes being conditioned within glass to be stored under simpler and safer conditions. However LWRs do not allow for plutonium multi-recycling: used MOX fuel can be stored for further recycling, in fast spectrum rectors, of the plutonium it contains.

Fast Neutron Reactors (FNRs) enables the expansion of nuclear energy all while meeting sustainable development criteria: resource saving and more complete waste management. Many countries are interested in this promising reactor type: sodium cooled FNRs, for which extensive feedback experience exists, corresponding to tens of reactors worldwide, is the most mature technology, heavy metals cooled FNR (lead or lead-bismuth eutectic) may be an alternate to sodium, gas cooled FNR (helium) may gather advantages of FNR and other applications than electricity generation.

The capacity to manage waste is characterised by the possibility to recycle indefinitely plutonium and maybe minor actinides, which are, after plutonium, the main contributors to long-term radiotoxicity. Plutonium multi-recycling is characterized by the breeding ratio, which represents the ratio between the plutonium generated from uranium 238 and the plutonium consumed. With a breeding ratio smaller than 1, a FNR operates in burner mode, i.e. it is used to consume plutonium and burn the other actinides produced in a LWR fleet; if the breeding-ratio is greater than 1, the FNR operates in breeder mode, and has the potential to be self sufficient with no recourse to a LWR fleet; the greater the breeding ratio, the greater the capacity of the reactors to provide fuel not only for their own future operation, but also to start new FNRs. Both modes are not antagonistic, those countries which favour a burner mode could move towards a breeder mode when they consider the timing appropriate: economic competitivity of FNRs, increase of uranium cost…

Today, three countries are engaged in the construction of sodium-cooled fast reactors: China is building the CEFR, an experimental 65 MW th (20 MWe) reactor that should reach criticality in 2009; India is constructing the PFBR, a 500 MWe prototype reactor, which should become critical in 2010, and Russia which is constructing BN 800 (800 MWe),

expected to reach criticality around 2012. These three countries have chosen the breeder mode; apart from them, various other countries are engaged in R&D programs on sodium-cooled fast reactors, and are considering new builds beyond 2020, namely France, Japan, South Korea; the United States wish to pursue an upfront R&D program, with projects for new builds pushed back by at least two decades.

Before deployment on an industrial and commercial basis, innovative components features and designs have to be developed and qualified:

For the general design, the loop concept is investigated in Japan while other countries focus on the pool concept. Simplification of design will be needed to meet economical objectives.

As for the fuel, oxide fuel has more feedback experience, but metal fuel is studied mainly because of its performances with regards to the breeding ratio as well as with regards to possibly simpler manufacturing process. Carbide fuel is also under consideration because of its high density, its high fusion temperature and good thermal conductivity. Fuel development should take into account manufacturing processes (remote handling in case of minor actinide presence), behaviour under irradiation in normal and accidental conditions, and compatibility with treatment processes.

Safety, especially behaviour under severe accidental conditions should be a field of major progress. Re-criticality should be well mastered. In support of safety, in service inspection & repair has to be improved to cope with the issues of sodium opacity and temperature of cold shutdown.

Energy conversion leads to the study of steam generators more resistant with regards to water-sodium reaction, or to the use of supercritical CO2 enabling higher yields.

Treatment and recycling are part of the system; hydro-metallurgical processes have reached an industrial level, and are being further investigated for additional improvements regarding minor actinide separation and non proliferation. Pyro-metallurgical processes are considered for metal fuels.

Proliferation resistance, like safety, should offer high guarantee levels in order to meet sustainable development targets.

Apart from national R&D programs that support all new projects, there exist international collaboration structures:

- The Generation IV International Forum (GIF), which aims at performing R&D on 4^{th} generation systems, among which sodium-cooled fast reactors. China, the Republic of South Korea, the European Union, France, Japan, the US and soon Russia are involved in the sodium-cooled fast reactor system arrangement.

- INPRO, launched in 2000 by the IAEA, gathers countries developing technology as well as user countries. By taking into account the needs of users all while defining common standards, especially in the fields of proliferation resistance, safety, waste management, INPRO program has largely contributed to setting up a shared methodology.

The INPRO program and the GenIV programs are complementary and have established links between one another.

Liquid metal coolants technology for fast reactors

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More than 50 years ago Academician A.I. Leipunsky initiated settling of liquid metals as coolants for nuclear power facilities (NPF) in the USSR. During this period requirements to the implementation of such coolants have been formulated. They covered characteristics and purposes of developed NPF. Sodium had been chosen as a coolant for the nuclear power plant (NPP) with fast breeder reactors and eutectic lead-bismuth alloy had been chosen as a coolant for nuclear-powered submarine (NPS) after comparison of coolant properties (physical, thermo-physical, chemical and others) with the consideration of the requirements for coolants.

Within the last years sodium coolant researches have passed a way from extensive laboratory investigations to long-term experience accumulated in designing, building and operation of NPF: starting from experimental nuclear reactor BR-5 (10) and experimental nuclear power facility BOR-60 to the industrial NPP BN-350 and BN-600. In further R&D works on technology and physical chemistry of sodium it allowed considering in a short time experience of industrial designing and operation. To fix upon sodium as a coolant has been made not only in the USSR, but also in all other countries where similar investigations were carried out (France, Great Britain, USA, Germany etc.).

As a result of this:

- physical and chemical characteristics of system “sodium-impurity-stucture (technological) materials” were studied: sources of impurity, their conditions and solubility in the coolant, kinetics of impurity interactions with the coolant and structure (technological) materials, behavior of impurity in NPF loops with account to the presence of disperse phase in the coolant, the possible negative consequences caused by the presence of impurity during NPP campaign;
- the allowable content of impurity in the coolant and protective gas is proved and the requirements to the coolant purity are developed;
- methods of clearing of the coolant and protective gas from impurity and monitoring of contamination both with sodium sampling and in a loop providing necessary purity of sodium are selected;
- it was developed cold traps (CT) for sodium cleansing of oxygen, hydrogen, tritium, iodine, products of structure materials corrosion (partially), for cleansing of cesium – graphite traps, and for the control over the concentration of impurity – pith indicators, sensor with diffusion membranes (DMS), electrochemical cells, special samplers.

Highly effective methods of regeneration of cold trap and also clearing of equipments and loop, in general, from remainder sodium and impurity are developed, with providing effective and economical repair works and successful further operation of nuclear facility.

The results obtained have provided not only successful operation of created NPF during scheduled term with nominal parameters with high installed capacity factor but also have played an important part in a substantiation of resource prolongation for BOR-60 and BN-600. As a result, decisions on the continuation of building of BN-800 and the beginning of investigations on development of competitive NPP with high power (BN-K) have been made.

Works on sodium technology with reference to development of these NPP are directed to improvement of their economical characteristics and increased safety – modelling of physical-chemical, thermohydraulic and technological processes in sodium contours of NPF, research of mass transfer and accumulation of tritium in contours of NPF, methods of cooling and reliable tritium localisation, improvements of devices of impurity monitoring in sodium, development of modern methods and devices for cleansing sodium of impurity, destruction and processing of radioactive sodium, carrying out of complex testing of emergency protection systems of steam generators, a scientific substantiation of technological modes in sodium loops on higher parametres of the coolant.

The choice of eutectic lead-bismuth alloy as a coolant in NPS has been determined by the complex of its physical and chemical properties. Low chemical interaction activity with air, water, steam and high boiling temperature exclude the possibility of explosion and fire, onset of boiling in high power intense sections of NPF. Low working pressure in a contour raises reliability and safety, simplifies design and equipment manufacturing, make essentially easier working conditions of primary equipment.

At the first stage of researches scientific foundation, methods and devices of Pb-Bi technology for transport NPF have been developed. The complex of methods and techniques has been operated carefully with numerous test facilities, a ground-based prototype of natural transport NPF and was introduced to industrial plants. There are no breakdowns of developed technology on ship projects 705 and 705 K due to the coolant conditions during all periods of their work (80 reactor-years).

Last decades in Russia the investigations of heavy coolant are conducted for fast stationary reactors (SVBR, BREST, etc) and acceleration driven systems. Except lead-bismuth alloy it is suggested to use pure lead as the coolant (BREST-OD-300). Lead is attractive due to its profitability and low prices; its polonium activity is approximately 1000 times as less than that in lead-bismuth alloy. However, high melting temperature (327°C) causes some difficulties when Pb-Bi is used as a coolant. The conformity of basic physical-chemical processes in loops with lead and lead-bismuth coolants was experimentally approved. It allows using for investigation of lead the working out for lead-bismuth coolant.

As a result, the following issues have been developed for substantiation of heavy-metal nuclear power engineering facilities: scientific bases of treatment of coolant, new constructional materials, methods and devices for monitoring coolant quality, methods and devices for impurity removal from the coolant and loop surfaces.

As for new generation NPFs, that factors appeared, which define new technological problems of heavy metal coolants. They are scale, resource, configuration, regime parametres.

Results of complex researches of lead coolant technology allowed working out the project «Regulations on technology of the reference with the lead coolant in Reactor Facility BREST-OD-300». In this project the main actions for lead coolant technology at all stages of building, putting into operation and campaign of BREST-OD-300 are stated.

Plenary Session 4:

Keynote papers

Safety and materials

Design and Assessment Approach on Advanced SFR Safety with Emphasis on Core Disruptive Accident Issue

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A basic safety approach in designing sodium fast reactors (SFRs) is essentially the same as one taken in light water reactors (LWRs). The concept of defence in depth (DiD), as widely applied to the design of LWRs, shall be applied to the safety design of SFRs. With a primary emphasis on preventing and detecting abnormal occurrences, safety design measures shall be provided under postulated abnormal conditions. Those are the appropriate means to shutdown the reactor, cool the residual heat in the reactor core and contain radioactive materials within the reactor facility. Even though the philosophy involved in the DiD concept has been universally accepted, the SFR-specific issues shall be taken into account in technical implementation.

The recriticality issue on core disruptive accident (CDA) is highly important in the commercialization of SFRs. Because the fast reactor core is not in the highest reactivity configuration, the recriticality issue in CDA condition has been one of the major safety issues of SFR from the begining of its development history. The conventional safety approach to this issue is (1) to minimize the occurrence probability of CDA by utilizing, for example, two independent reliable reactor shutdown systems, and (2) to assess the mechanical energy release due to recriticality events assuming hypothetical CDA, confirming the integrity of reactor vessel and component against the estimated mechanical energy and/or loading due to burning of sodium that could be spilled out from the reactor vessel. The assessment method of the CDA has been improved from the very beginning Bethe-Tait model in 1956, which assumes the gravitational fall down of the core fuel, to the recent more mechanistic models such as SAS 4A and SIMMER-III code, which consider various material motion and phase-change mechanism based on in-pile and out-of-pile experiments. Even though their early designs consider CDAs directly in the safety design, the treatment in safety evaluation is Beyond Design Basis Events with best-estimate method and assumptions. The purpose of CDA analysis has been therefore to provide or confirm an additional safety margin of the plant strictly designed for Design Basis Events. [1]

Generation IV Nuclear Energy Systems are being developed under the initiative of Generation IV International Forum (GIF) begun in 2000. The SFR was selected as one of the promising concepts together with other five concepts. Three goals for the Generation IV nuclear systems have been defined in the safety and reliability as listed below.[2]

- *Safety and Reliability-1, Generation IV nuclear energy systems operations will excel in safety and reliability.*
- *Safety and Reliability-2, Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.*
- *Safety and Reliability-3, Generation IV nuclear energy systems will eliminate the need for offsite emergency response.*

From a viewpoint of DiD philosophy, for the purpose of eliminating the need for the fifth level, which is the off-site emergency response, we need to strengthen the safety design of the fourth level of DiD, which is severe accident management. On the other hand, there is the fact that emergency response plans have been already prepared in compliance with national laws and regulations in many countries. In this sense it is effective to provide design measures to mitigate postulated severe accidents within a plant and/or to provide sufficient grace period to reach core damage and/or containment failure for the recovery by operator and for the judgement of proclamation of emergency response by authority taking into account the characteristic of severe accident progression.

To effectively meet the Generation-IV systems goals, advanced SFR designs exploit passive safety features to increase safety margins and to enhance reliability. The system behavior will vary depending on system size, design features, and fuel type. R&D for passive safety will investigate phenomena such as reactivity feedback due to axial fuel expansion and radial core expansion, and design features such as self-actuated shutdown systems (SASS) and passive decay heat removal systems via natural circulation. A preventive part of the fourth level of DiD can be strengthened by provisions of passive safety features, but their function and effectiveness should be demonstrated.

The favorable passive safety behavior of SFRs is expected to virtually exclude the possibility of severe accidents with potential for core damage. Nevertheless, design measures to mitigate the consequences of severe accidents are being considered because of recriticality potential. This approach is consistent with the DiD philosophy of providing additional safety margin beyond the design basis. In particular the safety approach to eliminate the severe recriticality will be highly useful, because with this approach, severe accidents in SFRs can be simply regarded as similar to LWRs. In addition, it is much easier to achieve in-vessel cooling and retention of post-accident core debris in SFRs because of excellent heat-transport characteristics of sodium coolant. Achieving this level of safety should result in licensing and regulatory simplifications that may in turn result in reduced system cost.

The above deterministic safety approach is complemented by probabilistic safety evaluation, which verify design features that assure very high levels of public health and safety. A risk-informed approach in design stage is desired for attaining well-balanced safety design. Although reliability data and initiating event frequencies on SFRs are not sufficient, probabilistic safety assessment should be extremely beneficial for systematically comprehending the risk characteristics of a plant with repect to a risk potential. Design improvement can be effectively made in such a way of appropriately controlling and minimizing the risk.

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Structural Materials: New Challenges, Manufacturing and Performance

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This paper reviews international perspectives on materials, manufacturing and performance on structural materials for fast reactors. It is recognized that fast reactors with closed fuel cycle shall play an eminent and major role to realize energy sustainability. Large scale exploitation of fast reactors shall require meeting of sustainability requirements such as economic competitiveness, safe and optimized waste management, increased proliferation resistance, improved use of uranium and thorium and enhanced efficiency. The designers of these reactors along with material specialists have a key role in meeting the above mentioned criteria of sustainability.

The paper shall describe features of advanced sodium cooled fast reactors and the resulting requirements on the performance of the structural materials. The choice of the fuel governs the cladding material for these reactors. Considering the choice as oxide, carbide, metallic with or without minor actinides; cladding materials are chosen based on performance modeling, available experience and expertise. Improved varieties of 316 austenitic stainless steels and oxide dispersion ferritice-martensitic steels emerge as front runners to meet the requirements. The developments in Ti modified 316 austenitic stainless steels give enough confidence to take these fuels to burn-up of 120,000 MWd/t. It is inferred that the target burn-up of 250,000 MWd/t would only be achieved with oxide dispersion strengthened iron-chromium base steels. Research and development of modified austenitic stainless steels and ODS alloys in Japan, Europe and India would be described in the paper. For achieving high burn-up of 250,000 MWd/t, ferritic-martensitic steel has emerged as the choice for the wrapper material. The current status of development and the challenges shall be described in the paper.

In particular, the European contribution to this paper addresses the development and performance assessment of the preferred materials in a cross cutting context. Indeed, it has been shown that the previously mentioned classes of steels have been selected as reference materials for more than one Generation IV fast reactor system, where demanding operating conditions (e.g. high temperatures, high stresses and high irradiation dose, corrosive coolants) are envisaged. The cross cutting aspects are related to the fabrication of the structural materials, their welding/joining and their qualification, in terms of mechanical and corrosion resistance in appropriate conditions and under neutron irradiation. Finally, the experimental findings are supported by multiscale modelling activities.
The structural materials for the sodium cooled reactor include variation of low carbon 304 and 316 austenitic stainless steels and modified 9Cr-1Mo steels. The assessment of their performance for 60 years lifetime of the reactor poses challenges to the structural mechanics,

experts and material specialists. Some of these challenges and possible solutions would be described in the paper.

Cost competitive manufacturing of high performance components relies on modeling and state of art manufacturing technologies with sophisticated testing and evaluation methodologies. Current state of art manufacturing science and technology shall be described and discussed.

The Japanese contribution to this paper describes the latest achievements of research and development program on materials conducted as a part of the Fast Reactor Cycle Technology Development Project (FaCT Project) led by Japan Atomic Energy Agency for the Japanese demonstration fast reactor whose operation is envisioned in around 2025. In this context, pursuing improved safety, reliability and economic competitiveness of the demonstration fast reactor, extensive research and development is being conducted to apply new materials to major components. Particularly for the reactor vessel and internal structures, 316FR steel, fast reactor grade 316 Stainless Steel, will be used to make the vessel creep-resistant as much as possible. For primary coolant systems, internal heat exchangers, secondary coolant systems and steam generators, Modified 9Cr-1Mo steel will be used to simplify the design of coolant systems. Development of various evaluation technologies for 60-year design with emphasis on welded joints and associated code qualification efforts are being elaborated in a wide perspective.

The authors, based on the wealth of experience in Japan, European Union (particularly within the Euratom Framework programs) and India, aim to define the challenges, current status of art and set the agenda for R&D in the coming years and decades. It is hoped that the joint perspective would enable realizing expected criteria of sustainability envisaged through sodium cooled fast reactors and closed fuel cycles.

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Plenary Session 5:

Invited papers and a keynote paper

Fuels and fuel cycles

Fast Reactor Fuel Development in Japan

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Japan launched the Fast Reactor Cycle Technology Development (FaCT) project in 2006. The primary concepts of FaCT project are sodium cooled fast reactors, the advanced aqueous reprocessing and the simplified pelletizing fuel fabrication. In the FaCT project, the design studies and the R&Ds on innovative technologies regarding the main concepts are conducted to present the specific concepts of demonstration and commercial FR cycle facilities by around 2015. This activity will be followed by further developmental effort to realize the first commercial fast reactor before 2050 based on the experience of demonstration reactor to be constructed and operated in a meantime.

Japan constructed two fast reactors, Joyo and Monju, and Joyo has been being utilized as an excellent irradiation tool of fast reactor fuel development. Joyo has extended capability of irradiation test by applying various irradiation rigs including material test, fuel pin test, fuel pin bundle test and off-normal condition test such as fuel power-to-melt test. Joyo has achieved high burnup and high neutron dose in its fuel and material irradiation rigs. Monju is under preparation to be utilized for fuel sub-assembly demonstration of future fast reactor fuels. Joyo and Monju will fill a role of fast reactors to make a progress of fuel development for the FaCT project and future commercial reactors.

In the FaCT project, JSFR is a reference concept of sodium cooled fast reactors. Its core and fuel are designed to achieve high burnup and high core outlet temperature and to be capable of minor actinide bearing fuel loading with homogeneous recycling manner. Its reference fuel is oxide fuel and alternative is metal fuel. Aiming at achieving high burnup, oxide dispersion ferritic steel (ODS) cladding and PNC-FMS sub-assembly duct were selected as reference core materials. Fuel pin design is large diameter annular pellet fuel, which gives advantage of high fuel volume fraction in the core, low fuel smeared density to accommodate fuel swelling at high burnup and fuel fabrication consistent with low fuel smeared density design.

Fuel irradiation tests were planed to develop future fast reactor fuel in the FaCT project. They include MA bearing fuel irradiation test, material irradiation test of high burnup core materials, ODS clad fuel pin irradiation test, large diameter annular fuel irradiation test and fuel power-to-melt test. Material irradiation test and MA bearing fuel irradiation test have been already started. Short term irradiation test of MA bearing oxide fuel supplied important irradiation data such as early-in-life fuel restructuring of MA bearing fuel and MA redistribution as well as Pu redistribution. Irradiation test will continue to supply further irradiation data, which lead to sub-assembly demonstration irradiations in Monju.

Fuel property study is a significant issue of MA bearing oxide fuel. JAEA has been investigating MA bearing fuel properties by the out-of-pile experimental studies and the analytical studies. Major properties such as fuel melting point, thermal diffusivity and specific heat have been experimentally studied. Current results show the limited contribution of minor actinides on fuel properties for oxide fuels of homogenous recycling of minor actinides.

Fuel fabrication technology is also a key issue of future fast reactor fuel. The simplified pelletizing process under development aims at reducing the pellet fabrication processes due to less oxide fuel powder treatment processes and less organic additives. Its key technologies are homogeneous oxide powder supply by the micro wave conversion process of U-Pu solution, binder-less granulation and die-lubrication pressing. The micro wave conversion process was already established and development of other two technologies are in good progress. Irradiation behavior of fuel pellets made from micro wave conversion powder was preliminary investigated, too.

Metal fuel development is also in progress in Japan. CRIEPI play an important roll to promote the development. JAEA is collaborating with CRIEPI in the fields of metal fuel fabrication and irradiation test. Preparatory activities of Joyo irradiation test are in progress by domestic fabrication of U-Pu-Zr metal fuel.

International collaborations are effective measure of fast reactor fuel developmental effort. One of such activities with related to fuel development in the FaCT project is Global Actinide Cycle International Demonstration (GACID) project by CEA, DOE and JAEA as a part of GIF/SFR international collaboration. The GACID project aims at demonstrating the MA transmutation capability and MA bearing fuel integrity in a fast reactor core, using Joyo and Monju. The project consists of a phased approach in three steps, Np/Am bearing oxide fuel pin irradiation as step-1, Np/Am/Cm bearing oxide fuel pin irradiationas step-2 and Np/Am/Cm bearing oxide fuel bundle irradiation demonstration as step-3.

As a conclusion, fuel development for future fast reactors are in progress as a part of FaCT project in Japan. Developmental effort includes irradiation tests, fuel fabrication technology development and out-of-pile studies such as fuel property investigations. International collaborative effor is also an improtant part of such activities.

Fuels for advanced sodium cooled fast reactors in Russia: state-of-art and prospects.

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The thought-out and consistent program of sodium cooled fast reactor (SFR) development is implemented in our country: BR-1 (1954) → BR-2 (1956) → BR-5 (1959) →BR-10 (1973) → BOR-60 (1969) → BN-350 (1973) → BN-600 (1980) → BN-800 (under construction) → BN-1800, BN-1200 (under development). In the BR-5, BR-10, BOR-60, BN-350, BN-600 reactors the different fuel types (PuO_2, UC, UN, UPuN, UO_2, $UPuO_2$, alloyed and non-alloyed metallic fuels, inert-matrices fuels) have been irradiated and investigated. Currently the investigations are carried out in the BOR-60 and BN-600 reactors.

FUEL PERFORMANCE PROBLEM FOR ADVANCED CORES

BN-800. On the current stage the construction of the 4-th power unit of Beloyarskaya NPP with the BN-800 reactor is quite important in order to increase competitiveness and safety, creation of the components of fast reactor fuel cycle and decrease of its environmental impact. The BN-800 core design with MOX fuel has been developed. The decision is done to use vipac MOX fuel as a standard BN-800 fuel. The maximum fuel burn-up is about 10at%. The specific feature of the core designed is its reversibility, allowing the nitride core using.

BN-1800. The conceptual design of the BN-1800 reactor with pellet MOX fuel core (17at% of maximum burn-up - I stage and 20at% - II stage) and the technical proposal for the BN-1800 reactor with nitride core have been developed. The experience has shown, that the achievement of burn-up values corresponded to the doses > 100-120dpa is limited by the high swelling of austenitic claddings steels and corresponding rapid deterioration of its mechanical properties. Thereby for the advanced cores with doses >200dpa the non-swelling ferritic-martensitic steels are considered as cladding materials candidates. Two directions of BN-K cores development are considered with the account of irradiation stability of cladding steels: (1)“low”-temperature core with maximum cladding temperature no more than 650-660^{0}C, (2)“traditional” core with maximum cladding temperature 700^{0}C. For the “low”-temperature core the EP-450 (12Cr-2Mo-Nb-B) steel is proposed for the cladding. For the second direction the ODS steel is considered.

BN-1200. Currently the R&D work is under way on the development of commercial fast sodium BN-1200 reactor with the maximum using of already time-tested and scientifically based technical decisions, realized on BN-800 and BN-1800 designs. Very high burn-up is specified for BN-1200 core (as well as for BN-1800), and accordingly, very high level of irradiation dose. The possibility of MA utilization is assumed also. MA fuel of homogeneous type with small content of MA (up to ~3%) and of heterogeneous type with high MA content (with inert matrices) is innovative fuel. In parallel with traditional core options the heterogeneous cores with oxide and metal are under investigation in order to increase the breeding ratio.

PRINCIPAL RESULTS OF FUELS INVESTIGATION

MOX fuel. Two MOX fabrication techniques (vibropacking and pellets) have been developed in Russia. The pellet MOX fabrication methods have been developed in VNIINM, Moscow. Two methods were used for fuel powder preparation: mechanical mixing of initial PuO_2 and UO_2 (the MMO method) and mixed co-precipitation of U and Pu dioxides (the GRANAT method). The fabrication technique of vibropack MOX fuel has been developed in RIAR, Dimitrovgrad. The required quality granulate is received by combined electro-crystallization of UO_2-PuO_2 from molten alkali metal chlorides. The technology of vibropack MOX fuel fabrication by mechanical oxides mixing is developed also. The method enables to provide the predetermined Pu distribution and MA inclusion. The vibropacked co-precipitated MOX fuel is the BOR-60 driver fuel with maximum burn-up of 15at%. The peak burn-up of 30at% is achieved for BOR-60 several experimental assemblies. The experimental fuel assemblies with pellet and vibropacked MOX fuel have been irradiated at the BN-600 reactor. The maximum burn-up of pellet MOX fuel is ~12at%, of vibropacked MOX fuel – 10.5at%.

Nitride. The principal investigation results of two nitride cores of the BR-10 reactor are given. The summary of mixed nitride investigations at the BOR-60 reactor is given, including BORA-BORA program carried out in the frame of CEA (France) - ROSATOM (Russia) collaboration. The nitride maximum burn-up is 12.1at%.

Metal fuel. Two techniques for the alloyed metal fuel fabrication are under study in Russia: hot extrusion and casting method. Through the different programs including collaboration with KAERI, the BR-10, BOR-60 fuel pins with U-Zr, U-Pu-Zr have been fabricated. One full scale fuel assembly with U-Pu-Zr has been irradiated at the BOR-60 reactor up to 10at%. The pins with unalloyed U and U-Pu fuel (smeared density 13…17 g h.a./cm^3) have been fabricated and irradiated at BOR-60, BN-350 reactors.

Inert matrices fuel. In the frame of BORA-BORA program the irradiation and post-irradiation examinations of two pins with (PuZr)N and 2 pins with PuO_2+MgO in the BOR-60 reactor have been completed. Through the ISTC MATINE Project the measurements of high temperature creep and thermal stability, thermal conductivity of (PuZr)N have been done. The principal results of the investigations are given.

MA fuel. The possibility of Am recycling using the pyrochemistry process for inert matrix fuel fabrication has been studied (collaborative CEA (France) - ROSATOM (Russia) AMBOINE program). The program included the fabrication of BOR-60 experimental fuel pin with vipac $(UAm)O_2$ in the core and with vipac $UAmO_2$+MgO in the axial blankets. Investigations on Am/REE (rare earth elements) and MgO separation by selective precipitation in molten salts have been carried out also. In the frame of ISTC MATINE Project the possibility is considered on the fabrication of (Pu,Am,Cm,Zr)N fuel containing up to 10mol% Cm on the RIAR site. The electrolytic refining in the molten chlorides on the liquid metal cathode is proposed as the main flow sheet. The technical-economical estimations have shown the technical feasibility of the offered processes at the use of technical base, rules and technology requirements of RIAR. Through the DOVITA program the irradiation of $UNpO_2$ fuel to 20at% has been done at the BOR-60 reactor.

Calculation codes development In order to prove the fuel performance under conditions of advanced SFR cores the calculation codes have been developed or up-dated recently in IPPE, VNIINM, RIAR. The validation of the codes is carried using the BORA-BORA results as well. Some results are given.

Advanced Fuels for Fast Reactors

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In addition to traditional fast reactor fuels that contain Uranium and Plutonium, the advanced fast reactor fuels are likely to include the minor actinides [Neptunium (Np), Americium (Am) and Curium (Cm)]. Such fuels are also referred to as transmutation fuels. The goal of transmutation fuel development programs is to develop and qualify a nuclear fuel system that performs all of the functions of a traditional fast spectrum nuclear fuel while destroying recycled actinides. Oxide, metal,nitride, and carbide fuels are candidates under consideration for this application, based on historical knowledge of fast reactor fuel development and specific fuel tests currently being conducted in international transmutation fuel development programs. Early fast reactor developers originally favored metal alloy fuel due to its high density and potential for breeder operation. The focus of pressurized water reactor development on oxide fuel and the subsequent adoption by the commercial nuclear power industry, however, along with early issues with low burnup potential of metal fuel (now resolved), led later fast reactor development programs to favor oxide fuels. Carbide and nitride fuels have also been investigated but are at a much lower state of development than metal and oxide fuels, with limited large scale reactor irradiation experience.

Experience with both metal and oxide fuels has established that either fuel type will meet performance and reliability goals for a plutonium fueled fast spectrum test reactor,i both demonstrating burnup capability of up to 20 at.% under normal operating conditions,[1, 2,3, 4, 5] when clad with modified austenitic or ferritic martensitic stainless steel alloys. Both metal and oxide fuels have been shown to exhibit sufficient margin to failure under transient conditions for successful reactor operation [6, 7, 8, 9 10]. Summary of selected fuel material properties taken are provided in the paper.

The main challenge for the development of transmutation fast reactor fuels originates from goals for achieving high burnup, operating at higher temperature, and the incorporation of the minor actinides (Np, Am, Cm) into the fuels. High burn-ups will allow uninterrupted reactor operations over longer periods of time and consequently, reduction of spent fuel volumes, and eventually a significant fuel cycle reduction cost. High burn-ups are however associated with physical limitations which are primary due the swelling of the fuel and oxidation of cladding inner surface as well as the dimensional stability of core materials such as cladding and subassembly duct due to high fast neutron dose. Higher temperature operation also challenges the performance of cladding materials and hence advanced cladding materials are needed for high temperature operation. The irradiation performance database for (U,Pu)N mixed nitride (MN) fuels is substantially smaller than that for metal carbide (MC) fuels, and these fuels can be considered to be at an early stage of development relative to oxide and metal fuels. Compared to MC fuels, MN fuels exhibit less fuel swelling, lower fission gas release,; however, the problem of the production of biologically hazardous 14C in nitride fuels fabricated using natural nitrogen poses a considerable concern for the nitride spent fuel waste management.xiv Interest remains in nitride fuels due to the combination of high thermal conductivity and high melting point. [12,13].

The paper also addresses the technology readiness level (TRL) concept as applied to various fuel options.

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Fast Reactor Fuel Programmes in Europe

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Fast reactor fuel development in Europe was in integral part of the process in the realisation of the Phénix and SUPERPHENIX reactors. Both reactors used oxide fuel. The pellets were of different pitch, with the larger fuel pellets deployed in SUPERPHENIX having an annular form to limit the centreline temperature. In parallel during the 70s and 80s Europe had extensive fuel programmes on alternative fuels, namely the mixed metal carbides and nitrides. The latter (MX) fuels have high heavy metal density, favourable for Pu breeding, and high thermal conductivity, allowing their operation at about 30% of the melting temperature, as opposed to oxide or metal fuels which operate at about 80% of the melting temperature. Despite several intensive R&D programmes notably in Germany (in KNK II), these fuels were not deployed.

At that time, the potential of a fully closed actinide recycle was recognised. The minor actinides (MA) and the Pu should be separated from the spent fuel and recycled, so that a substantial decrease in the long term radiotoxicity of the waste is achieved, i.e. from 100,000 to 500 years. Within this concept, the SUPERFACT experiment, performed by CEA and ITU in the Phenix reactor, represents a major milestone. The oxide fuels tested consisted of dedicated targets with high MA content and no Pu, and fuels for full reactor core recycle, namely U-Pu MOX fuel with small (e.g.2%) MA content. The irradiation achieved an Am transmutation rate of 30%, and no abnormal behaviour was noted. Higher than usual helium production and release, due to the presence of Am, was observed.

In recent years, dedicated targets for actinide transmutation have been investigated within the framework of European programmes on partitioning and transmutation (P&T). Many such investigations have concentrated on the inert matrix fuel (IMF) concept. These fuels have no fertile U content, and thus have the potential to increase the transmutation rate. The lack of fertile U and the presence of MA combine to degrade the reactor safety coefficients, which could be offset by operating dedicated reactors in a sub critical mode, whereby additional neutrons to reach criticality are provided by spallation, when high energy protons impinge on suitable target. This is the basis of the accelerator driven system (ADS).

The first IMF experiment in Europe utilising the IMF concept consisted of a target ($AmAlO_3$ in a $MgAl_2O_4$ matrix) which was irradiated in the HFR Petten. A remarkable fuel volumetric swelling of 24% was observed and was caused by helium generation, and retention in the fuel. The operating temperature was relatively low and limited helium release to the plenum.

The FUTURIX irradiation programme in the Phénix reactor has just been completed. CEA and ITU prepared CERCER and CERMET (based on MgO and Mo, respectively) fuels for

this irradiation experiment. The fuels were irradiated for 235 equivalent full power days (EFPD) alongside metallic and nitride fuels produced at INL and LANL. PIE will be performed in 2010.

HELIOS is a dedicated experiment in the HFR Petten to investigate helium behaviour in IMF. The MA was always Am. The temperature in the samples during irradiation is increased by the addition of Pu to increase the linear rating, or in contrast by dispersion in Mo, which leads to a lower fuel operating temperature. Optimised inert matrix microstructure, with tailored open porosity favourable to helium release, is also being tested. The irradiation programme will be completed in 2010.

The Generation IV initiative is leading to a renewed interest in fast reactors in Europe and elsewhere. MOX is the fuel of choice for the start up core of a demonstrator reactor. Nevertheless, any design must permit the fuelling of this reactor with minor actinides. Two recycle modes for MA can be considered. In the homogeneous mode the minor actinides are added to the MOX fuel in small quantities, i.e. 2-5% depending on the need to introduce MA from existing LWR fleets. In contrast, the minor actinides can be recycled in a heterogeneous mode as targets in dedicated assemblies near the periphery of the core. Here the fuel should be composed of a suitable matrix and the MA, the latter with a loading of about 20%. As the Gen IV fast reactors must be self sustaining, the matrix should be fertile. Depleted UO_2 is an ideal choice, and has the additional advantage that it is reprocessable.

New irradiation programmes are now underway to investigate further these advanced oxide fuels. These are based on a variety of basic scientific studies to determine safety relevant properties such as thermal conductivity, and vaporisation behaviour. In Europe, the closure of the Phénix reactor on March 2009 results in a complete absence of suitable fast spectrum irradiation sources, leaving only material testing reactors (MTR) for such studies. The MARIOS irradiation experiment in the HFR Petten investigates helium release for dedicated $(U,Am)O_2$ disks, loaded in capsules held at dedicated temperatures, with low temperature gradients. The SPHERE irradiation programme compares the performance of pellet and SPHEREPAC fuel, composed of $(U,Pu,Am)O_2$.

For the future, European fuel R&D will be closely aligned to the strategic research agenda of the European Sustainable Energy Technology Platform (SNETP).

Recycle Strategies for Fast Reactors and Related Fuel Cycle Technologies

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1. Introduction

Fast reactors and related fuel cycle (hereinafter referred to as “Fast reactor cycle”) technologies have the potential of contributing to long-term energy security due to effective use of uranium and plutonium resources, and reduction of the heat generation and potential toxicity of high-level radioactive wastes by burning long-lived minor actinides (MA) recovered from spent fuels of light-water reactors and fast reactors.

Further, it is likely that fast reactor cycle technologies can play a certain role in non-proliferation as addressed in GNEP (Global Nuclear Energy Partnership). With these features, R&Ds toward their commercialization have been promoted vigorously and globally as a future vision of nuclear energy.

2. Recycle strategies in each country

In Japan, it is determined that after burning uranium in light water reactors, plutonium is recovered from spent fuel and used for light water reactors at the moment and for fast reactors in the future. In order to make it possible, Fast Reactor Cycle Technology Development (FaCT) Project has been promoted with a combination of oxide-fueled sodium-cooled reactors, advanced aqueous reprocessing, and simplified pelletizing fuel fabrication adopted as a main concept aiming at startup of a demonstration reactor around 2025 and commercialization before around 2050.

In France, a comparison of the basic specifications between an oxide-fueled sodium-cooled reactor and a carbide(or nitride)-fueled gas-cooled reactor has currently been promoted towards technological selection for a prototype reactor in 2012 in accordance with “The 2006 planning act on the sustainable management of radioactive materials and waste (Act 2006-739)” enacted in 2006. Based on the results, France aims at startup of the prototype reactor in 2020 and commercialization in around 2040. For reprocessing, methods which extract actinides collectively such as GANEX has been developed to enhance proliferation resistance.

Although the U.S. returned to R&Ds on recycling in the 2000s, it has made a major shift from fast reactor cycle related and specific R&Ds to rather a long-term, science-based research program in accordance with the establishment of a new administration in January, 2009.

Russia, China and India have promoted R&Ds independently depending on their own national conditions and nuclear energy policy aimed at completion of the closed cycle using fast reactors. In China and India, using MOX fuel at the moment, a shift to metal fuels is expected as countermeasures for the growth of energy demand and supply in the future.

Further, Korea addresses R&Ds on metal-fueled fast reactors and pyro-processing systems, which have a relatively high proliferation resistance.

Introduction of fast reactor cycle systems will be carried out independently by each country following the national conditions and nuclear energy policy. It should be then considered important to have globally common consensus relating to safety philosophy, concepts of proliferation resistance, TRU burnup and recycling and so on.

3. Multinational cooperation on future nuclear systems

Twelve countries and one international organization have joined GIF （Generation IV International Forum） aiming to share R&Ds and tasks on Generation IV reactor systems (six reactor systems) in which five are fast reactors, and they have been promoting cooperative development and information exchanges in the framework of GIF. A common ground of understanding GIF has been nurtured in terms of its sustainability (efficient use of resources, minimization and management of nuclear wastes), economic efficiency, safety/reliability and proliferation resistance, and its technology goals have also been clearly defined.

INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles) intends to help to ensure that nuclear energy is available in the 21st century in a sustainable manner, led by IAEA and engaging twenty nine countries and one international organization from both technology holders and technology users countries, has been improving INPRO methodology and developing institutions for introducing innovative nuclear systems including fast reactor cycle in the future.

GNEP, a voluntary international partnership, aims to expand clean, sustainable nuclear power worldwide in a safe and secure manner while responsibly managing nuclear waste and reducing proliferation risks, and one of its objectives is to develop, demonstrate, and in due course deploy advanced fast reactors that consume transuranic elements recovered from nuclear spent fuel.

Such multilateral cooperation has now been promoted and received achievements in each framework. If these frameworks of multilateral cooperation can be unified to a manageable level in the future and further developed, it will be expected that fast reactor cycle technologies can be available on a global scale.

4. Conclusion

For the development and utilization of fast reactor cycle systems, while respecting each country’s concept, it is essential to organize the technologies and concepts which should be in common globally and build a framework to make it standardizing. Utilization of existing frameworks such as GIF and INPRO are considered as effective to realize it. Furthermore, a vigorous promotion such as international cooperative developments enables formation of international consensus on major technologies for fast reactor cycle as well as saving of resources by infrastructure sharing.

Japan, as a non-nuclear-weapon nation, considers that we should play a great role in studying and creating a system where fast reactor cycle technologies can be used in peace and we would like to actively contribute to the international community. Finally, it is firmly confirmed that Japan will make a great effort to enable the global contribution as one of the few nations who have both experimental and prototype fast reactors.

Plenary Session 6:

Invited papers and a keynote paper

Retrospectives and advanced simulation

The French SFR operating experience

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In France, there are two Sodium-cooled Fast neutrons power Reactors: Phenix, a 250 MWe demonstration reactor operated by the CEA and EDF from 1973 to 2009 on the site of Marcoule, and Superphenix, a 1200 MWe prototype reactor operated by EDF (with some other European utilities) from 1985 to 1998 on the site of Creys-Malville.

During the last twenty years, the most significant events were:

- The four negative reactivity transients which occurred on Phenix reactor in 1989 and 1990, and the many studies that were carried out to explain these phenomena and demonstrate the safety of the reactor (1990-1993);
- The modification of the Creys-Malville secondary circuits and their environment, for protecting the reactor against large sodium spray fires (1992-1994);
- The French government decision to definitively shut down Superphenix in 1997;
- The lifetime extension of the Phenix reactor, that needed a number of modifications and inspections, in accordance with the updated safety requirements (1998-2003);
- The beginning of the decommissioning of the Creys-Malville plant, that needed the development of specific processes and methods;
- The good operation of the Phenix plant during its last irradiation cycles (2003-2009), and the completion of the experimental programme on nuclear waste transmutation.
- The final test campaign on the Phenix reactor (2009).

The operating experience of Phenix and Superphenix reactors and their incidents are analysed to draw lessons for the design and the operation of the future Sodium-cooled Fast neutrons Reactors. If the balance of the operation of both reactors is contrasted, in particular because of a different political context, the learnings from incidents are consistent on both reactors as well in scientific areas as in technological and operating experience feedback domains.

On core physics, fuel behaviour , reactor control, components design, core sub-assembly and component handling systems, materials and sodium technology, maintenance and in-service inspection methods, France has gained an important cognitive heritage. It is a strong basis to achieve the future requirements on risks prevention (technical, human and financial ones), on economics and on availability, that will be required for the future SFR.

THE LAST TWENTY YEARS EXPERIENCE WITH FAST REACTORS; LESSONS LEARNT AND PERSPECTIVE

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ABSTRACT

Importance of rapid growth of electricity generation capacity for development of India is well recognized. Energy planners are of the strong view that nuclear energy has to form significant portion in ensuring long term energy security for India. The Nuclear Energy Programme in India has been visualized to grow in three stages. Stage-I consists of natural uranium fuelled Pressurised Heavy Water Reactors. The known reserves in the country can support only 10 GWe of installed electrical capacity. The second stage consist of Fast Breeder Reactors. The energy potential can be realized to 530 GWe. Third stage is based on utilization of vast reserves of thorium to build reactors based on U-233 fuel. The first step in the introduction of fast reactors in the country was the construction of 40 MWth, 13.5 MWe Fast Breeder Test Reactor (FBTR) with the help of French Atomic Energy and industry. The reactor has been in operation since 1985 and is serving as excellent facility for research and development for irradiation of fuel and structural materials and has generated valuable data and experience for the successful operation of fast breeder reactors. Successful operation of FBTR has provided valuable feedback to launch the 500 MWe Prototype Fast Breeder Reactor (PFBR). Unlike FBTR, this reactor has been designed indigenously backed up by strong research and development. As PFBR would be head of a series of at least a few reactors, cost effective design has been made by decreasing the number of components like primary and secondary pumps as 2 each, increasing the design life to 40 years and detailed optimization studies for each component and system. The reactor design incorporates the lessons learnt from the operating experience of sodium cooled fast reactors and the designs in vogue at the time of finalization of PFBR design. Realizing a big jump from FBTR to PFBR, the manufacturing technology development of important nuclear components was undertaken with the participation of the Indian industries. The manufacturing technology exercise gave us the confidence that Indian industries will be able to manufacture the components to the required quality. PFBR was sanctioned in 2003. The reactor construction has progressed very well except for the incident of Tsunami in December 2004 which led to raft reconstruction. The safety vessel and sodium tanks have been erected and sodium transfer into the tanks has commenced. Civil works for nuclear island and other buildings are in advanced stage of completion. Main vessel integrated with core support structure and core catcher is ready for installation. Almost all nuclear components are either available at site or expected to be received in current year. An important feature of India's nuclear programme is close fuel cycle. Fast reactor fuel cycle facility which will cater to fuel supply for PFBR after reprocessing the spent fuel is also planned at Kalpakkam.

After initial operation of PFBR, it is planned to construct four more oxide fuelled fast breeder reactors of 500 MWe capacity by 2020. These reactors, now under design, will have enhanced safety features and improved economics. These reactors will incorporate the lessons learnt from the construction of PFBR, in particular on plant layout and strategy for reduction in construction time. To accelerate the rate of deployment of breeder reactors for meeting the future energy demands effectively, metallic fuels with short doubling time are envisaged for the reactors beyond 2020. Research and development studies have been initiated for two designs and a significant portion of FBTR driver fuel will be metallic fuel in the next 8 years.

This paper brings out on the FBTR operating experience; PFBR design, regulatory review and reactor construction status, and the perspective for future growth of nuclear energy in India through the fast reactors.

Last Twenty Years Experiences with Fast Reactors in Japan

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○ Fast Breeder Reactor (FBR) Development

Japan, which is poor in natural resources, has been developing FBRs with an unshakable national policy in order to secure domestic long-term energy resources. The “Framework for Nuclear Energy Policy of Japan” formurated in 2005 states that striveing for the commercial use of FBRs from around 2050 is one of a guidline for the promotion of nuclear power generation in the future. In the process of FBR development, we have stepped up from experimental reactor “Joyo” to prototype reactor “Monju”, and now we are promoting R&D on a demonstration reactor toward commercial reactors.

A council, which was established in 2006 by administrative authorities, utilities, vendors and JAEA aiming smooth transition to a demonstration process, agreed that it was necessary to carry out FBR R&D efficiently under a structure with clear responsibility. Then Mitsubishi Heavy Industries, Ltd. (MHI) was selected as a core company in April 2007 for R&D upto the startup of basic design for FBR demonstration reactor. In order to concentrate a responsibility, an authority and an engineering function on FBR development, MHI founded a new company Mitsubishi FBR Systems, Inc. (MFBR). This means that we established the framework for promoting FBR development in Japan based on a coordination between JAEA and MFBR cooperated with vendors and universities as a key element.

○ Experimental FR “Joyo” (140MWt)

Power Reactor and Nuclear Fuel Development Corporation (PNC, currently JAEA) started construction of the experimental FR “Joyo” in 1970. The purposes of the construction were firstly to design, construct and operate a loop-type sodium-cooled FBR (SFR) with domestic technologies, then to accumurate technical knowledge, and secondaly to carry out necessary irradiation tests of fuels and materials for FBR development.

Joyo attained first criticality with MK-I breeding core in April 1977, then it was modified twice in 1982 and 2003 to meet the requirment of enhancing irradiation performance, now it equips MK-Ⅲ irradiation core. Joyo conducted pin-scaled irradiation tests of mior actinide (MA) bearing fuel. However, Joyo has been shut down since May 2007 after an obstacle at a part of fuel assembly discharging equipment was found, and it needs repairing.

○ Prototype FBR “Monju” （714MWt, 280MWe）

To demonstrate that loop-type SFR can be used as a power reactor with the experience of Joyo applied, PNC started construction “Monju” in October 1985 and it attanined first criticality in April 1994. However, a sodium-leak incident occurred in the secondary coolant system on December 8, 1995 and its operation has shut down since then. Presently, entire system function test is in progress for resumption of operation through an overhaul and reconstruction.

The original mission of Monju is to demonstrate the reliability of FBR power plant through its operation and establish soduim handling technologies. Monju is now positioned as a core function in “the energy R&D centralization plan” that Fukui prefecture is promoting, and will be used as one of the main fields for R&D toward FR commercialization.

○ Research into commercialization

(1) Demonstration FR designed by Utilities

As for the design of demonstration FR in the next stage after Monju, Japan Atomic Power Co. (JAPC) became the center and promoted the design studies and innovative elemental technology development from 1985 to 1999. Developing the loop-type reactor technology to rationalize the design, top-entry loop-type SFR (approx. 660 MWe) was selected, and a construction cost was estimated around 1.3 times as much as that of a light-water reactor (equivalent to 1000MWe power plant).

(2) JSFR (Japanese Sodium-cooled FR)

In 1999, JAEA launched the "Feasibility Study on Commercialized Fast Reactor Cycle Systems (FS)" with domestic partners, such as utilities, vendors, and universities. The study evaluated advanced concept of FBR cycle systems with various coolants such as sodium, heavy metal, gases and water, and suitable fuel types such as oxide, nitride and metal, with viewpoints of safety, economics, reduction of environmental burden, efficient utilization of natural uranium resources, nuclear proliferation resistance, and technical feasibility.

The study developed into "Fast Reactor Cycle Technology Development (FaCT) project" started in 2006 by all-Japanese lineup led by JAEA. In FaCT project, the SFR with mix-oxide fuel was selected as the most promising concept of FR cycle system. Now we are progressing design works and development of elemental technologies toward our goals of decideing what inovative technologies should be adopted in 2010 and indicating images of commercialization and the R&D roadmap around 2015 aiming at the start of operation of a demonstration FR around 2025. With regard to fuel cycle systems, we are promoting R&D on advanced reprocessing and simplified fuel fablication, which have features of low decontamination, no handling Pu alone, and MA bearing fuel.

(3) Others (4S reactors, supercritical water reactor)

4S reactors (i.e., Super Safe, Small and Simple reactor) has a unique feature which does not need refueling over a long period of 30 years in addition to the intrinsic safety feature of FR whose reactor core can be downsized. Toshiba Co. and Central Research Institute of Electric Power Industry (CRIEPI) have carried out R&D on 4S reactors. Now, U.S. NRC is preliminarily reviewing design certification of 4S reactor.

Research on an innovative nuclear energy system, Supercritical Water-Cooled Fast Reactor (SCWR), has been promoted maily by the University of Tokyo who names it "Super Fast Reactor", using supercritical water as coolant which enables to improve generation efficiency and to simplify and compactify nuclear reactor systems.

○ International cooperation

In order to proceed with FR development both efficiently and effectively, Japan vigorously participates in international cooperation activities such as bilateral or trilateral cooperation between Japan, the U.S, and France as well as multilateral cooperation, such as GIF, GNEP, INPRO, etc. Further, for contributing to global peaceful use of FR development, JAEA works in cooperation with the project that Pu derived from dismantled nuclear weapon is burned with BN-600 in Russia (called BN-600 Vibro-Pack option).

○ Development of human resources

Research Institute of Nuclear Engineering, University of Fukui was established in April 2009, and will bring up engineers who will take a major role in FBR development for the future.

○ Path forward

As an advanced country of FR development, Japan would like to make all-out efforts to realize JSFR including early restart of Joyo and Monju.

Experience Gained in Russia on Sodium Cooled Fast Reactors and Prospects of Their Further Development

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The experience on sodium cooled fast reactors (SFR) gained in Russia in recent 20-30 years has been reviewed, and prospects of their further evolution have been analyzed taking into account the said experience.

Consistent strategy of Russia in the area of fast reactors in recent 50 years has given grounds for gaining the world's leading positions in SFR. The experience accumulated on SFR operation is especially impressive. It covers research reactors BR-5/10 (Obninsk) and BOR-60 (Dimitrovgrad), demo facility BN-350 which is now in the territory of Kazakhstan Republic (Aktau), and the first commercial power unit with fast reactor BN-600 (Zarechny). Total experience with SFR operation in the USSR and in Russia amounts to over 140 reactor years (1/3 of entire world experience of operation). The construction of BN-800 reactor and the development of advanced commercial large-size SFR design has been underway (BN-K).

The analysis of the experience with operation of BN-600 and BOR-60 reactors has been highlighted.

Statistical information has been presented on the operation indicators of the commercial power unit with BN-600 reactor, with data for recent years covered (load factor, energy production, service indicators achieved for the main elements of the reactor facility (RF), including the fuel burn-up level mastered). The program of systematic increase of fuel burn-up realized on the BN-600 has been described with principal characteristics of the BN-600 core modifications realized in the process of RF operation. A steady functioning of the BN-600 reactor and high economic indicators achieved testify to a reliable mastering of the SFR technology, which covers technology of sodium coolant, operation and replacement of the main sodium equipment (steam generators, intermediate heat exchanger, MCP). A high level of safety has been demonstrated for SFR, with minimum effects on the personnel and environment, both in terms of gaseous releases, and volumes of liquid and solid radioactive wastes.

The statistical data on deviations from normal functioning conditions that took place in the course of BN-600 operation, including that of sodium leaks and SG leaks have been reported; they show clearly that none incident associated with sodium leak has occurred during recent 15 years.

The information on the efficient work in recent 5 years aimed to the BN-600 lifetime extension for the next fifteen years has been presented, which is an evidence of both successful design&development, system, and technological solutions used in the project of BN-600 RF, as well as of viability and promising prospects of this reactor technology as a whole.

The section of this report describing the experience with operation of BOR-60 reactor presents detailed materials on the tasks resolved in the course of its operation, on the deviations from normal functioning modes that took place there. The main trends of R&D fulfilled on BOR-60 for the justification of SFR technology have been described (the material irradiation studies on various fuel compositions and structural materials, justification of various types of SG «sodium-water» and the systems for their control, research works in the area of sodium coolant technology, safety studies for SFR, etc.).

Research reactor BR-5/10 is an absolute long-liver among the SFR, its 50-year anniversary being marked early this year; now it is under preparation for decommissioning. The main results of operation of BR-5/10 reactor have been reported, as well as principal works fulfilled at the phase of preparing the reactor for decommissioning, and its current status.

The conclusion is made on a reliable industrial mastering of the SFR technology in Russia, including the sodium coolant technology; the tasks are formulated to be resolved at the next phases: the demonstration of closure of the nuclear fuel cycle based on BN-800 reactor; optimization of the characteristics and design solutions to be adapted in the subsequent SFR designs, with the aim of reducing their cost parameters to the level comparable with thermal NPP; increase of the SFR safety to the level that would meet the requirements formulated for the 4-th generation reactors; upgrading of the complex of technologies for decommissioning of SFR based on BR-10 reactor.

Advanced simulation for fast reactor design

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This talk broadly reviews recent research aimed at applying advanced simulation techniques specifically to fast neutron reactors. By *advanced simulation* we generally refer to attempts to do more *science-based simulation* – that is, to numerically solve the three-dimensional governing physical equations on fine scales and observe and study the holistic phenomena that emerge. In this way simulation is treated more akin to a traditional physical expermiment, and can can be used both separately and in conjunction with physical experiments to develop more accurate predictive theories on reactor behavior.

Many existing fast reactor modeling tools were developed for last generation's computational resources. They were built by engineers and physicists with deep physical insight -- insight that both shaped and was informed by existing theory, and was underpinned by a vast repository of experimental data. Their general approach was to develop models that were tailored to varying degrees to the details of the reactor design, using free model parameters that were subsequently calibrated to match existing experimental data. The resulting codes were thus extremely useful for their specific purpose but highly limited in their predictive capability (neutronics to a lesser degree). They tended to represent more the state-of-the-art in our understanding rather than tools of exploration and innovation.

Recently, a number of researchers have attempted to study the feasibility of solving more fundamental governing equations on realistic, three-dimensional geometries for different fast reactor sub-domains. This includes solving the Navier-Stokes equations for single-phase sodium flow (Direct Numerical Simulation, Large Eddie Simulation, and Reynolds Averaged Navier Stokes Equations) in the core, upper plenum, primary and intermediate loop, etc.; the non-homogenized transport equations at very fine group, angle, and energy discretization, and thermo-mehanical feedback based on both experimentally and computationally derived constituitive relations. Accurate coupling methodoliges and treatment of uncertainties for multi-physics systems are critical related areas of research. While in actuality there is no rigid distinction between such science-based and more calibration-based modeling approaches, this general advanced methodology is a signficant departure from the traditional approach.

This talk will give a broad, critical overview of advanced fast reactor modeling research in the context of its potential for 1) reducing uncertainties for existing fast reactor designs and 2) enabling the exploration of more innovative designs with reduced reliance on physical experiments. The degree to which sensitivity of the simulations to data, model, and geometic uncertaintes can be quantified for science-based methods will be discussed in some detail, and their potential overall role in reactor design will be addressed.

Young Generation Event:

Keynote lectures

Nuclear Power Based on Fast Reactors. Scientific Idea, Early Experience, New Start

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Summary

In the early 1950s the author joined the IPPE activities on the theory of reactors (First World NPP, nuclear submarines). In the late 1950s A.I. Leypunsky placed him in charge of the research for the NPP development effort. He was also the research supervisor for the BN-350 and BN-600 fast reactor projects (in the 1960s) and for the BN-350 reactor start (in the 1970s). Since the accidents at EBR-I and the E.Fermi NPP, these became the first successful embodiments of the Fermi idea (1944), that is, nuclear power based on fast reactors.

The BN-350 reactor operated for 25 years and the BN-600 is still in operation. However, none of the projects have been continued. As the result of comprehending (already at the Kurchatov Institute) the causes for the unsuccessful early experience, in the 1980s the author gave up the Fermi-originated concept of the «FR-breeder» started on the Pu from thermal reactors and embarked on the development (at NIKIET) of the BREST «inherent-safe FR» of a moderate power rating with BR~1 to operate on enriched U or Pu.

The consumption of U (and the separation work) to start the FR on enriched U is considerably below that for thermal reactors-generated Pu, and the FR natural safety properties with respect to accidents, wastes and proliferation resistance once the adequate technology is selected (nitride fuel of an equilibrium composition, on-site «dry» processing of fuel, Pb in place of Na) also make large NPPs much cheaper. High rates of Pu breeding are therefore unnecessary, while U is used in full with BR≈1, that is, 100 -200 times as effectively as in thermal reactors, so inexhaustible low-grade ores suit as well.

Fitting FRs with a Th-blanket in future will also provide Th-^{3}U fuel for FRs of small-sized NPPs for local needs. Still, the prime task of nuclear power will remain generation of electricity at large NPPs, where it is profitable to use FRs in closed fuel cycles. The growth of nuclear power will entail an increase in the share of electricity in total energy consumption (currently ~1/6).

The BREST technology in closed fuel cycle has been studied since the 1950-1960s for FRs and nuclear submarines, but has not been employed in FRs as these have been no longer built.

It will be for another 20 years or so that nuclear power will continue to rely on TRs built in the 1940-1950s for military applications and converted later to electricity production, still failing to offer solutions to its problems. It can be expected that FRs will be built during this period to solve all of the nuclear power problems, including fuel balance, safety and economics, and, ultimately, the power problems proper.

By overcoming the inertia of stereotypes and the decades of stagnation in thermal reactors of the 1940-1950s and by building new FRs, we will be able to meet the challenges the world is expected to face in the 21st century and solve the fuel and energy problems.

The estimates given in the report indicate to the feasibility of promoting the evolution of nuclear power up to ~10^5 GW(t), which will not to cause a major unbalance with the ~10^8 GW falling onto the Earth. Nuclear power of such a scale will have the capability to provide the energy supply for the population of the globe at the level of advanced countries with no further limits in terms of cheap fuel resources.

Important matters in realizing commercial FBR cycle

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Nuclear energy, one of the essential measures against increasing world energy demand and global warming is expected to expand in near future. The Fast Breeder Reactor (FBR) cycle technology is supposed to enhance the utilization of uranium resources for the long term. Many countries are interested in pursuing the technology development. Japan with scarce natural resources also has been developing the technology for decades, since its beginning of utilization of nuclear energy.

As a person who spent many years in the fields of nuclear power generation, I sincerely hope that you, young generation will work hard passionately and succeed in commercializing the FBR cycle. I would like to show you what I recognize the most important things in achieving that goal.

The FBR cycle can not be commercialized without establishing both reactor and fuel cycle technologies. When FBRs are commercialized, it is easily found that the issues to be required in nuclear energy like nuclear non-proliferation, transparency and public acceptance must be more important than now. FBRs can not operate without fuel cycle activities including reprocessing, mixed oxide (MOX) fuel fabrication and MOX fuel transportation and so on. These activities require more effective and efficient safeguards, strengthened security measures in order to prevent proliferation and public acceptance than Light Water Reactors (LWR) operation because of the handling large amount of plutonium in the FBR cycle. Additionally, it should be noted that nuclear reactors are not possible to be installed and operated without agreement and cooperation of local residents. It is likely that more efforts are necessary to get public acceptance for the FBR cycle facilites with large amount of plutonium than LWRs.

Treaty of the Non-Proliferation of Nuclear Weapons (NPT) requires that all the peaceful nuclear activities should be reported to the International Atomic Energy Agency (IAEA) and put under its safeguards. Japan has built safeguards technology and system during the development of fast experimental reactor “Joyo”, prototype “Monju” and cycle facilities and has been accepting safeguards inspection. Towards realization of the commercial FBR cycle, continuing efforts and further activities for the advancement of safeguards technologies such as safeguards by design. Development of safeguards technologies will be needed taking the large burden caused by the increased fuel cycle facilities into account.

For ensuring nuclear security, preparation of standard documents by the IAEA, conclusion of treaties in the international society and ratification the treaties and adoption those to domestic laws in each country has been taken. Japan’s FBRs and cycle facilities have adopted security measures based on international discussions. In the FBR cycle, MOX fuel transportation would be a fragile process from nuclear security perspective and should be strengthened in the realization of commercial FBR cycle.

Transparency of the plutonium utilization is also key issue. In light of global concern about nuclear proliferation, Japan has been reporting amount of stored plutonium to the IAEA based on the Guidelines for the Management of Plutonium stipulated in the IAEA INFCIRC549, publishing “The Current Situation of Plutonium Management in Japan”, which is a more detailed report about stored plutonium, and “Plutonium utilization plan”. The efforts to ensure transparency of the plutonium utilization to gain public acceptance must be more important for deployment of FBRs and cycle facilities with large amount of plutonium.

Agreement and cooperation of local residents are indispensable for continuous operation of nuclear reactors including FBRs. In addition to that, operation of FBR fleet needs deployment of cycle facilities and frequent MOX fuel transportation. This suggests that much efforts to ensure the recognition of safety and necessity by the local people are needed to earn their agreement and cooperation to accept FBRs and cycle facilities.

I expect that young generation in the world cooperates and competes keeping these important matters in mind and having a big dream and passion in realizing commercial FBR cycle. I would ask young participants in the young generation session to express and exchange opinions actively using this opportunity.

Parallel Session 1.1:

Innovative fast reactors: objectives and driving forces

Sodium Fast Breeder Reactor Development : EDF's point of view

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EDF contributes to sodium fast breeder reactor (SFR) development and to the French SFR trilateral program. EDF requirements as a utility are first outlined, based on the approach followed for the EPR reactor. Then, R&D contributions are presented in the areas of core physics, safety, technology innovations, materials, deployment and fuel cycle scenarios. Some issues of the "2020 French prototype" are discussed as seen by EDF.

Long term sustainable nuclear energy implies that 3 criteria be met : the safety level of a new generation of nuclear systems has to be at least equivalent to the level of the previous one; economic competitiveness has to be maintained; social acceptance for the back end of the fuel cycle has to be ensured. A deployment criterion for Generation IV systems is the time when uranium resources availability is questioned, provided these systems are competitive. EDF also recognizes that these systems can help to facilitate acceptance of long live radioactive wastes but this does not constitute a Gen IV deployment criteria in itself since transmutation cannot solve alone the waste issue for the long term.

EDF made a first approach of its requirements for SFRs. It can be summed up by a comparison with Gen III LWRs and former French SFRs, as it is presented in the table below.

GIF design goals		Gen III LWRs	Former French SFRs
Sustainability	Resource Utilization	++	=
	Waste Minimization	++	+
Economics	Life Cycle Cost	=	++
	Risk to Capital	+	++
Safety and Reliability	Operational Safety and Availability	=	++
	Core Damage	=	+
	Offsite Emergency	=/+	++
PR&PP	Proliferation Resistance	different	+
	Physical Protection	=/+	++

++ : significant progress; + : some progress needed; = : objective already met

EDF gathered engineering feedback from operations of Phenix and Superphenix. One of the results of this analysis is that dramatic improvements are required for the operational performances with reference to the preceding FBR projects in France and in particular in the field of :in service inspection and repair, components and fuel handling and maintenance.

EDF performs R&D as a utility, that is to say that EDF is not a designer, nor a vendor, nor a government R&D center. Its SFR R&D program is broken down in 5 projects : fast reactor and system studies; materials; deployment strategies and scenarios; fuel cycle; operations feedback analysis and requirements.

The main results of its research effort are outlined in the paper in the field of core physics, safety (safety methodology and objectives, taking into account sodium technological aspects; projects devoted to safety improvements), technology innovations, materials and deployment scenarios. For instance, the Septen engineering division has proposed a few technology innovations such as an integrated heat exchanger/steam generator for loop type reactors. In this component, physical separation between sodium and water is achieved at the cost of a liquid metal coupling fluid. EDF works also on fuel handling, on the seismic behaviour of the buildings, of the reactor and of the core, on R&D to improve ISIR techniques.

A big effort is also made on deployment scenario studies in the frame of the French energy policy. As required by the French law, EDF participates in the assessment of minor actinides recycling in SFRs (impact on the repository and on the cycle above ground- reactors and fuel cycle facilities).The first results of these studies will be briefly presented in the paper. From EDF's point of view, a partial recycling of minor actinides in heterogeneous blankets might be adopted if it is feasible and if technological and economic advantages outweigh possible drawbacks. Removing some minor actinides (Am in particular) from the glass canisters may reduce the heat load on the repository in case of cooling period constraint above ground but the heat load and the associated activity will have to be managed in more complex facilities above ground (reactors, reprocessing plants, fuel fabrication plants, reactor fuel handling section, fuel transport and interfaces).

In 2006, the French Parliament asked for a fast reactor prototype to be commissioned before the end of 2020.A key issue for EDF is to determine the role that such a prototype will play : a research reactor or a demonstrator reactor with industrial capabilities at a smaller scale? If it's the latter, EDF will wish to check whether it fulfils detailed specifications in terms of safety (core physics in particular) and most importantly in nuclear island layout, operations (availability in particular), ISIR and maintenance. Some requirements for this prototype if industrial are presented as seen today by EDF in all these fields.

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Progress on reactor system technology in the FaCT project toward the commercialization of fast reactor cycle system

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Japan Atomic Energy Agency (JAEA) is now carrying out "Fast Reactor Cycle Technology Development (FaCT)" project along with the approach to the commercialization of fast reactor (FR) cycle system as shown in Fig.1 [1][2].

The design targets including "Safety and Reliability", "Sustainability", "Economic Competitiveness" and "Nuclear Non-proliferation" have been established as the principle of specifications for fast reactor cycle technology at the deployment stage around 2050, to contribute to meeting the global needs which the 21st century has encountered more than ever before, such as the environmental protection and the remarkable increase of energy demand foreseen especially in developing countries. In accordance with those design targets, design study and the related research and development (R&D) on innovative technologies for Japan Sodium-cooled Fast Reactor (JSFR) have been in progress aiming at completion of the conceptual design stage by 2015. The demonstration reactor is planned to operate around 2025. An interim report is ready for issue in June, 2009 which will show the design specifications considered to be feasible at present to meet the requirements for the commercialization and the R&D results to support the feasibility, as well as the investigation on optional measures to take for some of the innovative technologies which may have a high technical hurdle to be realized.

A council was coordinated by five parties; Ministry of Education, Culture, Sports, Science and Technology (MEXT), Ministry of Economy, Trade and Industries (METI), electric utilities, vendors and JAEA to discuss and conduct the R&D schedule and path forward for demonstration of the FR cycle technologies. New advanced R&D bodies have been also structured in JAEA, i.e., Fast Breeder Reactor Plant Engineering Research Center in Tsuruga as a center of excellence for key FR technologies such as inspection and repair on the basis of operation experience of Monju, and a sodium component test facility in Oarai to develop and demonstrate the function of the components and cooling systems.

An international collaboration is emphasized as an effective accelerating force for FaCT, as the development of FR cycle technology actually needs a long-term effort and large resources. Generation-IV International Forum (GIF) is a representative multilateral collaboration framework where Japan has participated since the initial stage of GIF and actively cooperated on especially sodium-cooled FR system as a leading role in its development. Actually, the design targets for FaCT and those for GIF were provided so as to be consistent with each other. International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) is also a multilateral collaboration framework where the assessment study on the JSFR concept has been implemented through the INPRO assessment method.

Continuous challenge toward the commercialization of FR cycle system will be made by utilizing the design and R&D results as resources for the key milestone in JFY2010 to determine which innovative technologies should be adopted, together with preliminary conceptual design study results for the demonstration reactor.

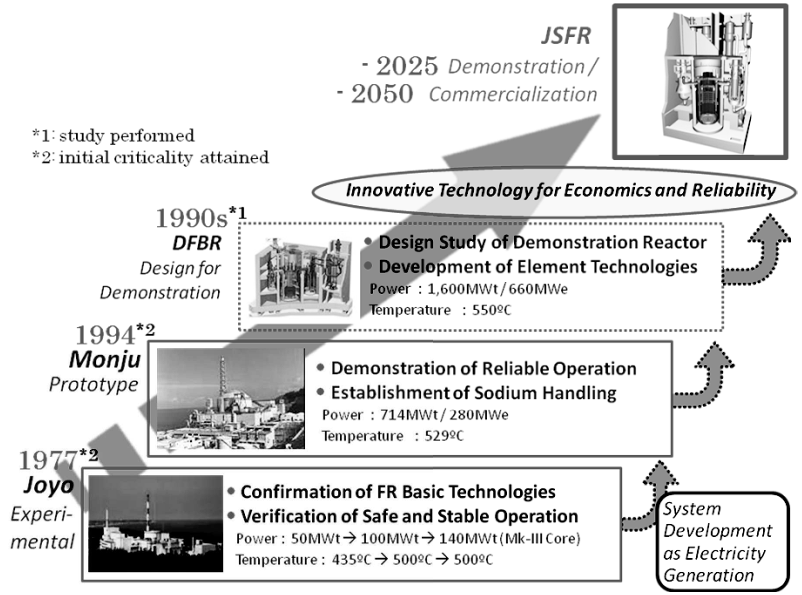

FIG. 1. Approach to the commercialization of fast reactor system in Japan

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Advanced SFR Concept Design Studies at KAERI

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Advanced SFR design concepts have been proposed and evaluated against the design requirements to satisfy the Gen IV technology goals. Two types of conceptual core designs, Breakeven and TRU burner cores were developed. Breakeven core is 1,200 MWe and does not have blankets to enhance the proliferation resistance.

According to the current study, TRU burning rate increases linearly with the rated core powers from 600 MWe to 1,200 MWe. Considering 1) the realistic size of an SFR demonstration reactor for the long-term R&D plan with the goal of a demonstration SFR construction by 2028, and 2) the availability of a KALIMER-600 reactor system design that was developed in the last R&D phase, a TRU burner of 600 MWe was selected.

The heat transport system of Advanced SFR was designed to be a pool type to enhance system safety through slow system transients, where primary sodium is contained in a reactor vessel. The heat transport system is composed of Primary Heat Transport System (PHTS), Intermediate Heat Transport System (IHTS), Steam Generating System (SGS) and Residual Heat Removal System (RHRS).

The heat transport system was established through trade studies in order to enhance the safety and to improve the economics and performance of the KALIMER-600 design. Trade studies were performed for the number of IHTS loops, the number of PHTS pumps, Steam Generator (SG) design concepts, energy conversion system concepts, cover gas operation methods, and an improved concept of safety-graded passive decay heat removal system. From the study, the heat transport system of Advanced SFR has design features such as two IHTS loops, a Rankine cycle energy conversion system, two double-wall straight tube type SGs, and a passive decay heat removal system.

In order to secure the economic competitiveness of an SFR, several concepts were implemented in the mechanical structural design without losing the reactor safety level. The material of reactor vessel and internal structure is a Type 316 stainless steel. The outer diameter of the reactor vessel is 14.5m, which is a very compact size compared to the other designs.

Various R&D activities have been performed in order to support the development of Advanced SFR design concepts and to update computational tools. These activities include validating neutronics analysis codes, PDRC experiment, the conceptual design of supercritical carbon dioxide (S-CO2) Brayton cycle system, Na-CO2 interaction test, under-sodium viewing technique, computerization of structural integrity evaluation, sodium technologies, and metal fuel technology.

Advanced fast sodium reactor power unit concept

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To ensure mass putting into operation of fast reactors after 2020, the Russian Federation develops commercial 1200 MW power unit with fast sodium reactor. The approved theoretically substantiated and practically proven technical decisions implemented in BN-350, BN-600, BN-800 reactor plant (RP) designs are used to the maximum when developing BN-1200. At the same time, new technical decisions are applied that improve safety and ensure high effectiveness of power unit and effective fuel utilization. Main technical characteristics of BN-1200 power unit are given in Table.

Characteristic	Value
Nominal thermal power, MW	2900
Electric power, MW	1220
Number of heat removal loops	4
Primary coolant temperature at IHX inlet/outlet, °C	550/ 410
Secondary coolant temperature at SG inlet/outlet, °C	527/355
Third circuit parameters: live steam temperature, °C live steam pressure, MPa feed water temperature, °C	 510 14 240
Steam reheating type	Steam
NPP efficiency (gross), %	42
NPP efficiency (net), %	40
Plant capacity factor, %	90
Operation mode	Base-load
Power unit service life, year	60

The approved technical decisions are the following:
- main layout and circuit designs of RP circuits;
- integral concept of RP primary circuit;
- reliable barriers preventing coolant leaks from circuits due to guard shrouds an other means;
- the most part of RP equipment;
- separate suction cavities of first circuit pumps with check valves at the outlet (loop layout of the first circuit);
- spent fuel assembly (FA) in-reactor storage.

To improve technical-and-economical indices of the power unit some new decisions were included in the design:
- casing-type steam generator design;

- bellows compensators in secondary circuit pipelines;
- substantially simplified refuelling system as compared with BN-600 and BN-800;
- use of boron carbide for in-reactor radiation protection that provides decrease of overall dimensions and shielding mass.

These additional technical decisions decrease sufficiently (by 45%) RP specific material consumption and cost as compared with BN-800 design. As per estimations, technical-and economic indices of BN-1200 power unit are comparable with those ones for new light-water reactor VVER-1200 (NPP-2006) of same power, which is developed for commercial construction.

Technical decisions enhancing BN-1200 safety:
- passive emergency protection, which includes absorber rods inserted into the core in case of reactor sodium temperature increase in addition to those hydraulically suspended in sodium flow;
- in-reactor emergency heat removal system with natural circulation of coolant in all circuits, which is connected to the primary circuit through the autonomous heat exchangers;
- elimination of primary circuit pipelines with high-activity sodium (relative to reactor vessel) that prevents the whole series of radiation-dangerous accidents;
- no intermediate sodium storage for FA unloaded from reactor that prevents accidents with radioactive sodium leaks;
- fitting out of the plant with the equipment localizing emergency releases from reactor during beyond design accidents that ensures efficient reduction of radiation accident consequences with severe core damage. One of the options being considered is application of the refueling box for that purpose.

The adopted technical decisions allow:
- sufficient decrease of severe core damage (up to 10^{-6} maximum for reactor per year) and respectively, level of activity release to the environment during beyond design accidents;
- elimination of evacuation and resettlement of population resident near NPP and substantial limitation of emergency planning zone as compared with those ones specified in actual regulatory documentation;
- maintenance of population radiation dose under design basis accidents by 20 times less than the allowable annual population radiation dose.

As per determining characteristics, the power unit meets the requirements for fourth generation NPP.

In view of approved decisions application, moderate R&D work is required to verify BN-1200 design on the basis of available experimental facilities. As new technical decisions are included in the design, it is necessary to develop some additional test facilities. In addition, the available computer codes shall be upgraded and new ones shall be developed to increase the level of design verification.It is possible to develop BN-1200 power unit design on the date that allows its implementation by 2020. The best site to construct first of a kind BN-1200 power unit is Beloyarsk NPP where BN-600 reactor is operated and BN-800 is under construction. Some of infrastructure required for a new block is already available (or will be developed in the near future) at this site.The change over to closed fuel cycle using BN-1200 power unit will start from plutonium accumulated in Russian PWRs. The analysis shows that plutonium quantity that will be bred in PWR-type reactors is enough to start up BN-1200 reactor series of 10.8 GW power by 2030.

Cores and Fuel Cycle of the Perspective Fast sodium-Cooled Reactor

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A perspective sodium reactor is under development in Russia nowadays. Initially, power level of 1800 MW (el.) was considered for this reactor. However, owing to many reasons, in particular, for transportability of the main plant by railway, the reactor power was later reduced to 1200 MW (el.). At the same time the base of the concept for the choice of the core parameters remained the same as for the 1800 MW power, including the following:

- low core specific power resulting in a decrease of the fuel lifetime and, consequently, a smaller annual consumption of fuel elements;
- enhancement of inherent self-protection: ensuring the sodium void reactivity effect (SVR) close to zero and a minimum reactivity margin for burnup;
- ensuring the reactor operation in different patterns of the closed fuel cycle organization: the use of plutonium from thermal reactor with and without MA for the first loading, recycling the own plutonium with/without breeding, burnup of own MA, etc.

Basic characteristics of the core of BN-1200 reactor approved for the current phase of designing have been reported.

The principle of layout with upper sodium plenum, like the BN-800 reactor type is preserved in the approved variant of the core for ensuring the SVR close to zero. It is an important feature of the core layout that fuel of one enrichment level is used. This approach simplifies the technological process of manufacturing the fuel elements and fuel subassemblies (SA) and the process of SA handling at NPP.

The Rules of nuclear safety (PBYa RU AS) were altered in 2008 in Russia, the requirement of negative reactivity coefficient from the volume fraction of coolant, i.e., the SVR close to zero, was withdrawn. This allows an extension of the area of optimal values for the core parameters, in particular, an extension of the core height and introduction of the top axial breeding blanket.

However, in spite of a reduced strictness of regulatory requirements, the question of changeover to the traditional layout of the core with top and bottom breeding blankets needs an additional analysis. In the core with upper sodium plenum accepted according to the condition of SVR close to zero, the core breeding ratio (CBR) of ~ 0.9 can be achieved, with total breeding ratio of ~ 1.2. In case of justification of the feasibility of the upper axial blanket the BR can be increased to 1.35.

The possibility of increasing the CBR and BR is considered in the report also owing to the increase of total volume fraction of fuel, as well as the variants of the core layout with positive SVR. A further increase of BR is associated with the application of high-density fuel, first of all nitride.

The value of critical loading of plutonium into the core is a principle issue for nuclear power, along with the BR. The intensity of commissioning of fast reactors can be determined by the critical loading level, which is increased in the decrease of heat release rate. However, as it has been shown by R&D results, this problem appears to be mitigated in the consideration of the fast reactors' fuel cycle closure.

The use of fuel decontaminated from minor actinides (MA) considerably simplifies the external fuel cycle of fast reactor and makes it cheaper, but it aggravates the problem of management of high-level

RW in entire nuclear power. At the same time the organization of fuel cycle with non-separated MA would allow a radical solution of the problem of utilization of the MA bred in thermal and fast reactors. The main results of calculation and theoretical investigations of MA burn-up as a by-function of the perspective fast power sodium reactor have been reported. These studies were concentrated mainly in two areas.

1) Utilization of outside MA. Two options have been considered - homogenous (MA put into MOX fuel) and heterogenous (MA burn-up in special burning SA arranged in the core heterogenously).
2) Utilization of internal MA. This task appears in the preservation of the “pure” production of fresh fuel for perspective fast reactors.

Obviously, the problem of MA utilization could be resolved only based on a comprehensive complex approach where technological potentials and economic issues play a determining role. However, the studies of physical aspects play an important role in the choice of engineering solutions for resolving the problem of MA utilization.

Heat release rate in fresh fuel after chemical processing is one of the key characteristics which influence the technology of fuel production in the closed fuel cycle. The most significant nuclides determining the heat release level in such fuel are curium isotopes (^{242}Cm, ^{244}Cm), ^{241}Am, and ^{238}Pu. Several options for the organization of closed fuel cycle that allow a considerable reduction of this level have been addressed in the report.

Advanced SFR concept based on PRISM and KALIMER

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The Sodium-cooled Fast Reactor (SFR) has been recognized as one of the promising nuclear options for generating electricity in terms of uranium resource utilization and reduction of radioactive wastes from nuclear power plants. This paper compares design features and identifies differences between KALIMER-600 and S-PRISM sodium-cooled reactors in order to explore the possibility of developing a novel SFR concept.

The design concepts of both S-PRISM and KALIMER-600 originate from that of the Advanced Liquid Metal Reactor (ALMR). ALMR is a SFR and was developed by a group of industry companies headed by then General Electic Company. Argonne National Laboratory (ANL) developed an electrometallurgical process for spent nuclear fuels to supply fuel for ALMR.

PRISM (Power Reactor Innovative Small Module) is a modular SFR based on the ALMR technology. The U.S. NRC has reviewed the preliminary safety information document (PSID) for the PRISM and has concluded that “no obvious impediments to licensing the PRISM have been identified” in the safety evaluation report (NUREG-1368).

S-PRISM is an improvement of PRISM for commercial potential. Each S-PRISM power block consists of two reactors producing 380 MWe, and is connected to a steam generator. The standard power plant design includes three power blocks producing a total of 2,280 MWe. Design improvements in S-PRISM include increased power output, multi-cell containment system, compact reactor building on single seismically isolated base pad, improved steam cycle efficiency, and improved design for the secondary side steam/sodium interaction.

KALIMER-150 is a sodium-cooled fast reactor developed by KAERI based on the ALMR technology, and KALIMER-600 is an improvement over the KALIMER-150 design. The improvements include reduced Intermediate Heat Transfer System (IHTS) pipe lengths, simplification of reactor internal structure, increased power, passive residual heat removal systems, and improved core performance. In contrast to KALIMER-150, which is modular, KALIMER-600 is a monolithic reactor with a thermal power of 1,523 MWt.

The reactor technologies represented in S-PRISM are compatable with the Intergral Fast Reactor (IFR) concept concived by Dr. Chuck Hill of ANL. Particularly, the metallic fuel technology is based on successful experience in the EBR-II and FFTF. Technologies for reactor and fuel cycle of KALIMER-600 are similar to PRISM or S-PRISM. The major difference is that KALIMER-600 is a monolithic design whereas the S-PRISM is modular. The modular concept of S-PRISM utilizes the advantage of factory fabrication, improved plant availability, and a shorter construction schedule to overcome the disadvantages of their relatively small size and lack of economy of scale. Also the small size of modular concept allows to use fully passive decay heat removal systems that simplifies the design and improves reliability. S-PRISM may be commercialized at a lower development cost than KALIMER-600, because there is no need for component scale-up and the design can be certified through the construction and testing of a single reactor module.

The S-PRISM operates at a lower pool temperature than KALIMER-600. S-PRISM adopts the RVACS (Reactor Vessel Auxiliary Cooling System) and the ACS (Auxiliary Cooling System) for residual heat removal, whereas KALIMER-600 uses the PDRC (Passive Decay Heat Removal Circuit) system. Both heat removal systems are passive and dependent on natural circulations. S-PRISM uses four electromagnetic pumps in the primary heat transport system, and KALIMER-600 uses two mechanical pumps. The different pump designs are due to different size and thermal power of a single reactor. S-PRISM and KALIMER-600 have different designs for the Sodium Water Reaction Pressure Relief System (SWRPRS).

KALIMER-600 has eliminated reactor internals, such as flow guide structure, internal primary pipes, internal fuel storage rack structure and radiation shield, in order to simplify the structure and minimize the flow blockage. KALIMER-600 also uses a detached skirt core support concept in order to facilitate ISI (In-Service Inspection).

The seismic designs for both reactors are almost identical. Both reactors are designed with the horizontal seismic base isolation system using High damping Laminated Rubber Bearings (HLRB) installed between reactor building lower slab and the basemat. However, the horizontal seismic isolation frequency for S-PRISM is 0.75 Hz, which is much higher than 0.5 Hz of KALIER-600. The higher horizontal isolation frequency results in a smaller relative displacement response between the isolator platform and the ground, but the efficiency of a seismic isolation effect can be significantly reduced when the input seismic load contains lower frequency components.

KALIMER-600 uses U-TRU-10Zr metal fuel only, whereas S-PRISM uses both metal and oxide fuels. Candidate cladding materials for high thermal creep being considered for KALIMER-600 and S-PRISM are Mod.HT9 and HT9M, respectively. KALIMER-600 is a radially homogeneous core, and TRU wt% in the inner/middle/outer cores are the same. S-PRISM has a radially heterogeneous core that has internal and radial blankets. The reactor refueling interval for S-PRISM is 23 months, and that for KALIMER-600 is 18 months.

KALIMER-600 cores are developed to achieve breakeven fissile breeding or TRU burning. S-PRISM cores are developed to have a TRU buring, fissile self-sufficiency or enhanced breeding to support a fast deployment scenario of fast reactors. Both reactors consider using metallic fuel integrated with the pyro-processing technology, where plutonium is never to be separated in pure form and minor actinides accompany the uranium-plutonium product. The fuel cycle facilities have a potential to be co-located with the reactors, eliminating most transportation, and thus both reactor designs in conjunction with the advanced fuel cycle technologies have better characteristics for proliferation resistance.

JSFR design study and R&D progress in the FaCT project

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Japan Atomic Energy Agency (JAEA) is now carrying out "Fast Reactor Cycle Technology Development (FaCT)" project which has followed the previous project, i.e., the Feasibility Study on Commercialized Fast Reactor Cycle Systems (FS) [1].

The FaCT project has implemented a conceptual design for Japan Sodium-cooled Fast Reactor (JSFR), which is an advanced loop-type reactor with a potential advantage of operation and maintenance capability as well as the perspective for the economical plant configuration [2]. Actually, the economic competitiveness with high reliability has been a crucial issue for the commercialization of sodium-cooled fast reactors and the replacement for light water reactors in the future. Innovative technologies, such as a two-loop cooling system including an intermediate heat exchanger (IHX) with high chromium ferritic steel even for a large-scale reactor as shown in Fig.1, and integration of a primary pump into the IHX, can significantly reduce the commodity of JSFR to attain the development target for the construction cost. The high chromium ferritic steel is also utilized to shorten and simplify the piping configuration. A compact reactor vessel (RV) has been also pursued in the design of JSFR with some innovative technologies including an upper internal structure with a slit where a fuel handling machine (FHM) moves horizontally.

Reliability assurance for JSFR is pursued by adopting a double boundary concept for all the sodium boundaries for the piping and vessels, a steam generator (SG) with double-walled tubes, and design accommodation and innovative technologies for an inspection and repair capability.

Safety enhancement is the premise in the design of JSFR. The safety design principle is that the design basis events (DBEs) shall be accommodated in a conservative manner and the consequences shall be terminated earlier by providing the rapid shut down system and ensuring the sufficient natural circulation capability for the decay heat removal systems. As for the beyond design basis events (BDBEs), Self Actuated Shutdown System (SASS) and natural circulation capability are provided for reactor shutdown and decay heat removal, respectively, to prevent the accident progression into core distruptive accidents. Furthermore, efforts have been made to avoid the severe energetics due to excursion by restricting the core design and providing the inner duct structures in the fuel subassembly to enhance the molten fuel discharge, so that the in-vessel retention capability can be reinforced providing the multiple layer core catchers.

Research and development on the innovative technologies has been in progress, and the results are being obtained and reflected into the design of JSFR. Design measures to prevent the vibration and wear of IHX tubes attributed to the mechanical vibration of the primary pump have been evaluated. The results of water tests on the in-vessel thermal-hydraulics have been reflected to take measures for the RV design to overcome gas entrainment and vortex-induced cavitations. Mechanical testing studies have been performed with a full scale mock-up test device of the new FHM to confirm the basic functions. Trial manufacturing for the components of the double-walled tube SG and development of an under sodium viewer technology for inspection of the reactor internal have been also progressing. To demonstrate the innovative technologies, a plan of a large-scale sodium test complex has been started to develop and demonstrate the function of the reactor components and the cooling systems.

In the FaCT project an international collaboration is emphasized to explore the possibilities of sharing and/or collaborating in the development of the innovative technologies. Generation-IV International Forum is a representative multilateral collaboration framework where Japan has played an active and leading role in cooperation on especially sodium-cooled fast reactor system. International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) is also a multilateral collaboration framework where the INPRO assessment method has been utilized for the assessment study on the JSFR concept.

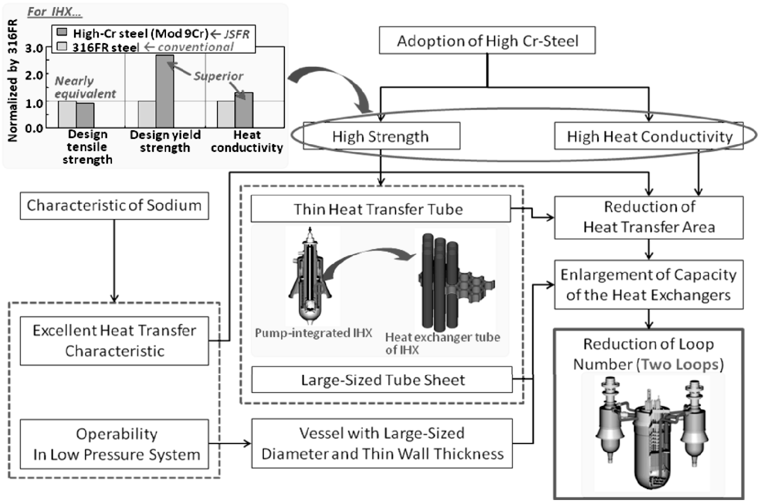

FIG. 1. Approach to reducing the number of heat transport loops

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Parallel Session 1.2:

Innovative fast reactors: objectives and driving forces

ELSY – The European Lead Fast Reactor

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The European Lead Fast Reactor is being developed starting from September 2006, in the frame of the ELSY (European Lead SYstem) project funded by the Sixth Framework Programme of EURATOM. The project, coordinated by Ansaldo Nucleare, involves a wide consortium of European organizations. The ELSY reference design is a 600 MWe pool-type reactor cooled by pure lead. The ELSY project demonstrates the possibility of designing a competitive and safe fast critical reactor using simple engineered technical features, whilst fully complying with the Generation IV goal of sustainability and minor actinide (MA) burning capability. The main objectives of the ELSY project are to show that:

- the adopted innovative design and technology achieves a very high safety standards,
- the fuel cycle can be closed,
- the non-proliferation resistance is enhanced,
- a high availability factor is reached,
- the economic competitiveness target is reached,
- the design is compliant with GIF goals

Sustainability was a leading criterion for option selection for core design, focusing on the demonstration of the potential to be self sustaining in plutonium and to burn its own generated MAs. To this end, different core configurations have been studied and compared. Economics was a leading criterion for primary system design and plant layout. The use of a compact and simple primary circuit with the additional objective that all internal components be removable, are among the reactor features intended to assure competitive electric energy generation and long-term investment protection. Low capital cost and construction time are pursued through simplicity and compactness of the reactor building (reduced footprint and height). The reduced plant footprint is one of the benefits coming from the elimination of the Intermediate Cooling System, the low reactor building height is the result of the design approach which foresees the adoption of short-height components and two innovative passively operated DHR (Decay Heat Removal) systems. Among the critical issues, the impact of the large mass of lead has been carefully analyzed; notwithstanding it has been demonstrated that the high density of lead can be mitigated by more compact solutions and improvement of the design of the Reactor Vessel support system, i.e. the adoption of seismic isolators for a full seismic-resistant design. Preliminary results of the reactor vessel and supports stress analysis indicate that an LFR larger than a medium-size plant (in the IAEA

classification) is potentially feasible. Safety has been one of the major focus all over the ELSY development. In addition to the inherent safety advantages of lead coolant like high boiling point and no exothermic reactions with air or water, a high safety grade of the overall system has been reached. In fact overall primary system has been conceived in order to minimize pressure drops and, as a consequence, to allow decay heat removal by natural circulation (note that this feature is essential for the unprotected loss of flow transient). Moreover two redundant, diverse and passive operated DHR systems have been developed and adopted. A sketch of the ELSY primary system configuration is shown in Figure 1.

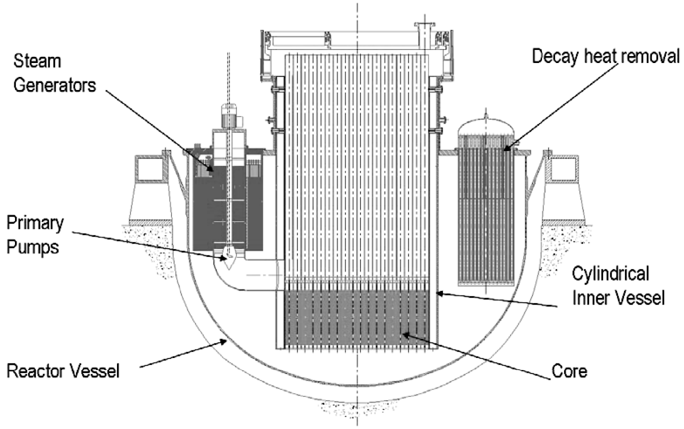

Figure 1 – ELSY Primary System sketch

The paper focuses on the main aspects of the proposed design for the European Lead Fast Reactor highlighting the innovation of this reactor concept, overall objectives as well as future developments. Main safety features of the proposed Decay Heat Removal systems will be presented. Some experimental results related to the development of appropriate materials or materials protection for the high corrosion environment of Lead are also briefly presented.

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SVBR-100 module-type fast reactor of the IV Generation for regional power industry

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Fast reactors (FR) with heavy liquid metal coolant (HLMC) were chosen as one out of six lines of development of the IV Generation innovative nuclear power systems (INPS).

The acronym SVBR-100 means **L**ead-**B**ismuth **F**ast **R**eactor with the equivalent electric power of **100** MW. The module-type fast reactor SVBR-100 with the HLMC (eutectic alloy of lead-bismuth) meets the main requirements to the fourth generation INPS worked out by GIF:

- **Efficient use of natural uranium energy potential.** The SVBR-100 reactor meets this requirement as in the closed nuclear fuel cycle (NFC) with the use of mixed uranium-plutonium fuel this reactor operates in the fuel self-production mode, with the core breeding ratio (CBR) slightly higher than 1.
- **Significantly higher safety level.** Due to the use of chemically inert lead-bismuth coolant (LBC) the SVBR-100 reactor meets this requirement owing to inherent self-protection of the reactor that operates at a low, close to the atmospheric, pressure.
- **Improved resistance to fissile nuclear material proliferation.** The SVBR-100 reactor meets this requirement due to the following factors: 1) there are no blankets where weapons grade plutonium can be accumulated; 2) uranium with lower than 20% enrichment is used in uranium oxide fuel; 3) a long core lifetime (7-8 years) is assured without any refueling and 4) any access to the fuel is prevented during the core lifetime.
- **Acceptable technical and economical parameters.** The SVBR-100 reactor meets this requirement owing to the following factors: 1) in SVBR-100 there is no need in many safety systems required for conventional reactors because of the high potential energy accumulated in the primary coolant of these reactors; 2) a high volume of production of this reactor can be achieved due to its low power level; 3) there is no need in R&D work and construction of a prototype reactor because the NPP power units of various capacity are based on the well-tested unified 100 MWe reactor module; 4) unloading of spent nuclear fuel (SNF) is performed cassette-by-cassette and the fresh fuel is loaded as a single cartridge.

The paper presents the main concepts of this innovative nuclear power technology based on the module-type SVBR-100 reactors and results of analytical studies on the reactor capable of

operating with various fuel types in various fuel cycles. A high safety level of this reactor is justified. Consideration is given to the issues of natural uranium consumption with the reactor start-up with the uranium fuel and further change for the closed NFC with the use of its own spent fuel. Possibilities of multi-purpose use of these reactors were analyzed, including their potential export in view of non-proliferation requirements.

Recent progress of Gas Fast Reactor program

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The GFR is considered by the French Atomic Energy Commission as a promising concept which combines the benefits of fast spectrum and high temperature, using helium as coolant. He properties are interesting with respect to safety: it is single phase (no threshold effect due to phase changing), chemically inert, and non toxic. It affords an optical transparency allowing potential improvements in temperature measurement, management for dismantling, and in-service-inspection. The voiding effect is limited, less than 1$, providing quasi-decoupling of the reactor physics from the state of the coolant. Nevertheless, Helium is a poor coolant, so that the GFR viability includes development of a refractory and dense fuel, and robust management of accidental transients, especially cooling accidents.

GFR feasibility is essentially linked to three demonstrations:
- the feasibility (fabrication, thermo-mechanical behaviour) of a refractory fuel,
- the safety architecture with appropriate systems for the prevention and a robust mitigation of accidental scenarios (especially depressurization),
- economic competitiveness.

The first one includes an experimental activity at the laboratory scale: completion of the results is expected by 2012-2015. The next step afterward will be the design, construction and the operation of a 50-100 MWth experimental reactor, the **Allegro** project (former ETDR), possibly as a European Joint Undertaking.

The full paper will recall the 2007 design choices and it will give an overview of the progress performed so far regarding the safety architecture and the safety evaluation.

The 2007 reference fuel technology is a ceramic plate type fuel element [1]. It combines a high enough core power density (minimization of the Pu inventory), plutonium and minor actinides recycling capabilities. Innovative to many aspects, the fuel element is a key issue in the GFR feasibility. It is supported already by a significant R&D effort also applicable to a pin concept that is considered as an alternative design.

Part of the safety architecture, the Decay Heat Removal (DHR) safety function is based on the control of gas circulation and gas inventory [2]:
- the primary circuit, the core in particular, were designed to ease the gas circulation and enhance the natural convection capabilities ;
- a close containment concept – functionally, a gastight envelope enclosing the primary circuit – was considered in order to limit the loss of pressure in case of depressurization (a "backup" pressure results from the equilibrium between the close containment and primary volumes).

The adequacy of the provisions retained in the design can be judged using a variety of deterministic and probabilistic methods. In general terms, the plant is deterministically

designed against the identified list of the operating conditions and a probabilistic assessment is also currently performed to verify that there are no vulnerable areas [3].

The preliminary acceptance criteria retained for the assessment of the Design Basis Accidents (DBA) will be explicited in the full paper as well as the selection of the reference situations that result from the combination of the initial state of the reactor (full power), of an initiating event (IE) and of the single aggravating failure inducing the most adverse effect on the consequence of the transient [3]. All the plant transients (about 30 cases) have been simulated using the CATHARE 2 computer code (Figure 1). The main results will be given and analysed.

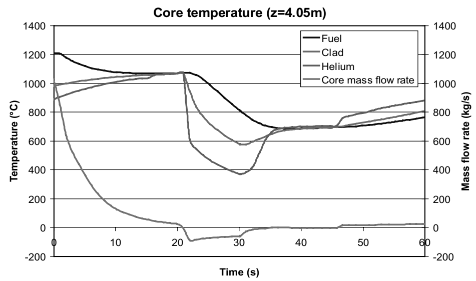

Fig. 1 - LB-LOCA, calculated core mass flow rate and temperatures

To a lesser extend, preliminary consideration on severe accidents will be presented : several families of severe accident scenarios leading to severe plant conditions have been preliminarily identified. An approach was proposed to distinguish those families depending on the integrity of the safety barriers, the magnitude and the dynamics of the phenomena induced by the accidents and the possible associated threshold effects.

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Future R&D Programs Using Monju

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Japanese prototype fast breeder reactor, Monju, will soon restart. Monju has three sodium loops with steam generators and a turbin-generator, 280 MWe. Monju is expected to demonstrate the sodium technology and the sodium-water heat exchange technology in Japan. It will be operated at full power operation during around 10 years after restart in order to accumulate operation & maintenance experience and to evaluate its design technology. The future R&D programs using Monju are shown in Fig.1.

After the system start-up test (SST), Monju will be operated under full power. In this stage, the main object of Monju operation will be to achieve its initial targets which were fixed when its construction was decided around thirty years ago. The targets are to demonstrate a safe and reliable operation, that is, accumulation of the operation & maintenance experience and evaluation of the design technology, and to establish sodium handling technology. For example, inspection and diagnosis technologies are important for maintenance of a sodium cooled reactor. An out-of-pile sodium test facility will be constructed near Monju in order to make many tests of inspection devices and research many chemical tests. At the same time, the activities for the performance improvement, for example, a new licence, will be prepared in order to utilize Monju as a R&D facility.

After the accumulation of operating experinece, Monju will be enhanced the performance as a R&D facility in order to demonstrate innovative technologies, for example, irradiation of advanced fuel, longer operation cycle, higher burnup. For this purpose, Monju will be needed to get a new licence and core modification. And Monju on-site non-destructive Post Irradiated Evaluation facility will be expected at this stage.

There were many R&D works in Japan with sodium out-of-pile facilities. All the experience were reflected in the design of Monju. Monju will demonstrate a handling of sodium technologies under power plant operation.

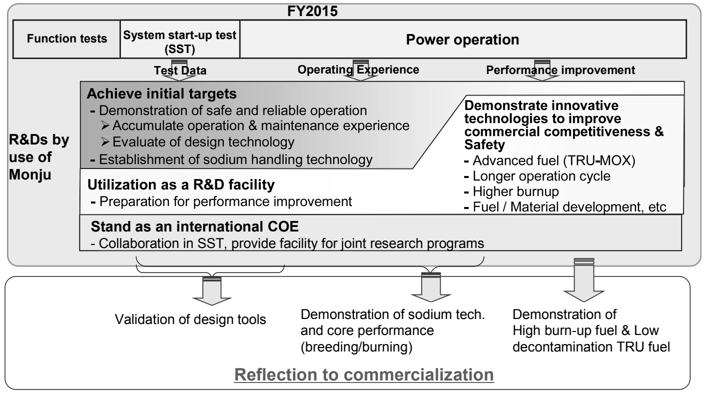

Fig.1. Future R&D Programs Using Monju

Design Features of Advanced Sodium Cooled Fast Reactors with Emphasis on Economics

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New incentives have recently been given by French authorities to promote the development of advanced fast reactors and fuel re-cycle options. Priority is given to Sodium cooled fast reactors (SFR) on which a significant experience exists. The objective of current R&D program carried out in tight collaboration between CEA, AREVA and EDF is to propose SFR concept(s) including innovative technologies and assess the associated industrial viability by 2012. The objective is also to propose the specification of a prototype that would qualify whole or part of the most promising selected options.

An overall specification was established by EDF for future Commercial SFR. The target for such SFR will be the same as contemporary LWR as regards availability factor, design life expectancy, maintenance, safety and electricity generation cost (investment + operation + fuel cycle). In addition, outstanding performances as regards energy resources sustainability are requested. The risk on investment must also be reduced, through adequate inspection and repair capabilities, and through a satisfactory public acceptance. Specific Proliferation Resistance and Plant Protection measures are also requested.

The present paper will give an overview of R&D orientations and efforts made to increase the SFR attractiveness, in line with preceding objectives:

- Enhanced safety is expected to be obtained by prevention and mitigation of severe accidents and a low vulnerability to external events and aggressions (use of a robust containment). The objectives are:

  - To prevent core damage by making an extensive use of the lines of defence approach, focusing on the design of a core with favourable reactivity coefficients, minimizing risks associated with sodium and diversifying and enhancing reliability of safety systems.

  - To mitigate consequences of core damage by making provisions against energetic criticality sequences resulting from core melt down and by ensuring a safe management of degraded configurations.

These enhanced safety features will contribute to minimize licensing and financial risks and will also take part to the enhanced public acceptance objective.

- To improve Sustainability, R&D efforts are put on closing the fuel cycle, optimizing the core for an optimal use of fertile and fissile materials while using highly performing U-Pu oxide fuel as a reference (dense fuel is evaluated as a longer term option).

- Different tracks are investigated to increase SFR economic competitiveness:

  - Minimizing plant capital cost by:

- simplifying the system and components design;

- performing R&D on ferritic-martensitic steels to evaluate how promising is this option to simplify and reduce costs of IHX, secondary piping and SG;

- improving the energy conversion system thermal efficiency;

- looking for a reduction of the size of the secondary circuit;

- looking for a mutualization, on the same site, of systems for fuel handling and component maintenance (when modularity is envisaged);

-optimizing the power level.

- Reducing generating costs, at the time of SFR deployment, with, for example, a closed Pu cycle with extended fuel burn up (development of high dose cladding materials), increased plant load factor, in particular through long cycle duration and reduced refuelling and component handling outage durations.

- Lowering operational costs, with a target that should be as low as those for Generation III+ reactors, including staffing requirements, maintenance/inspection, spare parts, repair/replacement, waste management costs, general services costs, and support from off-site facilities:

  - Systems need to be simplified to facilitate maintenance operations, reduce O&M costs and limit risks associated to human factors.

  - The control rod lifetime needs to be increased

  - Reactor design needs to be optimized to minimize personnel exposure.

  - A trade-off has to be found between prevention of Na issues and ease of access for maintenance operation.

  - The ability to carry out maintenance and inspection "on-line" has to be developed.

- Securing investment by means of an assured licensing (including a robust safety demonstration) and planning process, reduced investment and operating costs, enhanced In-Service Inspection and Repair (ISIR) capabilities and by taking decommissioning into account at the design stage.

Parallel Session 2:

Fast reactor coolant technology and instrumentation

Behaviour and monitoring of non-metallic impurities in liquid sodium

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Liquid sodium is the coolant of choice for fast breeder reactors. Liquid sodium is highly compatible with structural steels when the concentration of dissolved non-metallic impurities such as oxygen and carbon are low. However, when their concentrations are above certain threshold limits, enhanced corrosion and mass transfer and carburization of the steels would occur. The threshold concentration levels of oxygen in sodium are determined by thermochemical stability of various ternary oxides of Na-M-O systems (M = alloying elements in steels) which take part in corrosion and mass transfer. Dissolved carbon also influences these threshold levels by establishing relevant carbide equilibria. An event of steam leak into sodium at the steam generator, if undetected at its inception itself, can lead to extensive wastage of the tubes of the steam generator and prolonged shutdown. Air ingress into the argon cover gas would lead to an increase of oxygen in sodium and a steam leak at the steam generator would lead to an increase in oxygen and hydrogen concentrations at the secondary sodium circuit. A leak of hydrocarbon oil used as cooling fluids of the shafts of the centrifugal pumps of sodium systems are the sources of carbon and hydrogen impurities in sodium. Continuous monitoring of the concentration of these non-metallic impurities in sodium coolant would help identifying these ingresses at their inception. Yttria Doped Thoria (YDT) electrolyte based oxygen sensor is under development for monitoring dissolved oxygen levels in sodium. An electrochemical hydrogen sensor based on CaHBr-$CaBr_2$ hydride ion conducting solid electrolyte has been developed for detecting the steam leak during normal operating conditions of the reactor. A nickel diffuser based hydrogen sensor system using thermal conductivity detector (TCD) and Pd-doped tin oxide thin film sensor has been developed for use during start up or low power operations of the reactor. For monitoring carbon in sodium, an electrochemical sensor with molten Na_2CO_3-$LiCO_3$ as the electrolyte and pure graphite as reference electrode has been developed. Fabrication, assembly, testing and performance of these sensors both in laboratory and large sodium test rigs will be presented in this paper.

Sodium quality control; French developments from Rapsodie to EFR

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SUMMARY

In order to operate in reliable and safe conditions a Sodium Fast Reactor, it is necessary to master the quality of the coolant. The chemical control of sodium is performed versus the different chemical compounds : oxygen (corrosion control), hydrogen (detection of the sodium-water reaction), and to a less degree carbon (carburization, decarburization phenomena). Moreover, other detrimental effects could be avoided: plugging of narrow sections, loss of heat transfer efficiency in heat exchangers, contamination and dosimetry,…Oxygen and moisture are introduced mainly during handling operations; hydrogen is due to aqueous corrosion of the Steam Generator Unit and thermal decomposition of hydrazine, used to control the oxygen content in the water. Some specific events can occur and generates pollution of sodium: sodium-water reaction due to water ingress in the intermediate loop, oil ingress from mechanical pumps,…To mitigate the consequences induced by such events, it is necessary to carry out a purification campaign in order to reach the nominal O, H, C content in the sodium, respectively, in France [O]<3 ppm, [H]< 0.1 ppm and [C]<20 ppm.

The purification of oxygen and hydrogen is performed satisfactorily thanks to cold traps and mainly because of the fact that the solubilities are nearly nil for temperatures close to the fusion point, i.e. 97.8°C. The sodium is first cooled in a heat-exchanger-economizer to a temperature close to the saturation temperature of oxygen and/or hydrogen. It then flows through a cooler where it reaches a temperature below the saturation temperature. Sodium oxide or hydride crystals form and are retained on a packing which can be a stainless steel knit packing, Pall rings (in Germany),…The retention area may correspond to the cooling area. This process allows to obtain a large trapping capacity and efficiency if the design takes into account some specific rules based on the basic studies related to the mechanisms and kinetics of crystallization and also on some operational feedback from reactors.

The main goal of the primary purification systems is to maintain the oxygen content below 3 ppm: such a value is recommended with regards corrosion and more particularly its main consequences, mass transfer and contamination of primary vessel and components. For the intermediate loops, the objective is to maintain the hydrogen content below 0.1ppm: such a value allows to detect early the sodium-water interaction.

This paper deals with the developments of the purification systems for the primary and intermediate circuits of a SFR. Basis studies were carried out to understand the basic mechanisms of crystallization of sodium oxide and sodium hydride and to establish their respective nucleation and growth kinetics.

New concepts were developed for Phenix, Superphenix and EFR in order to provide higher efficiency ie equal to one, large filling rate and reliability in operation. The methodology of development, based on calculations, tests on mock-up, is the following one: definition of requirements: source of pollution and type: continuous or not, frequency of occurrence if incidental, identification of potential products, reactor operational conditions during purification campaign, choice of cooling fluid, sodium flow-rate, temperature range, strategy of replacement or regeneration,…In this paper we describe the strategy developed for three examples of typical purification systems .

Thanks to these developments, sodium purification was never considered as a key issue during steady state operation and purification campaigns necessary to deal with the main pollutions ie sodium-water reaction or air ingress. The sodium purification always allowed to fulfill the operational requirements.

Figure 1: Na2O crystals on mesh

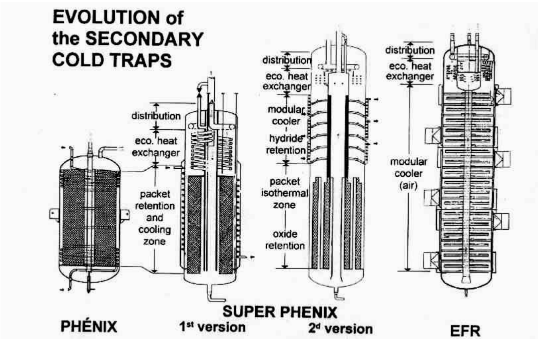

Figure 2: Evolution of the secondary cold traps

Some recent developments in the field of liquid metal measuring techniques and instrumentation

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Liquid metal cooling or liquid metal targets belong to innovative reactor concepts such as the sodium cooled fast breeder reactor or the lead-bismuth target in a transmutation system. The safe and reliable operation of liquid metal systems requires corresponding measuring systems and control units, both for the liquid metal single-phase flow as well as for gas bubble liquid metal two-phase flows. We report on some recent developments in this field.

Integral flow rate measurements are an important issue. We describe two new, fully contactless electromagnetic solutions and related test measurements at available sodium and lead loops. One of the sensors is of particular interest since its operation does not depend on the electrical conductivity of the liquid metal, hence it is independent on the melt temperature.

A development of the past decade is the local velocity measurement by application of the Ultrasound Doppler Velocimetry (UDV). It provides the velocity profile along the ultrasonic beam, and has the capability to work even through some channel wall. We report on measurements in liquid sodium at 150°C. For higher temperatures, an integrated ultrasonic sensor with an acoustic wave-guide has been developed to overcome the limitation of ultrasonic transducers to temperatures lower than 200°C. This sensor can presently be applied at maximum temperatures up to 700°C. Stable and robust measurements have been performed in various PbBi flows in our laboratory at FZD as well as at the THESYS loop of the KALLA laboratory of Forschungszentrum Karlsruhe, Germany (FZK). We will present experimental results obtained in a PbBi bubbly flow at 250...300°C. Argon bubbles were injected through a single orifice in a cylindrical container filled with stagnant PbBi. Velocity profiles were measured in the bubble plume. At the THESYS loop of FZK, stable velocity profiles were measured in a round tube of diameter 60 mm during a period of about 72 hours at temperatures between 180°C and 350°C.

Further, we report on the development of a contactless magnetic tomography of the mean flow in liquid metals. This method gives the full three-dimensional mean velocity distribution in a liquid metal volume. Results from a laboratory demonstration experiment will be presented.

Finally, a heat exchanger design will be presented working with an intermediate liquid metal in order to avoid the possible contact between a hot liquid metal and cooling water. It is installed at FZD at a lead loop where liquid lead of up to 500°C circulates. The room temperature liquid alloy GaInSn is used as intermediate melt, the heat flux is controlled by regulating the height of this melt in a gap separating the flowing lead from the cooling water.

Development of high sensitive and reliable FFD and sodium leak detection technique for fast reactor using RIMS

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An early detection of fuel failure and subsequent precise identification of the failed fuel subassemblies are important and indispensable for operational safety of sodium cooled fast reactors. They help to prevent radioactive contamination in the primary cooling system, to reduce the personnel radiation dose and also to improve the plant availability factor as well. The artificially blended xenon and krypton gas (tag gas) is used as the failed fuel detection and location (FFDL) of the prototype fast breeder reactor Monju. The tag gas with different isotopic ratio was enclosed in each Monju fuel cladding tube and is measured by means of the mass spectrometry in the event of fuel failure. Since the tag gas released in the argon cover gas region is eventually diluted under the ppb level, it needs to concentrate at about 10^5 times using cryogenic charcoal bed which requires an elaborate operation and time.

The high sensitive FFDL system is developed using laser resonance ionization mass spectrometry (RIMS) method[1]. The RIMS is suitable for the isotope analysis of the element of ultra trace level, since there is no increase of the background by interfering ion and isobaric interference except for measuring element, because the measuring element is selectively ionized. Developed RIMS system as shown in FIG. 1 consists of the variable wavelength laser system with pulsed optical parametric oscillator and reflectron type time of flight mass spectrometer (RETOFMS). Employing a pulsed supersonic valve for sample gas injection enhanced the sensitivity. In order to suppress the unwanted peaks, the mass gating unit was adopted to prevent the argon ions from reaching the detector of RE-TOFMS.

The RIMS was applied to measure the tag gas which is used for the pressurized steel capsule of on-line creep rupture experiment in the experimental fast reactor Joyo[2]. As a result, this system could analyze the tag gas isotopic ratios of about $10^0 \sim 10^2$ ppb level and the measured results could identify tag gas.

An in-pile fuel failure simulation test was conducted in the experimental fast reactor Joyo to evaluate the fission products behavior in the primary cooling system and cover gas, and to confirm the plant operation procedure in the event of fuel failure. During the sodium sipping operation for FFDL, the sample of cover gas was taken into the stainless steel container and xenon nuclides were analyzed by the RIMS system[3]. The RIMS system has detected stable xenon nuclides and ^{133}Xe of which absolute value was evaluated to be a few ppt using the

germanium semiconductor detector. This is the first measured data of actual fission gas in the sodium cooled fast reactor plant, and it shows the applicability of the RIMS system for the fuel failure detection using the stable radioactive xenon isotopes.

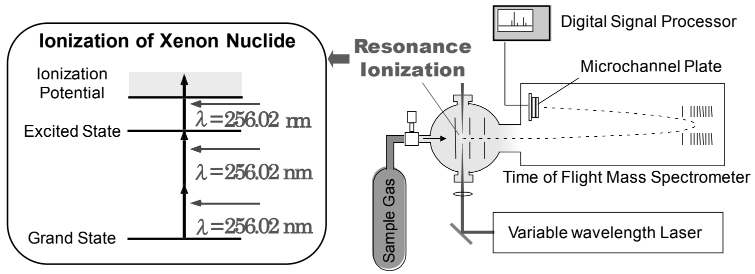

FIG. 1. Outline of RIMS system for FFDL system

In a demonstration fast reactor designed by the Japan Atomic Energy Agency (JAEA,) the gap volume between a reactor vessel and its guard vessel increases in proportional to the reactor size and nitrogen gas is filled the space in the double walled piping, resulting in the decrease of the sodium density. An innovative technology has been developed to selectively detect the sodium isotopes in the primary cooling system using RIMS. The target in this study is to improve the detection sensitivity of sodium isotopes up to two to three orders of magnitude lower than the current methods.

This research and development program consists of (1) investigating the detection process of sodium aerosol by RIMS, (2) manufacturing the prototype sodium detection system, and (3) testing its detection efficiency. The aerodynamic lens was newly introduced, which can transfer aerosols at atmospheric pressure into a vacuum chamber while increasing the aerosol density at the same time. Furthermore, the ionization process was applied by using the external electric field after resonance exciting from the ground level to the Rydberg level in order to increase the ionization efficiency[4, 5].

We performed the experiments using stable isotope ^{23}Na to evaluate the detection sensitivity of the prototype system as shown in FIG. 2[6]. Sodium aerosols of $10^{-2} \sim 10^2$ ppb were introduced into the aerodynamic lens from sodium aerosol generator, and aerosols were accumulated on the surface of the titanium tetrachloride plate for the set time using a chopper. After accumulation of aerosols, the plate was turned 180° and atomization laser was irradiated to its surface. After a delay, the resonance excitation laser was irradiated, and pulsed voltage was applied to ionize sodium atom. $^{23}Na^+$ ions were counted by means of the time of flight mass spectrometer. The preliminary test results using the stable isotope (^{23}Na) showed that prototype system could easily detect sodium aerosol of 100 ppb, equivalent to the sensitivity of the conventional detectors.

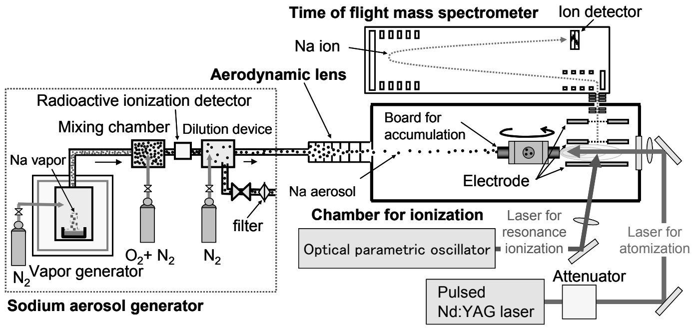

FIG. 1. Composition of Prototype Sodium Detection System

RIMS system will be an innovative system to improve the reliability FFDL system and Sodium leak detection system for the safety of fast reactor.

Present study includes the result of "the study of highly sensitivity technique for sodium leak detection using laser resonance ionization mass spectrometry to improve fast reactor plant safety" performed in JFY 2005 to 2008 entrusted to the Japan Atomic Energy Agency by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT.)

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Challenges and R&D program for improving inspection of sodium cooled fast reactors and systems

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Improvement of in service inspection and repair (ISI&R) capabilities has been identified as a major item of future SFR since ISI&R is strongly involved in safety analysis (number 1 defence line: verification of material and equipment state during reactor life), in economy reliability, and in investment protection (repair). Based on the feedback experience of sodium cooled reactors a pluri-annual R&D program was defined for the development of sodium fast reactors ISI&R capabilities. This specific R&D program is parted in four :

- o Conceptual design improvement of the primary circuit to limit the structures and components to be surveyed, to locate the sensible zones in accessible areas from outside or from inside (welded junctions) and to reduce the welds.
- o To develop the measurements and inspection techniques. Two axes are investigated: Continuous monitoring during operation (instrumentation) and periodic inspection tools (NDT during maintenance period).
- o To provide necessary accessibility and to develop remote controlling (robotic) with adapted carriers
- o To identify, develop and validate repair processes and techniques for repair interventions.

ISI&R objectives for sodium fast reactor primary circuits have been specified in 2008 with respect to the preliminary specifications issued by the French utilities (EDF) for the future nuclear plants (GENIV). Taking into account of the feed back knowledge, and the updated specifications, the main actions of ISI&R R&D program are the :

• Design : SFR conceptual design looks for improvement and innovative solutions (pool, loop, hybrid...) as regards ISI&R capabilities and accessibility.

• ISI&R techniques :

- Measurements and diagnosis during operation often require in sodium sensors able to support high temperature conditions (550°C). Interest was identified for telemetric measurement as used on Phénix.
- Direct examinations of the material state remain necessary and Non Destructive Techniques for periodic inspections are studied. UltraSonic (US) technique seems to be the basis, but EMAT is also investigated specifically to provide global diagnosis of the reactor vessel with few locations of a multi-array transducer. Possibilities for inspection from outside based on guided waves (through several internal structures) are also examined according to the Phénix experience (inspection of the core support shell). Transducers for these periodic inspections (under sodium at 180°C) are studied.
- Ultrasonic coupling at relatively low temperature has also to be optimized.

- Surface examination under sodium condition seems possible and could be used also for remote intervention (under sodium viewing).

• Simulation codes and experimental facilities : Corresponding to these numerous techniques, experimental tests are planned in water and then in sodium, meanwhile simulation codes (e. g. CIVA) must be developed / adapted to evaluate and compare the promising designs.

The main milestones of this ambitious R&D program are:

• Transducers validation (US telemetry, and NDT sensors) under sodium conditions in 2012,

• Definition of key elements of robotic equipment for under sodium intervention and first validation of repair intervention processes and techniques (cleaning, machining welding) in 2012.

R&D on Maintainance Technologies for FBR plants in JAEA
- The status quo and the future plan -

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The new inspection system have been applied to nuclear power plants in Japan since 2008 in order to improve their safety and reliability during operation. The performance of maintenance activities are evaluated through PDCA (Plan-Do-Check-Action) cycle and the utilization of the risk information, condition monitoring techniques, etc. is recommended in this system. Also, elongation of plant lives and high performance demands need the rationalization and sophistication of the maintenance program and technologies. Electric power companies owing LWRs already began to adopt advanced monitoring techniques, such as thermography, sampling of lubricants to check bearing integrity, etc. FBRs are under development but situation is as same as LWRs. However, unvisible and high temperature environments are added as the conditions that should be considered in developing advanced maintenance technologies, since the liquid metal coolant systems are adopted in many FBR plant designs. In this paper, the maintenance techniques now proceeding such as MONJU ISI equipments, and new promising monitoring systems, inspection techniques and repairing methods for the demonstration FBR are presented. FBR Plant Engineering Center was newly founded in JAEA Tsuruga region on April 1st 2009, one of whose main roles is to promote R&D of FBR maintenance technologies. The following figure shows the concept of the development of the FBR maintenance technology research from MONJU to the demonstration FBR and the R&D items.

1. Inspection, repair and replacement; The handling techniques of components containing sodium or in sodium are needed. Engineering elements of these techniques are visualization, remote control, automatic control, mitigation of radiation exposure, etc.

/ ISI equipments (a visualization equipment and a vehicle in liquid sodium) for the reactor vessel are being developed.
/ Verification tests for speed-up of inspection of tubes are proceeding.
/ Laser repair technique, remote control repair technique, etc. for dual piping and SG double tube are under development.
/ Disassembling, assembling, elimination of sodium, and inspection of the pump and IHX are the techniques to be developed.
/ FSW (Friction Stir Welding) is one of the expected repair techniques in sodium.

2. Monitoring ; Detection of water leakage in SG and monitoring in high temperature environment are key issues.

/ Development of sensors and enduranse tests in sodium is expected for SG monitoring.
/ Development of EMAT and ECT sustainable in high temperature environment is proceeding.
/ Very small sodium leakage detection system for dual piping system is to be developed.
/ Ultrasonic thermometer and flowmeter, hydrogen detector, and calibration method of sodium level gauge are important engineering elements.

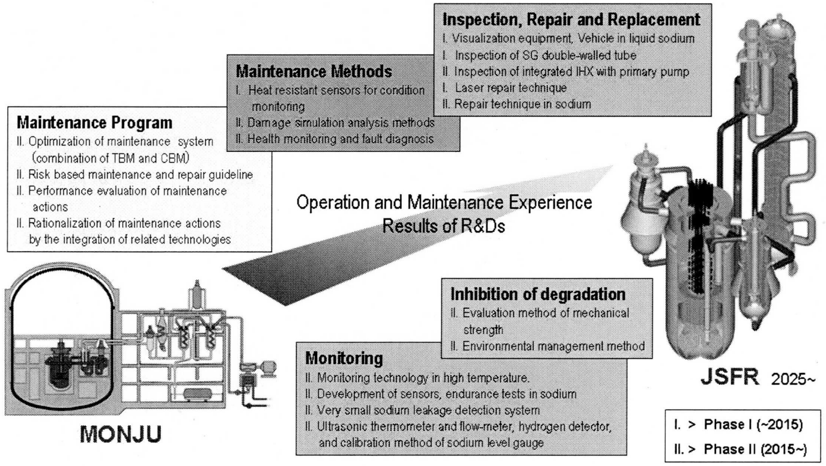
3. Inhibition of degradation; Degradation of materials is inevitable during plant life. Proper control of environmental conditions, strengthening of material surface, etc. are expected to contribute to elongation of plant lives.

4. Maintenance Methods; Time based maintenance (TBM) has been mainly used. However, it is needed to improve the maintenance strategy by introducing the condition based maintenance (CBM) in order to attain good performance from the point view of safety and reliability. R&D of condition monitoring methods are key issues.

/ Strain, temperature and vibration monitoring device, which can be used in high temperature environment, are effective. R&D of a heat-resistant FBG senser and its monitoring system is proceeding.

/ The developments of the damage simulation analysis methods of important compornents and a health monitoring and fault diagnosis system will be very effective.

5. Maintenance program and other techniques; Optimization of the maintenance system is very important in order to attain high performance and high quality of the plant operation. The combination of TBM and CBM and the integration strategy of the developed technology elements are key in making the maintenance program. Also, human factors should be considered in composing the system.



R&D on FBR Maintenance Technologies

Restoration Work for Obstacle and Upper Core Structure in Reactor Vessel of Experimental Fast Reactor Joyo

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The experimental fast reactor Joyo is the first sodium cooled fast reactor in Japan. Joyo attained initial criticality as a breeder core in April 1977 and has operated as a high performance irradiation test bed since 2003[1,2].

The 15th periodic inspection of Joyo commenced in May 2007 with the Fuel Handling Machine (FHM) being set up on the Rotating Plug (R/P) for refueling in June. When the R/P was taken down, measuring the load of the Hold-Down Shaft (HDS) revealed an abnormal decrease above the in-vessel storage rack (IVS). The HDS is a cylindrical FMH device that holds down the 6 surrounding subassemblies (S/As) which are adjacent to a withdrawn S/A. In order to investigate the cause of this, an in-vessel observation was conducted using a radiation-resistant fiberscope (RRF). As a result of the observations, it was discovered that the top of the irradiation test S/A “MARICO-2” (the material testing rig with temperature control) had bent onto the IVS (refer to Figure 1) as an obstacle, and had damaged the Upper Core Structure (UCS).

During the investigation of this incident, the in-vessel observations using RRF etc. took place at (1) the top of the S/As and the IVS for foreign material, (2) the bottom face of the UCS for damage under the condition with the level of sodium at -50 mm below the top of the S/As.

In-vessel observation techniques for a Sodium cooled Fast Reactor (SFR) are important in confirming its safety and integrity. Since an in-vessel observation for an SFR has to be conducted under severe conditions that include high temperatures (~ 200 deg-C) and high radiation doses (~ 400 Gy/h), and the primary sodium coolant has to be retained in the Reactor Vessel (R/V) to remove the decay heat, an in-vessel observation equipment has to be designed to not only tolerate the severe conditions but also be capable of being inserted into the sealed R/V through the fixed holes built in to the R/P and gain access to the observation areas.

The in-vessel observations were successfully conducted and the results provided useful information on incident investigations: (1) no loose part was found on the top of the S/As and IVS, but (2) the bottom face of the UCS had damaged. In addition, fundamental findings and the experience were gained during the observations, which included the design of equipment, operating procedures, resolution, lighting adjustments, photograph composition and the durability of the RRF under radiation exposure.

This incident necessitates the replacing UCS and the retrieving MARICO-2 for Joyo re-start. And conceptual procedure of restoration work has been almost established. In this discussion, high radioactivity of UCS, which becomes a critical in design of the cask, is one of important subject for development of handling device. In order to evaluate the activity of UCS, measurement of in-vessel gamma dose rate was carried out under the condition with the level of sodium at -50 mm below the top of the S/As. The measured value ranged 70 ~ 200 Gy/h at the top of the S/As in radial direction, and 0.1 ~ 200 Gy/h in axial direction. And it was confirmed that the main gamma-ray source was ^{60}Co. The results are used for the examination of shielding design of the cask of UCS.

The other action items in the restoration work, which are (1) design of UCS and MARICO-2 handling device, (2) reinforcement of buildings, (3) measures against sodium deposition etc., will be investigated in detail designs. The experience obtained through the restoration work in Joyo will provided valuable insights into possible further improvements and verifications for in-vessel inspection and repair techniques in SFRs.

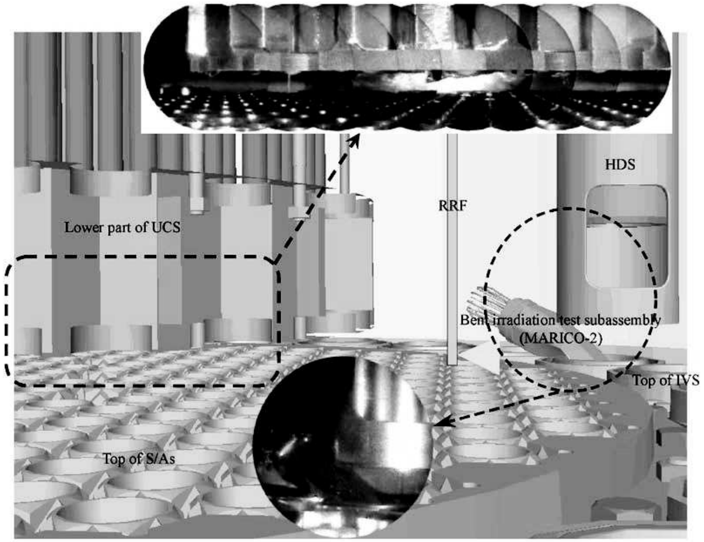

Fig.1 In-vessel observation results using an RRF

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Parallel Session 3.1:

Fast reactor safety: approaches and issues

Safety for the future Sodium cooled Fast Reactors

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The document is structured in two parts; the first one addresses the discussion of the safety objectives and principles applicable to the future Sodium cooled Fast Reactor (SFR); the second concerns the strategy and the roadmap in support to the realization of the SFR.

A) The safety objectives and principles for future reactors

The content of the first part reflects the works of AREVA, CEA and EDF concerning the safety orientations for the future reactors. The availability of such orientations and requirements for the SFRs has to allow introducing and managing the process which will lead to the detailed definition of the safety approach, to the selection of the corresponding safety options and to the identification and the motivation of the supporting R&D. As a preamble, the document thus reminds the objectives and the general principles envisaged for future reactors which are applicable to an industrial SFR. As for the safety objectives, those retained in France for the EPR™ are already very ambitious and guarantee a very high level of protection; further and prescriptive reduction of the risk with regard to this level is not justified and could even be counterproductive. That is why these objectives would be kept for future SFRs.

It is worth noting that the safety level for the EPR™ is reached through, among others:1) An extended design basis domain which, aside the classical "design basis", includes the treatment of severe accidents considering both their prevention and the mastery of their consequences. 2) The "*practical elimination*" of some situations which will be prevented and whose consequences will not be explicitly addressed by the design.

Generally speaking, the approach recommended for future SFRs insists on the necessity to realize a robust safety demonstration. Such demonstration shall primarily lean on the experience feedback as well as on the choices of design options and on the R&D programs which have to: 1) ease the identification and the management of the uncertainties and 2) contribute rejecting possible threshold effects.

The principles for the design shall remain based on a determinist approach and based on the defence in depth; this approach will have to take advantage of insights and indications brought by probabilistic studies (i.e.: on line simplified PSA).

A specific section addresses the concern for the treatment of the specific SFR severe accidents. The corresponding R&D is performed taking into account two possible tracks:

- The first R&D track will aim to "practically eliminate" the risk of whole core melting by implementing specific core and sub assembly (SA) design features. This very challenging track can include, for example, the search for significant progress in core or fuel design and significant improvements in core control systems.
- The second R&D track, equally very challenging, will aim to improve the mitigation provisions against hypothetical severe core accidents, still in the frame of a defence-in-depth approach. This can include, for example, the search for, robust containment designs, improved core catcher solutions, etc.

The results of these R&D program will allow choosing the most appropriate safety strategy as basis for the development of the future SFRs.

The main SFR's safety related themes and to the ways to insure the improvement for their treatment are identified. The selection of these "ways for improvement" leans on the experience feedback acquired by the design, the operation and the safety analysis of the Creys-Malville power plant (SPX1) as well as the Phenix plant, and of the SFR projects which followed: RNR1500 and EFR (European

Fast Reactor). The assessment by the European and French Safety Authorities is considered, as well as the international experience feedbacks from existing sodium-cooled reactors in operation.

It has to be noted that aspects connected to the proliferation resistance and the physical protection (PR&PP), to the radioprotection and to the non nuclear risks, as well as those related to the fuel cycle safety, are not addressed by the document.

B) Strategy and roadmap in support to the R&D for the future SFRs

A R&D program led jointly by CEA, EDF and AREVA has been written with the objectives to be able to produce by 2012 a preliminary report describing safety options for the future SFRs, and to deduce from it the safety options of the ASTRID prototype.

The prevention of accidents is based on the following orientations: to favour the natural behaviour of the plant, to reduce as far as possible common modes, to use very highly reliable systems. The selection of safety devices and provisions is based on consideration of efficiency with regard of both prevention and mitigation. If it could not be demonstrated that a given severe accident situation is practically eliminated, its consequences should be studied and mitigation means should be proposed and implemented. To help implementing the defence in depth principles, the line of protection method is used in preliminary design step.

The goal of a first part of the R&D program is to improve the methods and the tools for the safety studies, as for instance the development of a Probabilistic Safety Assessment (PSA) methodology which is appropriate considering the SFRs design stages and characteristics.

In order to define the R&D effort in support to the core design, and following the experience feedback, improved goals for the detection devices and the core's natural behaviour have been proposed for all the possible accidental situations (e.g., loss of flow, loss of heat sink, reactivity insertion, local accident, etc.).

Despite the fact that the list of situations which have to be practically eliminated is not yet finalized, a specific part of the R&D program is already organized in order to reach the objectives to practically eliminate a limited number of situations, in particular:

- The total loss of DHR systems. Works are on-going in order to propose dispositions for reducing as far as reasonable possible common modes, and for assessing the reliability of the DHR systems.
- Uncontrollable sodium-water reaction. In parallel to the development of suited steam generator, improvement of leakage detection is for example on-going.

About this last concern, other options are studied in order to exclude a sodium-water reaction by selecting an alternative working fluid: a gas based energy conversion system (ECS) or with an intermediate fluid different from sodium. In order to assess the impact of such energy conversion systems on the plant safety, a functional analysis has been done and, for example, for the former (gas ECS), strategies against the gas leakage into sodium are proposed.

For 2012, the objectives of the program which are related to the mitigation of severe accident are the following:

- First calculations of the behaviour of the new core and reactor designs for some severe accident conditions;
- Preliminary design of mitigation provisions favouring, for instance, boiling stabilization in the core, or the fuel ejection or the absorber insertion into the corium;
- Designs of internal and external core catchers suitable to the reactor designs, including the assessment of the relevancy of the use of sacrificial material;
- Definition of an experimental program (qualification of new fuels, qualification of a core catcher, assessment of the minor actinides' influence, etc.).

The objective of the last part of the program is the assessment of the relevancy of software used for Phenix, SPX1, RNR1500 and EFR, to verify that they are still pertinent and, when necessary, to elaborate the development plan for these codes.

Finally it is worth noting that, at the beginning of 2009 exchanges have started between the French SFR project and the French Safety Authority. Their objectives are to share the experience feedback on SFR and to exchange points of view on the R&D program.

Mitigation of Sodium-Cooled Fast Reactor Severe Accident Consequences Using Inherent Safety Principles

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Sodium-cooled fast reactors are designed to have a high level of safety. Events of high probability of occurrence are typically handled without consequence through reliable engineering systems and good design practices. For accidents of lower probability, the initiating events are characterized by larger and more numerous challenges to the reactor system, such as failure of one or more major engineered systems and can also include a failure to scram the reactor in response. As the initiating conditions become more severe, they have the potential for creating serious consequences of potential safety significance, including fuel melting, fuel pin disruption and recriticality. If the progression of such accidents is not mitigated by design features of the reactor, energetic events and dispersal of radioactive materials may result.

In the United States, accidents which have the potential for severe consequences usually are of probablity less than 1 x 10^{-4} per reactor year, intended to satisfy the U.S. Nuclear Regulatory Commission (NRC) goal of limiting accidents with any fuel melting to such low probabilities. Such severe accidents include the cateogry of Anticipated Transient Without Scram (ATWS) events mentioned above. Three accidents are usually analyzed to evaluate the reactor response in these cases; the unprotected (unscrammed) loss-of-flow (ULOF), where pumping power is lost and the pumps coast down, reducing coolant flow through the reactor core; the unprotected transient overpower (UTOP), where a control rod is inadvertently withdrawn from the core; and the unprotected loss-of-heat-sink (ULOHS), where the steam generator is isolated from the reactor in response to a turbine trip. For each of these accidents, there are several approaches that can be used to mitigate the consequences of such severe accident initiators, which typically include fuel pin failures and core disruption. One approach is to increase the reliability of the reactor protection system so that the probability of an ATWS event is reduced to less than 1 x 10^{-6} per reactor year, where larger accident consequences are allowed, meeting the U.S. NRC goal of relegating such accident consequences as core disruption to these extremely low probabilities. The main difficulty with this approach is to convincingly test and guarantee such increased reliability. Another approach is to increase the redundancy of the reactor scram system, which can also reduce the probability of an ATWS event to a frequency of less than 1 x 10^{-6} per reactor year or lower. The issues with this approach are more related to reactor core design, with the need for a greater number of control rod positions in the reactor core and the associated increase in complexity of the reactor protection system. A third approach is to use the inherent reactivity feedback that occurs in a fast reactor to automatically respond to the change in reactor conditions and to result in a benign response to these events. This approach has the advantage

of being relatively simple to implement, and does not face the issue of reliability since only fundamental physical phenomena are used in a passive manner, not active engineered systems. However, the challenge is to present a convincing case that such passive means can be implemented and used. The purpose of this paper is to describe this third approach in detail, the technical basis and experimental validation for the approach, and the resulting reactor performance that can be achieved for ATWS events.

There are several key components of the reactivity feedback for a fast reactor, including Doppler, coolant density, axial thermal expansion of the control rod driveline, axial thermal expansion of fuel and cladding, and radial thermal expansion of the reactor core. Several of these are widely accepted as being well understood, and they have been included in safety analyses for licensing of sodium-cooled fast reactors. However, others, such as the axial thermal expansion of the control rod driveline and the radial thermal expansion of the reactor core, are more controversial. The focus of the discussion in the paper will be the theoretical basis for these reactivity feedback mechanisms, and the experimental validation in reactors such as FFTF and EBR-II that demonstrates that such mechanisms can also be convincingly used in safety analyses for predicting the reactor response.

Following this discussion, examples will be provided demonstrating the response of sodium-cooled fast reactors to the ULOF, UTOP, and ULOHS accidents. Both oxide and metal fuel examples are provided, showing that the inherent safety concept can be applied in both cases, and that it is possible to completely mitigate any serious consequences for such accidents, with benign termination of these events.

The discussion will then move to analyses of even more serious initiating events, where additional failures are postulated to occur. Such accidents have a probablity of occurrence of less than 1 x 10^{-6} per reactor year, and may result in energetic events within the reactor vessel than can be large enough to fail the reactor vessel and threaten the integrity of the containment building. Analyses will be presented that demonstrate that the use of inherent safety principles may be able to lessen the consequences of a given event, with the result that the probability of an accident that could threaten containment integrity could be reduced even further. Desirable design characteristics will be identified for enhancing the reactor performance in such events, leading to the development of a reactor system with greatly increased safety and lower potential risk to the public and the environment.

Ways to the Nuclear Power Renaissance and Vital Risk Free Fast Reactors

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Analysis of the basic reasons of NP stagnation is presented. It was shown that the real ways to renaissance can be launched due to eliminations of all essential menaces and vital risks for the account of the set of the deterministic reactor/fuel cycle "contra-risk" properties.

Let us to use the definition that only those risks of a technology are referred to **vital** ones which are able to withdraw it behind the acceptability limits or to restrict the scale of the industrial application. Then for current NP, vital risks are:

threats of the catastrophic accidents, dangerous risks of weapons material proliferation, risks of extremely long storage of radioactive wastes, threats of enormous investment losses at the conditions of the limited capitals and of a monetary inflation, the effect of a "progressing deadlock" due to limits of the fuel reserves.

It means that to achieve overall acceptability of NP it is necessary to eliminate all these vital risks and this task is difficult a priori. Similar idea has been proposed and analyzed earlier as the main instrument of the radical enhancement of reactor safety applied to some innovative concepts of fast reactors of the next generation as, for example, it has been proposed in MSBR, RBEC, BREST-projects. The new goal is determined as the deterministic elimination or the guaranteed/acceptable suppression of all vital risks/menaces simultaneously. This is considerably more difficult, however, really possible due to the following means applied to innovative fast reactors (called hereafter as Vital Risk Free Fast Reactors –VRF-FR) with different coolants:

1. *no catastrophic accidents* property arrives if reactor cores are "self-defended", particularly if there are no large and positive void effects;
2. *risks of weapons material proliferation* are considerably suppressing if fuel cycles do exclude enrichement of the feed as well as fuel re-enrichment after reprocessing;
3. *risks of the storage of long-lived wastes* could be significantly minimized for the account of the "waste risk free fuel cycle" which is considered keeping a mixture of the residual actinides and lanthanides in reactors intended for their incineration as well as of transmutation of the most dangerous long-lived fission products; there are sufficient "working places" for incineration/transmutation if the NP park is growing;
4. *radical decrease of the investment risks* due to significant shortening of the NPP construction time if they consist of a set of autonomic modules of precise shop-made fabrication (in this case, each module can be launched in operation independently), cheap reactor fuel inventory and if safety systems of modules are simplified.

Problem of fuel reserve shortage does not exist for fast reactor parks, at least for a foreseeable time.

NP with VRF-FR is promised to be very promising and long lived. By large scale using, NP renaissance would be visible even in the nearest future.

Recent analyses [1-4] showed that this task can be solved on the base of existing/soon forthcoming fast reactor technologies and, this is important, for the account of neutron balance radical enhancements. The necessity condition – "supplementary" production of neutrons in cores became

the decisive factor for realization. It can be fulfilled both for the account of neutron spectrum hardiness and of parasitic neutron capture reduction, but not due to the fuel feed enrichment use.

For example, it was discovered [2] that the effective way of suppression of the positive void reactivity effects in fast reactor cores with non-enriched fuel feed is to follow the principle – to grow the K_∞ for increasing the neutron leakage from cores. Otherwise, the application of "economic" (respecting neutronics) fuels in the modular reactor configurations is recommended.

The number of supplementary neutrons required for each of means realization (and, totally, for whole task solution) is evaluated. This number depends on reactor structure as well as on coolant type: it varies in the interval of 0.3 – 0.5 neutrons/fission. In order to do this, there are several solutions: either using "dense/tight" core compositions with advanced fuels (nitrides based on N^{14} and N^{15}, carbides, nanoceramic fuel, ….) and coolants with the smallest neutron consumption (Pb-208,…) as well as the fuel components with low neutron captures (Cl^{37}) for some molten salt VRF-FR

or/and subcritical systems with relatively modest external neutron sources (ADS, thermonuclear hybrids).

Estimation of economical risks showed that the investment expenditures could be reduced considerably (about one order in the magnitude) if traditional units are replacing by modular ones.

NP with VRF-FR park is expected to be very promising and long lived. By large scale application, NP's renaissance would be visible soon in the nearest future.

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Design Features to Achieve Defence-In-Depth in Small and Medium Sized Reactors

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According to the categorization adopted by the International Atomic Energy Agency (IAEA), small and medium sized reactors are those with the equivalent electric output less than 700 MW. In most of the cases, deployment potential of SMRs is supported by their ability to fill niches in which they would address market situations different from those of currently operated large-capacity nuclear power plants. Ensuring adequate defence-in-depth (DID) is important for reactors of smaller output because many of them are being designed to allow more proximity to the user, specifically, when non-electrical energy products are targeted. Based on the activities recently performed by IAEA [1], the paper provides a summary description of the design features used to achieve DID in the eleven representative concepts of SMRs. The SMRs considered included reactors of different types — pressurized water reactors, pressurized boiling light water cooled heavy water moderated reactors, high temperature gas cooled reactors, fast reactors, and non conventional very high temperature designs. The design descriptions were structured to follow the definitions and recommendations of the IAEA safety standard NS-R-1 “Safety of the Nuclear Power Plants: Design Requirements” [2]. The focus of this paper would be on sodium cooled and lead cooled fast reactors — the 4S-LMR (Japan) and the SSTAR and the STAR-LM (USA).

An enveloping design approach for all SMR designs considered is to eliminate as many accident initiators and/or to prevent as many accident consequences as possible, by design, and then to deal with the remaining accidents/consequences using plausible combinations of the active and passive safety systems and consequence prevention measures. Incorporation of the inherent and passive safety features is in several cases facilitated by smaller reactor capacity and size. However, the design solutions for the active and passive safety systems are, generally, capacity-independent.

For the considered sodium-cooled 4S-LMR and lead cooled SSTAR and STAR-LM concepts, the designs incorporate optimum sets of reactivity feedbacks and other inherent safety features, provided by design, to effectively de-rate certain accidents and combinations of accidents that are potentially severe in reactors of other types. Specifically, the intention of the designers is to de-rate transient overpower events.

In the 4S-LMR, the corresponding features include negative whole-core void reactivity effect, contributing to defence in depth level 3, and the absence of control rods in the core, with power being controlled via a feedwater flow rate in the power circuit. The burn-up reactivity compensation is then performed with an active system based on a very slow upward movement of the pre-programmed radial reflectors, with no feedback control. Should a reflector get stuck, the reactor would operate safely for a certain time and then get “passively shut down” [1] by the increasing negative reactivity. At the same time, the drop of axial

[1] ‘Passive shutdown’ is used by the designers to denote bringing the reactor to a safe low-power state with balanced heat production and passive heat removal, with no failure to the barriers preventing radioactivity release to the environment; all

reflectors is a standard reactor shutdown feature. Altogether, the features mentioned above are unique to a small reactor size.

For the lead cooled SSTAR and STAR-LM, the inherent safety features contributing to prevention or de-rating of possible accidents are generally typical of the lead cooled reactor line. They include very high boiling point of lead; pool type design with a free surface of lead to allow removal of the gas bubbles from primary coolant before they enter the core; the guard vessel and reactor location in the concrete shaft; optimum sets of reactivity effects; and high heat capacity and small overall reactivity margin in the reactor core. It should be noted that some designers mention the unit size of the lead and lead-bismuth cooled reactors to be limited from seismic considerations. According to the studies performed in Japan, this size cannot exceed ~750 MW(e). Although still requiring further proofs, the “passive shutdown” capability of the lead cooled reactors is facilitated by smaller reactor size.

The designers of the SMRs addressed foresee that safety design features contributing to DID levels 1 – 4 could be sufficient to meet the objective of the DID level 5 “Mitigation of radiological consequences of significant release of radioactive materials”, i.e., that emergency planning measures outside the plant boundary might be reduced against those foreseen for larger capacity plants. For all SMRs addressed, the designers expect that prototype or first-of-a-kind plants could be licensed according to the current regulatory norms and practices in member states. Further advancement of regulatory norms could then facilitate design improvements in the next plants, specifically, as comes to a reduction in the emergency planning requirements. Risk-informed approach to reactor qualification and licensing could facilitate licensing with reduced off-site emergency planning for smaller reactors, once it gets established.

Within the deterministic safety approach it might be very difficult to justify reduced emergency planning in view of a prescribed consideration of a postulated severe accident with radioactivity release to the environment owing to a common cause failure. Probabilistic safety assessment (PSA), as a supplement to the deterministic approach, might help justify very low core damage frequency (CDF) or large early release frequency (LERF), but it does not address the consequences and, therefore, does not provide for assessment of the source terms.

Risk-informed approach that introduces quantitative safety goals, based on the probability-consequences curve could help solve the dilemma by providing for a quantitative measure for the consequences of severe accidents and by applying a rational technical and non-prescriptive basis to define a severe accident.

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relying on the inherent and passive safety features only, with no operator intervention and active safety systems being involved, and no external power and water supplies being necessary, and with the grace period infinite for practical purpose.

Parallel Session 3.2:

Fast reactor safety: approaches and issues

Generation IV International Forum Risk and Safety Working Group: Terms of Reference, Accomplishments, Current Activities & Perspectives

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The Generation IV International Forum (GIF) is a collaboration of governments of countries committed to joint development of the next generation of nuclear technology. A selection of six reactor technologies – four of which with fast spectra - has been identified on the basis of being clean, safe and cost-effective means of meeting increased energy demands on a sustainable basis, while being resistant to diversion of materials for weapons proliferation and secure from terrorist attacks [1]. The Generation IV International Forum (GIF) Risk and Safety Working Group (RSWG) was created to elaborate and recommend a homogeneous and effective safety approach applicable to the design and the assessment of these Gen IV systems.

The activities and work of the RSWG include, among others, the following: **1)** Identify and promote a common and consistent risk informed approach to safety in the design of Generation IV systems by: *a) proposing safety principles, objectives and attributes based on the Gen IV safety goals to guide R&D plans; b) proposing a technology neutral general framework of technical safety criteria and assessment methodologies; c) testing and demonstrating the applicability of the framework and assessment methodologies; d) proposing necessary crosscutting safety related R&D.* **2)** Provide consultative support on matters related to safety to GIF Systemss Steering Committee (SSCs) and other Gen IV entities which develop specific concepts and designs. **3)** Interact with the GIF proliferation resistance & Physical protection (PR&PP) Working Group (e.g. [3]). **4)** Interact with the NEA Multinational Design Evaluation Programme (MDEP) addressing the safety aspects of new reactor designs. **5)** Undertake appropriate interactions with regulators, IAEA and relevant stakeholders, primarily for the purpose of understanding and communicating regulatory insights to the Generation IV development

An effective and homogeneous approach to the safety of Generation IV systems must lean on a coherent and well-founded safety philosophy based, among others, on the following considerations [2]:

- ***Safety approach :*** The diversity of the Gen IV systems and the need for a homogeneous strategy applicable for the design and the safety assessment of these systems justify the development of an updated safety approach with the integral implementation of the defense in depth and in particular with consideration of possible severe plant conditions, and this since the very beginning of the design.
- ***Safety Objectives :*** The RSWG recognizes that the safety objectives applicable to the reactors of the third generation (e.g. AP1000 and EPR) are very ambitious and guarantee an improved level of protection reducing the level of risk in a demonstrable way. The RSWG believes that this achieved level is excellent and can be kept as a reference for future reactors. Meanwhile, although not formally required, further safety enhancements associated with Generation IV technologies, are likely possible through the implementation, early in the design process, of a safety that will be **"*built-in*"** within the design rather than **"*added on*"** to the system architecture.
- ***Defense in depth :*** Defense in depth (DiD) and its principles remain the foundation for the design. Compared to the current practices complementary objectives are suggested to achieve a DiD that is exhaustive, progressive, tolerant, forgiving, and well balanced.
- ***Risk-informed approach: design & assessment :*** The design process as well as the safety assessment should be driven by a "risk-informed" approach supported by deterministic insights, the use of PSA techniques and – as needed – by complementary tools to measure the degree of achievement of the DiD implementation. The increased robustness of the safety demonstration is the natural complement of this approach.

- ***Modeling and simulation*** : For Gen IV systems, in addition to prototyping and demonstration, modeling and simulation should play a large role in the design and the assessment.

A key objective for the RSWG is the capability to develop all the above safety considerations into technical requirements applicable by the designer. For example, the approach shall be able to correctly address concerns raised by specific fast reactor characteristics, such as: *the core which is not in its most reactive configuration; core void effect that could be positive; high power density; possible chemical reactivity for the coolant (e.g. for Na); possible opaqueness of the coolant (inspection of the immersed structures for Na & Pb); duration of the fuel element downloading*.

Besides defining and developing the approach, a tool is required to assess the capabilities of the Gen IV systems in terms of public risk aversion and safety margin enhancement, to assist designers, safety analysts and regulators. The purpose of the assessment is to the guide the design process to incorporate safety features and improvements, to provide help for the identification of vulnerable systems, and to quantify benefits accruable from improvements. The assessment may also be used to help support licensing and regulatory processes. The methodology is tentatively called the Integrated Safety Assessment Methodology (ISAM). It is envisioned that the ISAM will be used in three principal ways:

- **Drive the course of the design evolution** : The ISAM is intended for use throughout the concept development and design phases, with insights derived from the ISAM serving to actively drive the course of the design evolution.
- **Support risk and safety comparisons** : The methodology can be applied at any point in the design evolution from the conceptual development phase through the final design phase to support risk and safety comparisons of various options for the nuclear systems.
- **Measure the level of safety and risk** : The ISAM can be applied in the late stages of design maturity to measure the level of safety and risk associated with a given design relative to some safety objective or licensing criterion.

The integrated methodology consists of five distinct complementary analytical tools and stages which are ordered around the last one, the Probabilistic Safety Assessment. The tools/stages are the following: *Qualitative Safety Requirements/Characteristic Review (QSR); Phenomena Identification and Ranking Table (PIRT); Objective Provision Tree (OPT); Deterministic and Phenomenological Analyses (DPA); Probabilistic Safety Analysis (PSA)*

Together, these tools contribute meeting five objectives: 1) *Demonstrate how well Gen IV safety goals are met*, 2) *Show how compliance with specific safety requirements are demonstrated*, 3) *Differentiate alternative design solutions and quantify benefits*, 4) *Identify gaps in knowledge and areas requiring R&D*, and 5) *Be simple and cost effective to use*. Complete information on each of the elements will be presented within the full FR09 conference paper.

The development of the methodology is in progress and should be finalized by the end of 2009. Once the methodology available, the following stage will be the preparation of one or some examples of application (likely a Sodium cooled Fast Reactor) which will be collectively led by the RSWG and the concerned GIF SSC.

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Basis of Technical Guideline for FBR Fuel Safety Evaluation in JNES

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The guideline for safety evaluation of LMFR MOX fuel is now being studied. The draft version of the technical guideline will be completed with the current Monju fuel and will be reviewed in order to list up additional safety issues when the guideline is applied with the LMFR MOX fuels of Monju up-graded core and of next Japanese demonstration fast reactor.

It is considered that a lot of items shall be evaluated in order to confirm fuel safety during normal, transient and accident conditions. Fuel safety means to maintain its functions and to keep its integrity. These items are identified in this paper with three major functions, which are essentially requested for reactor safety, and the major functions are reactor shutdown, core cooling and containment of radioactive materials. Total of 13 items for LMFR fuel safety evaluation are identified and total of 17 items for LWR fuel safety are also identified. Safety Review Plan, which is published by United State Nuclear Regulatory Commission (US-NRC) [1], is referred in order to list up the items for LWR fuel safety evaluation. The items for both LMFR fuel and LWR fuel safety evaluations are compared in Table 1 together with specific materials and irradiation level. The safety items which are different between LMFR and LWR are shadowed by yellow. As shown in this table, the number of the items for LMFR fuel safety evaluation is less than that for LWR one. It might be considered that safety of LMFR fuel can be more easily evaluated as compared with safety evaluation of LWR fuel, especially concerning the functions of core cooling and containment. This means that irradiation behavior of LMFR fuel during irradiation might be simpler than that of LWR fuel since coolant pressure is low and boiling temperature of coolant is much higher than temperature at normal operation and the fuel behavior is mild under the transient over power (TOP) and loss of flow (LOF) conditions for LMFR [2]. It might be concluded that the items to be evaluated for LMFR fuel safety can be listed up without any serious omission.

The design analysis is mostly applied as fuel safety evaluation procedure, and analysis by computer modeling code is taken for predictions of fuel performances such as fuel centerline temperature, cladding mid-wall temperature, fuel pin diameter increase, cladding creep damage and so on. A simplified method using design equations is also taken for stress and strain calculations.

Acceptance criteria and/or design limits are determined based on many kinds of experiments, such as operational experiences in foregoing LMFR, irradiation tests and out-of-pile experiments. For example, the criteria and/or limit of fuel pin diameter increase should be determined based on the operational experiences in foregoing LMFR.

This paper will describe the fuel safety issues and assessment for LMFR fuel safety evaluation with reference to LWR fuel.

Table 1 Comparison of items for LMFR fuel safety evaluation with those for LWR fuel one.

Specific Materials and Irradiation Level	LMFR fuel	LWR fuel
Specific Materials	MOX Pellet SUS Cladding Sodium Coolant	UO_2 Pellet Zircaloy Cladding Water Coolant
Irradiation Level	2.3x10^{23} n/cm^2 (E>0.1MeV) 80GWd/t	1x10^{22} n/cm^2 (E>0.1MeV) 55GWd/t
Principal functions for Reactor Safety	**Items for LMFR fuel safety evaluation**	**Items for LWR fuel safety evaluation**
Reactor Shutdown	Fuel Assembly Deformation	Fuel Assembly Deformation
Core Cooling	Fuel Assembly Floating	Fuel Assembly Floating
	Fuel Pin Diameter Increase	Fuel Rod Ballooning
	Fuel Slumping	Violent Expansion of Fuel
		Generalized cladding Melting
		Cladding Embrittlement
Containment	Overheating of Fuel Pellet	Overheating of Fuel Pellet
	Pellet Cladding Mechanical Interaction	Pellet Cladding Interaction
	Stress & Strain	Stress & Strain
	Cladding Creep Damage	Internal Gas Pressure
	Cladding Temperature	
	Cladding Fatigue Damage	Cladding Fatigue Damage
	Wear mark	Fretting
	Bundle-Duct Interaction	Hydriding
	Duct-Duct Interaction	Cladding Collapse (Densification)
		Bursting
		Excessive Fuel Enthalpy
		Overheating of Cladding (DNBR, MCPR)

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Development of Integrated Analytical Tools for Level-2 PSA of LMFBR

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As same as to light water reactor, JNES (Japan Nuclear Energy Safety Organization) has devoted to prepare the analysis tools for PSA to liquid-metal cooled fast breeder reactor (LMFBR) to make safety evaluation from regulatory side. The developted tools consist of a group of safety analysis computer codes and an analysis method called PRD (Phenomenological Relationship Diagram) to qualify logically the probability distribution at the branching points in event trees. So far the tools have been used to evaluate the effectiveness of accident management measures of Monju proposed by the owner and the tools are under further development to describe the event progresses more realistically. One of the objectives of this improvement is to construct data bases of the Emergency Response Support System (ERSS) for Monju by conducting many application analyses to the conceivable scenarios after initiating events. The present paper introduces the function of each tool in the synthetic analysis system coupled with the accident scenario and presents points for future improvement.

Figure 1 shows the phase transitions of severe accidents of LMFBR and the role of each analysis tool. In (i) the plant response phase, the temperature of sodium in the primary cooling system begins to rise due to the power to flow mismatch. In cases of gradual temperature increase such as PLOHS (protected loss-of-heat sink), the sodium boundary will fail by the high temperature creep. If boundary failure does not occur, the sodium will lastly boil. The temperature and the pressure changes during the plant response phase are analyzed by the NALAP-II code. NALAP-II also calculates the SCDF (structural cumulative defect factor), that is an index of high temperate creep, of the key locations in the plant, however, the application is limited to the parts whose geometry are modeled by a cylindrical wall [1]. Hence, for the analysis of components with complicated shape that require the consideration of buckling, structure analysis codes, ABAQUS and FINAS, were used to analyze the large shape deformation. Due to the high temperature of the coolant, the fuel pin cladding also suffers the high temperature creep. Once the cladding fails, the fission products (FPs) kept inside of the fuel pin will release into sodium. Then, all noble gases and a part of volatile FPs will transfer to the cover gas region. The remained FPs in sodium will circulate in the primary cooling system. During the circulation, some part of FPs will deposit on the inner surface of the components. Such transfer behavior of FPs is analyzed by the ACTOR code.

The progress of (ii) the core disruption phase is analyzed by the AZORES code. In the phase the core will melt and the molten core materials will fall on the bottom of the reactor vessel and cause the melt-through of the vessel wall. High temperature and high pressure in the containment vessel are caused by the sodium-concrete and the hydrogen burning. On the other hand, the neutronics of the various geometries of the disrupted core is solved by the ARCADIRN-FBR code [2]. Using the neutronics data, the APK code performs the analyses of the super-prompt criticality and the prompt criticality phenomena. The derived information, such as the debris temperature, is used in the AZORES calculation of the core debris-concrete reaction.

For ATWS (anticipated transient without scrum), the progress of the core disruption phase is quite different from the initial events with reactor scrum, such as PLOHS, particulally in the early time peroid. Devalopment of an analysis code named ASTERIA was initiated to analyze ULOF (unprotected loss-of-flow).

In (iii) the containment vessel (CV) response process, the timing of containment failure and the pathes of FPs to the environment is important for the evaluation of the amount of the released FPs into the environment. These items are also analyzed by the AZORES code using the aerosol behavior module.

PRD provides an analytical perspective that starts from the highest level in an event tree (an event at a branching point) and travels back to elemental events at the lower levels. The value of an elemental event may be represented only with a probability distribution. By evaluating the transition of such uncertainties of elemental events, it is possible to estimate the distribution of probability of the top event by PRD.

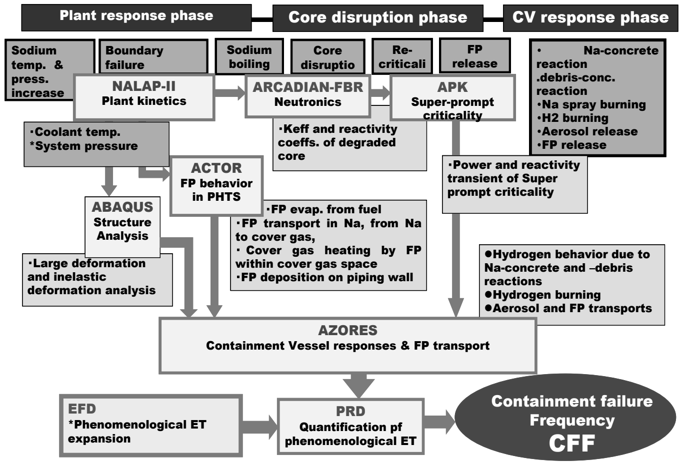

FIG.1. Phase transitions of severe accident of LMFBR and the synthetic analysis system being prepared by JNES

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Uncertainty Analysis for Unprotected Loss-of-Heat-Sink, Loss-of-Flow, and Transient-Overpower Events in Sodium-Cooled Fast Reactors

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Typically, reactor safety analyses utilize a deterministic approach where uncertainty is accommodated by assuming pessimistic values for input parameters that are important to safety. This paper considers a stochastic approach for explicitly including uncertainty in safety parameters by applying Monte Carlo sampling coupled with established deterministic reactor safety analysis tools. Similar analyses have been proposed in the past, but in these analyses a limited number of deterministic calculations were used to determine response surfaces for the outputs of interest and the response surfaces coupled to the Monte Carlo sampling[1]. The Monte Carlo approach yields frequency distributions for reactor safety metrics (e.g., peak fuel temperature) that can be compared to performance limits, allowing for an improved determination of safety margin and a clear determination of which safety parameters affect the transient response.

Example analyses have been carried out for an 840 MWth, sodium cooled, advanced burner reactor to demonstrate the effect of uncertainty in selected input parameters on the uncertainty of various outputs for calculations of unprotected (failure of reactor shutdown mechanisms) combined loss-of-heat-sink and loss-of-flow, loss-of-flow, and transient overpower events. The reactor has metallic fuel and a conversion ratio (transuranic production rate/transuranic destruction rate) of approximately 0.5 and is similar in design to the reactor having a conversion ratio of 0.25 considered by Cahalan, et al[2]. Mean values for the stochastic input parameters were taken to be best estimate values and standard deviations in the range of 10% to 20% of the mean values were assumed. The specific stochastic reactivity coefficients considered are the coolant temperature reactivity feedback, the Doppler coefficient, fuel axial expansion feedback, core radial expansion feedback, and control rod driveline expansion feedback. In addition to the reactivity coefficients, the loss-of-heat-sink calculations assume that the temperature at which a flow coastdown initiates and the rate at which heat removal decreases are stochastic. All stochastic parameters were assumed to have normal probability distributions and to be uncorrelated.

All the results considered in the present analysis are based calculations for each of 10,000 independent samples of the stochastic input parameters. The number of samples required in a more complete analysis would depend on the purpose of the analysis. For example, initial calculations might consider many more stochastic input parameters and use fewer samples to identify the most important parameters. Then calculations with a larger number of samples might be carried out using a smaller set of stochastic input parameters. Rather than go through a more detailed screening process, the choice of stochastic parameters used in the current analysis was based on previous experience.

Frequency distributions for several output parameters were obtained. The frequency distribution for the peak fuel temperature in an unprotected loss-of-flow transient is shown in Fig. 1. 99% confidence limits were estimated based on binomial probability distributions and

are shown in the figure. Also shown is the estimated frequency based on a log-normal distribution having the same mean and standard deviation as the results shown by the histogram. It is of interest to note that even though the input parameters were assumed to be independent of one another and to have normal probability distributions, the distribution for the peak fuel temperature does not resemble a normal distribution. In this case, even a log-normal distribution provides a poor approximation to the observed frequency distribution.

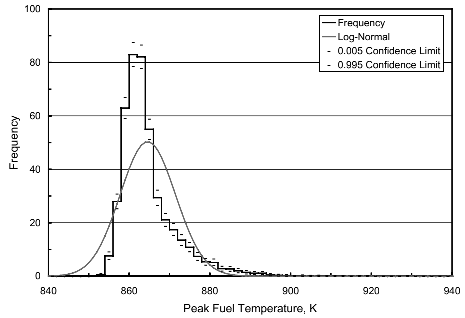

Fig. 1. Frequency distribution for the peak fuel temperature in an unprotected loss-of-flow transient.

Similar plots have been made for the peak fuel temperature for loss-of-heat-sink and transient overpower transients and for the peak coolant outlet temperature for each of the three considered transients. In some of these other cases, a log-normal distribution provides a better approximation to the observed frequency distribution, but in no case does the log-normal distribution fall within the 99% confidence limits over the entire range of temperatures.

Scatter plots of the peak fuel temperature against the various stochastic input parameters show that for the loss-of-heat-sink and loss-of-flow transients, the most important parameter is the core radial expansion feedback coefficient. For the transient overpower case, the peak fuel temperature depends most heavily on the total worth of the moving control rod and on the fuel axial expansion feedback coefficient.

The approach to uncertainty described provides for the estimation of probabilities for violating safety boundaries and should be useful in a risk based regulatory environment. It has the advantage of not requiring any substantial rewriting of existing the safety analyis computer codes. Future work should consider a larger set of input parameters and give more attention to appropriate probability distributions and possible correlations.

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Sodium void reactivity effect influence on the prospective fast neutron reactor safety and concept approaches

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After the Chernobyl NPP accident, Soviet Union regulations have been provided by zero integral sodium void reactivity effect (SVRE) requirement for the BN-800 fast neutron reactor which was under final design stage at this time period. To comply with the requirement, a new established reactor core concept was developed. The concept quintessence is provision of an above-core sodium cavity. Sodium density reduction or sodium non-availability at the cavity results in negative reactivity. It was stated out that sodium cavity specifications could be selected in such a manner that zero integral void reactivity will be guaranteed after boiling-off of whole fuel assembly sodium inventory. At next study stages, the designers started to implement the requirement for next nuclear reactor generations. The prospective high-power sodium-cooled fast neutron reactors were designed to implement the reactor cores featured by zero sodium void reactivity effect. Nevertheless, the idea resulted in reducing the reactor core height and some hindrances in optimizing the high-power reactor core because of considerable size increase of delivery plate, rotating plug, decrease of the reactor control rod efficiency etc. The SVRE problem is persistent for the beyond the design basis accidents when considerable amount of the reactor sodium inventory could be boiled out. Despite of many study runs on the design basis accidents of the fast neutron reactors (including the sodium-cavity-provided reactors as well), the designers have not acquired some enough evident requirements for specifying the positive SVRE value. Besides, beyond the design basis accident probability resulting in large-scale in-reactor sodium boiling off is very small one. Accident sequences which could result in the scenario for extant and future reactors, they should include more than ten failures for passive/active actuation type items.

The issue produces steadily some discussions on practicability of implementing the prospective reactor core concepts with zero or near to zero SVRE value. It should be mentioned as well, that valid NPP safety regulations do not require any mandatory compliance with the zero SVRE value. If the concept will be denied, several reactor core specifications would be considerably increased. To study these opportunities, three BN-1800 reactor type core model were analyzed for different type core height (85, 100, and 120 cm) and diameter and related SVRE value. The computational runs proved, the zero SVRE value requirement denial offers more enhanced high-power reactor core layout. Acquired data demonstrates that reactor core height up to 100 cm results in considerable (about as large as 2 times) decrease of fuel burnup reactivity margin and (about as large as 1.5 times) decrease of needed RCR drives. After all major parameters (excluding the SVRE value increased up to ~ 1% $\Delta K/K$), the reactor core layout is most favorable one.

ULOF type beyond the design basis accident consequences for the given reactor cores were at the same time investigated. It was stated, that only SVRE value reduction is not sufficient to ensure reactor self-protection under this accident occurrence. The reactor self-protection could be achieved when some additional features for enhancing negative reactivity effects due to in-core sodium heat-up are put into action. But, the features could be developed for a non-zero SVRE value reactor as well.

Therefore, practicability of this intentional reactor parameter deterioration to achieve targets which could be got by another way as well, without considerable technical and economic performance losses, is under question.

Finally, one could conclude: practicability of a fast reactor core concept with zero or non-zero SVRE could be proved by means of a large-scale technical and economic optimization studies. The concept denial offers wide opportunity to improve the NPP economic features; therefore, the opportunity should be posed under mandatory examination. The Paper considers and estimates major factors contributing to the optimization procedures as well.

Safety design requirements for safety systems and components of JSFR

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As a part of the conceptual design for Japan sodium-cooled fast reactor (JSFR), safety design requirements of JSFR is now being established in collaboration with JAEA, JAPC and MFBR. The basic safety approach is based on the development targets of a fast reactor cycle technology development (FaCT) project, i.e., the deterministic design approach based on the defense-in-depth philosophy, in which prevention and mitigation against BDBEs are considered as well as against DBEs, is supported by the probabilistic evaluations[1][2][3]. In order to embody a safety design, a higher level safety principle was broken down into a set of design requirements for each safety related system, structure and component (SSC). This paper will present an output of the safety requirements for safety related SSCs of JSFR.

In the course of this study, related safety principle and requirements for Monju, CRBRP, PRISM, SPX, LWRs, IAEA standards, development targets of FaCT project, goals of GIF and basic principle of INPRO etc. were taken into account to develop a next-generation global-standard safety requirements. It is stressed that preventive and mitigative measures against BDBEs shall be considered from early design stage. The attached figure shows outline of the safety requirements. The systems and components to be studied are divided: (1) core and fuel, (2) reactor vessel and its internal structures, (3) primary coolant system, (4) intermediate coolant system, (5) decay heat removal system, (6) reactor shutdown and reactivity control systems, (7) safety protection system, instrumentation and control systems, (8) electric power system, (9) fuel handling system, (10) steam and power conversion systems, (11) containment system and reactor building, (12) auxiliary systems. These requirements can also be categorized into some levels. Although the lower level expresses design requirements specific for JSFR, the higher level can be commonly applied to sodium cooled fast reactors.

This paper will describe outline of such safety requirements and also some detail of the requirements for some major safety related SSCs of JSFR. Here show some examples of the requirements.

As regards core and fuel, achievement of higher fuel burn-up and longer continuous operation period as well as fuel breeding is major design objective. Low decontaminated minor actinide content oxide fuel is used as driver fuel. The core and fuel shall be designed so as not to exceed acceptable fuel design limits, with the aid of associated safety functions such as reactor shutdown system, even under abnormal plant conditions. The core shall be designed to have the negative power reactivity coefficient throughout the fuel life time in the core. This means positive coolant temperature coefficient shall be compensated by some negative reactivity components such as Doppler coefficient. Mitigation of energy release due to core disruptive accidents (CDAs) shall be considered in the core and fuel design, namely, limitation of sodium void worth and installation of inner subassembly duct for molten fuel discharge are introduced.

The reactor vessel of JSFR has relatively small diameter and no nozzle structures for connection of pipelines. The hot leg and cold leg piping are inserted through the roof structure of the reactor vessel. The reactor vessel and its internal structures shall be designed to withstand thermal loads under postulated transients, seismic loads, neutron irradiation and so on. They shall be designed for in service inspection. Sodium leak detection shall be installed for early detection of postulated sodium leak events. As for coolant hydraulics, flow induced cavitation and gas entrainment shall be prevented. A plate structure for core debris suspension and cooling shall be installed at the bottom of the reactor structure as a design measure against CDAs.

Decay heat removal system shall be designed as passive operating system in order to achieve both system simplification and higher reliability. The decay heat removal system consists one Direct Reactor Auxiliary Cooling System (DRACS) and two Primary Reactor Auxiliary Cooling System (PRACS) in order to have diversity. They shall be designed to assure stable coolant flow rate. By means of opening the dampers in the air coolers, they start their operation. Therefore, it is important to get higher reliability of the dampers. Redundancy and/or diversity shall be taken into account in the design of dampers. Freezing and leakage of sodium shall be addressed in order to achieve higher reliability than in conventional designs.

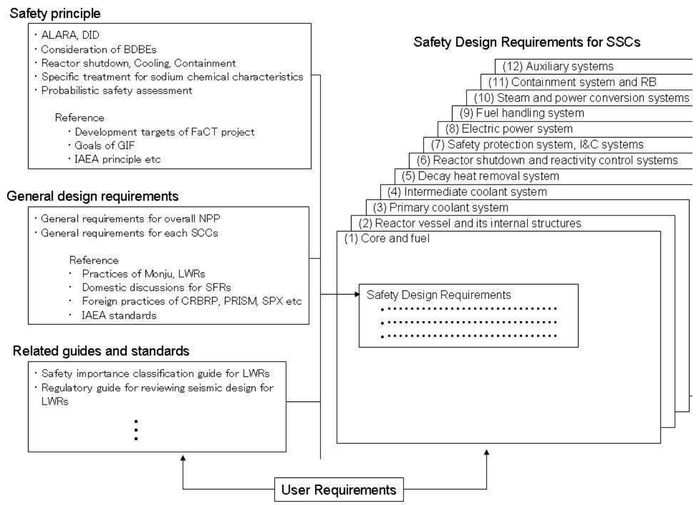

Figure ; Outline of the safety requirements for JSFR

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Parallel Session 4.1:

Fast reactor structural materials: achievements and new challenges

Advanced Materials for Nuclear Reactor Systems: Alloys by Design to Overcome Past Limitations

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Advanced materials have the potential to improve reactor performance via increased safety margins, design flexibility, and fast reactor economics and overcome traditional limitations. Increased strength and creep resistance can give greater design margins leading to improved safety margins, longer lifetimes, and higher operating temperatures, thus enabling greater flexibility. Improved mechanical performance may also help reduce the plant capital cost for new reactors both by reducing the required commodities (with concomitant reductions in welding, quality assurance and fabrication costs) and through design simplifications. However, successful implementation requires considerable development and licensing effort. Modern materials science tools such as computational thermodynamics and multiscale radiation damage computational models in conjunction with rapid science-guided experimental validation may offer the potential for a dramatic reduction in the time period to develop and qualify structural materials.

There are many requirements for all nuclear reactor structural materials, regardless of the exact design or purpose. All requirements for a materials use in an advanced fast reactor system must be considered and carefully weighed. These factors may include material availability and cost, ease of fabrication and joining, long-term stability, mechanical performance, thermal properties, neutronics, corrosion and compability performance, radiation tolerance, and code qualification status. Only through careful evaluation of all factors and a thorough trade analysis will the most promising candidate materials be chosen for further development. It is important to note that there is no ideal material that is best for each of the considerations listed. Indeed, all candidate materials have advantages and limitations. The most promising alloys, which allow the best performance, are also the least technically mature and will require the most substantial effort. These trade-offs must all be weighed carefully.

Recently, a program to provide advanced structural materials for fast reactor applications was initiated wihtin the United States. A thorough down-select process was conducted to weigh the requirements and benefits for all classes of structural materials. Four alloys were identified for further development. These include two ferritic-martensitic steels (NF616 and NF616 with special thermomechanical treatments) and two austenitic stainless steel alloys (high-temperature, ultrafine precipitation strengthened steel (HT-UPS) and NF709). NF616 is a 9Cr advanced ferritic/martensitic steel originally developed for super-critical boiler applications, while the HT-UPS alloys are 14Cr-16Ni austenitic stainless steels that were developed in the late 1980s by the U.S. Fusion Reactor Materials program for improved radiation-resistance [1]. Common ferritic-martensitic and austenitic stainless steels such as HT9 and 316, respectively, are traditional and proven materials for sodium fast reactors. However, the selected alloys offer considerable improvements in strength and creep resistance over these more mature steels and yet maintain other critical properties at the same level.

The superior performance and potential for improved reactor performance is illustrated for HT-UPS, D-9, and the traditional 316 SS in the figure below which shows the allowable operating regime in stress-temperature space[2-4]. (Alloy D9 is an advanced austenitic steel that was developed during the United States National Cladding and Duct Development program in the 1970s and 1980s). The maximum stress limit at 50 to 550ºC (423 to 823 K) is defined as 1/3 of the ultimate tensile strength, which is a more conservative design limit than 2/3 of the yield stress for stainless steel. The stress limit at higher temperatures is defined as 2/3 of the creep rupture strength at 10^5 hours. The HT-UPS

steel offers an additional 165 MPa over 316 SS at 500ºC. This increased strength at a constant temperature will allow reduced section sizes, and allow for longer lifetimes under stress, greater safety margins, or any combination of these. In addition, improved performance may also enable increased operating temperatures at a fixed stress. In the case of HT-UPS, an increase of ~150ºC is possible with no reduction in allowable stress, unless other reactor design limitations intervene.

Additional improvements in alloy performance are possible by using modern materials science techniques to further optimize properties and microstructures for specific conditions and components.

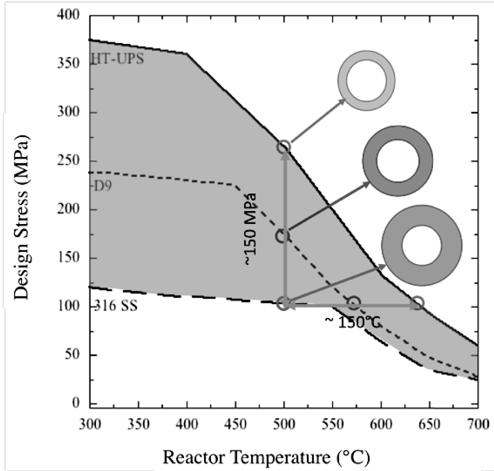

Figure 1: Temperature-design stress curves for 316 SS, D9, and HT-UPS steels. Higher strength can reduce the volume of materials required for components. (reproduced from [2], concept after [3-4])

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Pulsed E-beam modified FeCrAlY corrosion barriers for future fast reactor systems

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Compatibility problems of structural steel materials with the coolant are one of the key issues for the development of lead alloy fast reactors. A lead alloy cooled intermediate heat exchanger is investigated for future sodium cooled fast reactors to minimize the risk of sodium water interaction. Dissolution of metallic components into the lead alloy and extensive oxide scale growth are challenges that have to be met for both systems.

Oxide scales formed on steels act as diffusion barriers for cations and anions and offer therefore the best protection against dissolution attack. At higher temperatures, above 500°C, austenitic steels show severe dissolution attack, while thick oxide scales grow on ferritic martensitic (F/M) steels [1]. Thick oxide scales protect the steel from dissolution, but heat transfer, which is an important property for fuel pins is drastically reduced. A well understood measure is the alloying of strong oxide formers like aluminium (Al) into the surface that combine protection against dissolution with slow growing oxide scales. An Al content between 4 and 10 wt% is known to be sufficient for selective formation of alumina scales that grow only very slowly due to low diffusion coefficients for cations and anions [2]. In addition alumina is highly stable in lead and lead alloy environment also at higher temperatures.

Alloying Al into steel surfaces was done using the pulsed electron beam facility GESA. Pulsed electron beams (PEB) with an kinetic electron energy of 120 keV, a pulse duration of up to 50 µs and an energy density < 50 J/cm^2 are used to melt a thin Al layer (10 – 15 µm) deposited on the steel surface together with the outer part of the steel. The energy and duration of the electron beam can be adapted to achieve a melted depth of about 30µm. Due to the volumetric energy deposition materials melt adiabatically resulting in a high coolant rate of up to 10^7 K/s [3]. Such surface alloyed steels show there superior corrosion resistance in exposure tests at 650°C for 10000h's. However, the turbulent nature of the alloying process and the resulting reduced homogeneity of the Al distribution is one problem of this procedure. Therefore as a second step an Al containing alloy FeCrAlY with about 7 wt% Al was sprayed with a thickness of about 25 µm on the surface using low pressure plasma spraying. Using the GESA this layer was melted together with some µm of the steel and a perfect intermixing of the boundary layer was created. After the treatment the FeCrAlY layer is densified, smoothened, perfectly bonded to the substrate and the Al content is reduced to about 4wt% in average.

Such surface modified FeCrAlY layers on T91 steel were tested in flowing lead bismuth eutecit (LBE) at 480, 550 and 600°C. At 600°C thin alumina oxide scales develop and protect the steel from dissolution. At lower temperatures, however, at some places thick oxide scales like on T91 original steel develop. Detailed examination showed that the Al content at such places dropped due to spraying problems and the PEB treatment to values < 3 wt%. Therefore, the Al content in the sprayed powder was increased to about 11 wt% resulting in a

final composition after PEB treatment having 5 to 7 wt% Al. First tests show the optimized formation of thin alumina scales especially at lower temperatures.

Beside corrosion properties such surface barriers should not show any negative influence on the mechanical properties of the substrate. To evaluate the influence LCF tests in air and LBE, pressurerized tubes and creep tests are performed. No degradation of LCF properties due to the surface modification process and due to the LBE was found [4]. Pressurerized tubes of T91 original steel showed an increased oxidation rate under internal pressure. The surface modified tubes show the same thin alumina scales also observed without internal pressure. F /M steels showed in creep tests strong influence of the lead alloy on creep rate and rupture times (Fig. 1). The creep rate is up to 50 times higher in LBE compared to air. Formation of cracks in the oxide scale is the starting point for the reduction in creep strength. First results obtained with surface modified specimens show comparable behaviour in air and LBE (Fig. 1) and seems to confirm that the formation of thin alumina scales is a favourable corrosion protection barrier.

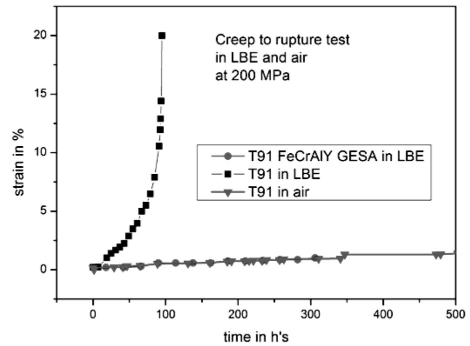

FIG.1. Strain in % vs time of T91 with and with out surface modified layer measured in creep to rupture tests in LBE and air.

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EP-450 Steel as Cladding Material for Fuel Rods for Fast Neutron Reactors

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Russian experience in operating sodium-cooled fast neutron reactors bears evidence of difficulties in achieving cost-effective burn-up of nuclear fuel of ~ 18-20% h.a., at which structural materials must withstand the damage doses of 160-180 dpa. The problem is in a weak resistance to radiation swelling of austenitic steels used as a cladding material for fuel rods. In Russia, extensive data on a high resistance of 12%-chromium steels to swelling [1] were obtained after irradiation of specimens in dozens of assemblies in the BOR-60 reactor up to damage doses higher than 130 dpa. The interest in ferritic-martensitic steels is strengthened by a number of advantages offered by their physical properties and indisposition for high-temperature radiation embrittlement with inter-grain destruction during creep.

The EP-450 ferritic-martensitic steel (13Cr-2Mo-Nb-V-B) is the most studied, well-developed and widely used one in Russian fast neutron reactors [2]. Performance of hexagonal FA wrappers was experimentally verified at damage doses of up to 160 dpa in the BOR-60 reactor and 90 dpa in the BN-600 reactor. Experience of using the EP-450 steel as a cladding material for fuel rods is less known, but is of great interest from the viewpoint of prospective improvement of chromium steels for the BN-type reactors.

The paper focuses on and summarizes the main results of post-irradiation examinations of 13 FAs with EP-450 steel claddings irradiated in fast neutron reactors. The maximum burn-up achieved during these tests made up 30% h.a. at a damage dose of 163 dpa. The results of the examinations formed the basis for the main conclusions.

Conclusions

1. The EP-450 steel as a cladding material for fuel rods for fast neutron reactors retains its high resistance to radiation swelling at damage doses of up to 163 dpa.
2. The main factors limiting reliable performance of fuel rods with EP-450 steel claddings are the maximum initial temperature of the claddings and the level of stresses in them caused by the gas and fuel pressure.
3. With a decrease of the maximum temperature down to 650°C the problems associated with high-temperature strength and corrosion damage via the iodine transport reaction mechanism can be overcome.
4. The use of the EP-450 steel as a cladding material in the commercial BN-600 reactor is impossible with the current technology for SFA storage.

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Development of SFR Fuel Cladding Tube Materials

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To increase the creep rupture strength of the cladding materials (9Cr-2W steel), new alloys were designed. Minor elements such as V, Nb, and Ta were added into the new alloys. The new alloys were prepared by a vacuum induction melting process.The alloys were hot rolled after a preheating at 1150^{o}C for two hours.The hot rolled plates were normalized at 1050 ^{o}C for two hours followed by air cooling and tempered at 750 ^{o}C for one hours followed by an air cooling.

The tensile tests were performed from room temperature to 700 ^{o}C at a strain rate of $2x10^{-3}$/sec. The creep tests were carried out at 650 ^{o}C under the constant load condition. The applied load changed from 110 MPa to 150MPa. The phase in the new alloys as a function of the temperature were calculated by using a ThermoCalc.The microstructures were observed by using a transmission electron microscope (TEM) and the elemental analyses on the particles were made by using an energy dispersive spectroscope (EDS) attached to a TEM.

The tensile properties of the alloys were evaluated by the uniaxial tension test. The V content was changed to investigate the effect of V on the mechanical properties of Ferritic/Martensitic (FM) steel. The 0.1V alloy showed higher yield strength than the 0.3V alloy regardless of the tensile test temperature. Tensile strenght also showed a similar tendency to the yield strength. But the elongation ot the 0.1V alloy was lower than that of the 0.3V alloy. The mass fraction of the V-rich MX particles may increase with the increase of the vanadium content. But the increase of vanadium content had no beneficial effect on the yield and tensile strength of the alloys.

The Ti was also added to form the stable Ti-rich carbide. The formation of Ti-carbide has a good effect on the mechanical properties of FM steel.[1] And the Ta had a similar effect with Nb, so Ta was added to know the effect of Ta on the mechanical propterties of FM steel. Tensile test results showed that the yield and tensile strength did not increased by the addition of Ti or Ta. Ta bearing alloy showed the lowest yield and tensile strength.

The 0.1V alloy had a higher creep ruputre strength than the 0.3V alloy. It is known that needle type V-rich MX particles have good effect on creep rupture strength of FM steel.[1] So vanadium content of FM steel was increased to form more V-rich MX particles. Thermocalc calculation results showed that the mass fraction of V-rich MX particles increased in high vanadium steel. But the increase of the V-rich MX particles had no good effect on the creep rupture strength of the FM steel. But further investigation will be continued to confirm the effect of V on the creep rupture strength.

Fig. 1 shows the creep rupture strength of Ti or Ta bearing FM steels. The creep rupture strength of FM steels did not increased by the addition of Ti or Ta. Ti or Ti addition had no good effect on the reep rupture strength of FM steels. The reference alloy showed a higher creep rupture strength than the Ti bearing and Ta bearing steels.

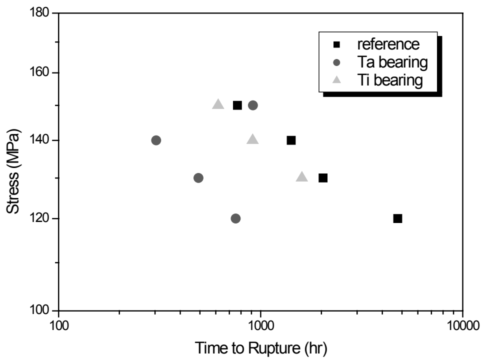

FIG.1 Creep rupture strength of new cladding materials

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Parallel Session 4.2:

Fast reactor structural materials: achievements and new challenges

Experimental Investigation of Strain Concentration Evaluation Based on the Stress Redistribution Locus Method

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High temperature fatigue tests of circumferentially notched specimens were conducted accompanying with local strain measurement by a capacitance type strain gage in order to investigate the applicability of the stress redistribution locus method to strain concentration evaluation.

Evaluation of local strain at structural discontinuities is an important technology in high temperature design rules of fast reactors because high temperature fatigue or creep fatigue damage usually initiates as cracks at such a locally high strained area. The Neuber's rule [1] is usually used in design codes such as ASME Boiler and Pressure Vessel Code, RCC-MR and so on. It has been pointed out that the Neuber's rule gives a conservative evaluation of strain concentration especially high stress conditions. The stress redistribution locus (SRL) method [2,3] had been proposed to improve the accuracy of local stress and strain evaluation for discontinuities. This study aims to verify the applicability of the method experimentally in order to improve and rationalize creep-fatigue damage evaluation technologies in the high temperature design rule of fast reactors.

Fatigue tests at room temperature and 550°C and creep-fatigue tests at 650°C were conducted in this study. Circumferentially notched specimens of which the elastic stress concentration factors, Kt, were 1.69 and 3.39 were employed. The material employed is 316FR steel. Strains at the notch root were measured by a capacitance type strain gage of which gage length is 1.0 mm. This gage can be used in high temperature, 700°C at the maximum.

The locus of stress redistribution for a structure can be obtained from elastic-creep analysis of it as the stress and strain behavior during dwell, and it has been reported that the locus depends on only structure so that the creep law and magnitude of loadings has little effects on it [2]. Inelastic analyses also conducted in order to obtain the loci of the specimens in this study and the inelastic stress-strain state is estimated as the intersection of the locus and inelastic stress-strain curve. Figure 1 shows an example of loci and a comparison between estimated and measured strain. Measured strains at the first cycle and middle life are compared with estimations by using monotonic and cyclic stress-strain, respectively. The bold line is the SRL master curve proposed in the previous study [2,3] and formulated as follows;

$$\tilde{\varepsilon} = \frac{1}{\kappa}\left(\frac{1}{\tilde{\sigma}} + (\kappa - 1)\tilde{\sigma}\right)$$

where $\tilde{\varepsilon}$ is the ratio of strain concentration factor, K_ε, to K_t, and $\tilde{\sigma}$ is the ratio of stress concentration factor, K_σ, to K_t. The coefficient κ is usually from 1.4 to 1.6, and this formula

corresponds to the Neuber's rule when κ is equal to 1.0. SRL method gives a small strain in comparison with the Neuber's rule with keeping conservatisms to the measures strain. The master curve also gives a good estimation for measured strain at middle life, though a too conservative estimation is obtained for the first cycle. Therefore, SRL method would improve the accuracy of strain concentration evaluation in comparison with usual methods such as the Neuber's rule.

The applicability of SRL method to relaxation behavior in creep-fatigue condition is also discussed in this study.

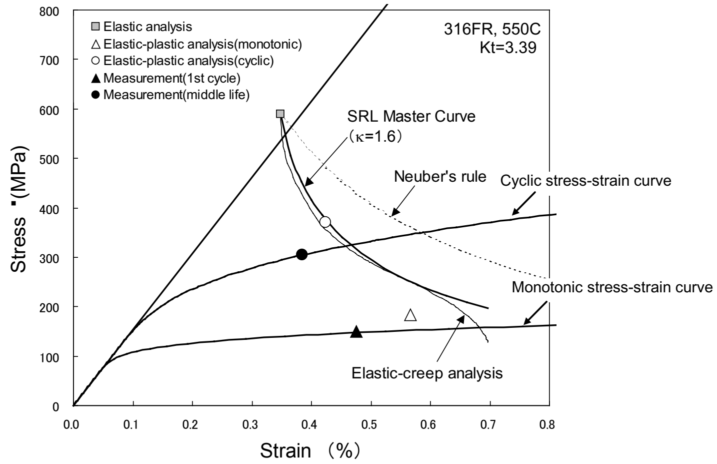

FIG. 1. Comparison of stress redistribution loci and estimated strains by SRL method with experimental results.

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Microstructural effect of solute addition for Fe-15Cr-20Ni steels irradiated in Joyo

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The relationship between the addition of minor element and the swelling behavior was investigated using nine heats of Fe-15Cr-20Ni austenitic model alloys, Fe-15Cr-xNi(x = 20, 25 and 30) and a PNC1520 steel irradiated at 480, 570, 620 and 700°C in the Joyo to doses of 20-56 dpa. Phosphorus and boron suppressed the appearance of swelling for these experimental irradiation conditions. Additions of titanium and niobium were effective for suppression of swelling at low temperature, below 570°C. Complicated microstructural dependence of nickel contents for Fe-15Cr-xNi alloys could be seen.

It is required that the core materials for fast breeder reactors (FBR) have good mechanical properties at high temperature and high dimensional stability against huge exposure to fast neutrons. Austenitic stainless steels have superior high temperature strength, but less swelling resistance compared with ferritic steels or martensitic stainless steels. The second-generation advanced austenitic steel for near-term application for FBR has been selected as Fe-15Cr-20Ni-0.25Ti-0.1Nb, designated PNC1520 by the Japan Nuclear Cycle Development Institute (JNC). In PNC1520, void swelling resistance and high temperature creep properties were improved by adjusting the amount of cold working and the minor elements such as titanium, niobium, phosphorous, boron. In order to make clear what element is effective in improving the stability of stainless steels, such as irradiation hardening, creep and void swelling under neutron irradiation, it is important to investigate the microstructural evolution of Fe-Cr-Ni alloys after neutron irradiation using model alloys designed to extract the effect of minor element addition. In the previous work, it was shown that the suppression of void swelling behavior could be seen for the PNC1520 alloys irradiated in Joyo and found that the radiation-induced precipitation, especially phosphide was main contribution for void swelling suppression at high temperature during neutron irradiation [1].

In this study, we used a set of Fe-15Cr-20Ni-M (M=B,C,Ti,Nb,P,Si and Mo) alloys to describe the effect of minor element addition on swelling behavior of Fe-15Cr-20Ni base alloys.

Nine different 15Cr20Ni-austenitic alloys modified by various minor solute element

additions , three different nickel contents of 15Cr-xNi-austenitic alloys and a PNC1520 steel were prepared. The amounts of minor element additions in the modified alloys were based on the content of minor element in PNC1520 steels. The specimen form was that of a 3 mm diameter, 0.2 mm thick TEM disk annealed at 1050°C for 2 or 3 min in a vacuum. Irradiations were conducted in Joyo at 480, 570, 620 and 700°C to doses of 20-56 dpa. Microstructural analysis was performed using a JEOL-2010. Precipitate identification was performed by combined information obtained from diffraction patterns, energy dispersive X-ray analysis (EDX) and two-dimensional ASID-EDX chemical analysis.

From microstructural analyses of Fe-15Cr-20Ni-M model alloys, two interesting features were obtained: (1) Si and Ti aggregation around the Frank loops and (2) suppression of void swelling in 0.004B-added model alloy at high temperature.

It is well known that the silicon, titanium and phosphorous additions can affect loop/network dislocation behaviors and that the silicon, titanium and niobium additions appear to enhance Frank loop formation in SA alloys. Aggregation of Si and Ti atoms inside the Frank loops and enrichment of Ni atoms around them were observed clearly. However, no Frank loop was observed in 0.1Nb-added alloys in any condition in this work, and large Frank loops were observed in 0.25Ti-added model alloys irradiated below 570°C. That is, titanium plays an important role for nucleation and growth of Frank loops. The void swelling in the 0.004B-added alloy during neutron irradiation was suppressed at high temperatures. However it is interesting that no precipitate, Frank loop nor coarsened dislocation network could be seen in the void-free specimens. A few faint traces of boron carbides were remained in the grain, but all of boron carbides were dissolved.

In the ternary alloy of Fe-15Cr-xNi alloy, the amount of swelling for the Fe-15Cr-25Ni alloys was the lowest among them irradiated in the range of 450 to 650°C up to 40 dpa. The void formation was suppressed following by reduction of dislocation density in the ternary alloys. The same tendency was also reported in the previous work [2], but the reason why 25% of nickel content for Fe-Cr-xNi ternary alloys irradiated at high temperature affects the reduction of void and dislocation formation is not still clear. Detailed information and discussion will be presented in the conference.

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Materials and code qualification needs for sodium-cooled fast reactors

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This paper summarizes the findings from the assessments of current status and future needs in code qualification and licensing of reference structural materials and new advanced alloys for sodium cooled fast reactors. Advanced materials are a critical element in the development of sodium reactor technologies. Enhanced materials performance not only improves safety margins and provides design flexibility, but also is essential for the economics of future advanced sodium reactors. Code qualification and licensing of advanced materials are prominent needs for developing and implementing advanced sodium reactor technologies. Nuclear structural component designs in the U.S. comply with the ASME Boiler and Pressure Vessel Code Section III (Rules for Construction of Nuclear Facility Components) and licensing is granted by the USNRC. As the liquid metal reactor (LMR) will operate at higher temperatures than the current light water reactors (LWRs), the design of elevated-temperature components must comply with ASME Subsection NH (Class 1 Components in Elevated Temperature Service). In the past licensing review for the Clinch River Breeder Reactor Project (CRBRP) and the Power Reactor Innovative Small Module (PRISM), the NRC/Advisory Committee on Reactor Safeguards (ACRS) raised numerous safety-related issues regarding elevated-temperature structural integrity criteria. Most of these issues remained unresolved today. These critical licensing reviews provide a basis for the evaluation of underlying technical issues for future advanced sodium-cooled reactors.

Major materials performance issues and high temperature design methodology issues pertinent to the LMR are addressed in this paper. The paper uses the ANL-developed reference design information and design concepts proposed by the four industrial consortia as a basis in the assessment of the major code qualification and licensing issues for the structural materials. The available database is discussed for the ASME Code-qualified structural alloys (e.g. 304, 316 stainless steels, 2.25Cr-1Mo, and mod.9Cr-1Mo), such as creep, fatigue, creep-fatigue interaction, microstructural stability during long-term thermal aging, and material degradation in sodium environments. An assessment of modified versions of Type 316 SS, i.e. Type 316LN and its Japanese version, 316FR, is also made to provide a perspective for codification of 316LN or 316FR in Subsection NH. Current status and data availability of four new advanced alloys, i.e. NF616, NF616+TMT, NF709, and HT-UPS, are also addressed to identify the R&D needs for their code qualification for LMR applications. For both conventional and new alloys, issues related to high temperature design methodology are described to address the needs for improvements for the design and licensing. Assessments have shown that there are significant data gaps for the full qualification and licensing of the structural materials.

Review of the existing database for Subsection NH materials has led to the conclusion that there is an extensive database and wide industrial experience with conventional austenitic stainless steels and ferritic steels. Though unique issues must be addressed for each

individual material, there are common challenges facing all these alloys under LMR operating conditions. The major issues for consideration are:

- *Verification of allowable stresses for the 60-year design life*
- *Long-term thermal aging effects*
- *Environmental Effects, in particular,* degradation of the reactor vessel when exposed to a flowing sodium environment due to transfer of carbon and nitrogen.

The advanced alloy development involves modification of existing alloys through minor compositional changes and thermo-mechanical treatments for improved high temperature performance. The conventional processes available for alloy development have been historically slow, taking decades to bring a new alloy to commercial market. However, given the computational tools currently available, a science-based approach for alloy improvement offers an opportunity to accelerate the alloy development process. Using computational thermodynamics and kinetic models, alloy composition and heat treatment can be optimized before alloys are even melted. Additionally, alloy compositions and treatments can be custom tailored for specific applications, which would minimize the number of trial heats required for optimization of the composition and also minimizes the tests needed to evaluate the properties. Moreover, such custom tailoring can be focused on the development of nanoscale features that are vital for improving the creep properties and irradiation resistance of the structural materials. Relative to materials that are ASME Code qualified, these techniques may allow for small modifications to approved alloys that may result in improved performance. Most probably there will be no need either for additional procedures for the Code qualification or for modifications to design methodology for the application of the new alloys. However, even if the original alloy (i.e., before modification) has sufficient database to address all qualification and licensing issues, the newly developed alloys (even with minor modification) have to undergo a detailed evaluation as if they are completely new alloys.

Four advanced alloys have been selected for further development in support of LMRs. These alloys are advanced ferritic/martensitic steels, NF616, NF616+TMT, and advanced austenitic stainless steels, NF709 and HT-UPS. There has been increased usage of advanced austenitic alloys and high-Cr ferritic steels in conventional power plants. Unfortunately, the understanding of long-term issues, such as creep, creep-fatigue and environmental effects are poorly understood. The neutron irradiation data of these advanced alloys are very limited, and the mechanical properties data in sodium environments are nearly none. Significant R&D and testing are needed for the reactor design and alloy qualification. Development and qualification of new materials for use in Section III Subsection NH presents a significant challenge to the design.

Analysis of the optimization of the secondary hot piping for a sodium fast reactor

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Mod. 9Cr-1Mo steel (T91) is a candidate material for sodium fast reactor (SFR) in particular for secondary piping. Its good conductivity, low thermal expansion regarding austenitic stainless steel used in the past reactor let the possibility to reduce the size of the loops. In order to validate this choice, it is necessary, firstly to verify that it is able to withstand the planned environmental and operating conditions, and secondly to check its supply, fabrication, welding and if it is covered by the existing design codes concerning its examination methods and mechanical design rules. A large R&D program on mod. 9Cr-1Mo steel has been undertaken in France, in order to characterize the behavior of this material and of its welded junctions in operating conditions of the Sodium Fast Reactor. In this program, the piping for the next sodium fast reactor SFR is optimized in order to minimize the size of the loop and so reduce the cost. In this way, two analysis on secondary hot piping design have been carried out with a stainless steel 316L(N) (used for the previous Sodium Fast Reactors Phénix and Super Phénix), and a mod. 9Cr-1Mo steel:

- The first analysis deals with the secondary hot piping of the Phénix Reactor. In this case the maximal temperature considered is 550°C.

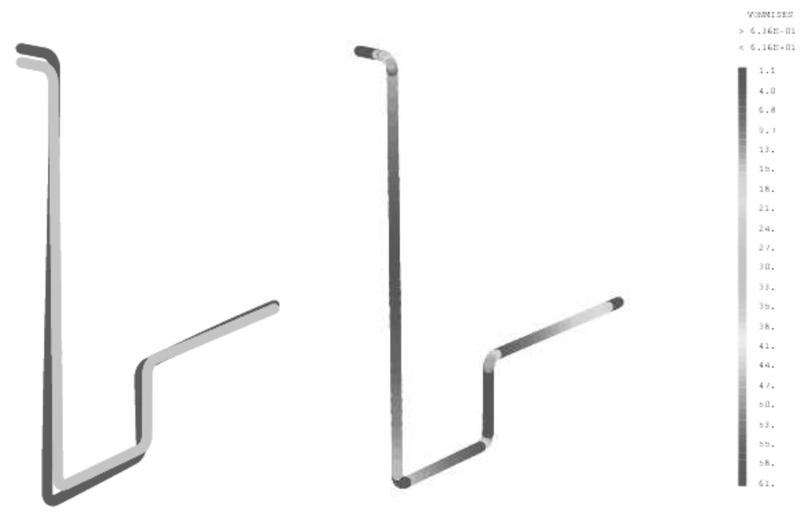

FIG. 1. Piping line of EFR – deformed piping line (with an amplification coefficient 10) and reference stress field under temperature range and displacements of components between 525°C and 20°C. CAST3M modelling for mod 9Cr-1Mo

- The second analysis deals with the secondary piping considered for the European Fast Reactor (EFR). In this case the maximal temperature considered is 525°C.

For both cases, design has been made for realistic operating conditions of EFR and Phénix for a period of 60 years. The analysis is based on the creep-fatigue damage and the application of the RCC-MR rules [1].

These two studies show that use of mod. 9Cr-1Mo steel has generally an advantage for moderate temperature (below 525°C). On the contrary, when the temperature is more important stainless steel 316L(N) presents lower damage than 9Cr steel. Indeed, thanks to advantageous thermal properties of mod 9Cr-1Mo steel, the stress state due to mechanical and thermal loading for this material is 20 to 30% lower than this of 316L(N) stainless steel. But for high temperature this benefit is too low to compensate the lower creep properties of 9Cr steel.

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Structural materials for russian fast reactor cores. Status and prospects.

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The energy strategy of Russia in the period up to 2020 contemplates a gradual introduction of a new nuclear energy technology based on the fast breeder reactors with the closed MOX fuel cycle. Further developments of nuclear power will demand inclusion of fast breeder reactors into the structure of NPPs.

Since 1980 in Russia at Beloyarsk NPP the only in the word commercial fast breeder reactor BN-600 is in operation. According to the plans the fourth power unit at Beloyarsk NPP with the fast breeder reactor BN-800 shall be put into operation in 2012. Under developments is a commercial sodium cooled fast breeder BN-1800.

The use of steel EP450 (12Cr13Mo2NbVB) as shrouds of FAs (96x2 mm) and cold worked steel ChS68 (06Cr16Ni15Mo2Mn2TiVB) as fuel claddings (6,9x0,4 mm) reliably ensured the fail-free operation of BN-600 reactor at the burnup of 11.2 % h.a. and the damage dose of 82 dpa. There is every reason to assume that the EP450 steel shrouds will not limit to reaching a higher fuel burnup.

Currently, for the BN-type reactors as promising structural materials for a staged increase in the fuel burn-up under consideration are austenitic and martensitic steels including those produced by the powder metallurgy method (ODS steels).

The main cause that restricts the burn-up of fuel clad in austenitic steels is their considerable swelling. This fact in its turn is responsible for the degradation of cladding short-time and long-time mechanical properties.

Consideration has been given to the principles of complex alloying and treatment of austenitic steels that make low swelling feasible at the irradiation doses of ~ 100 dpa. Currently experiments are under way in BN-600 reactor to validate the serviceability of austenitic steels as claddings: ChS68 steel up to ~ 90 dpa, EK164 steel (07Cr16Ni19Mo2Mn2TiVB) up to ~ 100 dpa.
As a cladding material that provides for the fuel rod operation to the damage doses of ~140 dpa under consideration are high temperature strength complex alloyed 12 % Cr steels EK181 (16Cr12W2VTaB) and ChS139 (20Cr12NiWNbVNB). Above all those steels differ from EP450 steel by extra alloying with carbon, tungsten and a little lower content of chromium. This alloying provides for the stability of the strengthening phases, the resistance to recrystallization processes as well as increases the high temperature strength characteristics in comparison to those of EP450 steel. BOR-60 irradiation tests and investigations of EK181 steel are in progress. Early in 2010 an experiment is to be carried on in the BN-600 reactor to irradiate two materiologic FAs (MFAs) having specimens of EK181 and ChS139 steels and of their modifications to reach the maximal damage dose of ~140 dpa.

With the aim of attaining the damage dose of ~180 dpa at JSC VNIINM dispersion strengthened 12% Cr steels (ODS steels) are under development using powder metallurgy method.

In progress are mastering the fabrication technology and complex out-of-pile investigations of fuel rod tubes from ferritic (based on EP450 steel) and martensitic (based on EK181 steel) ODS steels and their welded joints. The strainability of EP450-ODS steel was estimated. It is shown that the steel as cold worked to 60 % is capable of an adequately large elongation. The first experimental batch of thin-walled tubes 6.9×0.4mm in size was received from steel EP450-ODS at Institute of Inorganic Materials. The electron microscope examinations have established that the tube structure consist of ferritic grains within which uniformly distributed titanium-yttrium oxides of 7 nm in the average size are observable. Preparations are under way for BN-600 reactor testing materials of this class within two MFAs up to the maximal damage dose of ~ 140 dpa.

Development of Structural Materials for JSFR – Overview and Current Status

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This paper describes an overview and current status of the development of structural materials and core materials for Japanese Sodium Fast Reactors which is being conducted within the framework of the Fast Reactor Cycle Technology Development (FaCT) Project. Employing new materials is one of the most important key technologies that are necessary to achieve required level of safety, reliability and economic competitiveness of the Japanese Sodium Fast Reactors (JSFR). The new structural materials to be adopted in JSFR components are 316FR (Fast Breeder Reactor Grade Type 316 Stainless Steel) and Modified 9Cr-1Mo steel. The new core materials are oxide dispersed strengthened (ODS) and precipitation hardened (PH) ferritic steels. The development of these materials including the aspects such as the needs from plant design, material design, data acquisition, fabrication of materials of specific configurations planned for JSFR will be overviewed. Current status and path forward are also described.

316FR was developed in Japan by optimizing chemical composition within the specifications of SUS316 in the Japanese Industrial Standard. The optimization was performed from the viewpoint of maximizing the creep strength.

Modified 9Cr-1Mo steel was originally developed for the liquid metal fast breeder reactors by Oak Ridge National Laboratory (ORNL), US in 1970s. The steel has excellent heat transfer coefficient and low thermal expansion rate. In addition, the steel also has good high temperature strength, so that it is adopted for the material of primary and secondary coolant circuits, intermediate heat exchangers and steam generators. The steel has already been standardized in ASME boiler and pressure vessel code as Grade 91 steel and has a number of experiences of practical use in thermal power boilers, however it has not been used for nuclear power plants.

The design life of JSFR is 60 years. The verification of extrapolated creep strength must be performed by experimental results and by metallurgical approach to establish the material strength standard applicable for 60 year design. The following three measures are necessary and effective to materialize the 60 year design: 1) To collect extra-long term creep rupture data: The creep rupture tests longer than 200,000 hours will be conducted. 2) To demonstrate the validity of temperature-accelerated test by metallurgical approach: The metallurgical examination and quantitative analyses will be performed focusing on deformation mechanism, precipitation behavior and so on. 3) To develop monitoring techniques to confirm the safety margin: Techniques of surveillance or nondestructive examination will be developed to detect the invisible material degradation.

It is also important to estimate the creep strength of welded joints. Especially for modified 9Cr-1Mo steel, it is necessary to take "Type-IV" cracking into account. "Type-IV" cracking is generally

observed in the high temperature creep failure in the welded joints made of ferritic heat resistant steels. As far as the available creep data shows, the obvious creep strength degradation due to "Type-IV" cracking has not been observed in the welded joints of modified 9Cr-1Mo steel at 550C. However, there are not enough experimental data to demonstrate that degradation will not be tangible in SFR components for 60 years. Therefore, the creep strength of welded joints at 550C must be estimated based on temperature-accelerated test results taking appropriate conservativeness into account.

Regarding 316FR and Modified 9Cr-1Mo steel, elevated temperature materials strength standard (MSS) will be developed and will be incorporated in the elevated temperature design code published from the Japan Society of Mechanical Engineers (JSME).

ODS and PH ferritic steels are eligible for the core component materials to endure heavy displacement damages (250 dpa) and high burnup (250 GWd/t) for about nine years in commercialized sodium cooled fast breeder reactor (SFR) cores. Ferritic phase (bcc) is essential to prevent from void swelling, which emerges in austenitic phase (fcc) obviously over 100 dpa. In the Fast reactor Cycle Technology development (FaCT) project, we will develop large scale manufacturing technology for both ODS steel cladding and PNC-FMS duct tubes toward future mass production, and prove the tentative material strength standards by out-of-pile tests and a series of irradiation tests.

Nanometer size complex oxide dispersoids in matrix provide excellent dispersion hardening for ODS steels. Mechanical alloying and hot consolidation processes dominate the dispersoid morphology and resultant mechanical properties. It is necessary to control process parameters relevant to dispersoids. To establish MSS for fuel pin mechanical design, hundreds of specimens have been extensively examined for mechanical properties in air and stagnant sodium environments, and irradiation-tested in the experimental fast reactor JOYO to investigate irradiation effect on mechanical properties. In-pile creep rupture tests using pressurized tube specimens have been also carried out by the Material Testing Rig with Temperature Control (MARICO) in the JOYO. Both 9Cr- and 12Cr-ODS steel cladding tubes were assembled into fuel pins and have been irradiated since 2003 under a collaborative program between JAEA and Research Institute of Atomic Reactors in the BOR-60 in Russia.

PH ferritic steels such as PNC-FMS can be applicable for duct tubes and wrapping wire. Core support structures of the SFRs will be made from austenitic stainless steels, and therefore entrance nozzle of a fuel subassembly should be the austenitic. Design studies on the SFR cores in the FS consider that a PNC-FMS duct tube will be joined with SUS316 duct tubes at lower and upper ends. There are two methods to join; one is a mechanical joint with screw and the other is a dissimilar welding. We have selected the hexagonal tube-to-tube method with EB welding as the most promising. The tentative MSS for PNC-FMS was established by 1993, and irradiation tests in the JOYO and FFTF have been conducted to prove its validity.

Parallel Session 5.1:

Fast reactor fuel cycles

Strategies and National Programs of Closed Fuel Cycles: Russian Vision

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Nowadays, a transition to a full-scale closure of nuclear fuel cycle (NFC) is becoming an actual and key task for any long-term scenarios and nuclear power engineering development strategies.

Only successful solution of these tasks and effective industrial implementation of closed NFC are acceptable from the public and ecological viewpoint allows the transition to advanced development of nuclear power engineering both on a national and regional scale and at the world level.

It allows the transition of nuclear power engineering from accompanying and alternative component of the fuel energy balance to a primary and basic status, i.e. it makes nuclear power engineering a base for the world power engineering free of greenhouse gases and provided with resources available for millenniums (U-238 and in the future - thorium).

The report includes an expert review and analysis of strategies and national programs of leading nuclear countries (France, Japan, USA, China and India) compared to the Russian plans and programs for the period up to 2050 on the basis of the public press materials. It covers the following key positions and directions of the closed NFC development:

- Start-up dates, scope and objectives of thermal reactor SNF reprocessing
- Closed NFC technologies
- Dates of putting into operation of pilot industrial fast reactors
- Fast reactor fuel cycles
- Plans and dates of fast reactor building start-up
- Transition to high-density types of fast reactor fuel
- Utilization of Pu-bearing fuel in thermal reactors
- Nuclear Fuel Cycle closure on MAs
- RAW management strategy
- National policy, international cooperation and competition in the market of NFC services

In Russia, main system problems of large-scale nuclear power engineering relate to high and continuously growing volumes of SNF and a limited raw material base of the existing nuclear power engineering with thermal neutron reactors.

To solve these problems, Federal Target Program “Nuclear Energy Technologies of New Generation” was adopted in summer 2009.

Within the framework of the program, transition to a new technological platform (NTP) of the Russian nuclear power industry is to be provided. The NTP is based on the transition to U and Pu-closed NFC with Gen IV fast neutron reactors.

Developed on this base, innovative nuclear power engineering shall fulfill the Russian demands for energy resources for a historically observable time period, simultaneously solving the problem concerning a reuse of earlier accumulated SNF.

Figure 1 presents areas of investigations to provide a transition to NTP of the Russian nuclear power engineering.

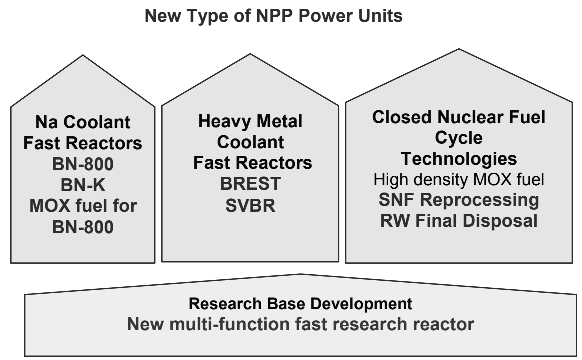

Fig.1 Areas of investigations to provide a transition to NTP of the Russian nuclear power engineering

Development of FBR Fuel Cycle Technology in Japan

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Japan Atomic Energy Agency (JAEA) has been conducting the Fast Reactor Cycle Technology Development (FaCT) project. In the FaCT project, the integration of the sodium-cooled fast reactor with oxide fuel, the advanced aqueous reprocessing and the simplified pelletizing fuel fabrication was selected as the main concept, because it was the most promising concept for commercialization. [1][2]

Status of R&D for FBR fuel cycle technology

The reprocessing concept is constructed based on the well established aqueous reprocessing. Some innovative technologies are adopted for the aqueous reprocessing in order to realize MA recycle as well as economical competitiveness. U/Pu separation technology of the NEXT system has some options according to the specification of reprocessing fuels and products as shown in **Fig. 1**. These processes are expected to be more efficient in costs, wastes management, and the nuclear non-proliferation. Main task of the NEXT process is to develop the equipments in engineering-scale with high reliability, criticality safety, high durability and remote maintainability. On the other hand, for newly applied processes such as U crystallization and extraction chromatography for MAs, there is a wide range of R&D tasks from the basic chemistry to the development of the engineering-scale equipment. In the FaCT project, six items have been identified as the main issues to be developed corresponding to each process step.

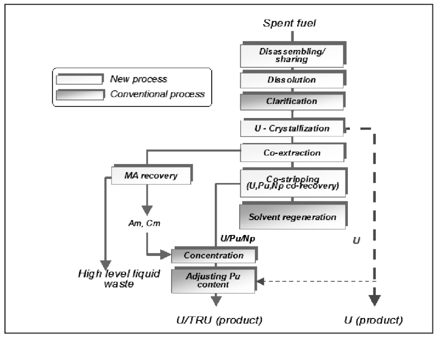

Fig.1 Process flow of NEXT process

Table 1 Development issues for fuel cycle technologies

[Advanced Aqueous Reprocessing]
Disassembling and Shearing
Dissolution
Uranium Crystallization
U,Pu and Np Co-extraction
MA Recovery
Salt Free waste Treatment
[Simplified Fuel Fabrication]
Conversion and Granulation
Pelletizing
Sintering
Studies of Fuel Physical Properties
In-cell Remote Handling Technology
TRU Fuel Handling Technology

In the simplified palletizing method process, innovative technologies are adopted to rationalize pellet fabrication process. The plutonium content adjusting is performed by solution mixing of Pu and U nitrate. Therefore, lots of powder treatment process can be eliminated. "Binder-less granulation process " and "die wall lubrication pressing " bring elimination of processes of binder powder mixing , de-waxing and de-gassing. However, adoption of simplified extraction process and MA recovery process allows the reprocessing products, the source material for fuel fabrication, to contain some

amount of FP and MA. Additional development issues were settled because high heat generation caused by decay heat of MA and high radio-activity of the fuel. Six main development issues are identified both for essential issues of the simplified pelletizing method process and for additional issues from MA and FP bearing.

Table 1 shows main development issues for reprocessing and fuel fabrication technologies development. Various investigation, laboratory scale hot tests, semi-engineering scale tests (cold and/or uranium), conceptual design study etc. are still going on. In 2008, JAEA has summarized current status of these development and rechecked the R&D deployment achieving goal targets at a check and review in 2010. These innovative technologies are decided to be adopted or not in 2010, on schedule. Furthermore, middle to long term R&D deployment has been under discussion considering urgent matters such as how to cope with the transition from LWR cycle to FBR cycle, fuel supply for the demonstration reactor etc.

Investigation on Transition from LWR cycle to FBR cycle

In Japan, the discussion of the next reprocessing plant, which will be operated from around 2050, will be started in the Japan Atomic Energy Commission from around 2010 as mentioned in "Framework for Nuclear Energy Policy" (Oct. 2005). Considering the transition from LWR cycle to FBR cycle, the LWR UO_2 SF and also LWR MOX SF, FBR SF would have to be reprocessed in the next reprocessing plant reasonably from the viewpoint of supply-demand balance of Pu and supply of the enough Pu products to FBR. The preliminary study and examination for transition from LWR cycle to FBR cycle has been conducted in cooperation with the related parties in Japan. The current results are the followings.

1) The necessity of the further examination for future nuclear power and fuel cycle deployment was identified, including investigation on transition of isotopic composition of nuclear materials.
2) The necessity to conduct the next reprocessing plant concept was identified (on the matter of to what extent LWR SF processing line and FBR-SF processing line should be apart or in common).
3) The preparation of process selection for the next reprocessing plant was conduct (investigation and evaluation of process performance and technical issues for several process candidates such as NEXT process, Co-processing, etc.).
4) The necessity to establish the roadmap for the long-term R&D scheme was identified.
5) The necessity to reinforce the study of proliferation resistance was identified.

At the current moment, the NEXT process is considered as the most promising concept for FBR SF reprocessing in the case of separated plant or at the FBR-equilibrium period. Meanwhile, for the transition period, from LWR cycle to FBR cycle, it would be necessary to investigate the possibility to reprocess by commoditizing plant and optimum process. Until 2010, above studies would be implemented in coordination with FaCT project. After the check and review of FaCT project in 2010 and the discussion of next reprocessing plant from around 2010, the framework for examination of the transition from LWR cycle to FBR cycle would be reviewed and restructured.

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Recent progress in advanced actinide recycling processes

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Nuclear power has the worldwide potential to curtail the dependence on fossil fuels and thereby to reduce the amount of greenhouse gas emissions while promoting energy independence of the different countries. The global energy context pleads in favour of a sustainable development of nuclear energy since the demand for energy will likely increase, whereas resources will tend to get scarcer and the prospect of global warming will drive down the consumption of fossil fuel. Therefore, retaining nuclear power as a key piece of the nation's energy portfolio strengthens French energy security and environmental quality.

How we deal with nuclear radioactive waste is crucial in this context. The public's concern regarding the long-term waste management made the French Governments to prepare and pass the 1991 and 2006 Acts, requesting in particular the study of applicable solutions for still minimizing the quantity and the hazardousness of final waste. This necessitates High Active Long Life element (such as the Minor Actinides MA) recycling, since the results of fuel cycle R&D could significantly change the challenges for the storage of nuclear waste. HALL recycling can reduce the heat load and the half-life of most of the waste to be buried to a couple of hundred years, overcoming the concerns of the public related to the long-life of the waste thus aiding the "burying approach" in securing a "broadly agreed political consensus" of waste disposal in a geological repository. It appears clearly that long-lasting nuclear options will include actinide recycling.

Within this framework, this paper presents recent progress obtained at the CEA/Marcoule on the development of innovative actinide partitioning hydrometallurgical processes in support of their recycling, either in an homogeneous mode (MA are recycled at low concentration in all the standard reactor fuel) or in an heterogeneous mode (MA are recycled at higher concentration in specific targets, at the periphery of the reactor core). Recovery performances obtained on recent tests in high active conditions of the Ganex process (grouped actinide separation connected to homogeneous recycling) are presented and discussed, as compared to the demands of P and T scenarios. New results concern also major improvements and possible simplifications of the Diamex-Sanex process, whose technical feasibility was already demonstrated in 2005 for americium and curium partitioning (heterogeneous mode).

In the coming years, next steps will involve both better in-depth understanding of the scientific basis of these actinide recycling processes, and for the new promising concepts, the studies necessary prior to industrial implementation of these processes.

Enhancing Minor Actinide Transmutation in ARR

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This paper presents a optimized homogeneous core concept of Advanced Recycling Reactor (ARR) toward enhanching Minor Actinide (MA) transmutation satisfying safety requirements, especially the void reactivity requirements. In pursuit of these objectives, following items are optimized: blanket type (AmO_2 or AmN), dimensions of axial and radial blanket, residual duration of radial blanket, and ratio of transuranics (TRU) to heavy metal.

In this study, "Optimization of MA transmutation" shall be defined as follows:

- Enhanced TRU burning rate per unit power generation satisfying the design requirements.
- Reduced recycling amount of the MA bearing fuel, while maintaining the large TRU burning capability and the enhanced MA transmutation capability.

The main specifications of core and fuel are a soudium cooled fast recator with 1180MW of thermal output, 70cm of core height, 150GWd/t of design based burn-up, 276 of subassembly in the active core, 331 of fuel pins per subassembly (including moderator pins), 82% of smear density in fuel, and three exchanging fuel batches for core fuel assembly.

In this design study the current requirements for the Japan Sodium-cooled Fast Reactor (JSFR) are applied. TRU contained in fresh fuel are Pu, Americium (Am) and Neptunium (Np). Curium (Cm) is not contained in the fresh fuel.

The group constants used are the fast reactor group constants JFS3-J3.3. For the calculation of the burning, the power distribution and the control rod reactivity, 3D-TRIZ diffusion code, TRISTAN developed and owned by Mitsubishi Heavy Industries, Limited (MHI), is used, while TRISTAN with18-groups of cross sections has been used for the reactivity coefficients.

Cases of calucations to optimize MA transmutation core is shown in Table I. It is clear from the nuclear calculation that the provided ARR has a significant Am transmutation capability, 81 kg/Tweh, satisfying the safety requirements, through the utilization of optimized Am blanket with long duartion time.

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TABLE I

Cases of Calculations for Optimized MA Burning Core

		CASE 1	CASE 2	CASE 3	CASE 4	CASE 5	CASE 6	CASE 7
Life time of blanket	Batch of Am blanket	3 batches/Rd	9 batches/Rd	→	→	→	→	→
Ax. blanket	Length	10cm	→	20cm	10cm	→	→	→
Radial blanket	Number	one column	→	→	2 columns	one column	→	→
Target material	Am blanket	MOX(Am50%,U50%)	→	→	→	MOX(Am20%,U80%)	AmN 100%	AmN 100%
Fuel life	Operation periods	780	→	→	→	→	→	810
Moderator pin	fraction	12%	→	→	→	→	→	15%
TRU	TRU/HM	50%	→	→	→	→	→	45%

*1 Average burnup is 50GWd/t and total time of cooling and waiting time is 6 years.
*2 Limit of fluence of fast flux: 5 E23/cm^2 (E.>0.1MeV)

U.S. Study on Impacts of Heterogeneous Recycle in Fast Reactors on Overall Fuel Cycle

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The heterogeneous recycle approach in which driver fuel assemblies (containing Pu or Pu+Np) and target assemblies (containing the remaining minor actinides) are used in the fast reactor core is being considered as an alternative to the homogeneous recycle approach in which all transuranic elements (TRU) are contained in the driver fuel and no target assemblies are employed. By this separation of the plutonium and minor actinides (MA) in the heterogeneous recycle approach, these actinides can be managed separately in the transmutation system and the fuel cycle.

Some advantages of the heterogeneous recycle approach include: (1) use of technology similar to existing recycle fuel fabrication and co-extraction processes for early deployment of advanced fuel cycle technology (permits time for additional R&D to find solutions to the handling of the high dose MA); (2) potential to confine the remote fabrication of MA-containing fuels with lower throughput to a dedicated sub-facility for fabrication; (3) easier fabrication of conventional recycle fuel form for driver assemblies (with the possibility that the first recycle of Pu or Pu+Np might not need to be remote); and (4) flexible management of MA loading in the core.

There are however potential issues with the heterogeneous recycle approach related to the confinement of the high radioactivity and heating of the minor actinides in compact assemblies. Preliminary evaluations (mostly core studies) of the issues associated with the use of heterogeneous recycle in advanced fast reactor systems have been performed by U.S. national laboratories. In a follow on study, an evaluation of the impacts of the approach on the technologies proposed for the fuel cycle (including reactor, separations, and fuel fabrication and performance) is being undertaken by the laboratories.

Issues currently being evaluated and to be discussed in the full paper include: (1) difficulty of recycling, handling, and fabricating target assemblies in the fuel cycle; (2) the state of target technology (fabrication, irradiation performance, etc.); (3) issues associated with MA storage (alternative to immediate use); (4) impacts on reactor performance and safety; and (5) impacts on the transmutation system dynamics and strategies. A quantitative estimation of the economic impact of target utilization is also planned.

Homogeneous versus heterogeneous transmutation in Sodium cooled fast reactors : comparison on scenario studies

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In the frame of the French law for waste management, scenario studies are carried out with the simulation software COSI to compare different options of separation and transmutation of plutonium and possibly minor actinides in the French fleet of reactors.

Sodium cooled Fast Reactor (SFR) is in France the best candidate for the GEN IV system to be deployed in a few decades. In these studies, these systems are supposed to be deployed around 2040.

The goal of the scenario studies is to evaluate the consequences of different options on the deployment of SFR from the plutonium inventory viewpoint, the actinides inventories, the possibility to transmute and incinerate the minor actinides (MA) in SFR and the capacity of cycle facilities.

In the SFR, the transmutation of MA can be achieved with various modes and waste management. The possible modes for transmutation are :

- The homogeneous mode where the minor actinides to be transmuted are directly mixed with "standard" fuel of the reactor,
- The heterogeneous mode for which the actinides to be transmuted are separated from the fuel itself, in limited number of S/A (targets) devoted to actinides transmutation [1].

As far as the studies will be achieved during year 2009, the paper will intend to present the results of 6 scenarios :

Scenario 1 : Pu recycling in the SFR,

Scenario 2 : same as scenario 1 and **minor actinides** recycling in SFR in **heterogeneous mode**,

Scenario 3 : same as scenario 1 and **Americium** recycling in SFR in **heterogeneous mode,**

Scenario 4 : same as scenario 1 and **minor actinides** recycling in SFR in **homogeneous mode**,

Scenario 5 : same as scenario 1 and **Americium** recycling in SFR in **homogeneous mode,**

Scenario 6 : same as scenario 1 and **minor actinides** recycling in **Accelerator Driven Systems** (ADS of EFFIT type)

The results of the scenario 1, 2 and 3 have been presented in detail in paper [2].

Each option has been evaluated in different dynamic scenarios taking into account the transition between the current nuclear reactor park and a SFR park, with the deployment of SFR in replacement of PWR.

The paper will present the impact of these options on the SFR core MA loading, the Plutonium and minor actinides inventories, and on the fuel cycle facilities.

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Parallel Session 5.2:

Fast reactor fuel cycles

Advanced Fuel Cycles and Fast Reactor Flexibility

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A most outstanding feature of fast neutron reactors (FRs) is the flexibility of the concept that allows to fulfil a wide range of different goals, e.g. to breed fissile material (the principal mission of a FR) or to burn TRUs or Minor Actinides (MA), for practically any Pu vector, MA content or MA/Pu ratio.The exploitation of this feature allows envisaging the progressive introduction of fast reactors to replace thermal neutron reactors to cope with sustainability and waste minimisation objectives by an appropriate tuning of FR breeding characteristics. Moreover, the fast reactor flexibility permits envisioning a transition phase devoted to the reduction of the built-up inventories of radioactive wastes (essentially MA), or even further to provide an effective bridge towards the use of an alternative hypothetical new energy source after a period of extensive use of fission nuclear energy: the large inventories of irradiated fuel in the fuel cycle can be reduced drastically with the use of low conversion ratio critical fast reactors. These "burner" fast reactors do not necessarily need to be new conceived as dedicated plants to burn existing spent fuel inventories, but they can be based on the same reactor design which has been developed during the phases of extended use of fast reactors by reverting the core from "iso-generator", or even "breeder", to "burner". The reversibility of a fast reactor core from burner to breeder was in fact already demonstrated within the CAPRA international program in the early nineties [1].

In the present work, we have investigated the role and characteristics of the FRs and their fuel cycle to be deployed in two very different scenarios:

a) A "global" scenario (presently under study within an OECD-NEA Expert Group), where the increasing energy demand worldwide, is met with the deployment of LWRs at first, and then successively of FRs. The key issue considered here is to determine the FR optimum breeding capabilities, in order to meet the energy demand without consumption of the Uranium resources down to a critical limit.
b) In the "regional" (European) scenario (previously studied [2] in the frame of both an EU project and the same Expert Group of the OECD-NEA mentioned above) the"double strata" strategy has been examined, in which Accelerator Driven Systems (ADS) were deployed at the regional level in order to eliminate the TRU legacy from certain number of countries phasing out the nuclear power and to stabilize the MA inventories in some other countries within the same region, that continue to rely on nuclear energy.

Further on we have explored the possibility to replace ADS with low conversion ratio (CR) critical fast reactors. In the case of scenario a), the objective was to replace starting in 2050 the entire LWR fleet by fast reactors as quickly as possible. The first calculation results are depicted in Figure 1, included is the total nuclear energy demand from 2005 to 2200 (according to the energy envelope assumed by the NEA Expert Group) and the subdivision in energy generated per reactor type. Plutonium stocks availability strongly affects the rate of FR deployment pace. A slow stepwise deployment of fast fleet was necessary in order to avoid the shortage of fuel. This is caused by the type of FR used in the simulations which is an "*isogenerator*" with a constant conversion ratio CR~1 versus time. It can be seen that the fast reactors cover full energy demand by 2200. However, the U consumption requires not only the use of known "conventional" U resources but also of almost all U coming from

phosphates. Thus it was attempted to replace the "*isogenerator*" FR with an effective breeder (CR~1.4).

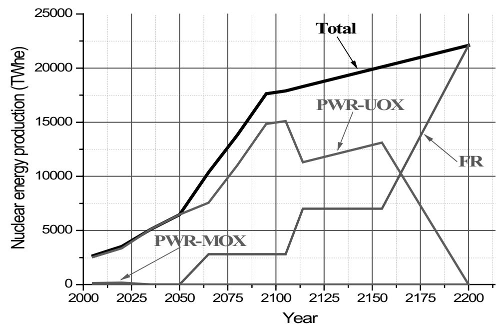

Fig. 1. Total energy demand and the share per reactor type

The results prove the dramatic improvement of the situation and the increased flexibility in FR deployment strategies. In the case of scenario b), results obtained in previous work [3] indicated that both metal and oxide fuelled critical fast reactor core can be designed in order to reach very high burning capabilities (i.e. very low conversion ratios). Moreover, it was shown that TRU fuels on a U matrix and not only inert matrix type of fuels can be used, in a large range of MA/Pu content ratios. This is a critical point, in view of the difficulty to develop a reliable inert matrix fuel, heavily loaded in MA, as indicated by several international programs. The full range of fast reactor explored in the present study is summarized in Table 1, that shows how the intrinsic flexibility of a fast reactor core, based essentially on physics features, can be used in very different contexts to provide a critical tool for future advanced fuel cycles. The full paper will provide a detailed analysis of the different scenarios.

Table 1. Top level reactor parameters

	FR isogenerator	**FR breeder with axial and radial blankets**	**FR burner**		**ADS**
Fuel type	$(U\text{-}TRU)O_2$	U-TRU-Zr	$(U\text{-}TRU)O_2$	U-TRU-Zr	TRU-MgO
MA/PU ratio	0.1	0.1	0.1/1.0	0.1/1.0	≈1.0
Pu content (%)	21.19	10.4.	(21.7-37.4)/ (22.3-44.3)	(17.3-32.5)/ (18-32.2)	45
Power (GWe)	1.45	0.6	0.6	0.6	0.154
Conversion Ratio	≈1	1.4	0.8/0.5	0.8/0.5	0.0
Cycle length (EFPD)	340	400	353/326	232/221	320

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Multi-regional Transitional Strategies Towards Fast Reactor Based Nuclear Energy Systems

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Sustainable nuclear energy systems are based on synergistic combinations of nuclear power plant (NPP) types with fast reactors (FRs) as a necessary component to address sustainability from a fissile resource and from a waste management perspective.

Though, the path forward to such sustainable nuclear energy systems is not yet clear, given impediments that need to be addressed including possible economics and socio-political concerns (e.g. proliferation risk). These impediments are regionally dependent given the differences among today's and future's nuclear energy systems in various parts of the world. That is, differences in spent fuel inventories, nuclear fuel cycle facility deployment rates, NPP-park composition and energy market organization with also differing socio-political support for nuclear energy.

An eight region representation of the world has been analysed using the nuclear fuel cycle dynamic scenarios systems code DANESS [1]. The representation involves a variety of nuclear energy system deployment paths per region with identification of fissile material exchange between regions which could facilitate achieving sustainability of the nuclear energy system at the world-level. Such regional representation allows for addressing the local competitiveness of nuclear energy which is crucial to achieving global energy sustainability via the nuclear fuel cycle.

The paper addresses the flexibility offered by FRs with varying conversion ratios concluding on the essential role that FRs have to play in 'regulating' the worldwide nuclear fuel cycle and especially the fissile material balance in the world. Therefore, FRs inherently represents an option towards energy sustainability with the potential to rebalance the fissile material inventory in the world, both in time and in space, and thus allow for addressing socio-political concerns on proliferation of such materials. The results of multi-regional global deployment scenarios that explore transition to FR strategies are presented in this paper.

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GEN IV deployment: Long term-prospective

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Nuclear power, already a significant source of electricity, has attracted renewed interest around the world in the past few years to meet the needs arising from demographic and economic growths. This trend occurs within specific circumstances, at a time when greenhouse gas emissions have to be sharply curbed and when security of energy supplies has to be enhanced. Nuclear energy is a technology currently available, adapted to both issues.

The objective of this study, achieved with the GRUS model (Management of the Uranium Resources under STELLA environment), is to assess how nuclear energy can answer an increasing need in electricity. It deals with the deployment of the various types of reactors, their rate and limits of installation, the constraint of Plutonium (Pu) availability, the impact on Uranium consumption as well as options allowing giving some flexibility to the various constraints.

We will analyze, in a world context, the feasible transitions between the actual fleet and a future one. The time scale will go until 2150 according to the lifespan of the reactors in order to cover two fleet replacements.

Most energy demand prospective scenarios give nuclear energy a big role to play. For example, for the IIASA scenarios we have chosen (A2, A3, B, C2), nuclear installed capacity would be increased by a factor of two to five up to 2050. That means, whatever the nuclear technology, around 2050, about 80 new GWe to be installed every year to satisfy the increase in demand in addition to the replacement of old reactors - except for strongly constrained scenarios requiring about 25 new GWe per year. Even so, around 2030, 20 to 45 new GWe are to be installed every year in order to meet the need for nuclear power.

Whatever the scenario, the 16 Mt of conventional uranium would be consumed before the end of the century (around 2070-2080 for scenarios for strong demand scenarios) and already engaged around the mid-century if the nuclear fleet were to be built up only with light water reactors. Even when taking into account the unconventional uranium (about 22 Mt more), the resources would be engaged before the end of the century.

Therefore the deployment of PWRs only is not a sustainable option in the long term. The deployment of FRs would thus be an answer to the resources issues linked to the long term development of nuclear technology.

However, the nuclear capacity that could be installed with FRs could be limited by Plutonium availability.

In any case, the third and the fourth generations would coexist all over the century.

Our results indicate that nuclear energy has to be boosted as soon as possible. It would be a mistake to wait for the fast reactors technology to be available, as the corresponding delay - notably in terms of Pu production - should never be caught up.

We have also identified the parameters that could provide flexibility in the deployment of FRs such as the launch date of the FRs, the breeding gain and the burn-up and done parametric studies.

Assessment of compatibility of a system with fast reactors with sustainability requirements and paths to its deployment

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This Study on the joint assessment of a nuclear energy system (NES) based on the Closed Nuclear Fuel Cycle (CNFC) with Fast Reactors (FRs) was implemented by Canada, China, France, India, Japan, Republic of Korea, Russia, and Ukraine between 2005 and 2007 in the framework of the IAEA INPRO Project. The main objectives of the study were to assess the NES CNFC-FR for satisfying criteria of sustainability, determine paths for its deployment, and establish frameworks and areas for collaborative R&D work.

The Joint Study was implemented in different steps. In its first step discussions were focussed on the analysis of country/region/world data along with possible national and global scenarios for the introduction of the CNFC-FR. Then technologies suitable for the INS were identified and reviewed, and finally a model for assessing of CNFC-FR was defined.

In the second step, the characteristics of the CNFC-FR were examined to assess its compliance with sustainability criteria developed in the INPRO methodology in the areas of economics, safety, environment, waste management, proliferation resistance, and infrastructure.

It was agreed to perform the assessment on the basis of a near term CNFC-FR using proven technologies, such as sodium coolant, MOX pellet fuel and aqueous reprocessing technology. Main results and findings of the study are summarized below.

The successful operation of several demo and demo/commercial FRs has demonstrated that FRs meet current safety standards. Results of safety analysis have also shown that the requirement to reduce for future FRs the risk of severe accidents by at least one order of magnitude can be fulfilled, if safety features are further enhanced through identified R&D. Probabilistic analysis has also demonstrated the ability of the Innovative Nuclear System (INS) to prevent the need for relocation or evacuation measures outside the plant site in case of a major accident. Thus, safety characteristics of near term CNFC-FRs are judged to be in compliance with safety requirements of sustainable energy supply.

The environmental effects of the demo and near term CNFC-FR are well within the performance envelope of current nuclear energy systems delivering similar energy products, with the lowest Green House Gas (GHG) emissions among them. The feasibility of excellent environmental and health preserving features of CNFC-FR has also been demonstrated by the

operation of demo/industrial CNFC-SFRs. The introduction of FRs with a closed fuel cycle in some countries might help to make most efficient use of nuclear fuel resources by using denatured uranium fuel and plutonium fuel, which can be generated in the FR blankets, if needed. The CNFC-FR would extend the utilisation of available uranium resources by more than sixty times and would be nearly inexhaustible. It can be considered as an energy resource suitable for a large scale national and global deployment.

Safe conditioning of waste arising from plutonium recycling is industrial reality today and an important practical milestone in reaching ultimate goals of the closed cycle strategy. The CNFC-FR has practically showed its potential to meet all today requirements related to waste management. With development and introduction of novel technologies for optimum management of nuclear fissile products and minor actinides, CNFC-FR would have break-through potential to meet the sustainability requirements related to waste management.

Proliferation resistance of the CNFC-FR due to realization of the intrinsic features could be comparable or higher than one of the Open Fuel Cycle (OF). The INS provides key technology for optimal utilization of fissile materials and elimination of their disposal in geological repositories thus providing a reliable background for applying extrinsic institutional arrangements. The Joint Study has judged that intrinsic features of the INS offer unique technological platform to meeting basic principles of sustainability in the domain of proliferation resistance. More efforts in further development of extrinsic measures still have to be done to provide conditions for utilizing these opportunities in transition to a new and higher level of the nuclear power proliferation resistance.

A legal nuclear framework in accordance with international standards has already been established in all countries participating in the Joint Study as well as an appropriate economic/industrial infrastructure. Efforts are being undertaken to enhance public acceptance, political support and inflow of human resources. The study came to conclusion that the CNFC-FR is a suitable technology for realization of a regional or multilateral approach to the assurance of the front and back end of fuel cycle services and transition to a global nuclear architecture that will provide new perspectives for growth of mature nuclear industries and at the same time facilitate the use of nuclear power by newcomers.

First of a kind CNFC-FR did not fully meet the economic requirements. This is the only area where the NES cannot comply with the INPRO demand. In accordance with the INPRO methodology recommendations, assessors have addressed the examination of possible improvements in the NES design and technology to meet the economic acceptance limits. Analysis of the study has shown that design simplification, increase of the fuel burn-up, and cost improvements via R&D along with construction in small series would result in competitive costs of FRs with thermal reactor NPPs and fossil-fuelled power plants. The Joint Study concludes that commercial CNFC-FR based on available technologies could be affordable in the medium-term in the countries mastering the technology.

The overall assessment has indicated that the medium-term CNFC-FR will confidently meet INPRO requirements of sustainable energy supply provided that the identified R&D (with focus on economics and safety) were carried out. Thus, transition to NES with increasing share of CNFC-FR should remain a primary objective and a driving force of the global nuclear power development strategies. At that, multilateral cooperation will be one of the most efficient paths to CNFC-FR deployment.

International Nuclear Fuel Cycle Centers in Global Nuclear Power Infrastructure

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Interest among the nations of the world to use nuclear energy is increasing due to economic, environmental and energy security reasons. The increase in the number of nations using nuclear energy might raise political risk of non-peaceful use of sensitive nuclear technologies. Therefore, additional measures should be taken in order to minimize risk of proliferation in connection with the awaited renaissance of nuclear power. The problem of nuclear nonproliferation is an extremely complicated one and in order to mitigate it different dimensions should be taken into account: political, technological and institutional.

Early in 2006 Russia proposed an initiative on global nuclear power infrastructure which will permit nondiscrimination access to nuclear energy of all interested countries observing requirements of nonproliferation regime [1]. The key element of such infrastructure should be system of International Centers (IC) to provide services of nuclear fuel cycle including at first stage uranium enrichment and later on management of spent nuclear fuel (SNF) under the IAEA control.

For effective management of SNF it is necessary to have developed technologies at least in four areas including fast reactors (FR) and closed fuel cycle technologies, SNF reprocessing, transuranium (TRU) fuel fabrication, nuclear waste management. At present the technology for only one area mentioned above have reached commercial level – LWR SNF aqueous reprocessing. Two other areas – technologies of sodium FR, MOX fuel for FR – have been demonstrated at semi-industrial level. Other technologies are still at R&D level – reprocessing of FR SNF, multi recycling of TRU fuel in FR, and nuclear waste management.

Business as usual scenario of ICs establishment for SNF management might be to wait until some nations commercialize all associated with FR and closed fuel cycle areas driven mainly by national interest in addressing uranium resource shortages. Obviously this way needs significant time for realization and first International Center might be implemented not earlier than by 2040-2050.

The authors propose for consideration another stage-by-stage approach. The main idea is to start at the first stage in organization of International Centers based on those elements of nuclear fuel cycle which have already been demonstrated or reached commercial level. It includes LWR SNF reprocessing, MOX fuel fabrication for FR, and sodium-cooled FR. In our opinion this approach may be realized in the nearest future.

This approach will solve problems of thermal reactors SNF especially for new countries worldwide willing to use nuclear energy, by concentrating plutonium in limited numbers of IC under the IAEA control. In this way ecological problem related to thermal reactor SNF will be solved as well. The base of such IC will be economical sodium-cooled FRs with

proved breeding ratio. At this stage SNF of FR is supposed to be stored in IC temporary storages until reprocessing technology and multi recycling of TRU fuel in FR are proved.

The principal structure of such an International Center providing nuclear fuel cycle services for nuclear power plants (NPPs) with light water reactors of 10 GW of installed capacity may be as presented on Fig. 1 [2].

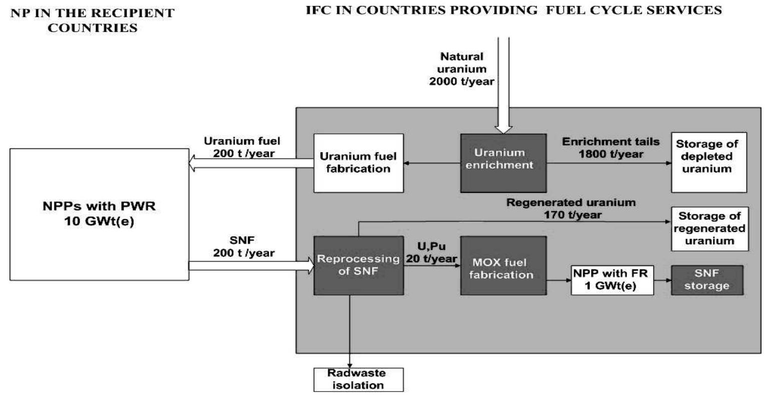

FIG. 1. Principal structure of an International Center in the mid-term perspective.

At the second stage for long-term perspective it is supposed that FRs deployed in a set of IC will solve the resource problem providing nuclear resources plutonium and uranium-233 for large-scale nuclear power comprising both thermal and fast reactors deployed worldwide. In this case altogether with ecological task connected with SNF management FRs will provide nuclear resources for the whole system of nuclear power. Fast reactors deployed in International Centers will use TRU fuel and have breeding ratio above 1. Fast reactors deployed in other countries besides International Centers are not supposed to have blankets with breeding ratio under 1.

At the first stage of International Center development the number of such Centers might be about 10% of total number of operated commercial nuclear units in the world. At the second stage in long-term perspective the number of International Centers should be substantially more to provide services in nuclear fuel cycle for world's large-scale nuclear power.

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Security and Control of Nuclear Material in PFBR

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In a developing country like India, Nuclear Power plays a very crucial role in sustainable energy development program. With five decades of research and development in the nuclear energy for application areas like medicine, agriculture, besides the major thrust area of power production, India is marching ahead with a highly ambitious nuclear energy program. The program spans all nuclear fuel cycle activities starting from exploration and mining, fuel fabrication, power production to managing spent fuel and nuclear waste. Our nuclear power program is carved in 3 stages with the ultimate goal of utilization of large quantity of thorium available in Indian Coasts.

The construction of Prototype Fast Breeder Reactor (PFBR) has been entrusted with a formation of a new special purpose vehicle, Bharatiya Nabhikiya Vidyut Nigam Limited (BHAVINI), a wholly owned Enterprise of Government of India under the administrative control of the Department of Atomic Energy (DAE). PFBR is the flagship project making us to enter to the second phase of our program. With the breeding technology it is expected to realize a huge power potential of 35 GWe of electricity bringing in energy security for the nation.

Even though the PFBR project is constructed within the high security complex of Kalpakkam, a hostile action by adversaries in the form of sabotage and / or theft or unauthorized removal of material can have adverse impact on the safety and health of workers, public and environment, emphasizing the need for stringent controls and measures towards proliferation, accounting, pilferage and sabotage. As the adversaries are more skilled, knowledgeable and forceful and can use stealth, force or deceit to carry out their malafide intentions, the best defense against such potentially catastrophic threats is to have a comprehensive integrated security systems backed up by strong administrative and operational measures.

In the integrated security systems of PFBR project, Physical Protection System plays a key role in preventing unauthorized entry of personnel to key areas (vital equipments location, active material storage etc) and hampering normal operation of the plant, removal of active material etc. The Primary objectives of the security system of PFBR is to protect the nuclear facility and nuclear fuel against acts which may endanger public by radiation exposure, protect the nuclear material against theft, prevent malevolent acts, permit only authorized activities in the protected areas, protect proprietary information, material and finally protection of employees and public around the nuclear facilities.

All the activities related to PFBR construction, commissioning, operation and maintenance are governed by Atomic Energy Act 1962, Atomic Energy factory rules 1996, Environment protection Act and other industrial regulations and acts of India. Security and

administrative measures including a well defined security organization and structure, design basis threat and engineered systems for security are defined both at the facility level and the apex level.

The Department of Atomic Energy has issued a comprehensive manual on security of nuclear facilities. The manual emphasizes the need for stringent controls and measures towards proliferation, accounting, pilferage and sabotage of nuclear material. The DAE Security manual provides the guidelines towards formulation of security organizational structure, reporting mechanism, interfacing with local law enforcement authorities etc. in addition to this, an independent body ‘Atomic Energy Regulatory Board (AERB)’ who does the supervision, licensing and regulation with regard to security of PFBR, had formulated regulation and guidelines through a manual on Security and specified Minimum requirements on security for all NPPs and enforces implementation. The security and the Physical Protection System (PPS) for PFBR were designed based on these documents and the guideline on Physical Protection of Nuclear Material and Nuclear Facilities issued by IAEA (INFCIRC/225 Rev. IV).

Design Basis Threat (DBT) forms the base document for all PP related measures for any nuclear facility. A comprehensive Facility specific DBT is formulated for PFBR also taking inputs from the IAEA’s implementation guide on Development, Use and Maintenance of Design Basis Threat and taking into consideration the national and local issues.

The Engineered system of security follows the design principles of defense in depth, multilayered system, balanced design and incorporating them in basic design & layout. The systems are devised on the principle of 4 Ds-deter, detect, delay and defeat and are realized by well defined technical, operative and administration measures suitably backed by our national law. For the above, a graded approach is adopted by classifying plant locations, defining physical limits of various security areas and installing multiple barriers and security checks. The type of the barrier is decided taking the Design Basis Threat into consideration. With the above concepts, entire plant area of PFBR is divided in multiple areas like protected boundaries, vital areas and inner area depending on their significance and importance from security point of view. The vital areas are identified with a scientific technique considering the plant safety assessment and events analysis. The strategic nuclear material is stored in the inner area inside the vital area. With the adoption of graded approach, The PP measures increases and becomes stringent as we move from outside to inside.

These multi layered systems are supplemented by a well defined automated access control system deployed to facilitate entry of authorized people and vehicles and prevent and detect unauthorized entry & exit attempts in all zones.

Manual and instrumented intrusion detection measures at multiple layers of physical barriers are provided with assessment of alarms. The system is integrated to various other systems including CCTV in such a way that in the event of an intrusion, the real time video of location of the intrusion will be displayed at selected display units, both inside as well as outside the plant. Multiple assessment methods are applied as per the requirement.

A redundant diverse physically different media based communication between central alarm stations; main plant control room, all security posts and response force are provided. There is also an inter communication between plant operation, security plant management and local law enforcement agencies.

All engineered systems are integrated to present the information in such a manner that it aids security to take appropriate actions.

Further there are well developed contingency plans to handle emergencies arising out of incidents related to radiological sabotage, theft of nuclear material, Intrusion into the facility, nuclear and other industrial accidents, fire and natural calamities. The contingency plan stipulates periodic emergency drills and reporting and corrective measures required for observed deficiencies.

Response force being the last line of defense interrupts advisory progress towards the target and neutralizes them before the advisory accomplishes the task. The composition & strength of Response Force, weapons and communications are worked out based on site specific DBT.

Adequate training is given on the engineered system operation, physical training for combat, security procedures, rules and emergency handling etc. All incidences related to nuclear security are brought to the notice of competent authority with a clear methodology of reporting.

A major contributor to enhancing the level of implementation and effective maintenance of PPS and protection of information is the increased awareness on the part of the personnel working within the plant, as to the value and importance of what they are doing. Hence through education and awareness training, employees are appraised and trained on the security support measures introduced to protect sensitive areas / information. Various training modules for various personnel ranging from security / response force to plant operating personnel based on individuals' specific job nature are prepared and delivered. In addition to the training, for plant operating personnel, licensing system specific to security aspects is also in place.

AERB review the security aspects of PFBR through various committees in hierarchical manner like Physical Protection Advisory Committee (PPAC), audit committee, technical review & update committee, Standing Group on Co-ordination and Review of Security Arrangements (SGCRSA) and implement requirements. As an integral part of physical protection, PFBR is also subjected to nuclear material accounting and control activities. Facility specific NUMAC cells are in place in the fuel fabrication units of PFBR who report to nodal cell on all aspects of nuclear material accounting. Further computerized systems of accounting for the fuel movements within PFBR are also under preparation.

In conclusion, it can be said that PFBR attaches a great importance to security of nuclear material and facilities from the very beginning. An integrated multi pronged approach for security of PFBR is adopted which gives high confidence in the effectiveness of the security of the nuclear material and the facility. These systems are constantly reviewed and updated taking into account the complex and dynamic changes in security scenario and making them an integral part of our nuclear energy program.

Since security is a subject matter of confidentiality, this paper highlights only the concepts covering objectives, fundamental principles, methodologies of physical protection and the graded approach without giving finer details.

Parallel Session 5.3:

Fast reactor fuel cycles

"Proliferation Resistance for Fast Reactors and Related Fuel Cycles: Issues and Impacts"

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The views expressed are the author's own and not those of the Los Alamos National Laboratory, the National Nuclear Security Administration, the Department of Energy or any other agency.

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The prospects for a dramatic growth in nuclear power may depend on the effectiveness of, and the resources devoted to, plans to develop and implement technologies and approaches that strengthen proliferation resistance and nuclear materials accountability. The challenges of fast reactors and related fuel cycles are especially critical, as they are being explored in the Generation IV International Forum (GIF) and the International Atomic Energy Agency's (IAEA's) International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) initiative, as well as by many states that are looking to these systems for the efficient use of uranium resources and long-term energy security. How do any proliferation risks they may pose compare to other reactors, both existing and under development, and the fuel cycles associated with them? Can they be designed with intrinsic (technological) features to make these systems proliferation resistant? What roles can extrinsic (institutional) features play? What are the anticipated safeguards requirements, and will new technologies and approaches need to be developed? How can safeguards be facilitated by the design process? These and other questions require a rethinking of proliferation resistance and the prospects for new technologies and other intrinsic and extrinsic features being developed that are responsive to specific issues for fast reactors and related fuel cycles and to the broader threat environment in which these systems will have to operate. There are no technologies that can wholly eliminate the risk of proliferation by a determined state, but technology and design can play a role in reducing state threats and perhaps in eliminating non-state threats. There will be a significant role for extrinsic factors, especially the various measures—from safeguards and physical protection to export controls—embodied in the international nuclear nonproliferation regime. This paper will offer an assessment of the issues surrounding, and the prospects for, efforts to develop proliferation resistance for fast reactors and related fuel cycles in the context of a nuclear renaissance.

Proliferation issues related to the deployment of Fast Neutron Reactors

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This paper is aiming at discussing specific proliferation concerns that could be raised by the industrial deployement Fast Neutron Reactors (FNRs).

Proliferation is defined in the nuclear context as the spread of nuclear weapons and the materials and technologies used to produce them. Therefore Proliferation Resistance (PR) has become one of the primary topics to address in the frame of the development of nuclear energy systems. In addition to this, Physical Protection (PP) of sensitive nuclear marerials and facilities has become a growing concern in recent years because of terrorism fears. In the initiatives to develop innovative nuclear energy system (Generation-IV, INPRO), PR and PP are key elements, along with economics, safety, sustainability and environment which has to be addressed. This paper will elaborate more on each of these notions (PR and PP) but it will deal mainly with PR and will focuss on FNR specificities with regard to PR. We just mention here that these specifities come from two main points:

- The neutronic of a fast neutron core is particularly well suited in producing plutonium (Pu) from U-238 with a very high proportion of Pu-239 (and hence a low content in Pu-240, Pu-242 and Pu-238). It can be mention that the same physics make it also adequate for minor actinides (MA) burning because of a favourable of fission / capture ratio for these minor actinides with neutrons in the fast energy range.
- The recycling of plutonium since the development of FNRs makes sense only if they are included in a closed fuel cycle to separate the uranium and plutonium (and possibly the minor actinide) so as to burn the U-238 transformed in Pu-239 (and possibly to "incinerate" the minor Actinides). As a matter of fact Pu-239 presents the highest neutron reproduction factor for fast neutron spectrum among all fissile isotopes and thus it is by far the best fissile isotope that can be used in FNRs.

On the other hand, a distinctive feature of FNRs is that there is no need to feed the reactor with enriched uranium. This means that when equilibrium of an FNR fleet is reached, enrichment facilities are no longer necessary, which is indeed an asset from proliferation point of view because enrichment technologies constitute one of the most concerns in this area. Another advantage of FNRs is that no plutonium is accumulated in waste to be disposed of since all plutonium is recycled in the system.

These elements underline the fact that it is necessary to investigate proliferation issues of FNRs through the whole "system", that is reactors concepts along with their associated fuel cycle facilities as a whole (from mines to ultimate waste) and their material flows. Therefore, the analysis presented in this paper will be based on this holistic approach.

To analyse proliferation issues of FNR systems, we must have in mind that nuclear non proliferation is a blend of :

- political commitments, such as treaties and multilateral agreements or multinational approaches,
- institutional arrangements also called "extrisic measures", such as international safeguard regime or export control mechanisms, and
- technical provisions also called "intrinsic measures", that "embeded" in the technology, such as technical features and operational modalties of facilities or nuclear materials that can impede their misuse (that is undeclared production or diversion of nuclear materials) for proliferation purpose or that facilitate implementation of extrinsic measures.

Only an appropriate and optimized combination of all these "barriers" is likely to minimize proliferation risks by establishing a robust and effective PR system. In that sense, we consider that PR design and assessment are calling for the use of the "defense in depth" which is widely and successfully used in nuclear safety. To complement this approach we will evoke PR assessment methodologies such as SAPRA which is developed in France and "PR&PP" methodology developed in the frame of the Gen-IV International Forum (GIF) which is currently implemented to assess proliferation resistance of an FNR system.. We will illustrate how these rational approches can help to "built" and assess PR systems for FNR, by discussing some particular aspects such as :

- Material "attractiveness": we will debate in particular on various means that could be implemented to reduce plutonium quality produced in core blankets of FNRs. In this analyse will examine the interest of mixing plutonium with minor actinides from a non proliferation point of view.
- Handling and control of fresh fuels and spents fuels : we will discusss the need to design a highly protected and reliable system able to identify one-to-one moves of all fuel assemblies or sub assemblies.
- Safeguardibilty of reactors and recycling facilities : we will examine the benefits that could be brought by a "safeguard by design" approach and more genarally by taking into account PR issues as early as possible in the design and deveopment of FNR systems (reactors and fuel cycle plants).

For these discussions we will borrow examples from the existing industrial experience of the comprehensive french fuel cycle.

At a more general level, we will deal with the issue of the location of recycling facilities by comparing benefits and drawbacks, from non proliferation standpoint, of a co-location of these facilities on reactor sites or, conversely, of centralized plants operating in the frame of a multi-national approach (MNA).

From these analysis, we will draw some conclusions that could pave the way to further studies allowing to enhance proliferation resistance features of FNR systems.

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Role of Safeguards in Proliferation Resistance for the Future Nuclear Fuel Cycle Systems

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A large amount of plutonium as well as high 239-Pu should be handled in the future fast reactor nuclear fuel cycle (FR-NFC), where very robust measures for nuclear proliferation-resistance (PR) have to be taken to prevent nuclear proliferation.

The proliferation resistant NFC impedes diversion by host states seeking to acquire nuclear weapons or other nuclear explosive devices. PR measures should be composed of intrinsic barriers of nuclear energy system and extrinsic, i.e. institutional barrier. PR on nuclear systems against the increase in nuclear diversion risk has recently been discussed in international nuclear societies such as INPRO and GIF, whereas the demand of the studies to pursue more effective and efficient Safeguards system increases and is being discussed in Safeguards communities. To find a good balance of extrinsic barrier and intrinsic one will come to be essential for NFC designers to optimize civilian nuclear technology with nuclear non-proliferation.

International Safeguards including Comprehensive Safeguards Agreement (CSA) and Additional Protocol (AP) is the most effective institutional barrier among other institutional measures in non-proliferation regime. It should strongly function for nuclear non-proliferation, particularly, in the countries where Integrated Safeguards (IS) is implemented, because it seems unlikely that abrogation of institutional systems or diversion of nuclear materials in such countries occurs under IS. The advanced Safeguards with high detectability can play a dominant role for PR in the states complying with full institutional controls, namely IS, whereas, some intrinsic measures should complement the entire PR system (See image in Fig. 1).

This paper discusses the role of Safeguards in entire PR framework for future Japanese FR-NFC.

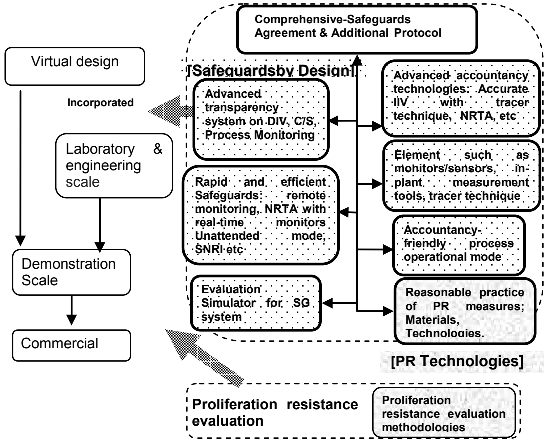

Fig.1 Design-based Robust Proliferation Resistant Nuclear Fuel Cycle (Example of High Detectable Safeguards System and Additional Proliferation Measures)

Parallel Session 6.1:

Fast reactor analysis: basic data, experiments and advanced simulation

Thermal and Hydrodynamic Fragmentation of a Single Molten Stainless Steel Droplet Penetrating Sodium Pool

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For the development of metallic fuel fast breeder reactors, the social acceptance (reactor core safety) is always an important problem. Although the probability of hypothetical core disruptive accidents (HCDAs) is quite low, it is indispensable to quantitatively evaluate the possibility of positive reactivity feedback neutronically and thermohydraulically coupled in the progression of HCDAs. For the termination of HCDAs, the molten core materials are required to be passively discharged from the core region. Therefore it is necessary to fully investigate the possibility and mechanism of thermal and hydrodynamic fragmentation due to the molten structural material-sodium coolant interaction.

As a basic study of FCI, Sugiyama *et al*. proposed the thermal fragmentation mechanism of sodium-entrapment type combined with the rapid release of latent heat within the droplet and jet [1,2], mainly based on the shapes of fragments and other evidences of interaction, such as the observation of a sodium microjet by using copper, tin or zinc as the metallic fuel simulants under the low superheating condition. Zhang *et al*. [3] conducted the experiments of a single molten copper droplet penetrating a sodium pool in a large range of T_i from below to above the melting point of copper, confirmed the same thermal fragmentation mechanism of sodium-entrapment type combined with the rapid release of latent heat and sensible heat within the droplet as within the jet and proposed that the fragmentation of molten metal jet can be evaluated by the fragmentation of a single molten metal droplet. Nishimura *et al.* [4] proposed the effect of hydrodynamic fragmentation with the high ambient Weber number (We_a>200), becomes predominant over that of thermal fragmentation under the low superheat condition (T_{sup} <165℃) by conducting the experiments of molten copper jet (20－300 g) penetrating sodium pool with the comparison of molten metallic fuels from Gabor *et al.*[5].

In the present study, in order to clarify the characteristics on the fragment sizes of a single molten stainless steel droplet and investigate the possibility and mechanism of thermal and hydrodynamic fragmentation under a wide range of superheat and ambient Weber number condition, the authors conducted an experiment of a single molten stainless steel droplet penetrating a sodium pool using an induction heating method newly developed. The experiment using the single molten stainless steel droplet (1－5 g) under a wide range of hydrodynamic conditions (56<We_a<586) was carried out in a wide range of superheating from 23 to 393℃,which corresponds to instantaneous contact interface temperatures(T_i) from 894 to 1086℃(far below its melting point), and in a narrow range of sodium pool temperatures from 295 to 337℃. In addition to the correlation on the fragment sizes of a single molten stainless steel droplet, the authors also reported the relationship of the present results with the previous results of a single molten copper droplet (1－5 g) reported by Zhang *et al.*[3] and the molten stainless steel jet (2.5 and 4 kg) reported by Schins' group[6,7].

The intensive fragmentations of the single molten stainless steel droplets were clearly observed even at T_i far below the melting point. The size distributions of molten droplet with

high We_a tend to be less than those with low We_a under the same thermal condition. When approximately We_a>250, the effect of hydrodynamic effect in fragmentation becomes predominant over thermal effect under the low superheat condition. Under the high We_a or high superheating condition separately, the size distributions of the single metal droplet and jet reported by Schins' group with 1000-fold difference in mass could keep very similar. The relatively larger size distributions of the single metal droplet only arise under the both low We_a and low superheating condition as shown in Fig.1. The results agree with the data of copper droplet obtained by Zhang *et al.*[3] and jet obtained by Nishimura *et al.*[4] very well. Moreover, it is found that the purity of sodium has little effect on the fragment sizes of droplet.

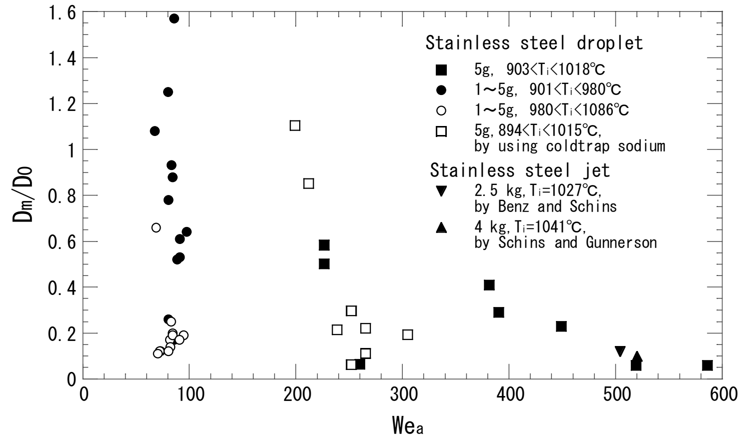

Fig.1 D_m/D_0 versus We_a of stainless steel droplet and jet fragments with different T_i

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Flow distribution and turbulent heat transfer in a hexagonal rod bundle experiment

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In the context of IP Eurotrans a series of experiments in electrically heated and unheated hexagonal rod bundles in water and lead bismuth eutectic (LBE) has been launched at the KArlsruhe Liquid metal LAboratory (KALLA) of the Forschungszentrum Karlsruhe aimed to quantify and separate the phenomena of turbulent heat transfer and flow distribution in hexagonal rod bundles with the final goal to describe the momentum and the energy transfer in a liquid metal operated fuel assembly. Consequently, the experimental program is composed of three major experiments that will be discussed in this paper:

In a first step the convective turbulent heat transfer of a turbulent lead bismuth flow along a vertically arranged, uniformly heated rod placed concentrically in an annular cavity is investigated at high power densities. This essentially thermally developing flow is studied experimentally by means of thermocouples, a traversable combined velocity-temperature sensor based on a Pitot-tube as well as thermocouple (TC) rakes consisting of 60 TC´s. The heat flux attainable is 100 W/cm² in a Reynolds number range from $6{\cdot}10^4$ up to $6{\cdot}10^5$. The experimental data exhibit that commercial CFD codes describe the temperature distribution adequately if the flow is mainly driven by forced convection and a fine mesh resolution is chosen. If, however, significant density gradients occur, mixed convection sets in even at high Reynolds numbers and significantly larger Nusselt numbers than numerically predicted appear. This is accompanied by an altered turbulence structure in the thermal boundary layer.

In the second experiment an isothermal 19-pin hexagonal rod-bundle assembly has been investigated in a turbulent water flow in a Reynolds number Range from $5{\cdot}10^3$ up to $9{\cdot}10^4$ covering both the transitional and the fully turbulent flow regime. Both pin to pitch ratio of 1.4 as well as the axial dimensions correspond 1:1 to the dimensions of the XT-ADS accelerator driven sub-critical reactor system considered in the IP-Eurotrans research program. Because of the opaqueness of the liquid metal flow the rod bundle water experiment is inevitable to gain information about the pressure drop of the bundle with spacer elements and the flow distribution in the sub-channels by means of Laser Doppler Anemometry (LDA) and Ultrasonic Doppler Velocimetry (UDV) technique. Due to the flow interaction with walls and tubes a non-isotropic momentum field is present in the reduced assembly, which yields an uneven flow rate in the individual ducts. The measurements of the static pressure loss of a single spacer and the complete test section show very good agreement of the loss coefficient with numerical predictions made by the sub-channel analysis code MATRA for the fully turbulent flow regime, whereas in the transitional flow regime with Reynolds numbers $< 3{\cdot}10^4$ the secondary flow leads to a rising loss coefficient.

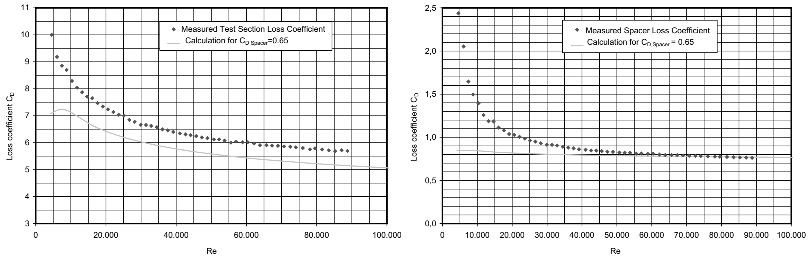

FIG.1. Measured loss coefficient and calculations for the complete rod bundle test section (right) and a single spacer (left).

The third experiment projected in the KALLA corresponds geometrically 1:1 to the water experiment and will be conducted in LBE using electrically heated pins. The heated length of this 19-pin hexagonal rod-bundle is 870mm with a heat flux of 100 W/cm² producing a total heating power of up to 430kW. The measurements cover a temperature range from 200°C to 400°C and Reynolds numbers up to 10^5.

Acknowledgements

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Phenomenon of Local Natural Circulation in a Circuit of Nuclear Power Plant

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The paper presents the results of experimental research into natural circulation modes (NC) performed on the BN-350 reactor, which allowed the phenomenon of local natural coolant circulation (LNC) emerging in sodium loops of reactor facility (RF) to be identified; the specific features of this effect were analyzed, as well as the conditions of its occurrence and the nature of its effect on the modes taking place in RF circuits.

A set of experiments was carried out at the BN-350 in 1996 with an aim to specify the integral thermoaccumulating and thermodissipating characteristics of individual loops and investigate sodium NC modes in the loops of RF primary and secondary circuits. The goal of these experiments was demonstration of a stable decay heat removal in the reactor at the expense of sodium NC in the facility circuits, as well as obtaining required experimental data for verification of the DINRUN computation code used for justification of admissible power levels of the BN-350 reactor. The analysis of these experimental results has lead to attention being paid to an important influence of local natural-convective coolant flows in some sections of RF circuits on the nature of circuit-wide natural coolant circulation (CNC) propagation.

Local natural-convective sodium flows due to a coolant temperature difference along the circuit length which are available even in case of zero CNC flowrate were found to emerge at individual sections of primary and secondary circuit loops along with CNC that are comparable with it in value. A characteristic feature of natural-convective coolant flows being described is the presence of fairly intensive heat transfer along the circuit at zero coolant flowrate over the circuit cross-section (Fig. 1). LNC was found to emerge both at horizontal and vertical sections of circuits.

The paper describes the experimental conditions and the main results obtained which testify that LNC occurs in the primary and secondary circuits of the BN-350 RF. Analysis was made of the BN-350 RF secondary loops configuration features as a result of which LNC emerging in the area of intermediate heat exchanger contributes to CNC reflux in the secondary circuit loops. Therefore it was shown that in the studies of NC modes in RF circuits a possibility of LNC occurrence should be taken into account, as well as its influence on CNC propagation (direction, CNC flowrate).

The paper provides a comparison between experimental results with the calculation results obtained with the use of one-dimensional code DINRUN, which incorporates a model that allows a heat transfer effect along the circuit at the expense of LNC to be taken into consideration. A satisfactory agreement between the experimental and calculation results was demonstrated.

Paper [1] presents the study of LNC circuits formation mechanisms at various sections of heat removal circuit (horizontal, vertical, inclined) and their interaction with CNC, and also states the criteria of coolant LNC occurrence under the CNC conditions both at horizontal and vertical circuit sections.

The research of LNC influence on the temperature state of RF circuits' equipment (pumps, valves, heat exchanger equipment etc.) is of special interest, not only in the NC modes, but also in the modes with a low flowrate of forced coolant circulation.

The need is shown in the further studies of interactions between both neighboring LNC circuits and LNC circuits with CNC flow including determination of the factors affecting the quantity of LNC circuits and their length at various sections of a closed circuit.

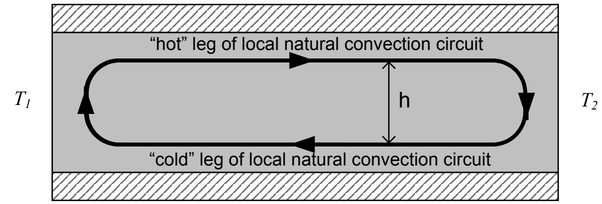

FIG. 1. LNC flow chart at a horizontal section of circuit ($T_1 > T_2$).

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The R&D test plan using sodium test loop for development of the 4S

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The 4S (Super-Safe, Small and Simple) would be able to supply about 10 MW of electrical power for 30 years without refueling. In the development of the sodium-cooled fast reactor such as 4S, the development of the equipment, the measuring instrument, and the system that greatly improves viability is demanded. Toshiba constructed the sodium test loop facility at the end of 2007. In addition, it is important that proving the function and evaluating the performance by using sodium in an environment near an actual plant condition. Verification and validation test of computer code used for safety analysis will be required for the reliability improvement. This report shows the sodium test loop facility of Toshiba and the test plans.

This facility allows practical temperature and flow velocity as same as 4S, and involves large test tank of 1 m diameter and 3.5 m height for immersion test. The development efficiency will be expected to improve by comparison between measured value and analysis by large scale computer system connected to network. This system can visualize the internal flow dynamics by using CFD codes using measured actual data, then, the designers and operators can recognize what is going on in the test loop. The safety of facility is secured by detection system of sodium leakage and earthquake, which lead to interlock automatically.

The near-term test plans using the sodium test loop is shown in Table 1. Pre-operation test had already finished. Some internal R&Ds are now going on. Electromagnetic flow meter was calibrated by volumetric method using large test tank. The heat exchanger, heater and cooler were confirmed to satisfy with design specifications by test operation. Toshiba has obtained some R&Ds funded by METI and MEXT. The design and fabrication for these R&Ds is now proceeding. For development of the fast reactor's equipment, performance and stability test of a large size electromagnetic pump, performance test of failed heat exchanger tube detection system for the double wall tube type steam generator, performance test of reflector systems and the other tests have been planed. Also, for validation and verification of safety analysis code, collections of the required data are promoted. In future, it's hoped that the sodium test loop facility is utilized not only 4S but also the other FBR demonstration reactors.

Table 1. The near-term test plans using the sodium test loop

		2009		2010
Ring type electromagnetic flow meter	(MEXT)	Design, Fabrication	Installation	Sodium test
10 m^3/min electromagnetic pump	(METI)	Design, Fabrication	Installation	Sodium test
Double wall tube type steam generator	(METI)	Design, Fabrication		Sodium test
Reflector cavity for 4S	(Internal R&D)	Sodium test		

Nuclear Data for Innovative Fast Reactors: Impact of Uncertainties and New Requirements

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The Working Party on Evaluation Cooperation (WPEC) of the OECD Nuclear Energy Agency Nuclear Science Committee established a Subgroup to develop a systematic approach to define data needs for advanced reactor systems and to make a comprehensive study of such needs for Generation-IV (Gen-IV) reactors.

The subgroup has been established at the end of 2005, and a final report has been published in 2008 [1]. A comprehensive sensitivity and uncertainty study has been performed to evaluate the impact of neutron cross-section uncertainty on the most significant integral parameters related to the core and fuel cycle of a wide range of innovative systems, even beyond the Gen-IV range of systems. In particular, results have been obtained for the Advanced Breeder Test Reactor (ABTR), the Sodium-cooled Fast Reactor (SFR), the European Fast Reactor (EFR), the Gas-cooled Fast Reactor (GFR) and the Lead-cooled Fast Reactor (LFR), the Accelerator Driven Minor Actinide Burner (ADMAB). These systems correspond to current studies within the Generation-IV initiative, the Advanced Fuel Cycle Initiative (AFCI), and the advanced fuel cycle and Partitioning/Transmutation studies in Japan and Europe.

State-of-the-art sensitivity and uncertainty methods have been used and they will be shortly described in the full paper.

The integral parameter uncertainties have been calculated at first using covariance data developed in a joint effort of several laboratories contributing to the Subgroup activity. This set of covariance matrices is referred to as BOLNA [2].

The discussion in the present paper is mostly focused on integral parameter (like keff, reactivity coefficients, power distributions etc) uncertainty due to neutron cross-section uncertainties. Fission spectrum uncertainties and the effect of resonance parameter uncertainty on Doppler have been also examined. The integral parameters considered are both related to the reactor core performances but also to some important fuel cycle-related parameters, like the transmutation potential, the doses in a waste repository or the neutron source at fuel fabrication.

The calculated integral parameter uncertainties, resulting from the initially assessed uncertainties on nuclear data, are probably acceptable in the early phases of design feasibility studies. In fact, the uncertainty on keff in less than 2% for all systems (with the exception of the Accelerator Driven System, ADS) and reactivity coefficient uncertainties are below 20%. Power distributions uncertainties are also relatively small, except, once more, in the case of the ADS.

However, later conceptual and design optimization phases of selected reactor and fuel cycle concepts will need improved data and methods, in order to reduce margins, both for economical and safety reasons. For this purpose, a compilation of preliminary "Design Target Accuracies" has been put together and a target accuracy assessment has been performed to provide an indicative quantitative evaluation of nuclear data improvement requirements by isotope, nuclear reaction and energy range, in order to meet the Design target accuracies, as compiled in the present study. First priorities were formulated on the basis of common needs for fast reactors and, separately, thermal systems. These priority items have been included in the High Priority Request List (HPRL) of the OECD-NEA DataBank.

The status of nuclear data uncertainties, as given from the initial uncertainty evaluation of the BOLNA covariance data compilation, and in particular the very low values for U-235, U-238 and Pu-239 fission and capture uncertainties, tend to indicate, in the case of the wide range of fast reactors considered in this study, a priority requirement for a drastic uncertainty reduction for some σ_{inel} (in particular for U-238, but also for Fe and Na), for the σ_{fiss} of higher Pu isotopes and in particular for Pu-241 (between ~1-500 keV) and for σ_{capt} of Pu-239 (~1-500 keV). These indications are valid for all fast reactors considered in this work, and which are representative of the current priorities of the Gen-IV and GNEP initiatives. Other requirements are of course associated to specific systems (as Si data for the GFR or Pb in the case of LFR and ADMAB).

Finally, in the case of ADS, where the fuel should be in principle heavily loaded with MA, tight requirements are found for some MA cross-sections, in particular for σ_{fiss} of Cm-244, Am241, Cm-245, Am-243, Cm-242, Am-242m, for σ_{inel} of Am-243 and for ν of Cm-244. For these reactions, the required accuracies are an order of magnitude below the present uncertainties. Concerning the major actinides, improvements are required for σ_{fiss} of Pu-241 (again ~factor 10), for σ_{fiss} of Pu-238 (~factor 5) and for ν of Pu-238 (~factor 3).

Very recently, an effort lead by BNL [3] has produced an improved version of the initial covariance data with an energy group structure of 33 groups (instead of 15), and revised uncertainties have been calculated on the wide range of Fast Reactor systems considered within Gen-IV and will presented in the full paper. Some typical results are given in Table 1.

ISOTOPE	Reactors											
	ABR Metal			ABR Oxide			SFR			EFR		
	Parameter			Parameter			Parameter			Parameter		
	K_{eff}	Na Void	Doppler	K_{eff}	Na Void	Doppler	K_{eff}	Na Void	Doppler	K_{eff}	Na Void	Doppler
Am241	0.03	0.17	0.10	0.03	0.20	0.11	0.06	0.56	0.20	0.03	0.17	0.10
Am242M	0.02	0.10	0.05	0.02	0.13	0.05	0.26	2.14	0.73	0.01	0.07	0.03
Am243	0.02	0.12	0.09	0.02	0.16	0.09	0.04	0.50	0.17	0.01	0.04	0.03
Cm244	0.07	0.76	0.42	0.15	0.81	0.54	0.17	2.60	0.67	0.04	0.17	0.18
Cm245	0.07	0.29	0.16	0.10	0.42	0.23	0.13	0.85	0.33	0.02	0.08	0.05
Cr52	0.02	0.23	0.07	0.02	0.14	0.05	0.02	0.66	0.08	0.01	0.11	0.05
Fe56	0.16	1.98	1.46	0.16	1.21	1.00	0.21	4.58	2.80	0.09	0.82	0.46
Na23	0.09	3.65	1.16	0.08	3.35	0.80	0.12	6.28	1.22	0.07	3.07	0.67
Ni58	0.00	0.02	0.03	0.00	0.02	0.02	0.01	0.04	0.03	0.11	0.32	0.20
Np237	0.02	0.14	0.08	0.01	0.07	0.05	0.05	0.70	0.22	0.01	0.03	0.02
O16	-	-	-	0.08	0.35	1.11	-	-	-	0.09	0.61	1.11
Pu238	0.26	0.17	0.77	0.28	2.40	0.83	0.62	6.49	1.69	0.15	1.22	0.51
Pu239	0.30	0.10	1.09	0.31	1.96	0.88	0.21	2.27	0.67	0.37	2.08	1.13
Pu240	0.35	0.12	0.55	0.38	1.44	0.54	0.53	3.09	0.78	0.37	1.15	0.55
Pu241	0.24	0.76	0.71	0.29	2.48	0.87	0.36	3.28	1.13	0.15	1.20	0.47
Pu242	0.13	0.29	0.41	0.18	0.74	0.41	0.27	3.10	0.67	0.08	0.24	0.18
U235	0.01	0.23	0.03	0.01	0.03	0.02	0.00	0.04	0.01	0.01	0.02	0.02
U238	1.03	1.98	3.39	0.82	3.45	2.52	0.25	2.59	0.84	0.95	3.55	2.77
TOTAL	1.21	3.65	4.41	1.09	6.66	3.58	1.08	12.83	4.32	1.13	5.66	3.48

Table 1. Unccertainties on three integral parameters and several fast reactors using new covariance matrix

A first analysis of these new results indicates that in general lower values are obtained with the new covariance values with the respect to those of BOLNA. This is mostly explained by uncertainties on MA and higher Pu isotopes. For this set of isotopes, while a better uncertainty has been provided for standard deviations, correlation values are not given. Therefore, a lower uncertainty is calculated. It is expected that in future releases of the covariance matrix this lack would be filled.

These results and the results of the nuclear data target accuracy assessment made in [1] indicate that a careful analysis is needed in order to define the most appropriate and effective strategy for data uncertainty reduction. It seems that a strategy of combined use of integral and differential measurements should be pursued in order to meet the requirements. Efforts in this direction are underway (see e.g. Ref. 4) and a new Subgroup has been established by the WPEC of the NEA-NSC in order to evaluate and compare different approaches and to make recommendations for future work in this field.

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JEFF-3.1.1 Nuclear Data Validation for Sodium Fast Reactors

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The JEFF-3.1.1 [1] Nuclear Data Library is the latest version of the Joint Evaluated Fission and Fusion Library. The complete suite of data was released in 2008, and contains general purpose nuclear data evaluations compiled at the NEA Data Bank in co-operation with several laboratories in NEA Data Bank member countries. JEFF-3.1.1 contains also radioactive decay data, activation data and fission yields data. It combines the efforts of the JEFF and EFF Working Groups who have contributed to this combined fission and fusion file. The library contains neutron reaction data, incident proton data and thermal neutron scattering law data in the ENDF-6 format.

The aim of this paper is to present the status of the validation of this library using the Monte Carlo Code TRIPOLI4.5 for fast reactor calculations. To reach that goal, we reanalyse a selected set of integral experiments performed in MASURCA Mock-up at CEA/CADARACHE, in ZPPR mock-up at INL USA and in SUPERPHENIX power Reactor. These experiments are:

- The CIRANO [3] program in MASURCA (1994-1997) was meant to extend the validation of ERANOS (code, schemes, data libraries) to Pu-burning fast reactors (CAPRA project) via the progressive substitution of fertile blankets by steel reflectors.
- The ZPPR10A experiment proposed in IRPhE of the NEA Data Bank. This experiment is complementary of the first one because sodium void effects have been measured and are available.
- Several experiments made during de commissioning of SUPERPHENIX that give information on different types of critical states of the core.

All these experiments are modelled with the TRIPOLI code to avoid most of the errors due to deterministic models and to focus only on the nuclear data biases. An example of the SUPERPHENIX core modelling is given on the figure 1.

Ongoing analysis shows the capability of the new JEFF3.1.1 nuclear library to predict the SFR neutronic behaviours. From this work and from the qualification work performed with ERANOS2 [4], some required improvements on nuclear data are highlighted. The results of this analysis will be given in the final paper.

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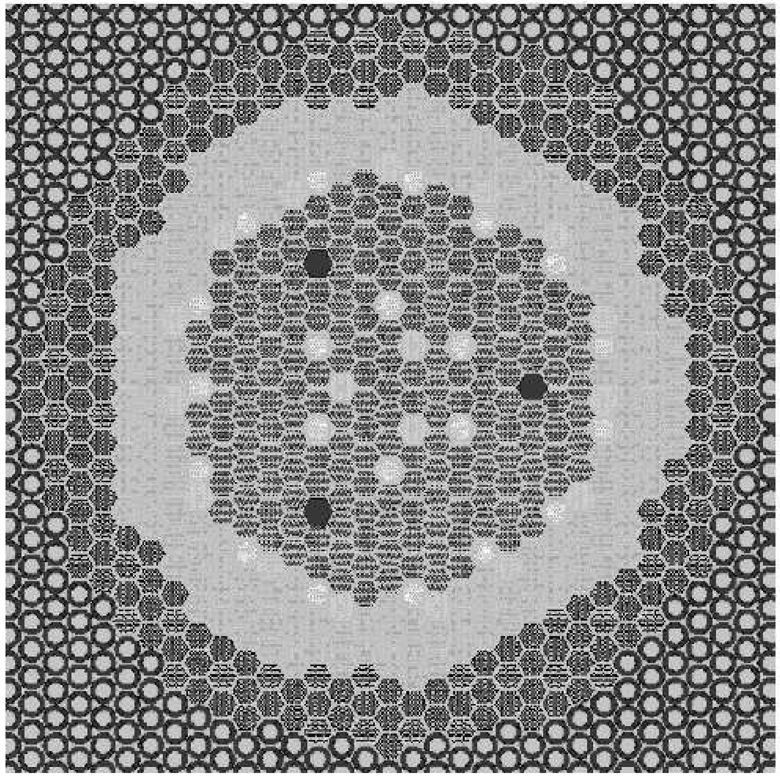

Figure 1: TRIPOLI4 model of SUPERPHENIX start up core

Sensitivity Coefficients for Fast Reactor Core Analysis

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Sensitivity coefficients of neutronics characteristics with respect to cross section changes are very useful for the evaluation of the effect of nuclear data change [1)], the cross section adjustment [2)] and the uncertainty evaluation[2)].

In the analysis of fast reactors, the sensitivity coefficients have to be calculated in multi energy groups to consider the cross section behavior in resonance energy range and fast energy range.

Therefore, it is desirable to develop a method to extend and/or condense energy group structures in calculating sensitivity coefficients.

In this paper, a method is developed to extend and/or condense energy groups, and the usefulness of the method is shown by numerical calculations in a fast reactor.

Let S^g and S^G be the sensitivities of a neutronics characteristics R with respect to the g-group cross section change $d\sigma^g$ and the G-group cross section change $d\sigma^G$. The superficies g and G denote the fine and broad energy groups. The broad group cross sections σ^G is expressed by using the fine group cross sections as follows:

$$\sigma^G = \sum_{g\in G} \phi^g \sigma^g \Big/ \sum_{g\in G} \phi^g , \qquad (1)$$

where ϕ^g is the g-group neutron flux. Using this relation between σ^G and σ^g, the sensitivity for the fine energy group, S^g, is related to that for the broad energy group S^G as follows

$$S^g = \omega^g S^G . \qquad (2)$$

Using the above equation, the sensitivities for the fine energy groups are calculated from the broad group sensitivity coefficients, when the fine group flux is given.

Furthermore, when the fine group sensitivities S^g are known, the broad group sensitivities S^G are given by

$$S^G = \sum_{g\in G} S^g . \qquad (3)$$

This equation is obtained by summing Eq. (2) over g included in the broad energy group G.

Equation (3) is an ordinary one, and Eq. (2) is a very convenient equation when calculating the fine group sensitivity from the broad group sensitivity.

Next, we treat the transport effect in sensitivity coefficient calculations. In fast reactors, the

neutron transport effect defined by the difference between transport and diffusion theory calculations becomes large.

Usually sensitivity coefficients are calculated based on the diffusion theory and may have some errors. So, we have developed the sensitivity calculation code SAGEPT by using the generalized perturbation theory based on the transport theory.

The validation test of the SAGEPT code has been performed by comparing the sensitivity coefficients calculated by SAGEPT with those calculated by the direct transport calculation.

Figure 1 shows an example of sensitivity coefficients calculated by SAGEPT. The sensitivity of sodium void reactivity, the reactivity change induced when sodium in a reactor core is voided, is shown together with the result of the direct transport calculation and that of the diffusion theory calculation. The result of SAGEPT agrees well with the direct transport calculation, and this good agreement shows the validity of SAGEPT. Furthermore, the diffusion theory calculation underestimates the sensitivity in absolute value by 20-30% in energy groups with resonance structures compared with the SAGEPT result. This shows the usefulness of the transport theory application to sensitivity calculations.

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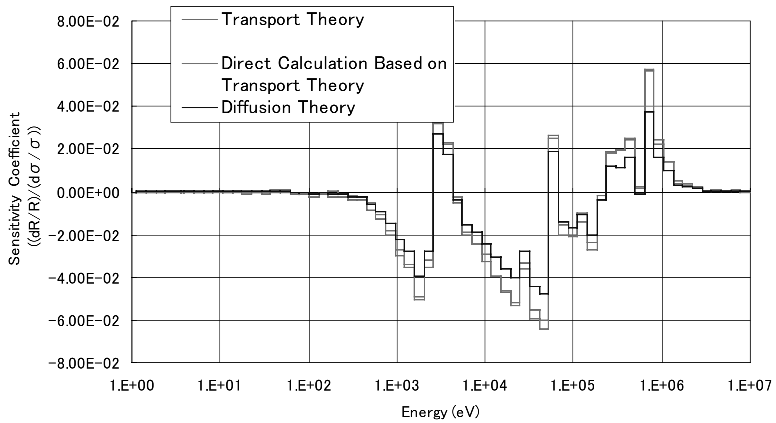

Fig. 1 Sensitivity of sodium void reactivity with respect to U-238 capture cross section

Parallel Session 6.2:

Fast reactor analysis: basic data, experiments and advanced simulation

REVIEW OF THE RECENT *FAST* PROJECT ACTIVITIES RELATED TO GEN-IV FAST REACTORS

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The FAST (**F**ast-spectrum **A**dvanced **S**ystems for Power Production and Resource Managemen**T**) project is an activity performed in the Laboratory for Reactor Physics and Systems Behaviour of Paul Scherrer Institut in the area of fast-spectrum reactor behaviour with an emphasis on the comparative analysis of the Gen-IV fast-spectrum systems. The project currently comprises 4 professionals, 1 post-Doc, 4 PhD and 3 MS students. The main purpose of the project is to create a centre of fast reactor competence in carefully chosen research areas in order to conduct studies within international frameworks aimed at safety enhancement of the fast-spectrum systems considered for construction in Europe. In more specific terms, the goal of the project is to develop and maintain ability to provide a unique expert analysis in the three main areas, namely neutronics, thermal hydraulics and fuel behaviour, with the use of the three main measures, namely: use of a unique computational tool, integration into international programs and organization of an efficient team, for the three Gen-IV fast-spectrum reactor systems, namely sodium-, gas- and lead-cooled reactors (see Fig. 1).

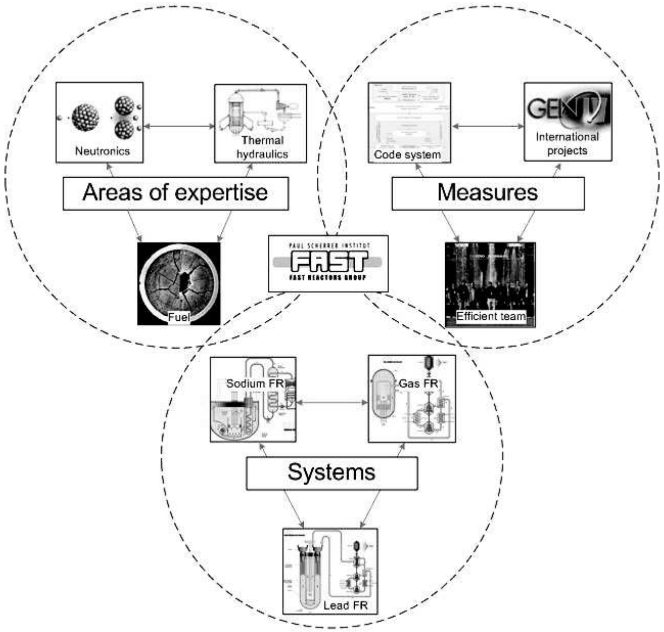

Fig. 1. FAST project concept

One of the main priorities of the project is to keep development and improvement of a calculational tool for simulating static and dynamic behaviour of the core and the whole reactor system of advanced fast spectrum concepts with different system configurations, core designs, fuel forms, coolant types,

etc. [1]. A code system of this complexity is particularly attractive in the context of core and safety-related studies of the advanced fast reactors being proposed by the Gen-IV International Forum (GIF). Using this code system, it is possible to analyse in a systematic manner a wide variety of both equilibrium fuel cycle conditions and transients. In addition, through the modelling of the whole reactor system, it is possible to assess those phenomena, which depend on the direct interaction between the core and primary/secondary systems. This code system (see Fig. 2) allows for comparing different aspects of the GIF systems, e.g. efficiency of burning minor actinides, behaviour in accident conditions, etc. Three important direction of developments of the FAST code system started in 2008 were 1) the elaboration of the new model for cross-section generation in transient analysis of fast-spectrum systems [2] (implemented in the PARCS code and coupled to the dedicated ERANOS procedure) as well as 2) the extension of the TRACE code to sodium two-phase flow simulation [3] (PhD study). The two latter studies together with the EQL3D analysis [4] of the sodium fast reactor equilibrium cycle [5] became an important milestone for the project, indicating the strengthening of the sodium fast reactor activities in the FAST project.

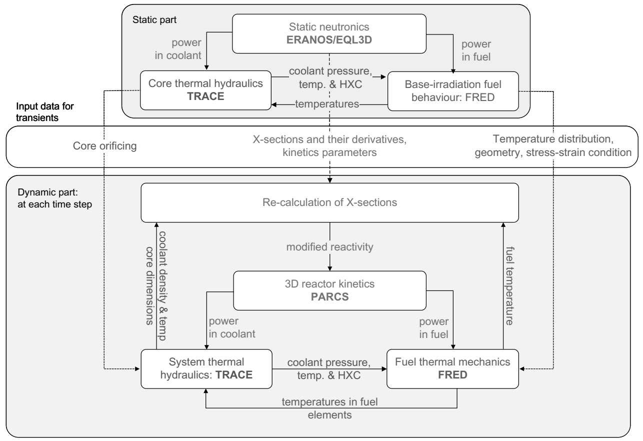

Fig. 2. FAST code system diagram

In order to provide a framework for the FAST project it is necessary to participate in international collaborations and in 2009 the project started to work in the EU FP7 ESFR project (European Sodium-cooled Fast Reactor). Since 2005 the FAST project has been participating also in the GIF Gas Fast Reactor (GFR) project both through a Swiss national contribution and as part of the EU FP6 Gas Cooled Fast Reactor [6]. The national contribution was performed in collaboration with the CEA through a CEA/PSI GFR bilateral agreement. In addition, the FAST project represents PSI on the GIF GFR design and safety project management board and the GFR steering committee. The examples of the work recently performed on gas cooled fast systems are: 1) 3D analysis of gas-cooled fast reactor core behavior in control assembly withdrawal accidents [7] (PhD study); 2) Development and application of an advanced fuel model for the safety analysis of the Gen-IV gas-cooled fast reactor [8] (PhD study); 3) Heavy gas injection in the Gen-IV gas fast reactor to improve decay heat removal under depressurized conditions [9] (PhD study); 4) Preliminary design of a Brayton cycle for an autonomous decay heat removal in the Gas Fast Reactor [10] (PhD study); 5) Calculational investigation in support of the low-temperature gas cooled fast reactor design.

The paper presents overview of the FAST project concept as well as brief descriptions of goals and main results of the specific studies mentioned above.

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K. Mikityuk

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Coupled Multi-Physics Simulation Frameworks for Reactor Simulation: A Bottom-Up Approach

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1. Introduction

Fast reactor cores rely heavily on coupling between neutron transport, heat transfer, fluid flow, and structural mechanics. Simulating the behavior of fast reactor cores from first principles will only succeed if this coupled behavior is accounted for in the simulation. Reactor core simulation has been performed for decades, and has resulted in many codes and modeling techniques validated with regulatory agencies worldwide. More recently, codes have been developed for many of the constituent physics (CFD, neutron transport, etc.) which scale to the largest parallel computers. Modeling fast reactor cores will require coupling or comparison between both types of codes. This coupling will need to account for the design-level resolution of the physical domain and solution data defined there.

Coupling high-fidelity domain models and simulation data produced by both existing and new physics implementations is accomplished using an integrated simulation framework. Previous approaches to framework design have been to develop data structures for the mesh and solution data and services which operate on those data structures, requiring the various physics modules to access those data structures directly. However, this type of "top-down" approach requires source-level access to the physics modules, which must be modified for the new data structures. This is a severe limitation for fast reactor modeling, due to both the large body of existing codes already validated for reactor applications which would need to be re-written, and because of the difficulty of using proprietary, non-open codes with this approach.

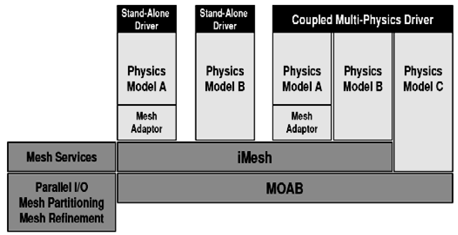

Figure 1: SHARP framework for coupled multi-physics simulation.

In this paper, we present a "bottom-up" approach to multi-physics frameworks, where we first develop common interfaces to simulation data, then adapt existing physics modules to communicate through those interfaces. Interfaces are provided for geometry, mesh, and field data, and are independent of one another; a fourth interface is available for relating data between these interfaces. Physics modules read and write data through those common interfaces, which also provide access to common simulation services like parallel IO, mesh partitioning, etc.. Multi-physics codes are assembled as a combination of physics modules, services, interface implementations, and driver code which coordinates calling these various pieces. The framework being constructed as part of this effort, referred to as SHARP, is shown in Figure 1.

2. Simulation Services Connected to the SHARP Framework

We are developing and/or assembling the various services required to perform end-to-end simulation of fast reactor core behavior and connecting them to our framework:

Mesh interface: We use the MOAB mesh database. MOAB provides a complete implementation of iMesh, a common interface to mesh data being developed by the ITAPS consortium in the US. MOAB supports parallel load/save of mesh and associated data, and communication of mesh and data in parallel.

Mesh generation: Several methods are supported for performing mesh generation. First, the CUBIT toolkit is used to develop geometric models and generate mesh for individual core assemblies, and optionally for whole-core meshing. The MeshKit library is also being developed, to provide meshing algorithms like copy/move/merge (for assembling a whole-core mesh from individual assembly type meshes) and mesh extrusion (projection of a 2D mesh into the third dimension).

Solution transfer: Parallel solution transfer between meshes, crucial for multi-physics modeling of fast reactors, is being developed in MOAB. Solution data at mesh vertices or elements in a source mesh can be interpolated onto a target mesh of the same or different element types, according to linear or spectral basis functions. Source and target meshes can be distributed across the same or different groups of processors. This service scales well to at least 32 processors. The overall organization of this service will easily accommodate other interpolation bases and normalization techniques.

Visualization & data analysis: Visualization and data analyses are critical parts of the simulation process, and rely on access to both mesh and field data. The Visit visualization tool has been enhanced to read mesh and solution data from iMesh, and is compatible with MOAB through that interface. Visit has been shown to scale to thousands of processors, visualizing data from analyses run on hundreds of thousands of processors.

Other capabilities: Other capabilities compatible with the SHARP framework, but not described in detail here, include parallel partitioning, parallel IO, geometric modeling and relating geometry to mesh data. These are all useful services for various types of multi-physics reactor analysis.

3. Physics Capabilities Being Coupled to SHARP

As part of the SHARP project, several physics codes at ANL have already been modified to read mesh from the iMesh interface, including:

Nek (CFD): Nek models CFD using a spectral element method. Meshes are relatively coarse, with the basis function in each element supported on a grid of NxNxN points. Nek is written in Fortran 77, and scales to over one hundred thousand processors. Nek has been modified to read mesh from the iMesh interface. Quadratic elements (27-node hexes) are used as input, and the N^3 spectral points are generated using a quadratic fit for bounding edges and faces.

UNIC (Neutron transport): UNIC models neutron transport using several discretization methods, including spherical harmonics, discrete ordinates, and method of characteristics. UNIC can operate on linear or quadratic tetrahedra or hexahedra. UNIC is written in Fortran 90, and scales to tens of thousands of processors. UNIC has been modified to read mesh and boundary conditions through iMesh.

Other physics codes have been connected to MOAB/iMesh which are not part of the SHARP effort. These codes include MCNP-DagMC, a CAD-based radiation transport code; and Cooper, for computing radiation hydro-dynamics. These codes each demonstrate a different type of integration with SHARP, but are not described further here.

4. Future Plans

In the area of physics, an integrated multi-physics code modeling both CFD/thermal-hydraulics (based on Nek) and neutron transport (UNIC) will be developed this year, with structural mechanics incorporated later. Enhancements to the solution transfer service will be made to improve accuracy and conservation options in the service. Continuing efforts will be made on expanding the types of physics coupled to SHARP, to improve the accuracy and conservation options available for solution coupling, and to improve scaling and parallel performance of the resulting codes.

Computational software package for analyzing the fast neutron reactor safety: Its improvement and development prospects

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Currently, ROSATOM institutions (and RF SSC IPPE first of all) are developing a software package executing major fast sodium-cooled neutron reactor safety analysis. This study field includes subroutines which permit to carry out 3-D calculation runs for inter-related unsteady neutronics, thermohydraulics, thermomechanical in-reactor and whole-facility accident processes involving sodium boiling, in-core steel and fuel meltdown and relocation with account of multi-component multi-velocity thermally-unsteady state models, in-core corium containment, reactor pressure vessel and in-core structure strength under emergency conditions, software for computing sodium fire and in-SG sodium-water interaction consequences, on-site radioactive contamination propagation and estimating the feasible NPP near-site population irradiation.

The paper does not discuss the software purposed for analyzing technological aspects of fast neutron reactor safety (e.g. sodium combustion, sodium-water interaction in a steam generator) and software for emergency state reactor neutronics parameters. The paper issue is an integral software package needed to carry out whole in-core and NPP thermohydraulic processes computation.

Named computational codes were developed after large experimental data array which have been acquired during several decades at the domestic and foreign reactors and test benches and thereafter accumulated from public domain publications available now. These experimental data constitute the analysis base; they have been implemented to create so-called flow pattern diagrams, closing relations of the computational codes. These computational codes have been proposed by the RF SSC IPPE and have been successfully implemented when reviewing and processing experiment run data acquired in the course of the emergency/transition state studies at BR-10, BN-350, BN-600 reactors, domestic facilities, and after analysis of some dedicated foreign reactor and facilities (e.g. SUPER PHOENIX, RAMONA, TREAT and some others). These computational codes have been implemented for international fast reactor emergency benchmark studies executed by specialists of many nations.

Therefore, available software stock could create an initial base for development of new improved computational codes. Starting from this point, one should provide a modern computational complex for developing new NPP generation, for validating their safety, and for ensuring best performance characteristics of extant nuclear power units which are under operation now. These computational codes should implement all modern IT achievement, incorporate integrated high-speed codes, enhance calculation analysis efficiency and quality, reduce labor input needed for acquiring final computed data. The code package will provide considerable economic effect both by means of selecting more optimal design and project solutions for the next nuclear power unit generation as well as by improving safety and reliability of the nuclear power units which are under operation now. This code development activities will foresee IBRAE Russian Academy of Sciences - Nuclear Safety Institute participation as well.

Some project and design solutions proposed for NPP with fast reactor, Generation IV, ingress outside of available practice and experience, therefore some additional experimental study runs are needed to examine different fast nuclear reactor safety issues. These experimental studies should be purposed both for validating new project decisions and for verifying improved computational codes which will be implemented to review safety aspects of the nuclear power units.

Rational development of the new future computational codes for ensuring new NPP with fast neutron reactor design, their safety validation, and related experimental studies will be carried out in several directions as follows:

- development of improved computational codes for providing reactor facility and whole NPP design and project activities based on modern technology approach which takes into consideration detailed complex review of neutronics, thermohydraulics, physical and chemical processes, thermomechanical features of a reactor and other reactor facility components;

- code development to optimize project and design solutions for the NPPs under development from a nuclear safety point of view;

- integral computational code development for reviewing and validating safety of the NPPs with fast neutron reactors;

- development of computational tools for providing information maintenance in the course of operation, damage diagnostics, residual operation lifetime estimation of the reactor core components and reactor facility equipment;

- experimental safety studies on sodium-cooled fast neutron reactors of Generation IV.

Abovementioned development of the computational code complex is an urgent task for creating new technological platform of our domestic nuclear engineering.

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Neutronics Code Development at Argonne National Laboratory

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As part of the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program of U.S. DOE, a suite of modern fast reactor simulation tools is being developed at Argonne National Laboratory. The general goal is to reduce the uncertainties and biases in various areas of reactor design activities by providing enhanced prediction capabilities. Under this fast reactor simulation program, a high-fidelity deterministic neutron transport code named UNIC is being developed [1]. The end goal of this development is to produce an integrated neutronics code that enables the high fidelity description of a nuclear reactor and simplifies the multi-step design process by direct and accurate coupling with thermal-hydraulics and structural mechanics calculations.

Current fast reactor analysis tools such as the fuel cycle analysis packages REBUS-3 [2] and ERANOS [3], and the safety analysis package SASSYS [4] contain neutronics packages built around multi-step averaging techniques (spatial homogenization and energy collapsing). These approximations vastly reduce the total space-angle-energy degrees of freedom required for nuclear reactor analysis and provide reasonably good solutions for most fast reactor design and analysis calculations. However, they have limitations in providing reliable answers for difficult reactor physics problems (e.g., the reactivity feedback due to core radial expansion). Additionally, it is desirable to reduce the uncertainties and biases in various areas of reactor design activities with the enhanced prediction capabilities that higher fidelity solvers provide. We therefore have a long term goal of replacing the multi-step averaging approximations by progressively more accurate treatments of the entire space-angle-energy phase space with sufficiently fine-grained levels of discretization. Given that high-fidelity transport calculations are not required in all areas of reactor analysis, we also desire an analysis tool that can allow the user to start at the current level of reactor analysis and transition smoothly (i.e. with familiar input/output) to less crude homogenization approaches and eventually to the fully heterogeneous descriptions. In this way, we intend UNIC to allow a reactor analyst to choose the desired level of approximation appropriate for their computational resources and analysis goals. This transition fundamentally assumes corresponding improvements in computing capabilities and continuous modernization of the code package to be compatible with the new computing technology.

To date, we have invested a significant amount of effort in creating large scale parallel modeling tools for the coupled simulation aspects of this project. More recently we have also started updating many of the legacy tools already available through REBUS-3/DIF3D system. With regard to the neutronics component, we have focused on developing three solvers for the neutron transport equation: PN2ND, SN2ND, and MOCFE [1]. These solvers are based on the finite element discretization of the spatial domain and are intended to handle truly arbitrary geometries and solve very difficult reactor physics problems where the legacy tools are known to produce results of questionable accuracy.

PN2ND and SN2ND are based upon the second-order even-parity transport equation, where spherical harmonics are utilized in PN2ND and discrete ordinates are used in SN2ND for the angular approximation. These solvers were designed to utilize 100,000 processors or more on reasonably-large problems, although only the SN2ND solver has had good performance on such high processor counts. These tools will allow for less severe (sub-assembly and pin-level) homogenization schemes that are appropriate for coupled simulation studies. Guided by the recent successes with SN2ND, we have begun the more intensive and difficult process of developing a multi-grid preconditioner such that the time-to-solution can be reduced. Within the next year, we hope to produce a tool that is competitive with the existing, comparable sequential tools on small size problems, although the better parallel scalability in our tool should allow us to solve larger problems in a shorter amount of time.

The MOCFE solver is based upon the first-order transport method of characteristics and treats both two- and three-dimensional geometries. The three-dimensional MOCFE solver can be used to model the explicit geometry with high fidelity, but its slow computational performance currently limits its application for such problems. Consequently, we are currently focusing on developing massively parallel computational algorithms and hope to make better use of it in the future. In addition to this work we are also creating a two-dimensional capability in MOCFE in accordance with the development of a new multi-group cross section generation code MC^2-3 (based on MC^2-2 [5], which is discussed in a companion paper. Thus far we have integrated MC^2-3 with UNIC and we are currently working on replacing the relatively crude geometrical approaches in MC^2-2 with the higher fidelity capabilities of MOCFE. We will discuss the ongoing development of MOCFE and its recent achievements on parallel machines.

In summary, the application scope targeted for UNIC ranges from the homogenized assembly approaches prevalent in current reactor analysis methodologies to explicit geometry, time dependent transport calculations that are directly coupled to thermal-hydraulics and structural mechanics calculations in reactor accident simulations. The creation of a single solver that can perform all of these calculations and still be competitive with the wide range of analysis tools already in use is somewhat formidable, especially considering the limited amount of manpower dedicated to this project. Further details and our motivation surrounding the development of specific solvers will be discussed in the full paper.

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Status of ERANOS-2 code system validation for Sodium Fast Reactor Applications

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The aim of the paper is to present the ERANOS-2 [1] code system associated to the European JEFF3.1.1 [2] library, and there capabilities to predict the behaviours of Sodium cooled Fast Reactor applications by the analysis of various experiments in fast neutron spectra: reaction rates, criticality, PIE of irradiated separated-isotope samples, void coefficient reactivity measurements.

The JEFF-3.1.1 Nuclear Data Library is the latest version of the Joint Evaluated Fission and Fusion Library. The complete suite of data was released in 2008, and contains general purpose nuclear data evaluations compiled at the NEA Data Bank in co-operation with several laboratories in NEA Data Bank member countries. JEFF-3.1.1 contains also radioactive decay data, activation data and fission yields data. It combines the efforts of the JEFF and EFF Working Groups who have contributed to this combined fission and fusion file. The library contains neutron reaction data, incident proton data and thermal neutron scattering law data in the ENDF-6 format.

The new release of the European Reactor ANalysis Optimized code System, ERANOS 2.2 has been developed and validated to establish a suitable basis for reliable neutronic calculations of current, as well as advanced FBR cores of the GEN-IV International Forum. The latest version of the ERANOS code and data system, ERANOS 2.2, contains all of the functions required for reference and design calculations of the fuel cycle behaviours of Liquid Metal Fast Reactors (LMFR's), with extended capabilities for treating advanced reactor fuel subassemblies and cores of Gas Cooled Fast Reactors (GCFR's).

The analysis of several types of experiments with JEFF3.1.1, associated to ERANOS2.2, shows the good predictability of these calculation tools for criticality calculations and fuel inventory prediction. The first series of experiments used is critical core experiments, such as CIRANO or ZPPR10 [3], made in zero power reactors like MASURCA or ZPPR. The second series is more dedicated to fuel cycle behaviours: PIE experiments of fuel pin (TRAPU experiment) and separated-isotope samples irradiation (PROFIL [4] experiments) both in LMFBR reactor PHENIX.

From this Qualification work, some required improvements on nuclear data are highlighted, as well as the need for new specific integral experiments.

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Benchmark analyses for BN-600 MOX core with minor actinides

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The IAEA has initiated in 1999 a Coordinated Research Project (CRP) on "Updated Codes and Methods to Reduce the Calculational Uncertainties of the LMFR Reactivity Effects". The general objective of the CRP is to validate, verify and improve methodologies and computer codes used for calculation of reactivity coefficients in fast reactors aiming at enhancing the utilization of plutonium and minor actinides (MAs). For this purpose, three benchmark models representing different modifications of the BN-600 reactor UOX core have been sequentially established and analyzed, the benchmark specifications being provided by IPPE.

The first benchmark model is a hybrid UOX/MOX core [1], with UOX fuel in the inner core part and MOX fuel in the outer one, the fresh MOX fuel containing depleted uranium and weapons grade plutonium. The second model is a full MOX core [2], similar MOX fuel composition being assumed; a sodium plenum being introduced above the core to improve the core safety. The third model is analyzed in the paper. The model represents a similar full MOX core, but with plutonium and MAs from 60 GWd/t LWR spent fuel after 50 years cooling (thus assuming a so-called homogeneous recycling of MAs in a fast system). This option is the most challenging one (compared to those analyzed earlier in the CRP) as concerns the reactor safety since an increased content of MAs, in particular americium, and higher (than Pu239) isotopes of Pu leads to less favourable safety parameters. On the other hand, existing uncertainties in nuclear data for MAs and higher Pu isotopes may lead to relatively high uncertainties in the computation results for the considered model.

The benchmark results include core criticality at the beginning and end of the equilibrium fuel cycle, kinetics parameters, spatial distributions of power and reactivity coefficients provided by CRP participants and obtained by employing different computation models and nuclear

data. Sensitivity studies were performed at CEA and JAEA to better understand the influence of variations in different nuclear data libraries on the computed results. The obtained results are analyzed in the paper.

The benchmark participants are in agreement as concerns qualitative variations in safety parameters due to employing reactor grade plutonium and minor actinides (instead of the fuel considered earlier). In particular, the absolute value of the Doppler constant decreases appreciably, while the positive core sodium void effect increases. On the other hand, relative variations (due to using different computation models and data) in reactivity coefficients - including the Doppler constant - are found to be high compared to those observed for the models investigated in the past.

For the latter models, ULOF and UTOP analyses were conducted in the past in the CRP framework. They show how variations in the computed parameters may affect the results of transient simulations. Similar simulations, bit in a smaller scope (only for ULOF, while considering only three sets of parameters which relate to close to maximum, minimum and average values of the Doppler constant), were also performed for the core containing MAs. Their results are analyzed.

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Analysis of Core Physics Test Data and Sodium Void Reactivity Worth Calculation for MONJU Core with ARCADIAN-FBR Computer Code System

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A computer code system ARCADIAN-FBR has been developed by utilizing existing analysis codes and the latest Japanese nuclear data library JENDL-3.3 for the analysis of core characteristics of fast reactors [1]. In order to verify the applicability of ARCADIAN-FBR, the experimental data obtained in the MONJU core physics tests were analysed. In the analysis, the continuous-energy Monte Carlo code MVP in the ARCADIAN-FBR was used and the core characteristics, such as criticality, excess reactivity, isothermal temperature coefficient and control rod worth, were evaluated. The results of analyses indicated that the Monte Carlo code MVP using the nuclear data library JENDL-3.3 can predict the core parameters of MONJU with good accuracy [2]. The validity of MVP code for fast reactor core analysis was confirmed through the verification using MONJU experimental data.

As an application of ARCADIAN-FBR for the analysis of fast reactor core, the sodium void reactivity worth, which is an important parameter in the safety analysis of fast reactors was analysed for MONJU core. MONJU was shut down in 1995 and the operation was suspended over 13 years. During the long-term suspension, nearly half of ^{241}Pu changed to ^{241}Am with short decay constant. For the restart of MONJU, a number of initial MOX fuels were replaced by the another MOX fuels which contains higher plutonium enrichment. The sodium void reactivity worth is strongly affected due to the accumulation of ^{241}Am. Therefore, the effect of ^{241}Am accumulation on the sodium void reactivity worth was investigated for MONJU core.

Using the Monte Carlo code MVP, the sodium void reactivity worths were calculated for the two MONJU cores, one is the initial designed core without accumulation of ^{241}Am and the other is ^{241}Am accumulated core for the restart [3]. The sodium void reactivity worth was calculated when 100% of sodium within the wrapper tube in the active core was voided. Fig. 1 shows the sodium void reactivity worths for the two MONJU cores. As shown in Fig. 1, the sodium void reactivity worth of the ^{241}Am accumulated core is almost twice of the core without ^{241}Am accumulation. The threshold fission cross section and neutron build-upfactor of ^{241}Am significantly increase over the 100 keV. Therefore, the spectral effect due to voiding is significantly larger in the ^{241}Am accumulated core compared to the core without ^{241}Am accumulation. As a resulut of calculation, it was confirmed that the accumulation of ^{241}Am significantly influences on the sodium void reactivity worth and hence on the safety analysis of sodium-cooled fast reactors.

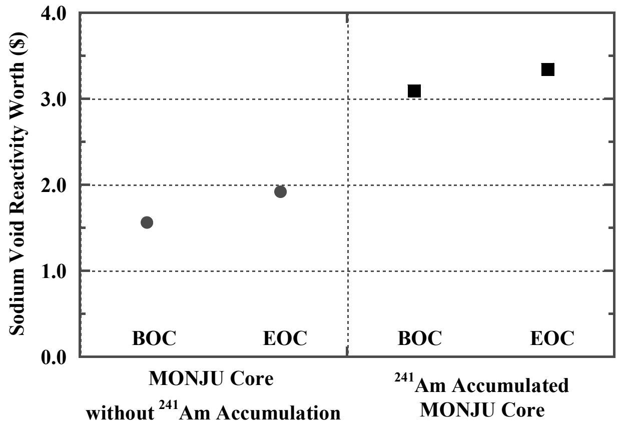

FIG. 1. Sodium void reactivity worth for MONJU core

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Parallel Session 6.3:

Fast reactor analysis: basic data, experiments and advanced simulation

Thermodynamic aspects of FeCr swelling under Helium irradiation

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Abstract - *Helium can accelerate the nucleation of cavities in FeCr based steels proposed as fuel cladding in advanced Gen-IV reactors. A detailed understanding of the thermodynamic aspects of Cr and He segregation is required to open the path to designing swelling resistant microstructures. Using atomistic simulations, we study Cr and He segregation at grain boundaries and surfaces as a function of Cr composition. We use a novel numerical approach based on a variance-constrained semigrand-canonical Metropolis Monte Carlo code and a new ternary FeCr-He potential based on the Fe-He and Cr-He potentials developed by K. Nordlund's group. We present preliminary results on FeCr swelling under Helium irradiation.*

I. INTRODUCTION

Fe-Cr alloys with 9-12% Cr content are the base matrix of advanced ferritic/martensitic (FM) steels envisaged as fuel cladding and structural components of Gen-IV reactors, and in future fusion power plant first wall and blanket structures. These steels show good mechanical properties and good resistance to swelling. However, Helium can accelerate the nucleation of cavities in FeCr based steels and a detailed understanding of the thermodynamic aspects of Cr and He segregation is required to develop the capability of designing swelling resistant microstructures.

We have developed a formulation of an empirical interatomic potential that incorporates the complexities of the thermodynamics of the FeCr system, adding He as a third element in the alloy, using results for Fe-He and Cr-He interactions developed by K. Nordlund's group. We use a variance-constrained transmutation ensemble[1] implemented in a massively parallel hybrid Molecular Dynamics/Metropolis Monte Carlo code[2] to study precipitation of He and Cr in grain boundaries.

II. THE INTERATOMIC POTENTIAL

A few years ago we proposed a formalism inspired in the CALPHAD approach[3] to describe the thermodynamics of alloys based on a Redlich-Kister polynomial description of the heat of formation of binary mixtures[4]. This formalism, the Concentration-Dependent Embedded Atom Model (CD-EAM), is well suited to describe multicomponent alloys as combination of binaries mixtures.

From the perspective of composition dependence, the case of FeCr-He is particularly easy since He is a chemically inert insoluble element whose description does not require a composition depndence. We therefore propose a ternary potential where Fe-Cr intersctions are composition dependent and Fe-He and Cr-He are just pair potentials. For the Fe-He and Cr-He pair potentials we use results from Nordlund et al.[5]

III. PRELIMINARY RESULTS

We present a study of the thermodynamic forces controlling Cr segregation at free surfaces and grain boundaries (GB's) by systematically exploring the Cr composition in thermodynamically equilibrated

samples containing free surfaces and twist GB's. Our results show how Cr is depleted at these defects, effect that for the case of free surfaces can be rationalized in terms of surface formation energies of Fe and Cr. Cr depletion is clearly observed in a surface-orientation averaged spherical sample. The figure below shows the surface composition versus bulk Cr composition at different distances from the surface. The figure shows Cr depletion at the surface at all compositions.

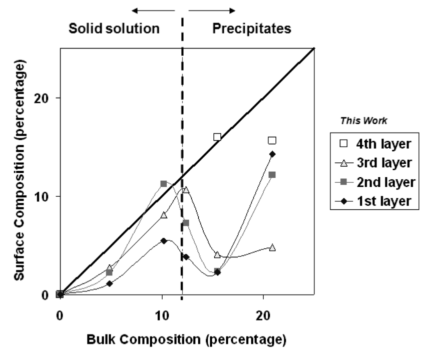

Fig. 1. Surface composition vs bulk Cr composition at different distances from the surface. The composition of a spherical FeCr single crystal shows Cr depletion at the surface at all compositions.

Preliminary results for Cr and He segregation at twist grain boundaries are currently underway.

IV. CONCLUSIONS

Our work represents a first step in the development of modeling capabilities to describe Cr and He segregation kinetic effects induced by radiation.

ACKNOWLEDGMENTS

The authors gratefully acknowledge Prof. K. Nordlund for making available his He potentials before publication. Work performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract DE-AC52-07NA27344.

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Turbulent liquid metal heat transfer along a heated rod within an annular cavity

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Many fast reactor systems are using liquid metals as coolant due to their neutronic properties and their relatively high boiling temperature and high thermal conductivity, which is expressed by a small molecular Prandtl number Pr. Especially the latter allows high core power densities at simultaneously small lateral temperature gradients, which is favourable in terms of reduced thermal stresses.

One of the most critical issues in the design of a fast reactor fuel assembly is a sufficient heat removal capacity to keep the cladding temperature within acceptable limits range. One coolant option are heavy liquid metals such as Lead and its alloys, e.g. the low melting $Pb^{45}Bi^{55}$ alloy which exhibits a high boiling point but a low kinematic viscosity ν. Although small viscosities yield at moderate velocities already quite large Reynolds numbers the high thermal conductivity causes a scale separation of the thermal and the viscous boundary layer. For pure forced convective flows the energy transport is entirely determined by the velocity field and hence the temperature is acting as a passive scalar. This allows computing the temperature field using commercially available computational fluid dynamic codes (CFD), which assume an analogy of momentum and energy transfer the so-called Reynolds analogy, if an adequate numerical resolution is chosen, see [1].

However at high power densities and even moderate flow velocities a transition from a forced to a mixed convective flow occurs as the liquid proceeds along a single fuel pin because of buoyancy. Buoyancy effects alter the velocity distribution, the lateral heat transport and additionally change the statistics of the momentum and temperature field, which demands a sophisticated modelling of the turbulent heat fluxes. The onset of mixed convection is given if Re is of $O(Gr^{1/2})$ where Gr is the Grashof number, which is easily attained for heavy liquid metals even at large Re.

This article presents a combined experimental and numerical study of a thermally developing turbulent lead bismuth flow along a uniformely heated fuel pin simulator (d_r=8.2mm) in a circular tube. The study focuses on the detailed investigation of the mean and temporal data of the momentum and energy field along the fuel pin (length l=870mm) at a fixed $Re=7.7\cdot 10^4$ ($2m^3/h$) for Pr=0.022 and a given heat flux of $q''=40{,}6W/cm^2$. Within the experiment local temperature and velocity distributions are recorded by traversable thermocouples and a Pitot-tube.

The experiment shows that already at the axial position $z/d_r=26.2$ close to the pin buoyancy alters the shape of the velocity profile. As the fluid proceeds along the rod the pin adjacent buoyant velocity monotonically increases, due to the rising pin temperature, which is reflected by a rising turbulence intensity of the axial flow component, compared to a pure forced convective flow. This yields an increased axial turbulent heat flux and moreover an increased lateral energy transport. While for $z/d_r \leq 26.2$ the turbulent temperature fluctuations are

determined by the turbulence of the velocity field, in the axial range from $26.2<z/d_r\leq 96.6$ the wall near fluctuations increase mainly due to the fluid acceleration caused by buoyancy. For $z/d_r>96.6$ the dimensionless intensity of the temperature fluctuations saturate, which is an indication that the wall near velocity field is already dominated by buoyancy. This analysis is supported by the fact that the dimensionless fluid wall interface temperature, which represents the inverse of the Nusselt number, attains an almost constant value. Numerical investigation indicates similar observations related to buoyancy effect on velocity field. However, Results show that temperature field is less influenced by buoyancy, mainly due to the high thermal conductivity of the fluid.

The experimental data show that even at Reynolds number above $Re>5\cdot 10^5$ and at moderate mean pin powers along a single fuel pin a flow transition from a forced to a mixed convective flow occurs, which yields an enhanced the heat removal. Hence, the currently used Nusselt number correlations to design fuel assemblies, represent a rather conservative assessment. Although the occurrence of mixed and buoyant flow phenomena demands the quite challenging development of anisotropic turbulent heat flux modelling as the comparison of the numerical with the experimental data reveal, the performance of heavy liquid metal cooled reactors can be significantly enhanced and the safety features considerably improved if these tools would be available.

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Modeling of Thermal Stratification in Sodium Fast Reactor Outlet Plenums During Loss of Flow Transients

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Advanced, high-fidelity neutronic and thermal-hydraulic modeling capabilities are being developed under the SHARP simulations framework at Argonne National Laboratory. In addition to the efforts to develop new coupled core neutronics and thermal-hydraulics models, a project to couple robust systems models with CFD-based high-fidelity plenum models is also underway. The purpose of this coupling is to better characterize transient natural circulation flow patterns and thermal stratification in reactor plenums during low-flow events such as protected and unprotected loss of flow accidents.

The need for coupled systems and CFD models was highlighted in a recent comparative study between the SAS4A/SASSYS-1 safety analysis code used in the United States and the CERES safety analysis code used by CRIEPI in Japan.[1] The comparison was made for the purpose of verification of the CERES code. Comparisons of the steady-state conditions show very good agreement for both protected and unprotected loss-of-flow transients. During pump coast-down periods at the beginning of each transient, temperatures and flow rates continue to show excellent agreement. However, beyond the initial parts of the transients, where natural circulation behavior begins to dominate, the results for temperature and flow begin to show differences.

Thermal stratification in the outlet plenum has been identified as a contributor to the natural circulation flow discrepancies between the two codes. Figure 1 shows the impact of outlet plenum treatment on the IHX primary inlet temperature during a protected loss of flow accident. When a simple mixing model is assumed (no stratification, bottom curve) the initially colder outlet temperature from the core lowers the average plenum temperature that is seen at the IHX inlet. On the other hand, when a simple, three-layer stratified model is used (top curve) the IHX is only exposed to the top, hot layer during the first hour of the transient. By comparison, the two-dimensional plenum treatment computed by CERES shows an intermediate temperature profile.

Incorrect predictions of IHX primary inlet temperatures affect the predictions of natural circulation flow rates in the primary loop. This, in turn, has a significant impact on peak coolant and fuel temperatures observed during the transient. In the above comparison, peak core outlet temperatures following the scram differed by approximately 75°C, with the higher prediction slightly above nominal outlet temperatures.

Under the present work, the commercially-available computational fluid dynamics code Star-CD is being coupled with the SAS4A/SASSYS-1 system code. In this coupling approach, SAS4A/SASSYS-1 provides realistic boundary conditions, based on whole-plant responses during a transient, to a detailed, three-dimensional plenum model that can accurately represent flow patterns and thermal stratification. This paper will then demonstrate that capability by reevaluating the transient accident sequences described above and comparing the results with

the two-dimensional results from the CERES analysis. The coupled code capability will later be validated using primary system data from the Phenix end-of-life tests.

Acknowledgements

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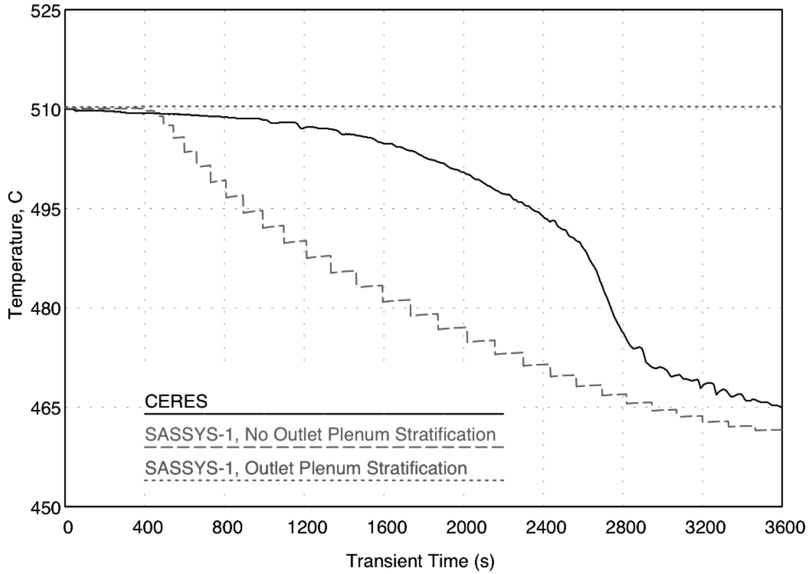

Figure 1: IHX Primary Inlet Temperatures With and Without Thermal Stratification in the SAS4A/SASSYS-1 Model Compared to the Results Predicted by CERES.

RANS Simulations of Turbulent Diffusion in Wire-Wrapped Sodium Fast Reactor Fuel Assemblies

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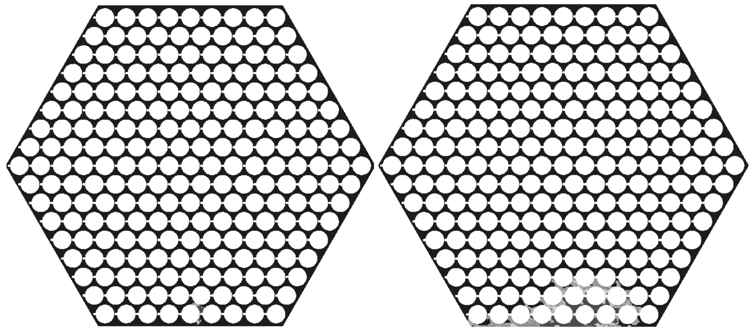
As part of a broader effort to develop an advanced, integrated multi-physics simulation capability for the design and analysis of future generations of nuclear power plants, the development of an integrated multi-resolution thermal hydraulic analysis tool package has been initiated. [1] Initial development efforts have focused on high-fidelity highly-scalable Direct Numerical Simulation and Large Eddy Simulation methods for prediction of detailed flow and thermal distributions in localized regions of a reactor core. [2]

To aid in prioritizing investment of resources and to begin to establish the mechanisms for communicating data between resolution levels, the range of applicability of each level of resolution is being evaluated through benchmark comparisons between the developed codes and the commercial RANS-bases CFD code Star-CD, beginning with simulations of the fueled region of a single wire-wrapped sodium-cooled fast reactor fuel assembly. [3,4] From a design perspective, one potentially significant advantage of higher resolution simulation of the fuel assembly is improved predictions of the exchange of coolant between individual flow channels, which is the primary mechanism for subchannel-to-subchannel heat transfer. Initial comparisons of Large Eddy Simulation predictions using the spectral element code Nek 5000 and Reynolds Averaged Navier Stokes predictions using the commercial finite volume code Star-CD suggest that the lower order RANS methods can likely be used to predict the hydrodynamic behavior within the assembly with similar accuracy to the LES simulations.

In this work, predictions of turbulent diffusion in a large wire-wrapped bundle using the commercial CFD code Star-CCM+ will be compared with legacy experimental data examining coolant mixing in a 217-pin wire-wrapped bundle. In the isothermal experimental studies, a saline solution is injected into a single channel of a full-scale assembly test section using water as a surrogate fluid. The diffusion of the saline plume is measured using conductivity probes placed at intervals throughout the assembly. The plume is modeled as a massless scalar in the CFD using several different turbulence models. Figure 1 shows the diffusion of the injected massless scalar at heights of 10cm and 70cm above the injection

point. Initial qualitative comparisons of the predictions with the experimental data show that the general trends of the measured concentration profiles are predicted.



(a) (b)

Figure 1. Predicted plume dispersion in a 217 wire-wrapped fuel assembly at a height of (a) 10cm and (b) 70 cm above the injection site.

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The Multi-Dimensional Analysis Method Development for KALIMER-600 using MARS-LMR CODE

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In this study, KALIMER-600 (Korea Advanced Liquid Metal Reactor), originally designed by KAERI (Korea Atomic Energy Research Institute) is described and analyzed by MARS-LMR CODE in multi-dimension for analysis of PDRC (Passive Decay heat Removal Circuit) performance and pressure drop. The sodium circulates mainly by the force of natural convection during operation of SFR (Sodium-cooled Fast breeder Reactor)[1]. In this phenomenon, the pressure drop occurs in the pool by lots of reasons such as friction and gravity caused by the structures and shape of the sub-channels (wire-wrap, spacer grid and etc) in the core region.

The important modeling features for the evaluation of natural circulation characteristics include a correct description of pressure drop and a heat transfer models for various situations.[2] In this paper, the friction factor *k* for each control volume is estimated from the calculation results of CFD (Computational Fluid Dynamics). For validation, CFD results were compared with the previous experiment data and are applied to provide the necessary parameters for MARS-LMR CODE.

With the aid of CFD results, MARS-LMR CODE is modified. The previous MARS-LMR CODE was constructed in one dimension giving a series of volumes and nodes. We have subdivided the UIS (Upper Internal Structure) into multi-dimensional shape and created 3-dimensional components for the parameters we are concerned about, to be more accurate in performing the analysis, to build more stable code, and to provide more reliable data.

In previous study, KALIMER-600, which is consisted with a lot of components in RV(Reacot Vessel) such as IHX(Internal Heat Exchanger, DHX(Decay Heat Exchanger), UIS(Upper Internal Structure) and Pump, was analyzed by MARS CODE in 1 dimension. Therefore, it is not able to describe some phenomena. For example, some part of the hot sodium which has passed core is met UIS, IHX or Pump. Thus, the heated sodium flow is changed because of those. However, in 1 dimensional analysis, the heated sodium has one way to hot pool except consideration of structure factors. On the other hand, the hot pool is modified to 3 dimension pool and considered representive components included UIS in muli-dimension analysis. The effect and contribution of this multi-dimensional analysis method is that it can suggest a sound base for the safety and performance assessment to show more reliable analysis. In addition, this result will be useful to understand not only pressure drop but temperure and velocity distribution in the pool.

The code is still under development. The hot pool in 3 dimension and the each compoments in 1 dimension do successfully input at code, whereas the junction of those parts are not accomplished. The connection of all components is only needed in the present.

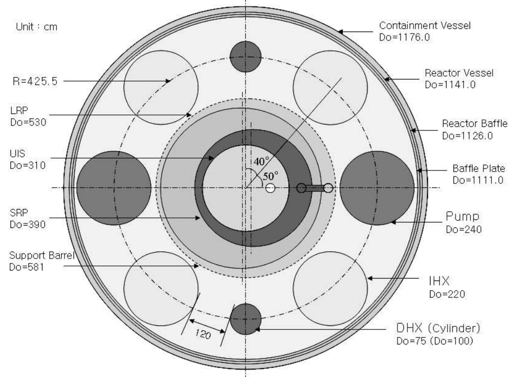

FIG 1.Top view of KALIMER-600 core desig [4]

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Parallel Session 7.1:

Advanced fast reactor fuels

Fast neutron reactor fuel cycle research programme at the Joint Research Centre

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Although uranium resources are large and diversified, they are not infinite. Considering the NEA/ IAEA Red Book estimated exploitable resources of 15 million tonne uranium, and the current annual consumption rate of about 67000 tonne uranium per year, the resources are sufficient for the next centuries. However, this optimistic view in term of resource utilization might be reversed if one considers the anticipated nuclear expansion. Thus, the deployment of a new generation of reactors (the Generation IV fast neutron reactors, FR) with a closed fuel cycle leading to a better use of natural resources needs to be prepared. Another sustainability aspect concerns the reduction of waste volume and hazards to be achieved by the recycling of all actinides.

Three main FR systems are being considered now in the international community, namely those cooled with sodium, helium or lead. Moreover, the molten salt reactor, in its fast neutron version, is also under development and considered for its sustainability potential using the thorium/uranium cycle. A large R&D effort is being undertaken at international level, and in particular in Europe. It is coordinated within the Generation IV International Forum, in which the Joint Research Centre (JRC) represents the EURATOM interests. But JRC also contributes directly, through its multi-annual research programme. The main emphasis in terms of research budget is put on the fuel cycle of the FR designs mentioned above. It includes the development of fabrication processes for oxide, nitride and carbide advanced fuels (which include plutonium and minor actinides), their materials property determination, and irradiation behaviour. Emphasis is also put on the property studies of potential molten salt fuels. Moreover, extensive reprocessing and partitioning (using aqueous and pyro-metallurgy processes) experimental studies are performed on irradiated fuels.

The paper will present a global overview of these programmes and of their results, as well as projections of their future evolution, considering our European dimension, and our contribution to the Sustainable Nuclear Energy Platform.

Comparative review on different fuels for Gen IV Sodium Fast Reactors: merits and drawbacks

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Sodium cooled Fast Reactor (SFR) is in France the best candidate for a prototype of GEN IV system to be built as early as 2020. Due to the important and satisfactory feedback experience built upon oxide fuels and its industrial maturity on the one hand, and to its full characterisation in normal and accidental conditions on the other hand, mixed oxide fuel $(U,Pu)O_2$ is considered as the reference fuel for the core of the ASTRID prototype.

The objectives followed for the next generation SFR in terms of safety (mainly related to the minimization of reactivity insertion risk) are achievable with an oxide fuel in a large power core (3600 MWth) while implementing adequate designs features.

Nevertheless, innovative studies performed at CEA show that the use of a dense and cold ceramic fuel such as carbide might even improve the core performances. Carbide, which is currently preferred to nitride, is indeed a candidate fuel worth to be considered for Generation IV Sodium Fast Reactors.

Focused on some other key design parameters (such as high breeding capability, safety, expected performances of the fuel cycle based on pyrometallurgical processes), several countries are considering the metal fuel for the SFR either as a long term reference or as a challenger to oxide fuel. In this context, conceptual studies have been conducted to compare the metal fuel to the others.

In the framework of the reactor studies performed at CEA with the support of AREVA and EDF, the main design direction is based on an oxide fuel pin core which, although quite innovative, is based on current fuel technology. Another direction, more challenging, is being followed with carbide fuel pin or plate cores. This direction will require significantly more long term R&D but is worth being investigated since it could give more flexibility to achieve enhanced safety characteristics while preserving fuel cycle sustainability for future SFR.

The COCONS approach has been used to evaluate the core behavior during conventional unprotected transient conditions for each type of fuel. This method appears to be a useful tool in terms of assessing new SFR core options.

The paper summarizes the current status of our analysis while comparing the basic features of oxide, carbide and metal fast reactor fuels from the viewpoints of the core performances, in pile-behavior and front/back end impact.

Results of post-irradiation examinations of inert matrices fuels irradiated in BOR-60 reactor up to 19 at% of burn-up in frame of Russian-French BORA-BORA experiment

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Within the framework of agreement between CEA (France) and Minatom (Russia) on fast sodium-cooled reactors the experiment BORA-BORA has been carried out on irradiation behavior study of different fuels with high plutonium content in the BOR-60 reactor. For the period of 1996 – 2007 the experiment covered fabrication, irradiation in BOR-60 and post-irradiation examination (PIE) of the experimental pins with oxides, nitrides and inert matrices fuels. The dismountable assembly design used for the fuel irradiation allows the intermediate unloading of some fuel pins for PIE and prolongation of other pins irradiation up the higher burn-ups. The first irradiation stage has been completed on October 2002, the second – on May 2005. The post-irradiation examinations of fuels after the second irradiation stage have been completed on 2007.

The results of non-destructive and destructive PIE of fuel pins with 40%PuN+60%ZrN and 40%PuO_2 +60%MgO after the second irradiation stage are presented in the paper. The non-destructive investigations did not show any anomalies in fuels condition. The distributions profiles of Co-58 and Mn-54 cladding activation products for all pins have been corresponded to the neutron flux distribution. The Zr-95 and Ru-106 activation product distributions testified the absence of some anomalies for fissile components axial distribution, of fuel column disruptions and of big metallic fission products inclusions. The Cs-137 distributions confirmed the absence of its migration from the core to the axial blankets.

The results of fuels microstructure study, of fuels density measurements, of fuels crystal lattice parameters measurements and fission products distributions study are presented.

Fabrication and Quality Control of MOX Fuel for Prototype Fast Breeder Reactor (PFBR)

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Uranium-Plutonium mixed oxide (MOX) fuel for both thermal and fast reactors have been fabricated by Advanced Fuel Fabrication Facility (AFFF) , Bhabha Atomic Research Centre, Tarapur, India . MOX fuel bundles fabricated by AFFF have been loaded in Boiling Water Reactors (BWRs) and Pressurised Heavy Water Reactors (PHWRs) and have been discharged after successful irradiation. An experimental fuel subassemby containing 37 MOX pins is being irradiated in Fast Breeder Test Reactor (FBTR) at Kalpakkam near Chennai and has seen a burn up of more than 80000 MWD/T. MOX fuel pins containing 44% PuO_2 have been recently loaded as a part of the hybrid core of FBTR.

AFFF has now taken up the manufacture of MOX fuel pins for the Prototype Fast Breeder Reactor (BHAVINI) coming up at Kalpakkam . The core consists of 181 sub assemblies containing 217 MOX fuel pins each. It is required to fabricate nearly 40,000 MOX fuel pins (3 meter long) for the first core. The Prototype Fast Breeder Reactor is designed with two different fissile enrichment zones to be loaded with MOX subassemblies with a nominal composition of 21% and 28% of PuO2.The fuel pellets of required composition are made using conventional powder metallurgy processes. The pellets are annular with an inner hole of 1.8mm diameter and outside diameter of 5.5mm. AFFF has developed the technology of making annular MOX fuel pellets for PFBR and optimized conditions of fabrication. Multistaion rotary presses have been used for compaction of the pellets. The fuel pin consists of a MOX stack of 1000mm and axial blanket of deeply depleted uranium dioxide of length 300mm on either side. New techniques have been used at different stages of fabrication of the fuel pins namely pelletisation, welding and wire wrapping. Studies have been made to use laser welding technique for welding of endplugs. Automation has been introduced in a number of process steps in the fabrication line.

A detailed quality control plan is prepared based on the specifications and advanced process and quality control procedures have been incorporated to meet the stringent quality requirement of the fuel going out. A number of new and advanced techniques have been used to improve the quality and the reliability of the finished fuel pins. Neutron well coincidence counting technique has been used as a process control to confirm the compositon of the mixed powder. An automated pellet sorting and inspection system has been developed for inspection of MOX pellets. Use of passive gamma scanning and X-Gamma Autoradiography (XGAR)and Passive gamma Scanning with an annular scintillation detector of the finished fuel pins increases the reliability of the final product . An ultrasonic technique has been developed to check the integrity of the endplug welds.

This paper presents the details of fabrication of the MOX fuel for PFBR and outlines the quality control/process control procedures followed at AFFF to ensure the quality of the fuel and describes the new and advanced techniques developed for the fabrication and quality control of MOX fuel for PFBR

VIBROPAC MOX -FUEL FOR FAST REACTORS -EXPERIENCE AND PROSPECTS

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SSC RIAR

Since the end of 1981 RIAR has been carrying out integrated investigations and developments to validate fuel cycle of fast reactors BOR-60, BN-350 and BN-600. These investigations and developments involve fabrication, in-pile tests and postirradiation examinations of fuel pins and FA containing granulated vibropac MOX fuel. Plutonium of various grades and levels of FP purification has been used for this fuel production (WG - weapon-grade; PG - power-generating plutonium including the one with low level of FP purification and MA additives).

In present-day design of vibropac MOX fuel pin the most significant parameters, determining reliability of the fuel pin, are getter additive in the form of 5-10%wt of metal uranium particles and cladding material of the fuel pin. [1, 2, 3, 4].

Experience in vibropac fuel tests in the BOR-60 reactor showed that vibropac MOX fuel had satisfactory life-time characteristics even at a superhigh burnup, upto 30%h.a.

All EFAs operated in the BN-600 reactor for the specified life in accordance with the test program. The maximum thermal loadings of fuel pins in the FA were in the range of 39.5-46.8 kW/m, the maximum cladding temperature was within 627-703 °C taking into consideration uncertainty of parameters. The maximum dwell-time of EFA in the core was 569 eff. days. The peak achieved burnup was 10.6 % h.a., damage dose – 80.9 dpa. No violations of safe operation limits were observed during experimental FAs irradiation in the BN-600 reactor.

Table presents information about the quantity of manufactured fuel pins and FAs, their purposes and types of granulated fuel used in these FAs.

Table. Summarized data on fabricated FA

Fuel type	FA purpose (reactor)	FA quantity, pieces	Quantity of fuel pins, pieces
$UPuO_2$ (WG,PG)	BOR–60	427	15799
$UPuO_2$ (WG)	BPhS	8	1016
$UPuO_2$ (WG)	BN-350	2	254
$UPuO_2$ (PG)	BN–600	4	508
$UPuO_2$ (WG)	BN–600	30	3810

In most fuel pins (see table 1) fuel contained a "dedicated" additive - getter in the form of metal uranium particles (3 – 10 %wt).
Along with mass fabrication of the fuel pins and FAs (table 1), RIAR has developed procedures of producing granulated oxide fuels and fuel pins for fast reactors of the next generation. These are fuel pins containing recycled PuO_2, residual fission products (up to 8%wt), recycled MOX fuel, MA additives NpO_2, Am_xO_y (2 – 5 %wt), high in PuO_2 (up to

45%wt), etc. The fuel pins with these types of fuel are irradiated in the BOR-60 reactor, and in most cases they are incorporated in dismountable FAs, which makes it possible to perform interim nondestructive inspection of the tested fuel pins [5].
From results of RIAR research and developments a decision was taken about creation of facility for granulated MOX fuel production and BN-800 vibropac FA fabrication.

Development of Np and Am bearing MOX fuels for Japan sodium cooled fast reactors

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Homogeneous mixed oxide containing minor actinides (MA-MOX) has been developed for Japan sodium-cooled fast reactors. The fuel contains maximum 5% MA in total in which Am and Np are contained mainly, and the oxygen-to-metal (O/M) ratio of the pellets is needed to control to low value of less than 1.97 for high burn-up of 150GWd/t. As a part of the development, an irradiation test of $(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{2.00-x}$ was conducted in the fast reactor *Joyo*. The purpose of the test was to evaluate the redistribution and the microstructure change as a parameter of O/M ratio. Measurements of physical properties[1-3] and establishment of the pellet preparation technique[4] were carried out for the irradiation test. Based on their data, the pellets of $(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{2.00-x}$ (x=0.02 and 0.04) were prepared and irradiated, and the irradiation behavior was analyzed[5]. In this paper, the thermo-physical properties, the basic data for pellet preparation and the irradiation behavior were evaluated for MA-MOX development.

(1) Physical properties

Theoretical densities, melting temperatures, thermal conductivities and oxygen potentials of $(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{2.00-x}$ were evaluated for thermal analysis of the irradiated fuels, which were shown in Table 1 together with the data of MOX. The effect of MA addition on their properties were not observed significantly[1-3].

(2) Pellet preparation technique

The pellets were prepared by mechanical blending method. The sintering behaviour of $(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{2.00-x}$ was investigated as functions of temperatures and oxygen potential during the sintering. It was found that the sintering in an atmosphere of Ar/5%H_2 gas mixture added misture was needed to obtain homogeneous pellets. And then the pellets were annealed in various atmosphere to adjust the O/M ratio, and the O/M ratio adjustment technique was established[4].

(3) Irradiation test
The pellets of $(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{1.98}$ and $(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{1.96}$ were prepared for the irradiation test. The theoretical density ratio of the pellets was 93%TD±2%. The fuels were irradiated at 427-432kW/cm for 10min and 24hr. The irradiated pellets were analyzed by ceramography and electron prove micro analysis[5].

(4) Irradiation behaviour
In the irradiated pellets of $(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{1.98}$ and $(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{1.96}$, columnar grain and central void were observed, and the content of Am and Pu increased at the pellet centre. No change of Np was observed. Thermal analysis results showed that maximum temperature of the $(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{1.98}$ pellet was about lower than that of another one. But the extent of the redistribution and the microstructure change were larger in the $(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{1.98}$ pellet. The local oxygen potentials in the radial direction of the pellets were evaluated.

In the pellets of $(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{2.00-x}$, thermo physical properties were investigated, preparation method for low O/M homogeneous fuels was established, and the high linear heat rate irradiation tests were carried out successfully in *Joyo*. The data related to the redistribution and the microstructure change were obtained.

TABLE 1. Physical properties of MA-MOX and MOX

	Theoretical density (g/cm^3)	Solidus temperature (K)	Thermal conductivity at 93%TD at 1273K (W/mK)	Oxygen potential at 1273K (KJ/mol)
$(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{1.98}$	11.08	3029	2.422	-567
$(U_{0.66}Pu_{0.3}Am_{0.02}Np_{0.02})O_{1.96}$	11.03	3040	2.150	-589
$(U_{0.7}Pu_{0.3})O_{1.98}$	11.06	3046	2.478	-572
$(U_{0.7}Pu_{0.3})O_{1.96}$	11.02	3062	2.194	-594

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Parallel Session 7.2:

Advanced fast reactor fuels

Manufacture of Core Sub-Assemblies and Fertile Fuel Assemblies for Indian Fast Breeder Programme

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Utilization of the vast reserves of thorium in the Fast Reactors has been the prime goal of nuclear power programme in India. Nuclear Fuel Complex (NFC) has successfully fabricated axial and radial blanket assemblies for the Fast Breeder Test Reactor (FBTR) containing thoria. Development of technologies for the manufacture of thoria pellets involved characterization of powder obtained by oxalate precipitation and calcination with the addition of small quantities of magnesium during oxalate precipitation [1]. This has resulted in achieving desired sintered densities. Though the magnesia doped thoria has yielded specified densities by sintering the pellets at ~1600^0 C, experimental works on activated sintering of the thoria powders containing small quantities of Nb_2O_5 has yielded similar densities when sintered in air at much lower temperatures. Doping of ThO_2 by Nb_2O_5 is expected to give rise to oxygen interstitials or vacancies. A higher valency additive like Nb_2O_5 and an oxidizing atmosphere has resulted in substantially lowering the sintering temperature while a lower valency additive and a reducing sintering atmosphere requires higher sintering temperature. Increase in the diffusion coefficient of thorium is likely to be responsible for activated sintering [2].

Several experimental works on compaction of thoria powder revealed that the desired green density could be achieved at a moderate pressure of 140 MPa. An increase in inter-particle contact is evident by an increase in the green density and also in the specific surface area as the compaction pressure is increased. The compactability and sinterability characteristics seem to be affected by long storage of powder. Ball milling of powder prior to pre-compaction and granulation is found, in addition to breaking the agglomerates, to restore the original characteristics of the powder [3]. The technique of thermal etching has been successfully used for examining the microstructural features of sintered ThO_2.

Different varieties of stainless steel are employed for the manufacture of intricate components required for Fast Breeder sub-assemblies which are not easily machinable. Large number of precision machined components are fabricated through specialized machining, forming and welding techniques and finally assembled with the help of special jigs and fixtures. Several fabrication techniques were developed like Clad Tube Crimping, Button Forming of Hexagonal Tube, Bead Forming of Spacer Wire, Welding of Clad Tubes to End Plugs and Hexagonal Tube to Foot & Handling Head. Specialized Joining Techniques like Pulsed Current GTAW are employed for the fabrication of thin-walled Fuel Elements.

Developmental works are also undertaken for standardizing manufacturing techniques for Oxide Dispersion Strengthened (ODS) alloys for clad tubes of Fast Breeder Reactors, which will have an edge over conventional materials with respect to excellent resistance to void swelling and irradiation embrittlement and also capable of operating under severe conditions for extended periods.

The paper highlights various developmental activities carried out for the manufacture of core sub-assemblies for the Fast Breeder Test Reactor (FBTR) and the forthcoming Prototype Fast Breeder Reactor (PFBR).

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Performance Evaluation of Metallic Fuel for SFR

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U-TRU-10%Zr metallic fuel is a reference fuel for SFR. The fuel rod of SFR consists of a metallic fuel slug and a liquid metal thermal bonding within a kind of HT9 steel, like the Integral Fast Reactor (IFR) fuel concept.

Fuel performance evaluation is being performed in the following tasks; the fuel-cladding diffusion couple tests, the fuel irradiation test, the fuel design and the performance analysis model development.

One of the important issues in the metallic fuel development is the eutectic melting of fuel with cladding. Diffusion couple tests of U-Zr-X together with ferritic martensitic steels (FMS) such as HT9 or T91 were carried out [1].

Candidate materials for the diffusion barriers are being selected to eliminate the eutectic reaction. Diffusion couple tests of U-Zr-X were carried out for the barrier materials such as Zr, Nb, Ti, Mo, Ta, V and Cr. Among these barriers, V and Cr exhibited the most promising performance. Fig. 1 shows that there was no interdiffusion between fuel and cladding material.

Another concern is how to apply the barrier to the inner surface of the cladding. We have tried several methods such as CVD, plasma coating, and electroplating. Electroplating was selected to be most competitive in terms of cost and applicability.

Fuel irradiation tests plans to be performed in the HANARO research reactor. The design of the irradiation capsule was completed in 2008, and the irradiation capsule is to be fabricated in 2009. Irradiation test will start in 2010. The composition of the fuel slug is U-10%Zr-(0, 6 Ce).

Since SFR metal fuel is being developed in connection with pyro-processing, RE and MA would be included in the fuel. In this case, RE is precipitated in the U-Zr matrix, which might cause a variation of fuel properties.

Under an assumption of a RE-rich phase forming a macroscopic mixture with the matrix, the thermal conductivity was estimated for U-Zr-RE alloy. The thermal conductivity model was verified with the experiments.

The radial fuel constituent migration is a general phenomenon in the metallic alloys. This phenomenon may affect the in-reactor performance of metallic fuel rods.

The constituent migration model was developed and installed into the MACSIS code to simulate constituent redistribution. The radial profile of Zr and Am redistribution was calculated for the SFR metallic fuel,

The probability of cladding failure or damage during the steady state and transient conditions must be evaluated by appropriate predictive codes. The MACSIS code has been developing for the design analysis of metal fuel in KALIMER-600 [2]. MACSIS is a metallic fuel performance computer code for a fast neutron environment. Main structures of the code consist of a temperature profile calculation routine, a swelling/FGR calculation routine, and a deformation calculation routine.

The fuel design limit was investigated by performing a sensitivity analysis of CDF limits according to the plenum/fuel volume ratio, and the cladding thickness.

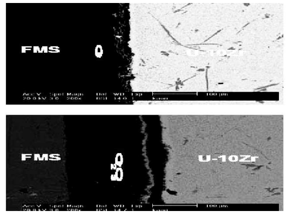

FIG. 1. Cr barrier vs. U-Zr diffusion couple test

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Development of Minor Actinide-Containing Metal Fuels

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Fast reactor metal fuels containing minor actinides (MAs) Np, Am and Cm and/or rare earths (REs) Y, Ce, Nd and Gd are being developed by CRIEPI in collaboration with JRC-ITU in the METAPHIX project. The test fuel pins including U-Pu-Zr-MAs(-REs) alloy rods in parts of the fuel stacks were fabricated and irradiated up to maximum burnup of ~10 at.% in the fast reactor Phénix. At present the post-irradiation examinations (PIEs) are in progress at JRC-ITU.

U-Pu-Zr-MAs(-REs) alloys of different composition were prepared for characterization experiments in advance of the test fuel pin fabrication. The miscibility of MAs in the U-19Pu-10Zr (wt%) alloy was considered to be quite high: 15wt% Np and 10wt% Am could be mixed homogeneously in the U-Pu-Zr alloy. On the other hand, Am-REs-rich precipitates were formed in the alloys if REs were present. The metallographic observation of various U-Pu-Zr-MAs-REs alloys showed that such precipitates are dispersed homogeneously in the alloy provided that the indivisial amounts of MAs and REs added are limited to less than 5wt%. Furthermore, the physical properties such as elasticity, thermal conductivity, solidus temperature or phase transition temperature of U-Pu-Zr alloys are practically unchanged after the addition of ≤5wt% MAs and ≤5wt% REs.

Based on the characterization results of the MA- and RE-containing alloys, standard U-19Pu-10Zr fuel stacks including segments of U-19Pu-10Zr-2MA-2RE or U-19Pu-10Zr-5MA/U-19Pu-10Zr-5MA-5RE alloy rods were fabricated for the irradiation experiment. A reference fuel pin of U-19Pu-10Zr alloy without MAs and REs was also fabricated and irradiated. The irradiation experiment was performed in the fast reactor Phénix (France) with the support of the Commissariat à l'Énergie Atomique (CEA, France). The irradiated metal fuel pins were discharged from the reactor at ~2.5at.%, ~7at.% and ~10at.% burnups in order to systematically examine fuel irradiation behavior and MAs transmutation rate. All the irradiations forseen in this study were completed by May 2008. The non-destructive examinatoins of the irradiated metal fuel pins revealed that no damage due to the irradiation had occurred.

The irradiated fuel pins up to ~2.5at.% and ~7at.% burnups were transported to JRC-ITU and are presently undergoing detailed destructive PIEs.

The results of the measurement of the axial distribution of gamma-ray intensity for the ruthenium isotope ^{106}Ru, which hardly moves in the fuel alloy, showed fuel stack elongations of 1.9-2.5% (9-12mm) and 3.5-4.1% (17-20mm) in ~2.5at.% and ~7at.% burnup fuel pins, respectively. Fuel elongation behavior was independent of MAs and REs additions.

Plenum gas analysis revealed that 45.7-51.3% and 63.1-68.2% of fission gases generated in the ~2.5at.% and ~7at.% burnup fuels, respectively, were released to the plenum; the total amount of fission gas generation was calculated using the ORIGEN-2 code. These results are consistent with reported data on the U-Pu-Zr test fuels irradiated in EBR-II [2] as shown in Fig.1.

The PIE results suggest that up to ~7at.% burnup fuel swelling and fission gas release of U-Pu-Zr fuels containing ≤5wt% MAs and REs are essentially the same as those of MA- and RE-free U-Pu-Zr fuels.

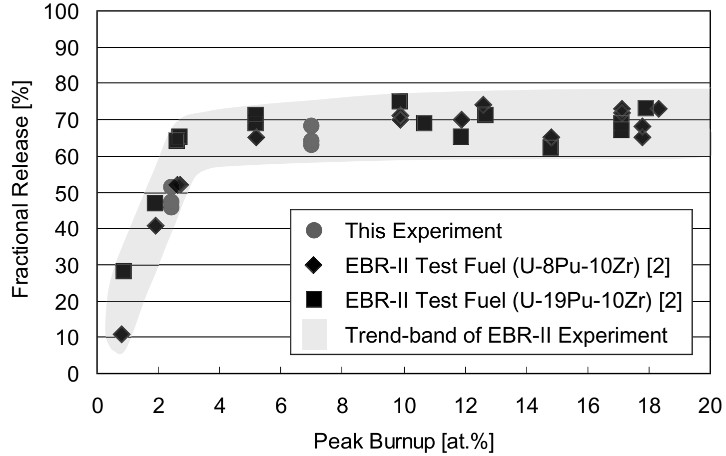

Fig. 1 Burnup dependence of fission gas release fraction.

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Carbide and Nitride Fuels for Advanced Burner Reactor

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Under the U.S. fast reactor program, reference and alternative 1000 MW_{th} Advanced Burner Reactor (ABR) core concepts were developed using ternary metallic (U-TRU-Zr) and mixed oxide (UO_2+$TRUO_2$) fuels [1]. Recently, mixed carbide and nitride fuels have been considered as fast reactor fuels on the basis of their high density, compatibility with coolant, high melting temperature, and excellent thermal conductivity although they are ceramic fuel like a mixed oxide fuel [2, 3]. Thus, the performance of the ABR core loaded with carbide and nitride fuels was evaluated in this study with an expectation that the carbide and nitride fuels can mitigate disadvantages of both metallic and oxide fuels in the ABR: favorable passive safety features in a severe accident compared to the oxide core, a higher discharge burnup compared to the metallic core, and a potential to increase thermal efficiency. All calculations performed in this study were focused on the neutronics characteristics, although the fabrication and irradiation experiences for carbide and nitride fuels are limited and some problems were observed in the reprocessing and irradiation of these fuels.

The mixed monocarbide and mixed mononitride fuels were selected as the alternative fuel forms and the ABR core concepts with these fuels were developed based on the reference 1000 MW_{th} ABR core concepts. For consistency, the potential design goals used in the reference ABR core concepts were also employed in this study: a 1000 MWth power rating, medium TRU conversion ratio of ~0.75, a compact core, one-year operational cycle length at least with a capacity factor of 90%, sufficient shutdown margin with a limited maximum single control assembly fault, and possible use of either metallic or any ceramic fuels in the same core layout.

The core layout and outer assembly dimensions of the reference 1000 MWth ABR core [1] were kept, but the intra assembly design parameters were varied to maximize the discharge burnup within the fast fluence limit of the HT9 cladding. The resulting design parameters such as active the core height (112 cm for the carbide fuel and 94cm for the nitride fuel), fuel volume fraction (35% for both fuels) are between the values of the metallic and oxide core concepts. The carbide fuel design adopted a low smeared density of 75% TD to avoid severe fuel cladding chemical and mechanical interfaces, while the smeared density of the nitride fuel was kept at the oxide fuel value of 85% TD. Sodium was considered as the thermal bond between fuel pellet and cladding. However, helium gas was also considered as the alternative option because helium is widely used in the oxide fuel.

Generally, the core performance parameters of the carbide and nitride fuels in the ABR core are between the values of the metallic and oxide cores. The neutron spectrum is softer than that of the metallic core, but harder than that of the oxide core. The fuel resident time of 60 months is longer compared to the metallic core, but shorter to that of the oxide core. The required TRU enrichment (25% for the carbide fuel and 23 % for the nitride fuel), average discharge burnup of ~100 GWd/t are also higher than that of the metallic core, but smaller than that of the oxide core. However, due to the high thermal conductivity (similar to the metallic fuel) and high melting temperature (similar to the oxide fuel), the margin to fuel melt

was significantly increased. This result indicates that the core could be more compact or the operating temperature could be increased further to reduce the plant cost. Compared to the carbide core, the nitride core requires more heavy metal loading to compensate the parasitic neutron absorption in N-14 and the increased neutron leakage from the shorter core.

The changed fuel composition affected the kinetics parameters and reactivity feedback coefficients, but the variations were minimal. The sodium void worth of the carbide and nitride cores are 5.9$ and 5.0$, respectively, which were smaller than that of the oxide core because of the increased neutron leakage. The Doppler constant (-1.3$ for the carbide core and -1.1$ for the nitride core) is smaller than that of the oxide core (-1.5$) mainly due to the lower average fuel temperature. The replacement of the bonding material by helium gas increases the fuel temperature of the carbide and nitride fuels by ~200°C, but overall impacts on the kinetics parameters and reactivity feedback coefficients were negligibly small. As a result, the carbide and nitride cores satisfy all sufficient conditions for core passive safety that were not met by the oxide core.

In conclusion, the core performance parameters and reactivity feedback coefficients of the carbide and nitride fuels in the ABR core were generally between those of the metallic and oxide cores. The good thermal conductivity and high melting temperature lead to a significant decrease in the average fuel temperature (comparable to metallic fuel), and hence provide large margin to fuel melt and favorable passive safety features without additional design fixes that were required in the oxide core concepts. However, including the limited experiences in fabrication and irradiation, the problems in reprocessing and irradiation require additional studies.

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Parallel Session 8:

Improvements in fast reactor components and system design

Design Concepts for Reactor Assembly Components of 500 MWe Future FRs

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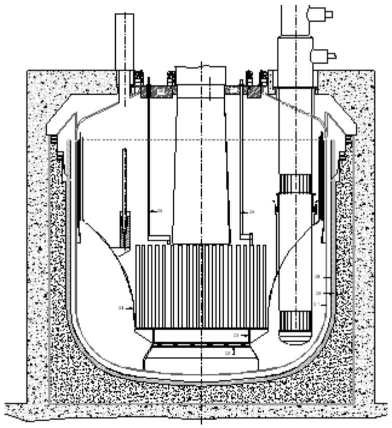
It is planned to construct four 500 MWe SFRs, as the logical followup of 500 MWe Prototype Fast Breeder Reactor (PFBR), with improved economy and enhanced safety. Among many measures to achieve economy, reduction of reactor assembly dimensions, component wall thicknesses and manufacturing cost is the main activity undertaken. Conceptual designs of reactor assembly components of various SFRs are critically looked at taking into account design and construction experiences. This exercise has yielded an innovative reactor assembly design, which is presented in the paper.

The improved design concepts are compact and symmetric welded grid plate without fuel transfer post, inner vessel integrated with fuel transfer post, thick plate rotatable plug, control plug integrated with small rotatable plug, torus roof slab, simplified fuel handling scheme with elimination of inclined fuel transfer machine, conical support skirt for reactor assembly with optimum support location to minimize the seismic moments and safety vessel made of carbon steel integrated with reactor vault liner. The improved design concepts have indicated significant economic advantages. To confirm the design, detailed thermal hydraulics and structural mechanics analyses are carried out and optimum dimensions and structural wall thicknesses are arrived at, to comply the design code RCC-MR (2007).



The paper discusses the basis of conceptual designs giving due considerations for emerging design concepts, analysis backups and further R&D required.

Fast neutron reactor plant equipment upgrading

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Owing to BN-350 and BN-600 operation, as well as BN-800 development, extensive experience has been accumulated for the entire equipment of fast neutron reactor plants. Possible technical solutions aimed at equipment upgrading to enhance safety and technical-and-economic indices of BN-1200 RP next generation have been selected on the basis of the analysis of available positive experience in fast neutron reactor development and operation in view of necessary R&D work.

Main circulation pumps of primary and secondary circuits

BN-800 and BN-1200 MCP-1 design is based on the decisions checked during long-term operation as a part of BN-600 RP (submerged centrifugal pump, bottom hydrostatic bearing). The main purpose of BN-1200 MCP-1 design upgrading is increasing of MCP-1 replaceable parts lifetime as compared with BN-800 design, which is first of all determined by impeller cavitation wear. To increase the impeller lifetime from 4.5 to 16 years, it is planned to change over to decreased rotation speed, to optimize the flow path and to strengthen its surface. MCP-2 design is also based on the decisions checked during long-term operation as a part of BN-600 RP (vertical centrifugal single-stage pump with single-sided suction impeller with bottom hydrostatic bearing and free sodium level in the pump). Except flow rate increase, BN-1200 MCP-2 characteristics do not practically change as compared with BN-800 MCP-2. Planned are the experimental studies of flow paths using scaled models, tests of first-of-a-kind MCP-1, and 2 samples with motor drives at water test facility.

Intermediate heat exchanger

The design solutions accepted for BN-800 and BN-1200 IHX are confirmed by their reliable operation in BN-600 reactor under similar conditions. The design is characterized by the central downcomer pipe for secondary circuit sodium and straight heat exchange pipes with compensating bends. During inspection of one of six IHX withdrawn from the reactor after 25 years of operation, no changes are revealed that can prevent their further operation. Practical absence of heat-exchange pipe material corrosion and wear at spacing points was confirmed. It allows reduction of heat-exchange tube thickness from 1.4 mm to 1 mm as applied to BN-1200 reactor and simultaneous increase of service life up to 60 years. This IHX design does not require additional experimental studies.

Control rod drive mechanisms

BN-800 CRDM design is substantially based on the solutions adopted for BN-600 reactor and is confirmed by operation experience. The basis for BN-800 CRDM design upgrading is application of step motor that simplifies the kinematic scheme and improves its reliability. BN-800 and BN-1200 CRDM designs are similar. The main difference is large amount of CPS control rods (~50 kg instead of 15 kg) that increases control rod mass and drive power. As per the requirements of regulatory documentation, pilot samples shall be tested and test facilities shall be correspondingly upgraded.

Built-in filter trap

In contrast to BN-600 and BN-800 designs, it is decided to arrange the cold filter traps in BN-1200 reactor vessel directly in order to eliminate the outer pipelines and, thus, to avoid radioactive sodium leaks. In this case the systems for servicing of the outer sodium circuit are eliminated. Two types of cold filter traps will be used – with recuperator (during reactor power operation) and without recuperator (at shutdown reactor). The main difference from BN-800 cold filter trap is the design features due to cold trap filter arrangement in reactor vessel. Both the mockup and pilot sample shall be tested at sodium test facility.

Air heat exchanger

As compared with BN-600 RP, BN-800 RP has the system with air heat exchanger to remove residual heat. BN-1200 AHX design is completely based on the solutions made for BN-800 AHX. The heat exchange surface is assembled of finned tubes. The only difference is in number of sections and removed heat capacity. The service life is increased from 40 to 60 years. The experiments are performed within BN-800 studies.

Steam generator

Section-modular SGs operate as a part of BN-600 RP. Similar SG is use in BN-800 RP also. These SG show high reliability and allow isolation of individual section in case interloop leak without reactor power decrease. However, this concept has significant disadvantages due to high specific metal consumption and branching network of pipelines with valves. As for BN-800 SG, there are no sodium reheat modules, and BN-1200 reactor design is a casing-type SG concept, which consists of two sections in each loop. In contrast to BN-600 and BN-800, BN-1200 SG is made of chromium steel. Application of new structural material allows reduction in heat exchange tube thickness from 3.0 mm to 2.0 mm with SG service life increase from 20 to 30 years. Application of casing-type design and increase of section power allows improvement of mass and size characteristics of SG and RP. The experiments will be performed at the sodium test facilities using models and tube bundle fragments.

Manufacturability

Well-developed cooperation of enterprises was established to fabricate the equipment for previous fast reactor designs. This cooperation is practically available till now. Most of enterprises have personnel experienced in this technology owing to following-up activities in BN-600 operation and upgrading, as well as participation in Chinese CEFR development and current BN-800 constructing.

During BN-1200 RP equipment fabrication, same manufacturers can be involved, who can accumulate additional experience in fabrication of similar BN-800 RP equipment by that time.

Development of elevated temperature structural design methods to realize compact reactor vessels

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Realization of a compact and simple reactor vessel (RV) is one of main issues in fast reactor cycle system technology development (FaCT) project[1]. Its size has a large impact on construction costs of plants, because it affects the layout of surrounded components and the building size. Main design requirements for the RV are 550deg operating temperature, 60 years plant life, low pressure and low dose irradiation. Under above conditions, main loads of reactor vessels are thermal stresses induced by fluid temperature change at transient operation as in the figure 1. Structures respond to them with elastic plastic creep deformation under elevated temperature conditions. It can induce incremental deformation and creep fatigue crack at critical portions around the sodium surface, thermal stratification layer and core support structures. Those phenomena are so complex that design evaluation becomes sometimes too conservative. Conventional reactor vessels have therefore wall protection systems against thermal loads, which enlarge diameter of vessels. Material is 316FR austenitic steel. The object of this study is development of precise elevated temperature structural design methods[2] to remove unnecessary protection devices from reactor vessels.

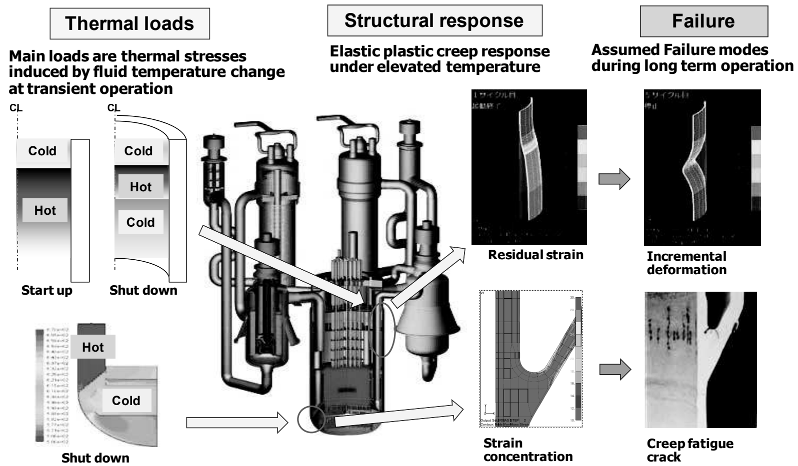

Figure 1 Assumed failure modes and their mechanism of reactor vessels

In order to improve total accuracy of design evaluation procedures, new methods are proposed with considering characteristics of reactor vessels for thermal load modeling, structural analysis and strength evaluation.

GUIDELINES FOR THERMAL LOAD MODELING

One of main difficulties of thermal load modeling is their inducement mechanism by interaction between thermal hydraulic and structural mechanics. Design evaluation requires envelope load conditions with considering scatter of design parameters. Proposed guidelines enable precise load modeling by grasping sensitivities of thermal stress to design parameters including thermal hydraulic ones.

GUIDELINES FOR INELASTIC DESIGN ANALYSIS

Guidelines are proposed to apply inelastic analysis methods for design of reactor vessels. There are so many influence parameters in inelastic analysis that conservative and unique solutions are hardly found. To overcome such difficulties, mechanism and main parameters of inelastic behaviors of reactor vessels were clarified. Guidelines give conservative results within the same mechanism as expected reactor vessels.

ELEVATED TEMPERATURE STRENGTH EVALUATION METHOD

Incremental deformation and creep fatigue strength evaluation methods were proposed. Accumulated strain is limited within no influence of fatigue and creep-fatigue strength. Taking design conditions of reactor vessels into account, creep fatigue evaluation considers strain concentration and an intermediate stress hold effect on creep-fatigue strength. Influences of thermal aging were also confirmed.

AKNOWLEDGEMENT

Present study includes the result of “Development of elevated temperature structural design method for fast reactor vessels and failed fuel detection and location system” entrusted to JAEA by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT).

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Challenges and Innovative Technologies On Fuel Handling Systems for Future Sodium Cooled Fast Reactors

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Fast Reactors have a unique capability as a sustainable energy source in terms of both utilisation of fissile material for energy production and minimisation of the nuclear waste, due to the hard neutron spectrum. As a result of a screening review of candidate technologies and in the frame of the international forum Generation IV, Sodium Fast Reactors (SFR) are amongst the selected systems to address the sustainability issues with a coherent set of innovative requirements. The guidelines for the definition of such innovative requirements are the Generation IV goals with significant improvements on economy, safety, environment, waste management and proliferation resistance as promising milestone towards a sustainable nuclear energy.

CEA, Areva and EDF have an extensive experience and significant expertise in Sodium cooled Fast Reactors over the past 40 years of R&D and feedback experiments. Some improvements are needed on the SFR to meet the GEN IV goals, and in particular the reduction of investment and operating costs: the Fuel Handling System (FHS) can be considered as an essential step in the reactor design. The reactor refuelling system provides the means of transporting, storing and handling reactor core subassemblies. The system consists of the facilities and equipments needed to accomplish the scheduled refuelling operations. The choice of a FHS impacts directly on the general design of the reactor vessel (primary vessel, storage and final cooling before going to reprocessing), its construction cost and its availability factor. Fuel handling design must take into account various items and in particular operating strategies such as core design and management and core configuration. Moreover, the FHS will have to take cope with safety assessments : a permanent cooling strategy to prevent fuel clad rupture, plus provisions to handle short cooled fuel and criteria to ensure safety during handling.

In addition the handling and elimination of residual sodium must be integer; it implies specific cleaning treatment to prevent from chemical risks such as corrosion or excess hydrogen production.

The objective of this paper is to identify the challenges of a SFR Fuel Handling System. It will then present the range of technical options incorporating innovative technologies under development to answer to the GENERATION IV SFR requirements.

Prospects for improvement of supporting systems of BN reactors based on BN-600 and BN-800 engineering experience

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Spent fuel assembly ablution system of is one of system which are only used for BN reactors. Besides sodium ablution in this system, a Spent fuel assembly is tested for leakiness wits the help of the defective fuel assembly detection system (DFADS-AC).

Summary of system work regime (process).

Ablution cell (AC) is filled with nitrogen. Spent fuel assembly is placed in SA.

1. Ablution regime in case when there are no suspicions of faulty sealing based on other shell hermeticity control readout. Steam ablution is conducted. After that SA is filled with nitrogen again. Spent fuel assembly is kept in nitrogen for heating-up due to residual heat. Then nitrogen is delivered from SA to gas DFADS-AC. If no fault is found in the sealing, spent fuel assembly conducted is washed by water which also delivered for control to water DFADS-AC.

2. Ablution regime in case when there are suspicions of faulty sealing based on other cover hermeticity control readout. First of all the control is performed in gas DFADS-AC. If there is no faulty sealing indication the series of ablution scheme is the same as described above.

The main feature of spent fuel assembly ablution system of BN-600 is that one DFADS-AC is used for two ACs. It means that the parallel work of two ACs is impossible.

Besides pipelines for delivery of steam and nitrogen to AC are united, which result is steaming of nitrogen pipeline. This leads to extra difficulties in parallel usage of two ACs.

Changes in the spent fuel assembly ablution system for BN-800 based on operation experience of BN-600:

- There are two gas DFADS-AC and water DFADS-AC, one for each AC;
- Separation of pipelines for delivering of steam and nitrogen to SA;
- Possibility of separate delivery of ablution substance to each SA;
- There is no usage of gas DFADS-AC in nitrogen drop to the technology blowing system ;

All performed changes in the current system make it possible to perform spent fuel assembly ablution in two AC simultaneously. It means that we will get double efficiency of the system.

Prospects for development (improvement) of the spent fuel assembly ablution system.

The scheme of DFADS-AC requires the specific equipment due to characteristics of radioactive control sensors. For example, the pressure reducer, aerosol filter and moisture separator. If sensors are developed which do not limitation of gas parameters, all mentioned elements can be excluded from DFADS-AC. This will considerably improve the reliability of the system.

Development of a large diameter electromagnetic pump and a back-up power supply system for the 4S*

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Toshiba has been developing a large diameter high temperature sodium-immersed electromagnetic pump adopted for the primary circulation pump of 4S (Super-Safe, Small and Simple). The 4S is a small-sized, sodium-cooled fast reactor, which needs no onsite refueling for 30 years. The electromagnetic pump (EM pump) is immersed in the reactor vessel as shown in Fig.1.

Toshiba has already developed several high temperature EM pumps with the flow capacities of 1m^3/min, 44m^3/min and 160m^3/min. These pumps were tested in sodium test apparatus and the pump characteristics were verified. These test results showed the potential that EM pump can be applied to a commercial sodium fast reactor. The EM pump for 4S has a 10.6m^3/min flow capacity, and it is within the range of manufactured experiences, while there are some technical issues remained as described below.

The first issue is the largest coil diameter in the ever fabricated EM pumps. The pump is located between the reactor vessel and internal structure of 4S. Therefore, the fabrication ability and accuracy of the coil have to be verified. The second is the concerns about the flow stability. The largest diameter of the EM pump makes the aspect ratio (pump annulus diameter / stator length) 1.2, which is larger than that of our manufactured experience 0.4. This flat geometry has the possibility to arise flow instability in pump flow annulus. The flow characteristic of pump has to be verified in sodium flow test. The third is 30 year integrity with the maintenance free structure in sodium. The reason is that 4S aims to be a non refueling plant during 30 year operation period, and upper plug of the vessel is also planned to be confined during the operation. The last issue is function of flow coatstdown characteristic. The characteristic is required in order to protect the fuel cladding in the plant trip. However small the fluid inertia is, EM pump needs to meet any flow coastdown curve required from the plant.

To resolve those issues described above, Toshiba has started the prototype test described below.

In order to verify the fabrication and performance of 4S EM pump, prototype pump coil is manufactured and the fabrication procedure and accuracy is evaluated. Secondary, in order to evaluate the pump flow characteristic, prototype 4S EM pump is designed. The size is same as commercial 4S pumps. The characteristics of the pump will be evaluated in the sodium test apparatus. In order to preserve integrity of the pump during the operation, high toughness should be provided to the components of the 4S EM pump. Then, the critical components to failure will be selected by probabilistic risk assessments (PRA), and the durability will be confirmed by endurance tests. Finally, in order to create the flow coastdown function, back-up power supply system is designed, which drives EM pump in the event of plant trip. The system is manufactured and evaluated by connecting to the prototype 4S EM pump in sodium test apparatus.

This report presents the fabrication of EM pump and its test results conducted under atmospheric conditions.

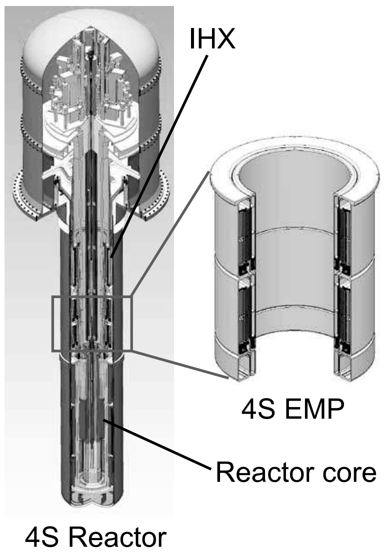

Fig.1 Schematic drawing of 4S Reactor and 4S EMP

Acknowledgement

The part of present study is the result of "Development of high temperature electromagnetic pump with large diameter and a passive flow coast compensation power supply adapted to GNEP medium / small reactors" entrusted "Toshiba Corporation" by the Ministry of Economy, Trade and Industry (METI) of Japan.

Research and development for the integrated IHX/Pump

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The proposed component is an integrated intermediate heat exchanger with primary pump into one vessel (integrated IHX/Pump) for the plant economy, reduction of material (steel) of the primary cooling system. It is one of advanced concepts of the Japan Sodium-Cooled Fast Reactor (JSFR) which Japanese Electric utilities, Japan Atomic Energy Agency (JAEA), Mitsubishi FBR Systems (MFBR) and Mitsubishi Heavy Industries, Ltd (MHI) have promoted in the Fast Reactor Cycle Technology Development (FaCT) project. This report summarizes the development plan, the evaluation method, and some results of scale model vibration tests which have been performed since 2005.

In order to confirm its feasibility, it is important to avoid the resonant vibration caused by integrating of the pump into the static IHX and to prevent excessive wear-out of the heat exchange tubes caused by contact with the tube support plates. Therefore, the evaluation method of wear property has been developed since the start of this concept. It combines fluid-structure interaction vibration analysis, work-rate analysis and wear-rate of tube material. The vibration analysis has been applied to low-vibration design of the component, and a lot of examinations using a scale model were carried out to confirm and improve the analysis accuracy.

The 1/4-scale model was selected for vibration tests to check the vibration characteristic. The eigenvalues and rated revolution of the model are about 4 times larger than the actual integrated IHX/Pump. The 1/4 scale is suitable for the tests because of the installation area, rotating speed and cost. The vibration caused by the pump was measured, and the character of frequency response was acquired.

The test results were compared with results of eigenvalue analysis and response analysis performed with a detailed model. We use an analysis code called "FINAS" which is a general purpose analysis code developed by JAEA from 1976. It is capable of static stress analysis, dynamic analysis, heat conduction analysis, etc., and can directly handle the vibration behavior of complex multi-cylinders with fluid between the cylinders.

In addition, research and developments are planned by 2010 for the flow evaluations, the vibrational evaluations, and the wear-out evaluation for IHX tubes.

Comparison of Pool/Loop Configurations in the JAEA Fast Reactor Feasibility Study

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JAEA conducted a feasibility study on commercialized fast breeder reactor cycle systems from 1999 to 2006 (FS). In the FS, various fast reactor concepts with various power levels, coolant materials and plant configurations were proposed and competed. In the large-scale sodium cooled reactor region, four nuclear industry vendors proposed each original advanced sodium cooled reactor concept in 1999. One was a loop concept which was named "JSFR" [1] later and the other three were pool concepts. The first competition among the four concepts showed that the economical competitiveness of JSFR is better than pool concepts. Therefore, in the 2000 study, one pool concept was selected and the selected pool concept was refined to compete with JSFR. In this appendix, the selected FS pool concept is described and the pool/loop comparative study in the FS is briefly summarized.

Schematic illustration of the reactor vessel of the FS pool is shown in Figure 1. Relation of reactor vessel diameters and electric output of various sodium-cooled reactor concepts are compared. The comparison shows that the FS pool concept has a smaller reactor vessel than recent conceptual large reactors such as SPX-2, SNR-2, EFR and BN-1600.

The material amount comparison shows that the reactor vessel and primary system material amount of the FS pool concept is heavier than that of the FS loop concept (JSFR) by approximately 250ton. Because the reactor vessel diameter comparison of various concepts shows that the FS pool concept is one of the most compact pool conficurations, the FS pool/loop competition is thought to provide comparative information between the most economical loop and pool concepts.

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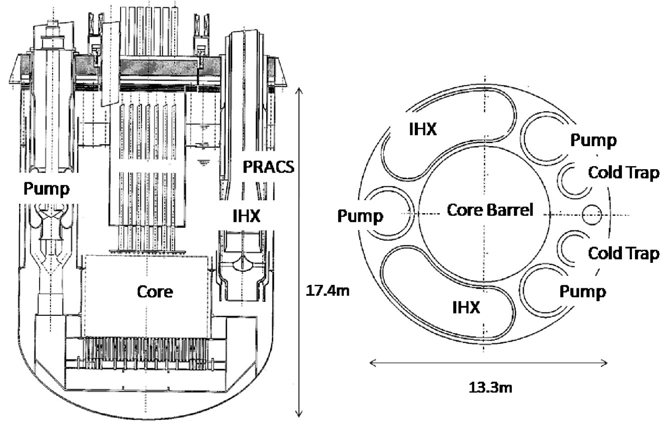

Figure 1 Reactor Vessel of the FS Pool Concept

Unsteady elbow pipe flow to develop a flow-induced vibration evaluation methodology for JSFR

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For an advanced sodium-cooled fast reactor (named Japan sodium-cooled fast reactor: JSFR) a two-loop cooling system is designed by adopting a large-diameter piping system with high coolant velocity. The high-velocity piping system brings a flow-induced-vibration issue. To address this issue, experimental and analytical studies have been carried out to grasp flow-induced vibration characteristics in the piping. A flow-induced vibration evaluation methodology that was preliminarily verified with the experimental data tentatively inidicated positive feasibility of the JSFR piping [1]. This paper describes the current status of flow-induced vibration evaluation methodology development for primary cooling pipes in JSFR, in particular emphasizing on recent R&D activities that investigate unsteady elbow pipe flow.

Experimental efforts have been made using 1/3-scale and 1/10-scale single-elbow test sections for the hot-leg pipe. First of all, experiments with water for the hot-leg piping were carried out using the 1/3-scale test section under rectified-flow conditions [2]. The 1/10-scale experiment indicated no effect of pipe scale by comparison to the 1/3-scale experiment. The next experiment using the 1/3-scale test section was performed to investigate the effect of swirl flow at the inlet. Although the flow separation region was distorted at the downstream from the elbow, the experiment clarified that the effect of swirl flow on pressure fluctuation onto the pipe wall was not significant as shown in Fig. 1. An additional experiment was intended to study the effect of elbow curvature. The experiments with water clarified that turbulence is weakened in an elbow with larger curvature than that of the JSFR .

For cold-leg pipe experiments, a test section with triple elbows is necessary to simulate the JSFR pipe. Since the interference of multiple elbows should be investigated to understand turbulent flow in the cold-leg pipe geometry, 1/15-scale experiments with double elbows were carried out to compare the single-elbow experiment with the same scale. The experiment showed that flow in the first elbow influences a flow separation behavior in the second elbow.

Simulation activities include Unsteady Reynolds Averaged Navier Stokes equation (U-RANS) approach with a Reynolds stress model using a CFD code and Large Eddy Simulation (LES) approach using in-house codes. A hybrid approach that combined with RANS and LES

was also applied using a CFD code. Several numerical results appear in this paper, focusing on its applicability to the hot-leg pipe experiments. These simulations reasonbly agreed with the experimental data using the 1/3-scale test section. The simulation also revealed that Reynolds number scarcely affects flow patterns and flow velocity distributions. The numerical results would be provided to the input data for the structural vibration evaluation of the piping.

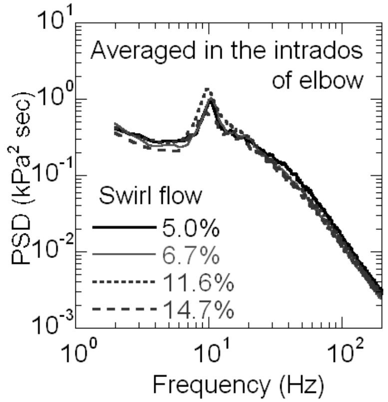

Fig. 1. Effect of inlet swirl flow on the power spectrum densities of pressure fluctuations.

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Parallel Session 9:

Past twenty years with fast reactors and experimental facilities: experience and prospects

Availability of Research and Test Facilities for Fast Reactor Development

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The OECD/NEA Nuclear Science Committee (NSC) has recently published the report of an Expert Group on "Research and Test Facilities Required in Nuclear Science"[1]. Within its scope the report considered Reactor Development, including Fast Reactors.

The Expert Group also created a database listing facilities relevant to its remit and this has been available for public access via the NEA website since Spring 2008 [2].

The paper discusses the findings and recommendations of the Expert Group. These focus on the status and anticipated developments in Fast Reactor technology in the medium term and makes comments on the needs for zero and low-power reactors and sub-critical assemblies to extend the knowledge of the skills base as well as research reactors and critical assemblies related to particular reactor designs.

The paper emphasises the Expert Group's recommendation that further federation of the financial, scientific and technical efforts of the OECD-countries could optimise available resources. The efforts within GNEP and the Generation IV International Forum are cited as demonstrations of willingness to participate in such collaborations.

The key role played by international institutions in the promotion of such co-operation between countries is stressed and as are existing synergies between NEA and IAEA activities.

The maintenance of the current skills base of very competent and experienced scientists and technical staff is discussed; in particular, the Frédéric Joliot / Otto Hahn Summer Schools on Nuclear Reactors Physics, Fuels and Systems (which in 2009 has focussed on Fast Reactors [3]) and the World Nuclear University are commended.

The paper remarks on the wealth of information that exists in the results from past experiments on irradiated fuels, operational reactor feedback, and sodium technology, and emphasises the issue of knowledge retention, for example through databases of older experiments such as IRPhE [4].

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Core Modification for the High Burn-up to Improve Irradiation Efficiency of the Experimental Fast Reactor Joyo

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The experimental fast reactor Joyo of the Japan Atomic Energy Agency (JAEA) has operated since 2003 with the upgraded MK-III core, which provides a high level of fast neutron flux of $4.0x10^{15}n/cm^2s$. Following the irradiation field characterization of the Joyo MK-III core in the first and second operational cycles, several important irradiation tests of fuel and material for fast reactors were successfully conducted in the next four operational cycles, These irradiation tests included mixed oxide fuel containing minor actinide (MA-MOX) and oxide dispersion strengthened (ODS) steel.

These tests utilized the capsule type irradiation rigs. These rigs contained up to six test fuel pins in each stainless steel capsule, The steel capsule has sufficient strength to withstand the pressure due to potential sodium-fuel interaction in the event of fuel melting or fission gas release from a high burn-up fuel pin failure. The fissile material content in this type of subassembly is smaller than that of the driver fuel, resulting in a decrease of core reactivity and fuel burn-up if many capsule type rigs are loaded at the same time.

In order to overcome this problem, the following countermeasures to compensate the core reactivity have been investigated so as to increase the core average burn-up (Fig.1) .

(1) Installation of zirconium reflectors around the driver fuel region to improve the neutron efficiency

The stainless steel reflectors are loaded in the Joyo radial and axial regions around the driver fuel region. Zirconium's nuclear characteristics reflect neutrons with less absorption compared to stainless steel or high nickel alloys. The survey calculation based on the JENDL-3.2 cross section set showed that one layer of radial reflectors containing zirconium at 65 % volume fraction of the total steel weight provides an excess reactivity increase of about 0.5 % Δk/kk'. In order to compensate the increased reactivity, the refueling batch number was changed and detailed core characteristics were evaluated. The calculated results showed that the installation of zirconium reflectors reduces the average number of fuel exchanges and increases the average fuel burn-up with small effects on other core characteristics such as power peaking factor, control rod worth and reactivity coefficients.

(2) Changing the control rod arrangement and replacing control rod with driver fuel to increase the fuel inventory

The control rod is usually designed to maintain adequate shutdown margin considering a variety of uncertainties such as calculation and measurement errors, variation of the core size, boron burn-up, and asymmetric effect in the Joyo MK-III core. However, extensive experimental data and core management experience through the core performance test[1] showed that the calculated control rod worth based on the JUPITER standard fast reactor analysis method with the bias correction agrees with the measurement within 5 %. Therefore, the design margin is found to be reduced from 18 % to 10 % so as to reduce the number of

control rods from 6 to 5. This could increase the core average burn-up as much as 10 GWd/t and the number of fuel exchanges was reduced as much as 11 %. Linear heat rate and cladding temperature were within the reactor safety criteria in any case. In order to achieve the increase of fuel burn-up, the future R&Ds are needed. The PIE works are now under progress for the highest burn-up MK-III driver fuel of 86.0 GWd/t to obtain the fuel design data.

These countermeasures will be applied to the Joyo core management in the future to enable more innovative irradiation experiments using the capsule type rigs. The detailed core design and the work to obtain the new license are now underway in conjunction with the enhancement of the Joyo's irradiation function. The core modification and concepts will be useful for the various contributions of Joyo, which has the prominent high neutron flux, to the research in future energy systems and basic science and to conduct the material irradiation tests required for the Generation-IV systems and other advanced reactor systems.

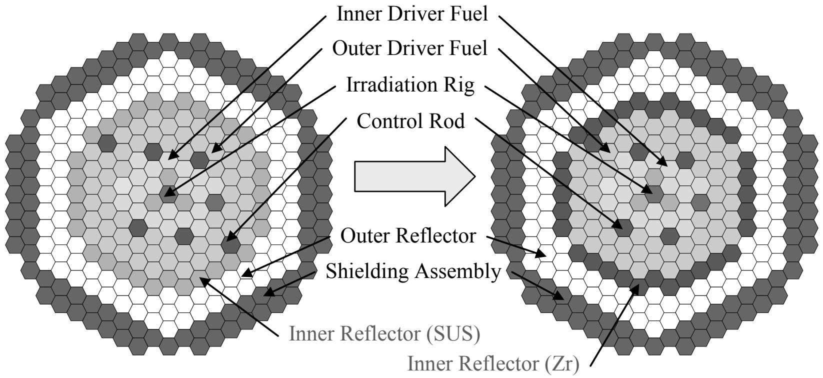

Figure 1 Core Modification

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Scientific Design of Large Scale Sodium Thermal-Hydraulic Test Facility in KAERI

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A large scale sodium thermal-hydraulic test facility is being designed for verification of the advanced design concept of the passive decay heat removal circuit which is a safety grade residual heat removal system of KALIMER-600. In the test, its cooling capability during the long- and short-term periods after reactor trip is evaluated and also the produced experimental data will be utilized for the assessment and verification of the safety and performance analysis codes. In this paper, the design feature of the test facility is presented along with the design method which is applied for preserving the global and local thermal-hydraulic phenomena.

The main test section of the experimental facility is composed of a primary heat transport system and a passive decay heat removal circuit which are scaled-down from the KALIMER-600 design. The main test section includes all major components reflecting the real configuration. The preliminary concept of it is shown in Figure 1. Auxiliary fluid systems such as an intermediate heat exchanger gas cooling system, a sodium supply/purification system, a heat loss compensation system, a power supply system and a gas supply system are included in the experimental facility to provide the various environment for performing the experiment.

In order to represent important thermal-hydraulic phenomena in the passive decay heat removal circuit as well as the reactor system, the main test section is designed complying with proper scaling method for geometric, hydrodynamic and thermal similarities. The major thermal-hydraulic phenomena which will be investigated in the test are the natural circulation through a primary heat transport system and a passive decay heat removal circuit, and the multi-dimensional flow distribution in the reactor pool. In order to reproduce these phenomena, the overall design parameters are drawn by the global scaling criteria which are produced based on the one-dimensional balance equations of mass, momentum and energy, and the boundary condition at the interface between fluid and solid.

Also, the important components such as heat exchangers, primary pump and simulated core are designed by preserving the local thermal-hydraulic phenomena related with the corresponding components.

Overall scaling of the facility is 1/125 for volume and 1/5 for height. The overall height of the facility is about 20 m. The reactor vessel height and diameter are about 3.6 m and 2.3 m, respectively. The reactor core is simulated by electrical heaters of 1.91 MW capacity which corresponds to a 7 % of the scaled full power. Sodium is used as a working fluid and its inventory of the main test section is approximately 15 tons. Operating temperatures of the system are preserved in the experiment. In the test, the natural circulation cool-down capability by the passive decay heat removal circuit in conjunction with the reactor system will be investigated for the design basis events such as the loss of flow event, the loss of heat sink event and the reactor vessel leak event.

Starting with the basic design of the test facility in 2008, its installation is scheduled to be completed in the middle of 2012. The main experiments will begin in 2013 after the startup test in 2012.

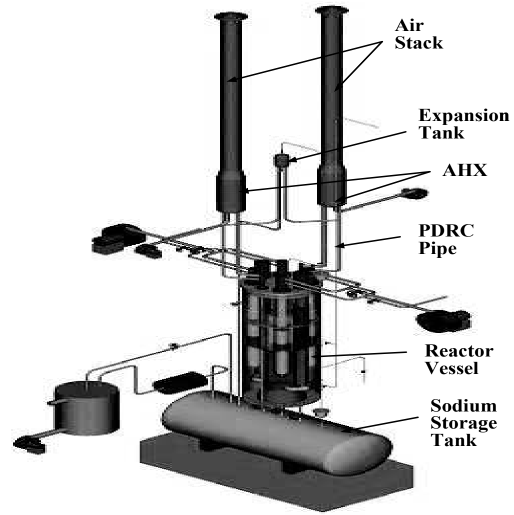

Figure 1. The preliminary concept of the main test section

Status and Basic Lines of Development of Experimental and Material Science Base for Fast Reactor Technologies

A.V. Bychkov

SSC RIAR

The objective of nuclear power engineering in Russia is production and supply of cheap electric and thermal power to the utilities for the foreseeable future providing nuclear, physical, environmental and technical safety in the scope that meets the public needs. The large-scale nuclear power engineering should solve the key problems of ensuring energy safety of the country, improving the quality of life of the population and stabilization of the international situation.

Comprehensive technological solutions for construction of fast reactors based on the new structural materials, new fuel types, which can be integrated into the industrial implementation of the global fuel cycle, are proposed as the basis for the development of advanced nuclear power engineering.

The industrial implementation of the global fuel cycle in Russia provides for a complete recycle of all actinides, as well as recycle of fast reactors SNF.

Experimental and material science base for the fast reactor technologies built in the days of MINSREDMASH, MINATOM and developed nowadays by ROSATOM has unique technical capabilities.

The report describes the Russian experience gained by the nuclear power enterprises in operation of the experimental complexes and facilities. An attempt is undertaken to highlight the short- and long-term tasks which will be addressed in the context of the further development of the experimental and material science base for fast reactos technologies.

The report includes also an expert review of national experimental complexes and facilities of nuclear countries which have the fast reactor experience.

Parallel Session 10:

Fast reactor knowledge management, education and training

International Fast Reactor Knowledge Organization System

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For three decades, several countries had large and vigorous fast breeder reactor development programmes, which had their peaks by 1980. From that time onward, Fast Reactor (FR) development generally began to decline and efforts for FR reactor development essentially disappeared by 1994.

This development stagnation continued until 2003. In September 2003, in Resolution GC(47)/RES/10.B, the International Atomic Energy Agency (IAEA) General Conference recognised the vitality of nuclear knowledge.

The loss of FR knowledge has been taken seriously and the IAEA took the initiative to coordinate the efforts of the member states in the preservation of knowledge in FRs. In the framework of this initiative, the IAEA intends to create an international inventory combining information from different member states on FRs and organized in the knowledge system in a systematic and structured manner.

Fast reactor knowledge organization system (FR KOS) is based on the FR taxonomy [1], which is a basis of the description of the whole FR area.

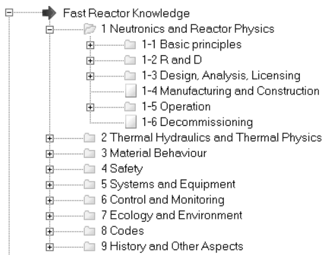

FIG.1 Top Levels of Fast Reactor KOS

The core elements of the FR-KOS concept are:

- the purpose for which the FR-KOS is created;
- the knowledge domain covered by the system;
- the approach to knowledge mining in FR-KOS;
- the target customers of the FR-KOS.

The main purpose of FR-KOS is to assure the preservation of the FR knowledge and experience gained in various countries in a form that would facilitate the effective search, knowledge mining and use of the stored information.

It is assumed that an electronic repository would not only preserve knowledge on FRs, but would also facilitate the classification of FR information accumulated in different countries and the exchange of this information between countries.

Global scientific community acknowledges that FRs are promising and essential for tapping the huge potential of nuclear power to contribute to the economic development of the world. This may be seen from the development of FRs in Asia (Japan, India, China, Republic of Korea), as well as the revival of interest in FRs in the developed countries within the framework of International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and Generation 4 programmes.

This is the evidence that FR technology will undoubtedly play an important role in the future.

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The French Sodium School : Teaching Sodium Technology for the present and future generations of SFR users

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This paper provides a description of the French Sodium School located in France (at the CEA Cadarache Research Centre) and of the Fast Reactor Operation and Safety School (FROSS) created in 2005 on the Phenix plant. It presents their recent developments and the current collaborations throughout the world with some other nuclear organizations and industrial companies. The very recent courses implemented to answer to the future need of Generation IV SFR concept and design, are also presented.

The initial objective of the sodium school was to form engineers and operators able to work on sodium fast reactors or on supporting R&D activities. The sodium school history can be resumed in several key dates.

- 1975 : Creation of the Sodium School at Cadarache (Training of Phenix plant teams)
- 1980 : Accreditation by EdF (Electricité de France – French national electricity supplier) : Training of Superphenix plant teams
- 1984 : School opened to foreign companies or utilities (Training for SNR300 team - Germany)
- 1995 : Partnership with the INSTN (French Nuclear Teaching Institute)
- 1998 : With the decision to stop the SUPERPHENIX reactor, the sodium school has defined a new set of modules more orientated towards decommissioning (theory and practice).
- 2002 : Cooperation with JAEA (Japan Atomic Energy Agency – Japan) to provide 37 lectures at Monju reactor (program scheduled on 1 week per year during 5 years)
- 2005 : Collaboration with FROSS
- 2006 : With INSTN, a new course is under preparation based on SFR design and main options.
- 2008 : The Sodium Fast Reactor Design and Component module is now operational for French people and organized jointly by the INSTN and the Sodium School (maximum 2 session of 2 weeks per year).

Complementary to the Na school held in Cadarache, FROSS training objectives are to answer to the training needs of other international partners involved in the development of Sodium cooled Fast Reactors (SFR. The training objectives of Phenix-based FROSS are to share Phenix over 33 year experience of FBR operation and provide, in English, a formation on : Safety and organizational aspects of SFR operation, Sodium technology,Circuit and plant operation, with emphasis on safety and commissioning aspects, Normal, incidental and accidental instructions.

For all the aspects linked with sodium safety, FROSS is associated with the French sodium school, where part of the training takes place. Depending on the initial knowledge and experience in managing installations with sodium coolant, training sessions of 2 and 3 weeks are presently organized.

The sum of courses provided by CEA through its Sodium school and FROSS organizations is a unique valuable amount of knowledge of Sodium Fast reactor design, technology, safety and operation experience and practical exercises provided for the national demand and since the last five years extensively opened to foreign countries. Over more than 30 years, this organisation has demonstrated its flexibility in adapting its courses to the changing demand in the Sodium Fast reactor field, and in association with the PHENIX and SUPERPHENIX plants, can adapt its teaching techniques using specific theoretical and practical courses and lectures.

Human Development in Japan and Abroad Using the Prototype FBR "Monju" Towards the Next-Generation Age

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Japan is engaged in research and development for an innovative FBR, targeting commercial operation to start around 2025 via the FaCT (Fast Reactor Cycle Technology Development) project. To prepare for the new FBR age, INITC has been working on human resource development using Monju towards the next-generation age not only for Japan, including companies and students, but also for the world aiming at becoming a base of the international educational training.

Effective mastering technology requires educational training harmonized through lectures and exercises, consequently, providing such education needs preparing substantial hardware and developing software. As for the preparation of hardware, the Fast Reactor Training Facility [1], which consists of two training facilities for systematically teaching the technologies for sodium handling and maintenance related to FBR technology, was built in May 2000, based on the lessons learned from the December 1995 sodium leak accident [2]. As regard in to operator training, the hardware of MARS (Monju Advanced Reactor Simulator) [3] was reconstructed corresponding to the sodium leak safety measures, and simulator function was also improved in order to advance operator training.

The following advances [1] concerning the software were also carried out based on the teachings obtained from the leak accident as well as hardware: a) Strengthening operator training framework by establishing educational training guidance, improving operation manuals and introducing an operator training evaluation system (SAT: Systematic Approach Training) [4]; b) improving training curriculums related to plant system engineering and sodium handling technologies, which cover teaching from the basis to the expert levels. A total of 26 training courses were established, such as 8 simulator courses, 5 FBR plant system engineering courses, 6 sodium handling courses and 7 maintenance courses. And each course has been continuing even while Monju operation has been stopped for over a decade.

Moreover, INITC is holding two types of international training courses every year, i.e., the "International Safety Training Course" to help spreading nuclear safety technology in Asian countries and the "International Sodium Technology Training Course" for trainees from China and U.S.A. Besides these activities, student educational trainings are also held periodically to enhance the understanding of the nuclear technology and increasing the interest and care for nuclear field: a) a university student educational seminar that is one of the collaborative activities with CEA, France; b) the Monju Education that is a part of the environment and energy education for primary and secondary students in the local community. The variety of the above mentioned educational trainings will contribute to the development of the human resource in Japan and abroad, towards the next-generation age.

Table 1. Outline of Monju Educational Training Framework

Division	Number	Description	Main Target
FBR Operation Technical Training	8 Courses	8 types of simulator training courses (classified individual courses, family course, refresh course, etc.).	Operator
FBR Plant System Engineering Training	5 Courses	Cover from the basis to expert knowledge of FBR plant technologies by one fundamental course and four advanced courses.	Engineer, Operator and Maintenance Worker
Sodium Handling Technical Training	7 Courses	Various unique courses for sodium properties, sodium loop operation, extinguishing sodium fires, sodium piping leak, sodium treatment, etc.	Operator, and Maintenance Worker
Maintenance Technical Training	8 Courses	Diverse maintenance courses using four Monju specific maintenance training models and four conventional maintenance technical training models.	Maintenance Worker

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Knowledge Management in Fast Reactors and related Fuel Cycles

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The 21st century is ushering in a new phase of economic and social development which can be referred as “Knowledge Economy”, in which knowledge has become the key asset in determining the organization’s success or failure.

The IAEA defines knowledge management as an integrated, systematic approach to identify, manage and share an organization’s knowledge collectively in order to help achieve the objectives of the organization.

Nuclear technology is very complex and a highly technical endeavor. It relies on innovative creation, storage and dissemination of knowledge. The nuclear energy is characterized by long time scales and technological excellence. Nuclear knowledge management is a critical input to nuclear power industry, the associated fuel cycle activities and nuclear applications in medicine, industry and agriculture.

Realizing the importance of knowledge preservation in the area of fast reactor technology, IAEA had given a consultancy work to Argonne National Laboratory to study and suggest the means of knowledge management. The IAEA initiative seeks to establish a comprehensive inventory of fast reactor data and knowledge for the fast reactor development in the coming years. It was suggested that the knowledge regarding important disciplines like fuels & materials, reactor physics and core design, operations, the demonstration of safety should be preserved.

Various countries have initiated the fast reactor knowledge preservation activities. In France, CEA,EDF and Framatome ANP have initiated liquid metal cooled fast reactor knowledge preservation project that deals with R&D aspects and Superphenix design. European Fast Reactor collaboration(MASURCA,SNEAK,ZEBRA) has preserved the zero power critical experimental data in the SNEDAX database. Japan has started a comprehensive knowledge preservation program including the capture of “Human Knowledge” based on interviews. In Russia steps are initiated to preserve fast reactor knowledge regarding BFS-1, BFS-2 and KOBR and post irradiation experience. In UK a super archive was prepared. In USA, TREAT and ZPPR data are currently on a magnetic tape and hard copies with some transfer to electronic files. It is therefore subject to loss. Hence selected ZPR and ZPPR log books are being scanned and selected critical configurations are being preserved.

It is needless to emphasize that in R&D organizations like Indira Gandhi Centre for Atomic Research(IGCAR) with a mandate to conduct broad based multi disciplinary programme of scientific research and advanced engineering directed towards fast reactor technology and associated fuel cycle facilities, knowledge management plays a vital role. It also helps in our vision to achieve world class leadership in the fields of Fast Reactor technology and related Fuel Cycles. Also, India would like to achieve energy security through Fast Breeder Reactors.

IGCAR has been operating a Fast Breeder Test Reactor(FBTR) successfully for the last 23 years with a unique Pu-U carbide fuel. The Centre has developed and nurturing world class expertise in the areas of fast reactor engineering, reactor safety and analysis, sodium technology, materials development and characterization, non destructive evaluation, in service inspection, reactor instrumentation, computer modeling etc. The centre had successfully reprocessed the Pu-U carbide fuel from FBTR of 150,000 MWd/t burn up. A lot of knowledge has been created in these domains and is being effectively managed and utilized.

Learning from 380 reactor years of knowledge base of international experience and knowledge accrued from our own Fast Breeder Test Reactor through successful operation for 20 years and with major engineering experiments in fast reactor technology conducted, IGCAR has indigenously designed 500 MWe Prototype Fast Breeder Reactor(PFBR) and the reactor is under construction. With creative management of knowledge of the centre a Fast Reactor Fuel Cycle Facility is being built.

The Centre has adopted and practicing a knowledge management policy covering both the forms of knowledge viz. explicit and tacit knowledge. All the old design reports, training documents, internal reports pertaining to FBTR were scanned and soft copies have been stored in web based knowledge portal of the centre. In addition, the design reports, internal reports of PFBR along with the journal publications, presentations, progress reports are stored in knowledge management server. The tacit knowledge of the retiring employees is being collected through exit interviews, reports etc. and stored.

The paper gives a few examples of success stories of knowledge management, especially, how the knowledge accrued over 23 years of successful operation of FBTR has been stored and disseminated effectively to ensure that none of the problems that occurred in the initial stages of operation of the reactor have not occurred again. The innovative experiments and knowledge accrued in the materials development and it’s utilization in the PFBR design have been highlighted. The expertise gained in handling sodium from small quantities in the beginning to 1750t for PFBR through tankers is explained. The success story from reprocessing spent nuclear fuel of 0.3% Pu with 7 GWd/t burn up and conducting related R&D to reprocessing of nuclear fuel of 70% Pu and 150 GWd/t of burn up is mentioned. The experience of how the in house reactor design knowledge has been channelized to develop a Full Scope Replica type Operator Training Simulator is explained.

The success of any knowledge management project depends on the **Passion** and **profound belief** that knowledge management is not only worthwhile but it is in fact **a way of living**. Also, it is to be believed that knowledge management is not a destination but a journey.

POSTERS OF SESSION 1:

Innovative fast reactors: objectives and driving forces

Design study on the Advanced Recycling Reactor

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The design study on the Advanced Recycling Reactor (ARR) has been conducted. This paper presents the pre-conceptual design of the ARR that is a loop-typed sodium cooled reactor with MOX fuel.

International Nuclear Recycling Alliance (INRA) takes advantage of international experience and uses the design based on Japan Sodium-cooled Fast Reactor (JSFR) as reference for FOA studies of US DOE, because Japan has conducted R&Ds for the JSFR incorporating thirteen technology enhancements expected to improve safety, enhance economics, and increase reactor reliability.

The targets of the ARR are to generate electricity while consuming fuel containing transuranics and to attain cost competitiveness with the similar sized LWRs. INRA proposes 3 evolutions of the ARR; ARR1, a 500 MWe demonstration plant, online in 2025; ARR2, a 1,000 MWe commercial plant, online in 2035; ARR3, a 1,500 MWe full-scale commercial plant, online in 2050. INRA believes the scale-up factor of two is acceptable increase from manufacturing and licensing points of view. Major features of the ARR1 are the following:

The reactor core is 70 cm high and the volume fraction of fuel is approximately 32%. The conversion ratio of fissile is set up less than 0.6 and the amount of burned TRU is 45-51 kg/TWeh. Decay heat can be removed by natural circulation to improve safety. The primary cooling system consists of two-loop arrangement and the integrated IHX/Pump to improve economics. The steam generator with the straight double-walled tube is used to improve reliability. The ARR1 is co-located with a recycling facility. The overall plant facility arrangement is planned assuming to be constructed and installed in an inland area. The plant consists of a reactor building (including reactor auxiliary facilities and electrical/control systems), a turbine building, and a recycling building. The volume of the reactor building will be approximately 180,000 m^3.

The capital cost for the ARR1 and the ARR2 are estimated. Also, the construction schedule and regulatory and licensing schedule are estimated. Furthermore, the technology readiness level and the technology development roadmap are studied and identified to be ready for commercial deployment.

European Lead-cooled SYstem core design: an approach towards sustainability

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The neutronic design of the European Lead-cooled SYstem (ELSY) [1] – a 600 MWe Lead-cooled Fast Reactor (LFR) for the transmutation of radioactive waste together with electricity generation developed within the 6th EURATOM Framework Programme – has been carried out for a open square-lattice core configuration characterized by innovative Finger Absorber Rods (FARs) which act both as regulation and shutdown control system.

The overall core layout [2], mostly defined by complying with mechanical and seismic constraints, has been optimized in order to obtain a flatten (up to 1.2 maximum-to-average ratio) power/Fuel Assembly (FA) distribution. A trade-off between a fuel Burn Up (BU) as high as 100 GWd/t and an acceptable Breeding Ratio (BR) has been also achieved.

The need to guarantee an as flat as possible power/FA distribution has also suggested to adopt an innovative control system for the regulation of the criticality swing during operation, i.e. a number of small B4C rods (FARs) distributed – one per FA – on a selected subset of assemblies throughout the core. The use of many rods, slightly inserted into the active zone, results into both a small perturbation of the neutron flux during operations and a sufficient margin of antireactivity available for scram.

In ELSY the power-to-flow ratio is locally adjusted by changing the fissile (Pu) enrichment at different radial positions in the core, taking also into account the actual dimensioning and positioning of the FARs system. This has imposed an iterative design procedure in order to define the proper zoning and enrichments for power/FA distribution flattening, guaranteeing also the effectiveness of the FARs for the reactor control.

The result of this iterative design procedure has led to a core segmented into three Pu enrichments zones made of 56, 50 and 56 FAs, characterized by 14.45, 17.53 and 20.50 $^a/_0$ of Pu respectively. The criticality swing during the cycle (1.25 y in 4-batches) has resulted in some 900 pcm (k_{eff} = 1 at End of Cycle, EoC, with all regulation FARs withdrawn), also due to a net BR = 0.94 [3].

For the compensation of the cycle swing, 38 FARs – positioned between the intermediate and the outer fuel zones to maximize their effectiveness – have been foreseen, providing the needed anti-reactivity by a 30 cm insertion into the active zone. Their complete insertion has been evaluated to provide further 2500 pcm for the reactor shutdown.

Two independent safety systems have been also introduced for scram, in order to provide the required reliability for the reactor control: a second set of 32 FARs and a set of 8 CRs, both acting passively by gravity into empty channels.

The core design of ELSY has also shown that it is possible to realize an "adiabatic" reactor, i.e. a reactor self-sustainable in Pu and burning its own generated MAs. This is in compliance with the sustainable goals of the Generation IV systems. Such a system would base its operation upon the net "conversion" of either Natural or Depleted Uranium (NU or DU) into Fission Products (FPs) only.

The analysis of the ELSY cycle showed that the build up of the MAs behaves almost exponentially towards an equilibrium concentration of some 0.7% on the Heavy Metals (HMs) mass. Such a concentration is expected to fulfill the safety requirements for the ELSY adiabatic configuration.

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LEAD-COOLED FAST REACTOR (BREST) WITH AN ON-SITE FUEL CYCLE

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Summary

Out of a great many of new power technologies, fission nuclear power is the only realistic way to stop the growth in extraction and combustion of fossil fuel. However, this is something to be achieved only through the nuclear power to have by the mid-21st century the capacity an order of magnitude as high as the current level.

Nuclear power of such a scale will necessitate a new nuclear technology which is required to provide:

- transition to power with unlimited fuel resources;
- economically competitive nuclear power through reducing the cost of building and operating NPPs of a high inherent safety level with highly efficient utilization of fuel and generated heat;
- elimination of severe accidents with radioactive release which require evacuation of population, to be achieved primarily through combining inherent safety, passive protection features and impossible loss of lead coolant;
- an environmentally safe closed fuel cycle with in-pile combustion of minor actinides and radiation-equivalent disposal of radioactive waste;
- creation of nuclear proliferation barriers by way of eliminating uranium enrichment and plutonium separation facilities.

These problems are solved in the BREST lead-cooled fast reactor.

The initial stage plan is to build an NPP with a demonstration reactor and an on-site fuel cycle to verify designs, try out processes involving lead using as the coolant and study the behavior of the reactor and its systems and components in different modes, including resistance to anticipated operational occurrences and accidents simulated on the reactor. This will be followed by a rapid transition to a fast lead-cooled power reactor of 1200 MW(e) featuring two-circuit heat removal from the core to the turbine with supercritical steam parameters.

The state of the activities to develop the NPP with lead-cooled fast reactors and an on-site fuel cycle is presented.

A core with a moderate power rating has been considered. This will have a uranium-plutonium mononitride fuel, a lead reflector, a widely spaced fuel lattice, levelled-off and stabilized lead and fuel cladding temperatures, a high breeding coefficient CBR ~1, negative power and temperature reactivity coefficients and a negative void effect. Emergency processes have been analyzed and found not leading to prompt-neutron runaway of the reactor, loss of coolant, fires and explosions to involve fuel damage and catastrophic radioactive release even when the external barriers are broken and all active safety features fail. Perfect utilization of fuel and high efficiency, simplification of structures and elimination of complex engineered safety and accident localization systems are prerequisites for reactor economic competitiveness.

Basic NPP, reactor, on-site fuel cycle and RW processing structures as well as the results of experimental studies to support the approaches taken are presented.

P-DEMO for Demonstration of Fast Spectrum Transmutator PEACER

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The concept of proliferation-resistant, environment-friendly, accident-tolerant, continual and economical reactors (PEACER) has been developed by the NUTRECK, aiming at converting all spent nuclear fuels(SNF) into Low and Intermediate Level Waste(LILW) by integrating an advanced pyrochemical recycling and heavy liquid metal cooled fast reactor technology into a multi-national transmutation facility. It is proposed that the PEACER mandate - transmutation leaving no high level waste behind – can be demonstrated by a multi-national facility designated as P-DEMO.

P-DEMO is an integral pool-type reactor cooled by LBE coolant in primary cooling system with approximately 8m in height and 4m in diameter. P-DEMO will be started up with initial 19.75% enriched U-Zr (38-62 w/o) metallic fuels and its normal operation entirely relies on the natural circulation capability of liquid LBE. Total power of P-DEMO is 100MW thermal equivalent to 35MW electric power rating and is to be located within an underground containment facilities with strict security control and physical protection. One of the most important design features of P-DEMO is a central test assembly for fuel and materials qualifications which is cooled by LBE forced convection in the independent square assembly.

As shown in Fig. 1, the primary heat transport system is sealed by double wall reactor vessel for excluding the loss of coolant accidents from potential accident scenarios. Natural circulation also eliminates potential risk of loss of flow accident scenarios in primary cooling system. Double containment also protects the emission of radioactive isotopes in the case of significant abnormal conditions. If the hypothetical core melting accidents, the molten core distribution and cooling structure designated as COASIS will spread and solidify the molten core to curb the situation into stable long-term cooling mode with the help of RVACS. This COASIS is placed in the bottom of reactor vessel.

Twelve helical type steam generator modules are equipped at the above downcomer where the cooled coolant flows downward to the core. Heated LBE storage tank and cover gas control system (CGCS) are established on the upward reactor enclosure and transporter system for refueling also is in this region. The CGCS controls the dissolved oxygen level in LBE by flowing hydrogen and oxygen and traps radio-toxic polonium vapors. For the all potential accidents, the reactor vessel air cooling system (RVACS) is designed to remove decay heat by natural circulation of air. The primary shutdown control assemblies are established above the core held by magnetic drive mechanisms during normal operation. They are dropped with the loss of magnetic power into the core by gravity upon a demand. In the above core, the secondary shutdown assemblies are located with different holding mechanism, self-actuated bi-metal latches, and are dropped into the core in the case of accidents accompanying coolant temperature rises.

P-DEMO operates at the low temperature range of 300 ~ 450℃ in order to assure no corrosion problem for structural materials with LBE coolant throughout the 60 years of lifetime. Considering large density and inertial mass of LBE coolant, a three dimensional seismic isolation system is employed for both reactor and the preprocess units to assure their structural robustness

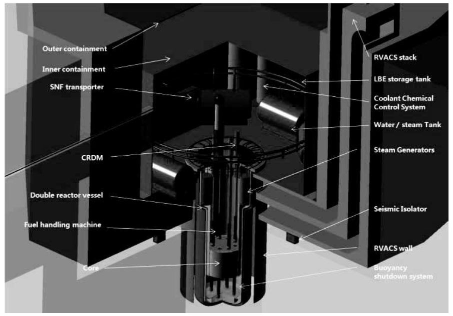

Figure 1 Three Dimensional Design of P-DEMO

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Conceptual Design of 1,000 MW_{th} Inherently Safe Fast Reactor (ISFR)

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ISFR (Fig.1) is a boiling heavy water fast reactor, which is of process inherent ultimate safety (PIUS) type [1, 2]. Unlike PIUS, however, the primary system of ISFR does not contain boron. ISFR may breed fuel in the core. The volume ratio of moderator to fuel is 0.643, while the power density is 78.3 kW/ l in the core. Heavy water was chosen as the coolant of ISFR, since not only it has a smaller slowing down power than light water, but also it produces photo-delayed neutrons. The fuel contains minor actinides which act not only as burnable poisons, but also as fertile materials and hence help reduce k_{eff}^{max} in a reactor life. This is preferable, since ISFR has neither a chemical shim nor control rods. ISFR has not only a positive void coefficient, but also voids already at the steady state. Thus, from a viewpoint of safety and stability, it is necessary for time constant τ_{α} of the positive void feedback to be sufficiently large and also to install initially closed two-way check valves (TCVs) at the lower honeycomb. Time constant τ_{α} increases as gap conductamce h_{gap} decreases. At present, h_{gap} is assumed to be ~ 1. kW/(m^2・^{O}C), which can be materialized by using a mixture of He and Ar as the gap gas.

Being closed at a steady state, the TCVs can be used as passive switches to turn off the pump motors, and the reactor control system as well as heaters in an accident. Thus, on an accident occurrence, when the absolute value of pressure differece across the TCVs $(\Delta p)_{TCV}$ exceeds 2 kPa, one half of the TCVs will open to introduce cold inner pool water into the circulating primary system (CPS) and hence the void fraction wll decrease to induce a negative reactivity which overrides the positive feedback soon enough to shut the reactor down. ISFR has the passive safety shutdown system (PSSS) (Fig. 1) [3] in the secondary system, which is another application of the PIUS configuration modified with initialy closed TCVs at the lower honeycomb. ISFR has other passive features such as heat pipes, which transfer decay heat to the atmosphere after a reactor shutdown.

Both large τ_{α} and initially closed TCVs help each other to make it possible to design control logic that enables ISFR to perform power level shifts. What plays the main role in the logic is a main coolant pump (MCP) speed control. Let q(t) and $q_C(t)$ be the relative reactor power and its target to be specified by the reactor operator, respectively. For example, $q_C(t)$ should be unity for a constant power operation. Suppose that q(t) is greater than $q_C(t)$. Then, what we should do to get $q(t) = q_C(t)$ is to increase MCP speed ω. Because, as ω increases, the discharge pressure of the MCPs increases and hence void fraction in the reactor decreases to induce a negative reactivity. The MCP speed control alone does not suffice, but other controls also are needed. Among them is a reactor inlet valve control to keep the primary TCVs being closed during a normal reactor operation.

It has been confirned by calculations with the THYDE-NEU code [4] that, with the help of the reactor control system thus designed, ISFR can shift its power, provided that $q_C(t)$ changes quasi-statically so that $|(\Delta p)_{TCV}|$ can be kept less than 2 kPa.

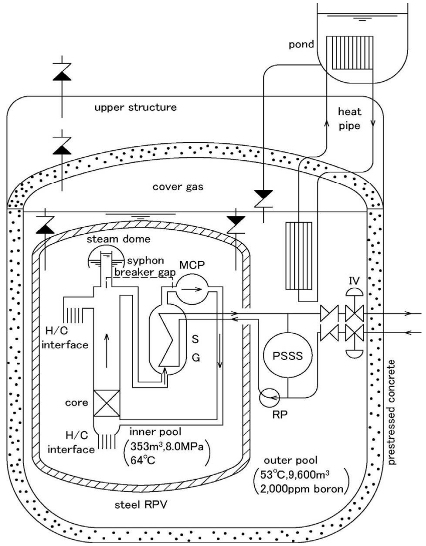

Fig.1 Schematic of ISFR

A steam generator tube rupture (SGTR) can be regarded as a design basis accident of ISFR. A calculation with THYDE-NEU for a double-ended rupture of 500 SG tubes gives the following results. Soon after the accident occurrence, the reactor power rises due to the positive void feedback. First, owing to disturbances brought about by the accident, one half of the secondary TCVs open not only to induce an upward flow through the PSSS, but also to trip all the pumps as well as the reactor controls. Next, owing to the MCP trip, one half of the primary TCVs also open to introduce the cold inner pool water into the CPS. The resultant negative reactivity is soon enough to override the positive void feedback. Hence, the the reactor power now turns around and then monotone decreases to the decay heat level, as a natural circulation persists in the primary system. Finally, a loss of on-site power occurs to isolate the SG-PSSS and hence the natural circulation tends to be established also in the secondary system.

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Advanced Light Water Reactor with Hard Neutron Spectrum for Realizing Flexible Plutonium Utilization (FLWR)

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An advanced LWR concept with hard neutron spectrum named FLWR is a BWR-type reactor with a core consisting of hexagonal-shaped fuel assemblies with a triangular tight-lattice fuel rod configuration. It has been proposed in order to ensure sustainable energy supply in the future based on the well-experienced LWR technologies.

The reactor concept of the FLWR is designed to utilize the most of the existing Advanced Boiling Water Reactor (ABWR) plant system. Therefore, only the core concept is new. The FLWR aims at effective and flexible utilization of uranium and plutonium resources by adopting a two-stage concept of core designs. The core in the first stage of FLWR is for intensive utilization and preservation of plutonium with no degradation of the isotopic quality of plutonium based on the experience of the current LWR and MOX utilizations. The one in the second stage realizes sustainable multiple plutonium recycling with a high conversion ratio over 1.0. When the technologies and infrastructures for multiple recycling with MOX spent fuel reprocessing are established, the core of the first stage proceeds to the second stage by only changing the fuel assembly design in the same reactor system.

The design target for the FLWR core is to achieve a high conversion ratio with keeping negative void reactivity characteristics, which is one of the most essential features for realizing a stable and safe operation in BWR-type reactors. The representative core designs of a 1,356MWe class large-scale FLWR are summarized in Fig.1. The core design of the first stage is a high conversion type core with a conversion ratio of 0.84, in which the gap width between fuel rods has been set to be 3.8mm by keeping the smooth technical continuity from the current LWR. In the second stage core, the gap width between fuel rods is reduced to 1.3mm and the fissile plutonium enrichment in the MOX fuel is increased up to 18 wt%. The rod diameter is increased and the rod pitch is kept to be the same as at the first stage. By adopting a short and flat core configuration with axial and internal blankets of depleted uranium as well as increasing the void fraction in the core, the second stage core has been evaluated to have a fissile plutonium conversion ratio of 1.04, which is defined as the ratio of the amount of fissile plutonium in the discharged fuels to that in the loaded fuels. The recycling characteristics in the second-stage FLWR core have also been investigated in detail. Based on these investigations, multiple recycling use of plutonium by using the FLWR core is considered to be feasible even under the advanced future reprocessing schemes with low decontamination factors.

In addition to the representative core designs mentioned above, advanced concepts of fuel and core designs have been developed for both stages to enhance flexible plutonium utilization in the FLWR. One of those advanced concepts is combined use of MOX and enriched UO_2 rods in the fuel assembly for realization of a high fissile-plutonium cnversion ratio of around 1.0 in the first stage. The advantage of the concept named FLWR/MIX is that the required plutonium enrichment in the MOX rods is rather low so that the fuel can be fabricated in a facility for the current MOX fuel utilization. The discharge burn-up can be designed to be

within the range of the current MOX fuel utilization. Therefore, it is expected that breeder-type operation in the first stage of FLWR could be realized under the framework of the current UO_2 and MOX fuel technologies and related structures. The new concept for the second stage is use of re-enriched reprocessed uranium from LWR spent fuels in the internal blanket region of the core. The internal blanket of re-enriched uranium can reduce the plutonium inventory to attain the same performances as the core design with depleted uranium blanket and attain a high conversion ratio of around 1.1, due to the enhancement of conversion characteristics in the internal blanket region. The proposed concept can contribute to realize a smooth and speedy transition from the LWR fuel cycle to the sustainable fuel cycle, as a core design for the transitioning period during which a sufficient amount of uranium reprocessed from LWR spent fuels is available as resources.

For realization of the FLWR concept, the core cooling capability in the very tight lattice fuel rod arrangement is one of the most important issues to be clarified. For this investigation, some critical heat flux tests with 7 to 37 rod bundles have been already conducted under the wide ranges of the experimental parameters such as the system pressure, the core inlet water temperature, the water flow rate, the radial power distribution and so forth, including the same pressure and temperature as in the ABWR plant. The results obtained so far have shown enough heat removal capability from the tight-lattice bundle with the rod gap width of 1.3 mm.

Studies on the core designs for both stages of FLWR as described above shows that the concept of FLWR is feasible and promising, and it is expected that it can realize flexible plutonium utilization in the future circumstances of the nuclear fuel cycle.

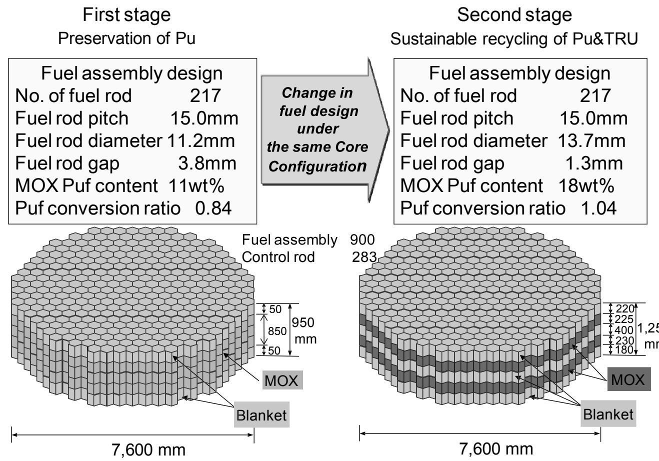

Fig. 1 Schematic of the representative core desigh for a large scale FLWR(1356MWe)

Minimum burnup required for sustainable operation of fast reactors without reprocessing

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Intellectual Ventures is pursuing [1] the vision of Edward Teller et al. of a Travelling Wave Reactor [2]. Except for the initial critical fissile fuel loading this reactor type is to be fuelled with fertile material. The basic principle of this reactor concept is to have a sufficiently high breeding gain to enable building up the fissile concentration in blanket elements to make their k_∞>1.0, before the core elements reach their radiation damage limit. If this principle can be achieved, it will be possible to operate fast reactors on fertile fuel feed without need for fuel reprocessing. The objective of the present work is to estimate the minimum burnup and minimum dpa required for achieving such a sustainable operation.

The first part of the study searched for the minimum burnup required for sustainable operation when using metallic uranium fuel in which the uranium is alloyed with 7.5 wt % zirconium. The analysis was performed for a simplified one-dimensional multi-zone slab system that is subjected to reflective boundary conditions. This idealized reactor is assumed to operate in equal length cycles; at the end of a cycle the highest burnup fuel zone is removed, each of the other equal-volume fuel zones is shuffled, and a fresh fertile fuel is loaded into the other side. An equilibrium cycle is searched such that k_∞ at the BOEC will be 1.0. The variable parameters of this study were the number of zones, zone thickness and the cycle length. The minimum required burnup of 13.4% was obtained for a system thickness of 80 cm made of 2 cm thick zones. The thicker the zones the larger becomes the minimum required burnup and the longer is the cycle length. The corresponding dpa is 250. Figure 1 shows k_∞ evolution of this system. The cycle length is, approximately, 4 months and the burnup reactivity swing is slightly above 1%. To obtain a longer cycle length at the same power density one would have to use thicker zones the consequences of which will be somewhat larger minimum required burnup and burnup reactivity swing.

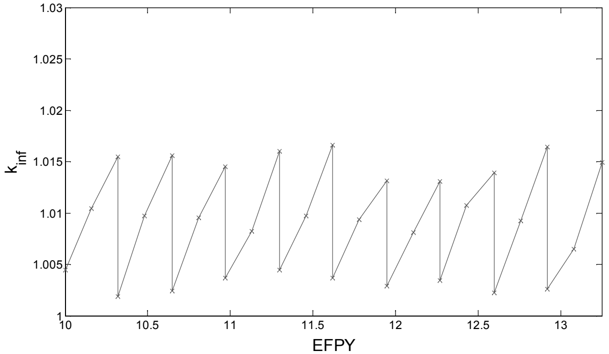

FIG. 1: Evolution of k_∞ for several equilibrium cycles for the system made of 40 slabs of 2 cm (the X scale is arbitrarily taken to start at 10 EFPY)

The second part of the study compared several types of possible fuel materials – metallic uranium and thorium, uranium carbide, uranium nitride, uranium oxide, thorium hydride and a dispersion fuel [3]. The analysis was performed using an approach similar to that described above. The system total thickness was 80 cm and made of 10 equal volume zones. This relatively coarse zone mesh was used in order to save computer time as we are searching for the relative ranking of the required burnup and accumulated radiation damage rather than for their absolute minimum values. It was found that the metallic uranium fuel provides the smallest minimum burnup of ~16.3%. However, nitride fuel having nitrogen enriched in ^{15}N that requires about 20.3% burnup accmulates just about the same number of dpa. All the other fuel types analyzed require larger burnups and their clad is subjected to a larger number of dpa.

The third part of the study estimated the minimum required burnup from a one dimensional finite core. The core is divided into 24 equal volume zones. At the EOC, the highest burnup zone is discharged, the 23 other zones are shuffled and a fresh blanket zone is loaded at the outermost zone. It is found that, when using U-(10$^{w}/_{o}$)Zr alloy fuel, the minimum required burnup is 20%. A somewhat smaller burnup is required if instead of depleted uranium one uses for the feed LWR spent fuel from which the fission products were removed and that was converted to a metallic alloy with Zr.

The minimum burnup required in finite cores is presently under investigation.

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Production Enhancement and Quality Degradation of Pu Produced in FBR Blankets

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For smooth deployment of FBR in near future, securing economy and non-proliferarion are pivotal factors. This study has two main objectives related to these two key factors. One is to enhance Pu production efficiency in blanket region of fast breeder reactor core. The other is Pu composition degradation to improve the proliferation resistance of FBR fuel cycle. The former contributes to reduce amount of blanket fuels to be fabricated and reprocessed to gain the same quantity of Pu, consequently it is expected to improve the fuel cycle ecomony. The composition of Pu generated in FBR blanket generally has very high quality, namely, equal to weapon or super grade. The latter objective is to deteriorate the quality of generated Pu by adjusting neutron spectrum in the blanket region and the initial composition of blanket fuels.

A conceptual core design of MOX fueled, sodium cooled FBR with 1500MWe rating is performed by using SLAROM and CIRATION code with nuclear data set prepared for fast reactors of JFS-3-J3.2. The initial fuels in the active core region contains 5% of minor actinides produced in UO2-fueled PWR with enrichment of 4.1% and burnup of 43GWd/t after 10 years cooling. A index FIR (fissile inventory ratio at EOC and BOC) is used to measure the net fissile balance that is suited for fuel cycle mass balance analysis in addition to the BR (breeding ratio) in the standard definition.

The moderator material used to tailor the neutron spectrum in the blanket region is ZrH. The depleted uranium pins in the radial blanket are replaced by ZrH pins in the volume fraction range from 0% to 30%. By increasing ZrH pins, the Pu production per unit mass of UO2 in blanket increased from 0.08kg-Pu/kg-UO2 to more than 0.1kg-Pu/kg-UO2. The FIR also slightly improved by replacing small fraction of UO2 by ZrH pins however it turned for the worse in the higher range more than 5% of ZrH.

Loading ZrH into blanket fuel assembly drastically affects the Pu composition at EOC. The quality of Pu is generally classified into super(<3%), weapon(3-7%), fuel(7-18%), reactor(18-30%) and MOX(>30%) garade based on the content of Pu-240 in Pu [1]. By refering this classification, the quality of Pu generated in the blanket is deteriorated from super grade to fuel grade if 10-15% ZrH is loaded into blanket.

Side effects caused by ZrH pin loading are also studied in terms of sodium void reactivity and power distribution. The loading of ZrH sligtly decrease the sodium void reactivity however the impact is not significant. The power density at EOC in the radial blanket is almost doubled by loading 30% of ZrH compared with no ZrH loading case.

The effects of moderator (ZrH) loading into radial blanket on Pu production efficiency and Pu quality deterioration are studied intending for better economy and enhanced proliferation resistance of FBR fuel cycle. The loading ZrH contributes to improve Pu production efficiency however FIR slightly improved just in the low range of ZrH (<5%). The impact of small ZrH loading on sodium void reactivity and power distribution is not significant and those can be handled by core design optimization.

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A feasibility study of hydrogen production by HTE coupled with SFR

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The nuclear energy can be an alternative to a fossil fuel as a primary energy source by producing not only electricity but also various secondary energy. As a secondary energy career, Hydrogen is expected to be an excellent candidate as a view point of resources and environment. It emits no carbon dioxide like fossil fuel combustion and it can be made from water which exists abundantly on the earth. From a viewpoint of long-term energy reservation, the hydrogen production by sodium cooled fast reactor (SFR) will be an important technology in the future.

High temperature steam electrolysis (HTE) coupled with SFR is an appropriate technology to produce hydrogen. The HTE can be operated under wide range of temperature without carbon dioxide emission since water is an only feedstock. (1)

Hydrogen production by the HTE can be explained by the fact that electrolysis of water occurs by giving energy (Δ H) to a solid oxide electrolyte cell (SOEC) with high temperature steam, as shown in equations below.

$$H_2O \dashrightarrow H2 + 1/2O_2 - \Delta H$$

$$\Delta H = \Delta G + T\Delta S$$

In these equations, ΔG is the Gibbs free energy change that is added as electricity and $T\Delta S$ is the heat energy that is added as heat. Hydrogen production efficiency (ϕ) is defined as bellow.

$$\phi = HHV / (W / \varphi + Q)$$

Here, HHV is a higher heating value of hydrogen, $W\ (= \Delta G + \alpha)$ is an electricity consumption. $Q\ (= T\Delta S + \beta)$ is a heat consumption, α is an extra electricity consumption in addition to the net electrolysis reaction. β is an extra heat generated in the electrolysis reaction. φ is a power generation efficiency of turbine generator

Figure1 shows a schematic drawing of hydrogen production system by the HTE coupled with SFR. The nuclear reactor generates heat, and turbine-generator converts a part of this heat to electricity, and then the residual heat is transported to the HTE system. The electricity is supplied via the rectifier to the SOEC in the HTE system, and it is also sent to power grid.

As a result of an analysis of heat and mass balanec on the HTE flow sheet coupled with SFR, maximum hydrogen production capacity is estimated around 150 Nm^3/hour per MWth of reactor power with 40% power generation efficiency *(φ)* of turbine generator. And hydrogen production efficiency (ϕ) is estimated around 40%. This hydrogen production efficiency is higher than conventional water electrolysis like alkaline water electrolysis coupled with SFR. This shows the HTE has a ptential to be one of the canditates for the future long-term energy reservation.

This paper shows system analysis of the HTE coupled with SFR and a technical feasiblity of hydrogen production by HTE coupled with SFR.

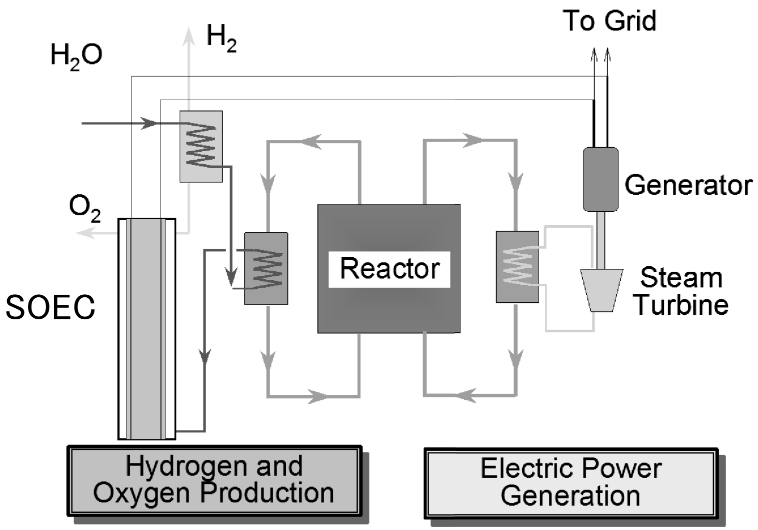

Figure 1 Hydrogen production by HTE coupled with SFR

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Risk Assessment of a Dimethyl Ether Steam Reforming Hydrogen Production System by an Advanced Reactor

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Hydrogen is regarded as a clean fuel because it does not pollute when burned with air. In the case of commercial use, there is a need to research how to produce hydrogen more efficiency and large scale. Although there are some methods of hydrogen production, it can be considered that the heat of the nuclear reactor is promising method. In the recent studies on the hydrogen production with nuclear power, there has focused on the technical issues[1][2]. Therefore, the object in this paper is the risk assessment of hydrogen production plant by Dimethyl Ether (DME) steam reforming with the use of nuclear power.

First, suitable systems with the DME steam reforming plant were studied for FBR (Fast Breeder Reactor). An conceptual design of hydrogen production plant system is shown in Fig.1.

Next, FMEA (Failure Mode and Effects analysis) was performed to identify initiating events for suggested coupled plant systems. After identifying initiating events, event tree analysis (ETA) was performed to quantify the average frequency of an accident at this complex plant. Then, generic failure rates are cited by published data in nuclear information archives in Japan[3].

The result of the PSA, the safety of the DME steam reforming plant with nuclear power depends on a rupture of reformer and heat exchanger between hydrogen and DME by the result of FMEA. Event tree analysis shows that the average frequency of hydrogen or DME explosion (shown in Table 1) is 7.7×10^{-7} year^{-1} in the case of the rupture of the reformer and 1.9×10^{-8} year^{-1} in the case of the rupture of the heat exchanger.

Despite the simplified and conservative analysis, the frequencies of end state were small value. More plant operational data and experiences could help to refine the analysis. Although, results of this assessment could be reflected detailed designs of hydrogen production plant with nuclear power in the future.

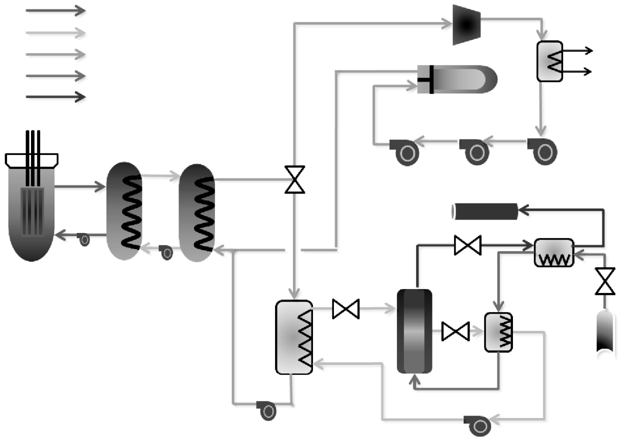

Fig.1 System diagram of coupling FBR and DME steam reforming plant

Table 1 Results of PSA – All end state frequencies

Initiating event	End state	frequency
rupture of reformer	Hydrogen or DME explosion	7.7×10^{-7}
hear exchange tube break	Hydrogen or DME explosion	1.9×10^{-8}
	Structual damage	3.7×10^{-9}

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Technical and economical assessment of Sodium-cooled Fast Breeder Reactors with increased cycle length

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Sodium-cooled Fast Breeder Reactors (Na-FBR) are currently being analyzed in the framework of the CEA-AREVA-NP-EDF collaboration within the GENIV Na-FBR Project.

The initial core design developed in the framework of this project, baptized SFR v2 [1], has a 5-batches core reload scheme, with a cycle length of 410 Equivalent Full Power Days (EFPD). This core design has a reactivity loss smaller than 1 pcm per day, so it already allows an increase of the cycle length compared to past FBR concepts (i.e. SuperPhenix, EFR [2]).

Nevertheless, an additional increase of the cycle length (reduction of the frequency of reloads) seems possible given the very low reactivity loss over the cycle of this core design. This attempt is driven by an objective of overall economic performance optimization, following the same trend as for the operation of PWRs.

This paper presents the on-going studies concerning the flexibility of the SFR v2 core design to improve its economical performances, by increasing the cycle length. Basically, the two following strategies were investigated:

- If the pellet design is the same as for the SFR v2b, the cycle length up can be increased (solutions up to 24 months have been explored) in innovative designs in which the fuel burn-up, the safety-related coefficients (Doppler, Na void worth) and the stability of the power shape during the evolution of the SFR v2b design are maintained, but with a penalty on the core behavior during an unprotected control rod withdrawal accident, which becomes less attractive that in the 5-batches configuration;

- With an innovative pellet design, the advantageous behavior of the SFR v2b core design during an unprotected control rod withdrawal accident is maintained, but with the penalty of a burn-up reduction (in order to ensure the same margin to MOX fuel melting as for the SFR v2b core) and with a slight increase of the core diameter.

This analysis shows that the increase of the cycle length can be counterbalanced by a possible worsening of the core behavior during the control rod withdrawal transient, possible burn-up reduction or core diameter increase.

In order to weigh the costs and the benefits in the framework of an economical optimization, a simple economical model was developed, taking into account the reactor cost (related to the core size), the cycle cost (related to the burn-up) and the plant availability (related to the cycle length and the fuel reload scheme). This model was made to sort different designs on the basis of a simple economical indicator, because of the difficulty – at the present stage of development of the reactor – to precisely determine the plant load factor. Finally, the core designs were sorted also on the basis of physical or technical considerations: a value of the main safety-related characteristics as close as possible as in the SFR v2b design (sodium void worth, reactivity loss over the cycle, margin to fuel melting in the hot spot), in order to keep the attractive features of this core design.

This study is to be considered as a step toward the definition of the future core design, which will result of a global optimization between economic performances and safety characteristics.

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Development of FR construction cost estimation method in FaCT (Fast reactor Cycle Technology development) project

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The economic competitiveness is the crucial issue for FR plant. Within the flow of FaCT, the commodity shall be reduced by introducing a compact design with the innovative technologies, although the ISI capability shall be required to ensure the plant reliability as the commercial reactor.

In order to evaluate the economy for the Japan Sodium-cooled Fast Reactor (JSFR), the account code named SCALLE (Sum of Cost Account Leading to future Logistics Economy) has been developed, in which the basic methodology is bottom up of component costs based on the amount of material data and the corresponding unit cost. The amount of material is evaluated based on the JSFR conceptual design study. The unit costs for materials and structures are derived from the value of public documents about FR and LWR design etc. mainly on the EEDB (Energy Economic Data Base) of U.S. DOE. Cost evaluation accounts are classified considering JSFR system configuration.

The code enables to take into account economical effect of learning and common use of balance of plant to evaluate a JSFR twin plant concept and NOAK (Nth of a Kind) as well as FOAK (First of a Kind). FOAK cost is assumed 30% of the overnigt cost based on the document published the Shicago univ. The learning effect factor defined as "ratio of cost reduction when the quantity of production double", and set 6% as equipment cost, 10% as site labor cost and 3% as indirect cost based on the document related to the Generation IV deliberation.

The difference between domestic and foreign prices and the price fluctuation are taken into account using economic indexes, for example, "Producer Price Index" piblished by US Bureau of Labor Statics whem we convert the value of equipment cost and material cost in US from document published year to wanted estimation year, and so on.

The validation and verification of SCALE code is carried out by evaluating the construction cost of past demonstration FR plant designed in Japan, and comparing evaluated value by SCALLE code with the evaluated value at the time.

As a examle of the JSFR construction cost estimation, the ratio of NOAK cost is 74% of FOAK cost. This code is used as basic information to evaluate cost of power generation in FaCT economics evaluation.

POSTERS OF SESSION 2:

Fast reactor coolant technology and instrumentation

The modeling of corrosion products mass transfer in circuits of LMFBR with sodium and lead coolant

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The purpose of modeling of corrosion products mass transfer in circuits of LMFBR with sodium and lead coolant is to evaluate the conditions for effective performance of coolant and structural materials, including the permissible levels of coolant temperature increase and concentration of impurities in coolant as well as to obtain quantitative data on output and accumulation of solid-phase impurities in the elements of hydraulic path of LMFBR circuits.

There were developed two types of models for the evaluation of corrosion products mass transfer in sodium-cooled circuits in one-dimensional approximation. The first model takes into account the dissolution and crystallization of impurities being transferred in sodium. The second model takes into account also chemical interaction of impurities in sodium. In both models the elements of the hydraulic part are simulated by round channels. The Lagrange system of coordinates is used in the mathematical description of processes in channels.

In the BN-600 primary circuit with the permissible concentration of oxygen in sodium the corrosion products mass transfer is evaluated using the first model, because there is no need for considering the chemical interaction of impurities. The evaluations showed that after approximately 400 passes through the BN-600 primary circuit there was established steady state disperse system in sodium with the parameters not changing with further passes. The quantitative characteristics (density of distribution and concentration of particles) have been obtained for the resulting self-preserved disperse system of corrosion products in the BN-600 primary circuit.

As calculational results have been obtained that the average size of corrosion products particles in the BN-600 primary circuit is about 0.02 microns; the bulk of particles in disperse phase does not exceed 0.04 microns in size. The concentration of an entire particle spectrum is about 0.0004 ppm.

From the evaluation, it follows that in the areas with the highest density of mass flux the rate of deposition formation is about 0.38 microns/year in the core and 0.08 microns/year in the intermediate heat exchanger at deposition porosity of 50 %.

In the core, the deposition formation takes place at the inlet region; at the core outlet there occurs dissolution of fuel assembly materials in sodium.

The data obtained testify that the mass transfer of structural material corrosion products in the BN-600 primary circuit under steady-state operating condition (at nominal parameters of the installation) will have no effect on performance characteristics of the primary circuit during the service life of the reactor.

When evaluating the mass transfer of corrosion products in sodium circuits with elevated concentration of oxygen in sodium, its chemical interaction with structural materials components should be considered, first of all with chromium and iron.

The second model considers the mass transfer of chromium in sodium in view of sodium/oxygen/chromium system formation and the reaction in coolant $Cr + 2Na_2O = 3Na + NaCrO_2$ as well as mass transfer of iron in sodium in view of sodium/oxygen/iron system formation and the reaction in coolant $Fe + 3Na_2O = 2Na + Na_4FeO_3$.

The mass transfer of chromium and iron was evaluated for a hypothetical simulating circuit with the temperature distribution along the length varying from 300°C up to 900°C. The system of equations describing the process and the applicable program module for its numerical solution have been proposed.

The calculations performed for oxygen concentration in sodium of 10 ppm showed that the minimum concentration of double chromium oxide $NaCrO_2$ in the near-wall region is attained at a maximum temperature of 900°C, while reaching the saturation value with the temperature decrease to 600°C.

It is evident from the calculation results that the rate of chromium leaching-out from the wall may be $1.5 \cdot 10^{-11}$ kg/(m^{2}s) (0.06 microns/year) in the extreme point at presence of 2.5 microns oxide film on channels wall.

The maximum flux of particles of $NaCrO_2$ formed in cold zone to the wall (a rate of deposit formation of $5.5 \cdot 10^{-11}$ kg/(m^{2}s) (0.14 microns/year)) corresponds to the peak sodium temperature of 900°C.

From the consideration of the change in concentration of iron double oxide Na_4FeO_3 in the near-wall region it follows that it does not reach the saturation in sodium along the full length of the circuit. The iron flux to sodium in the extreme point (900°C) is as great as $1.6 \cdot 10^{-7}$ kg/(m^{2}s) generally due to disintegration of Na_4FeO_3 on steel surface. However, the flux varies its direction in the cooled section, there occurs iron transition to the channel wall in the extreme point up to $2 \cdot 10^{-8}$ kg/(m^{2}s) that corresponds to the rate of deposit formation of about 80 microns/year.

The data obtained including the method for calculation of chromium and iron mass transfer can be used for the analysis of operation of LMFBR sodium circuits.

With reference to lead coolant, two models of steel oxidation have been developed, for the cases with prevailing mechanism of magnetite formation and formation of two-layer oxide film on steel surface. Based on the first model, the permeability constants for magnetite in lead were calculated using available experimental data.

The second model involves the mathematical description of oxidation process at simultaneous formation of magnetite and iron - chromium spinel layers of oxide film. The simultaneous solution of the equations obtained under the given boundary conditions allows the formation kinetics of each oxide sublayers to be calculated.

As it follows from the calculations for 650 ^{0}C, the magnetite mechanism of oxide film formation prevails at oxygen activity in lead from 1 up to 10^{-3}. At oxygen activity of 0.00013 and lower the iron-chromium spinel mechanism of oxidation is the only case.

The problem of modeling in one-dimensional approach of iron mass transfer in steel circuit with lead has been solved in view of the formation and transfer of suspended particles and chemical interaction of impurities in coolant as well as the formation of two-layer oxide film on steel surface. The mathematical description of the model, algorithm and computer program have been developed.

With reference to the BREST-300 primary circuit, the calculated data are obtained on distribution of suspension particles on sizes in lead coolant and rate of change in thickness of magnetite layer and steel wall of the circuit in extreme points in the core and steam generator. The release of corrosion products of structural material into the coolant of the primary circuit depending on the thickness of oxide layer has been estimated.

Large Size Sodium Purification Device used for producing nuclear grade sodium of CEFR

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Nowdays fast neutron breeder reactors in the world ,including China Experimental fast reactor(CEFR) choose liquid sodium as coolant. However the industrial sodium manufactured in sodium factories cannot be used directly as the reactor coolant because it contains various impurities. These impurities are harmful to the heat transfer properties, the mechanical and corrosive properties of the structural materials ,the nuclear properties, and even affect the reactor safe operation.

To guarantee the CEFR's safe operation, nuclear grade sodium quality control standard is developed. The industrial sodium is produced by Lan Tai Industry Stock Company(LTISC) in Inner Mongolia of China. Except for calcium and oxygen, the other impurities' content is less than the permitted content of impurities. In order to purify calcium and oxygen in sodium , Large Size Sodium Purification Device（LSSPD）was built by LTISC in Inner Mongolia in 2004. But the sodium purification technology,the analytical methods of oxygen and calcium in sodium and design were provided by China Institute of Atomic Energy(CIAE).The construction,commissioning and operation of LSSPD were finished by both sides.

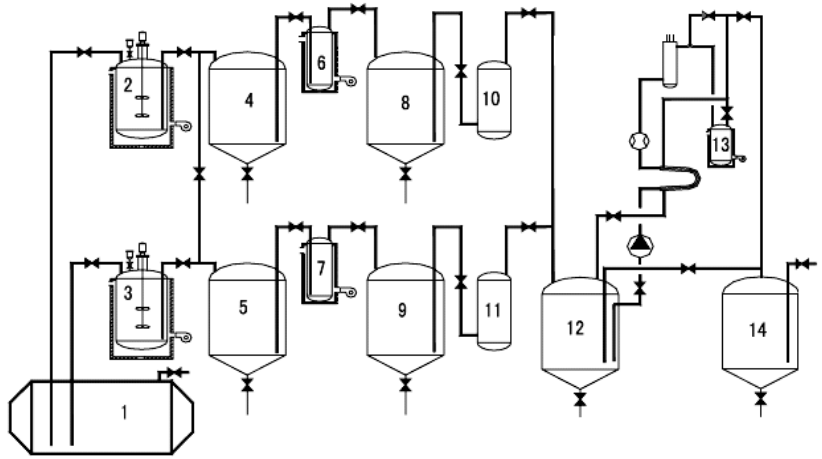
The flow chart of LSSPD is shown in Figure 1. It is composed of two same purification loops which respectively includes industrial sodium receiver, reaction tank, the first and second stage deposition vessels, the first and second stage filters,cold trap and nuclear sodium storage tank. Meantime argon gas system,vacuum system,instruments and control system are designed and installed.

Because it is easy for calcium to react with oxygen in sodium,calcium can be removed from liquid sodium by addition of sodium peroxide into reaction tank,low temperature deposition in the first and second stage deposition vessels,filtration of the first and second stage filters and cold trap purification system.Oxygen in the form of sodium oxide is removed in the process of purifying calcium.Key technical parameters are as follows:

Addition amount of sodium peroxide: 280g/t
Temperature and time of removing calcium reaction: 350℃,24h
Velocity of mechanical mixing: 100tr/min
Temperature and time of the first and second stage deposition vessels:Sodium is heated to 220℃,then decreased to 130℃.The whole process almost needs 24h.
Temperature of the first and second stage filters: 115℃～120℃
Sodium flow rate through the first and second stage filters: 1.5m^3/h
Cold temperature of cold trap: 115℃～120℃
Sodium flow rate through cold trap : (0.8—1.0)m^3/h
Sodium inlet temperature of cold trap: 220℃

In order to analyse calcium and oxygen in sodium,the secondary circuit sodium sampler is designed and manufactured by ourselves. It is based on the overflow sampling method.The analytical methods of calcium and oxygen in LSSPD's nuclear sodium are respectively established, i.e. Vacuum Distillation—Induced Coupling Plasma Atomic Emission Spectroscopy(ICP-AES) and Vacuum Distillation—Alkali Titrated by Acid.

LSSPD can produce 1.5t nuclear grade sodium per day. About 337 tons nuclear grade sodium was produced from Feb,2006 to May,2008, and filled into seven sodium storage tanks of CEFR. The device has provided a lot of precious technical parameters, design evidences and experiences of commissioning and operating successfully.



1—industry sodium receiver
2,3—reaction tank
4,5—the first stage deposition vessel
6,7—the first stage filter
8,9—the second stage deposition vessel
10,11—the second stage filter
12—NO.1 nuclear sodium storage tank
13—cold trap
14—NO.2 nuclear sodium storage tank

Figure 1 The flow chat of LSSPD

Development of a new electromagnetic flow meter in Sodium-cooled Fast Reactor

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<Introduction>

Electromagnetic flow meter (EMF) is usually applied to measure the flow rate in the heat transport system of a Sodium-cooled Fast Reactor. But it is difficult to apply such flow meter to the primary cooling system of a pool-type Fast Reactor, which has no pipings. Also currently designed loop-type Fast Reactor has only relatively short piping with high velocity, which isn't suitable to measure stable flow. The purpose of this study is to develop a new type EMF for Fast Reactor to measure flow at the annular flow pass of sodium components under high temperature and radiation circumstance. Fig.1 shows the configuration of a new type EMF. It consists of magnetic cores and exciting coils for forming a magnetic field perpendicular to an outer duct of flow area, and sevral pairs of electrodes for measureing a voltage generated by the movement of the sodium across the magneteic field. This EMF has superior characteristics such as design flexibility to extend flow range, and minimize the capacity of flow calibration facility which needs only one segment test.

<Remarks of elementary study of a new type EMF>

For a loop-type Fast Reactor such as JAEA Sodium-cooled Fast Reactor (JSFR), location of a new type EMF meter could be annular space of an integrated component that includes intermediate heat exchanger (IHX) and a sodium pump. For a pool-type Fast Reactor such as the 4S(Super-Safe, Small and Simple), the location could be the annular space of the electromagnetic pump outlet. Main specifications for each EMF are studied according to the operating temperature, flow range, the gap size of annular flow pass and so on. Targeted value of output voltage is approximately 5 mV.

<Numerical analysis>

Characteristics of the new electromagnetic flow meter is analyzed by coupling 3-dimensional numerical electromagnetic fluid analysis code (Falcon) and CFD code (Star-CD). As for the 4S, the analysis result shows that 1200 ampere turn (AT) of exciting coil current is required to get over 5 mV output voltage for flow rate 10.6m^3/min. Output voltage of flow meter increases monotonically according to increase of sodium flow, so basic feasibility is confirmed by this analysis.

<Sodium loop testing>

It is planned to perform the sodium loop testing of the new EMF test articles in the sodium loop at TOSHIBA. Two test articles will be made for sodium test, one is 6 segments test flow meters that model 360° area of annular flow, and other is 1 segment test flow meter

that models 60° area of annular flow. Each test article is approximately 2000 mm height, and outer diameter of flow area is 800 mm, flow gap of sodium coolant is 10 mm.

<Conclusion>

This report will show an elementary study of the new electriomagenetic flow meter and basic analysis results of the characteristics.

This study is the result of “Development of a new electromagnetic flow meter in Sodium-cooled Fast Reactor” entrusted to “Toshiba” by the Ministry of Education, Culture, Sports, Science, and Technology (MEXT) of Japan.

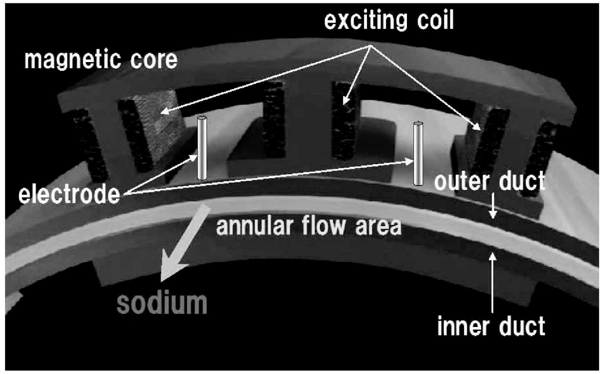

Fig.1 configuration of a new electromagnetic flow meter

Ultrasonic Flowmeter for JSFR

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The piping materials of the Japan Sodium-Cooled Fast Reactor (JSFR) at the commercialized stage, will be ferromagnetic materials Mod. 9-Cr steel. Therefore, it is not feasible to adopt the electromagnetic flowmeters used in conventional FBR plants.

This paper describes the developmental status of the ultrasonic flowmeter system(USFM) as a substitute flow rate measurement system to JSFR.

The features of the USFM are the following;

- In consideration of the double wall piping structure on JSFR, ultrasonic transducers should be installed directly on the surface of the inner primary coolant pipe. Therefore, the transducers should work properly under the temperature of 395 degrees C at the rated power, and be replaced by remote replacement system.
- The transducer remote exchange system should maintain with air tightness between the inner primary coolant piping and the outer piping during the normal plant operation, apply appropriate pressure to the transducers against the inner primary coolant piping, and exchange the transducers without removing the outer piping under the maitenance outage.
- Multi-pass propagation time method is effective for detection of flow rate in the short entrance region (e.g. in the short straight piping) and the requirements of the signal processing equipment are the following;
    - Linearity and repeatability of output signal : less than ± 2% of Full Scale
    - Fluctuation rate of output signal : less than ± 5% of median
    - Response : less than 0.3 s
- The USFM is designed as one of the Safety Protection System.

The potential use of an alternative fluid for SFR intermediate loops : selection and first design

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Abstract – Among the Generation IV systems, Sodium Fast Reactors (SFR) are promising and benefit of considerable technological experience, but improvements are researched on safety approach and capital cost reduction. One of the main problems to be solved by the standard SFR design is the proper management of the risk of leakage between the intermediate circuit filled with sodium and the energy conversion system using a water Rankine cycle.

This risk requires notably an early detection of water leakage to prevent a water-sodium reaction. One innovative solution to this problem is the replacement of the sodium in the secondary loops by an alternative liquid fluid, less reactive with water. This alternative fluid might also allow innovative designs, e.g. intermediate heat exchanger and steam generator grouped in the same component. CEA, Areva NP and EdF have formed a working group in order to evaluate different “alternative fluids” that might replace sodium. A first selection retained seven fluids on the bases of “required properties” as: large operating range (low melting point, high boiling point …), fluid cost and availability, acceptable corrosion at SFR working temperature. These are three bismuth alloys, two nitrate salts, one hydroxide melt and sodium with nanoparticles. Then, it was decided to evaluate these fluids through a multi-criteria analysis in order to point advantages and drawbacks of each fluid and to compare them with sodium. Lack of knowledge, impact on materials, design, working conditions and reactor availability should be emphasized by this analysis, in order to provide sound arguments for a research program on one or two most promising fluids. A global note is given to each fluid by evaluating them with respect to “grand criteria”, weighted differently according to their importance. The grand criteria were: thermal properties, reactivity with structures, reactivity with other fluids (air, water, sodium), chemistry control (including tritium management), safety and waste management, inspection maintenance and repair (ISI&R), impact on components and circuits, availability and cost, level of use. The impact on reactor availability and manageability and the level of knowledge on each fluid were estimated through the former criteria and introduced in the final evaluation as main criteria. The aim of this paper is to present the method of evaluation, the results obtained and the choice that have been made. The impact on design and operation are enhanced for the most promising fluids. It was found that sodium remains the most interesting intermediate fluid. However, Lead Bismuth Eutectic and sodium with nanoparticles also presents some interests and should be further evaluated.

The experimental study on wetting behavior between liquid sodium and various plated stainless steel under low temperature condition

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【Purpose】A liquid sodium is used as a coolant of fast breeder reactors because of its adequate physical properties such as high thermal conductivity and wide range of liquidas phase temperature. It is very important to understand the wetting mechanism between the liquid sodium and metallic materials, especially stainless steel, under the relatively low temperature condition (less than 350℃) compared with that of the normal operation from the viewpoint of design and development of an inspection equipment using a ultrasonic technique. For example, an under sodium viewer (USV) using a ultrasonic wave sometimes fails to function in the sodium pool unless the liquid sodium wet the outer surface of ultrasonic sensor made of stainless steel. To resolve this problem, the experimental study on wetting behavior between liquid sodium and various plated stainless steel have been conducted to improve the wetting between the sodium and the stainless steel.

【Examination】About 20mg of sodium were dropped on a specimen of various plated stainless steel type-304 ($50mm^{L}\times 50mm^{W}\times 2mm^{t}$) under the temperature condition of around 250℃ and the inert atmosphere condition (oxygen concentration: less than 0.02ppm/vol, dew point: less than -70℃). The plating conditions of specimens are listed in Table 1. The plating thickness on the specimens was measured by X-ray spectrometer (XRS) and the accuracy was within ±10%. The sodium spreading behavior on the specimen was visually observed and recorded by video camera, and the sodium spreading area (Δ S) on the specimen was evaluated in each second by the image analysis software. After the test, these specimens were rinsed by alcohol and distilled water, and these solutions were analyzed by chemical analysis to quantify the amount of dropped sodium and plated elements dissolved in the sodium. The post-test examinations of the specimens have been made by a scanning electron microscope(SEM) and an energy dispersive X-ray fluorescence spectrometer(EDX) at the part encroached by the liquid sodium.

【Result】

- The dropped sodium on the plated specimens of No.7 and No.13 spreaded to the almost whole surface within a few seconds. On the other hand, the droplet on the specimen of No.15 did not supread at all after 15 minutes.
- The wetting on Au surface plating was better than that on In surface plating.
- The wetting on thick Au surface plating was better than that on thin Au surface plating.
- The wetting on Ni foundation plating was better than that on Pd foundation plating or none foundation plating.

【Conclusion】The wetting behavior between liquid sodium and various plated stainless steel depends on not only surface plating elements but also the surface plating thickness and fundation plating elements.

Table 1: Plating conditions of specimens (○ indicates plating)

Plating element ※	Thickness (μm)	Specimen No.														
		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
Surface of stainless steel type-304																
Ni	5.0		○	○	○	○	○	○				○				
Pd	2.0	○		○	○	○			○	○	○	○				
Au	0.05	○	○	○	○	○	○	○	○	○	○	○	○	○	○	
Au	1.0	○	○		○		○		○			○	○	○		
In	2.0	○	○	○							○	○	○			

※oder from stainles steel type-304

Demonstration of Remote Field Eddy Current Testing of Double Wall Tube with Wire Mesh Layer

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A helical-coil-type double wall tube filled with a wire mesh layer is proposed for a steam generator (SG) of a new-type fast-breeder reactor in Japan in order to further increase reliability in operation. This double wall tube consists of inner and outer tubes and a wire mesh layer between inner and outer tubes. An inservice inspection (ISI) technology is required to put this double wall tube SG to practical use. We need to detect small defects on an outer surface of an outer tube during an insert of a sensor into an inner tube. A major candidate for this inspection is a remote field eddy current testing (RFECT). An objective of this development is to demonstrate a defect detection of a double wall tube with a wire mesh layer using a RFECT.

A RFECT was required increase of sensitivity and spatial resolution to detect a smaller defect. As a high sensitivity technique, to increase an indirect magnetic field intensity, we focused attention on increasing a direct magnetic field intensity in vicinity of an exciter coil and devised a method to concentrate a direct magnetic flux into an intense field by the use of an exciter coil with a magnetic material (flux guide) [1][2], additionally, applied a flux guide to a detector coil. As a high spatial resolution technique, we adopted a multiple detector coil arrayed circumferentially. We optimized a shape of a flux guide made of high magnetic permeability material by a magnetic analysis and made prototype coils. Two experiments, a measurement of a magnetic field (a voltage of a detector coil) distribution in a double wall tube with a wire mesh layer and defect detection tests, were performed to verify effects of flux guides and a multiple detector coil. According to the experimental results, the indirect magnetic field intensity (the voltage of the detector coil in the region of the indirect magnetic field) increased more than 100 times by the application of the exciter and detector coils with flux guides. Finally, we detected a smaller defect over the wire mesh layer by the adoption of the multiple detector coil.

We designed the exciter and multiple detector coils with the flux guides to increase the sensitivity and spatial resolution of the RFECT. It was confirmed the indirect magnetic field intensity increased more than 100 times. We demonstrated a smaller defect detection of a double wall tube with a wire mesh layer using a RFECT and verified flux guides and a multiple detector coil were effective for high-sensitivity and high spatial resolution of a RFECT. We considered that this RFECT also was available for inspections of other thick heat exchanger tubes.

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Development of the in-vessel repairing technology with friction stir welding method for FBR

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FBR is supposed to serve for 60 years in order to be economically competitive with LWRs. To guarantee the safety operation, establishing the in-service inspection and repairing technology for FBR components which are exposed to the liquid sodium is essentially important. Since the Friction Stir Welding (FSW) process has a possibility to repair the components in liquid sodium without draining the liquid sodium form the reactor vessel for repairing, it can be an attractive candidate method for the in-vessel repairing.

Material and shapes for the FSW tool were surveyed to satisfy the demands above. As the result, a tool with the protruding pin made of PCBN with 9mm diameter x 6mm length has been selected among some 30 types of tools. It was also found that load control method was more preferable rather than position control method for the in-vessel repairing machine.

FSW operations with the selected tool and the control method were successfully performed in argon gas environment and subsequently, in liquid sodium to SS316L austenitic stainless steel plates with an slit-type artificial defect which was filled with sodium. Fig. 1 shows an example of the FSWed specimens performed in the liquid sodium. No defects were indicated by the visual testing of the cross section and radiographic testing, and the 5mm depth x 0.5mm width of the artificial defect which can be detected by ISI in FBR was well repaired. Plunging downforce for FSW was not more than 30kN, which is significantly low compared to the conventional FSW for austenitic stainless steel in order to relax the requirement for the in-vessel repairing machine. Obtained optimum process conditions for liquid sodium were surveyed. The needed welding speed was much slower than that in the argon environment to input enough friction heat for compensating the heat removed to the liquid sodium. The FSWed specimens were heat treated with the conditions corresponding to the 40 years of FBR operation. Even though chromium carbides precipitations were observed in some area of some of the specimens, tensile strength of the repaired specimens was the same level as the base metal.

Weld bead and HAZ, which are more likely to have defects compared to the base metal, were also FSWed in the argon environment and it was found that the optimum process conditions were the same as the base metal.

Considering the repairing on the curved in-vessel components, FSW on the tilted plate and curved specimen were performed. Allowable tilt angle were the range of -0.5 - +8.0 degree which corresponds to about 370mm length on the in-vessel component with R=2500mm.

An concept of the in-vessel repairing machine has been established. The machine is inserted through the ISI hole to the inside of the reactor vessel and reaches to the place to be repaired with jointed-robotic arm.

Present study is the result of “Development of the in-vessel repairing technology with friction stir welding method applicable in liquid metal” entrusted to “Mitsubishi Heavy Industries, Ltd.” by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT).

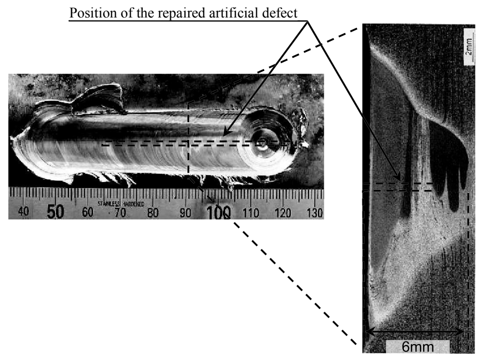

FIG. 1. An example of the FSWed SS316L in liquid sodium

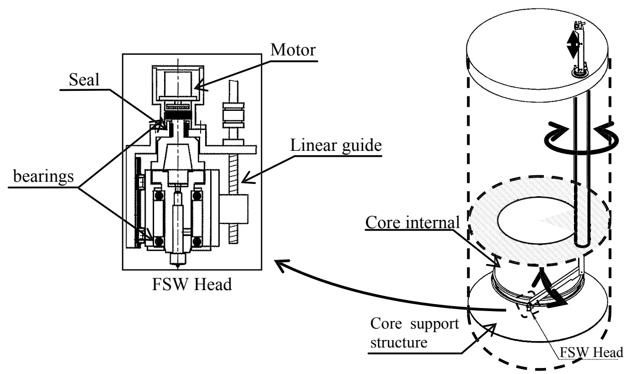

FIG. 2.Concept of the in-vessel repairing machine with FSW

In-Service Inspection and Repair Program for Commercialized Sodium-Cooled Fast Reactor

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Japan Atomic Energy Agency (JAEA) has promoted the Fast Reactor Cycle Technology Development (FaCT) project in cooperation with Japanese utilities. The Japan Sodium-Cooled Fast Reactor (JSFR) at the commercialized stage should have excellent prospects in operation and maintenance as well as the economy.

This paper describes the concept for the maintainability of JSFR which has been developed in the FaCT project.

We set the in-service inspection program in referring to that of the light water reactor and considering the characteristics of the SFR.

JSFR is designed with the double boundary structure filled with inert gas so that the safety can be secured by taking necessary measures against leakage of sodium after the leakage is detected even if the primary sodium boundary is damaged. Therefore, the continuous leakage monitoring is executed to the primary boundary as ISI in the regulatory framework. Voluntary inspection is also executed in consideration of a little operating experience with the SFR. In addition, visual test to detect deformations of or large cracks on the core support structure as well as the volumetric test to the heat exchanger tubes of the steam generator are performed.

The plant components of JSFR are designed to enable the above-mentioned inspections, and the inspection technologies are under development. Concerning to the reactor vessel, sensors that can inspect reactor structure in sodium of opacity and high temperature by using ultrasonic wave and the vehicle that can swim installing the sensor in sodium are being developed. And inspection holes to introduce the vehicle in the reactor are scheduled to be set up. To inspect the inside of the intermediate heat exchanger (IHX), it designs to pull out the pump shaft from the pump-integrated IHX.

Concerned with repair, we assumed the probability of the damage of each equipment and set the repair level appropriate for the probability. The damage with possibility to be generated several times in the plant lifetime is considered at the design stage so that the damaged parts can be easily repaired or replaced.

The environmental condition (temperature, dose rate and work space, etc.) for the inspection and the repair is considered. In addition, we are making an effort to secure the accessibility to the equipment as much as possible to correspond to the trouble not assumed at design.

Development of an ISI Robot for the Fast Breeder Reactor MONJU Primary Heat Transfer System Piping

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Background

The fast breeder reactor (FBR) “MONJU” carry out in-service inspection (ISI) in important components for safety. ISI of the primary heat transfer system (PHTS) piping is performed by sodium leak monitoring, a visual testing with ITV camera and a volumetric testing with ultrasonic[1]. The volumetric testing inspect maximum part of stress concentration in PHTS pipe by using ultrasonic. ISI use remote control robot on the grounds of high temperature (atmosphere 55 degrees C, pipe surface 80 degrees C) and radiation exposure condition (dose rate 10mGy/h, pipe surface dose rate 15mGy/h). Moreover, volumetric testing use tire type ultrasonic sensor on the grounds of a sodium boundary which chemically reacts with water and oil. Light-water reactors (LWR) can be inspected by ultrasonic that uses water and oil [2, 3].

Purpose

This development of inspection system is intended to use new control robot and new tire type ultrasonic sensor. The robot control adopt teaching control method. The target is reproducibility of less than ±5mm. The new tire type ultrasonic sensor adopt double oscillators, because of the multipath reflection wave from contact rubber etc., the noise level decreases and consequently S/N ratio well. The defective detection target was decided to be a depth 50% electrical discharge machining (EDM) slit from pipe wall thickness (t= 11.1mm) with a signal per noise ratio (S/N) not less than 2 (6dB).

Results and Conclusions

We developed a new inspection system (Fig. 1) for the in-service inspection of PHTS of the FBR “MONJU”. Moreover, we carried out performance test about new inspection system. The control performance of the new robot driving confirmed it was about less than 5mm by the experiment. The detection performance of new tire sensor confirmed it was detectable an EDM slit with depth 10% from pipe thickness and with a S/N ratio not less than 4.0 (12.0dB).

The robot and new tire sensor that developed as a result of the experiment confirmed the performance that was able to be used for the inspection was possessed.

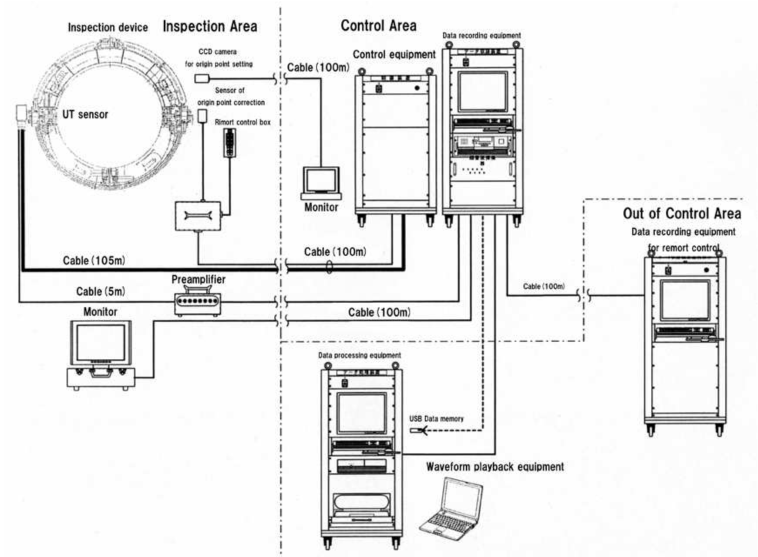

Fig.1 Inspection system

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POSTERS OF SESSION 3:

Fast reactor safety: approaches and issues

CAFE experiments on the flow and freezing of metal fuel and cladding melts (1) -Test conditions and overview of the results-

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For metal fueled fast reactors, assessment of the core disruptive accident (CDA) is necessary for both design and licensing. This assessment has generally addressed an initiating accident phase (fuel pin disruption and material relocation within subassemblies), followed by a transition phase (disruption of subassemblies and gross core-material relocation), and concluding in a post-accident heat removal phase (establishment of a stable, coolable configuration of core material). In order to assess the CDA, knowledge of the flow and freezing behavior of the molten core material as it moves axially through and beyond the core is required. This information is important for predicting fuel relocation during both the initiating and transition phases. For uranium-based metal fuel in stainless steel-clad fuel pins, the flow and freezing behavior of the core melt is complicated due to metallurgical interaction that will occur between the melt and the steel structures such as intact cladding, duct wall, or below-core structures. This interaction results in composition changes of the melt including eutectic formation and associated changes in the freezing temperature of the melt and other thermo-physical properties. The objectives of the Core Alloy Flow and Erosion (CAFE) experiments are to investigate the fundamental flow, metallurgical interaction, and freezing behavior of uranium-iron-type melts within iron-based trough-shaped flow channels and provide information that can support the development of mathematical models that describe the movements of molten fuel-bearing core materials during CDAs.

The CAFE experiment apparatus consists principally of the induction heating system, melt flow system (comprising the crucible, flow controller cup, trough-shaped flow channel, and catch cup), confinement and ventilation system, instrumentation and control system. Melt produced in yttria-coated crucible by induction heating was received at the flow controller cup so as to prevent splashing of the melt as the melt drops into the trough. Then, melt flowed down within approximately 660mm long inclined trough and was received by the catch cup located below the bottom of the trough. Flow was observed and recorded by three video cameras. Several thermocouples (TCs) were placed inside the crucible, near the flow controller cup, and on the top surface of the trough almost directly under the discharge tube of the flow controller. Twelve TCs were also attached to the bottom of the outside of the trough at equally-spaced intervals along the axial centerline of the trough. An optical pyrometer was also mounted to view the interior of the trough. Post-test examinations include cutting each trough transversely into numerous segments, weighing each segment, and taking photographs of both ends of each segment.

Summary of CAFE test conditions and results is presented in Table 1. Four UT series tests were conducted using molten uranium whose melting point is ~1400K. Two E1T series tests

were performed using U-Fe eutectic mixture whose melting point is ~1000K. In each test, 1 to 1.65 kg of melt was introduced into an inclined trough.

In the early tests, UT-1 and UT-2, melt flow down the trough was limited by rapid freezing and spillage over the sides of the trough. This was due to a number of factors (including small discharge tube diameter and trough angle), but the most important factor was the low temperature of the trough. In subsequent tests (E1T-1, E1T-2, UT-3 and UT-4), troughs were heated (up to ~870 K maximum) in order to prevent early freezing. In UT-3, in order to minimize trough penetration, melt temperature into the trough was decreased to just above the melting temperature. As a result, the penetration was only a few pin-hole-size perforations and no overflow from the trough occurred. In UT-4, although penetration occurred in the top part of the trough due to higher melt temperature than that in UT-3, more of the melt flowed completely down the trough.

In E1T-1, upon release of the U-Fe melt into the heated trough, most of the melt flowed easily down the trough and entered the catch cup. It should be noted that approximately 22% of the total melt had penetrated the winged top of the trough where it entered the trough. After the test, a fairly uniform layer of frozen melt remained inside the trough. In E1T-2, although flow and freezing behavior looks similar to that in E1T-1, the trough was completely penetrated in three areas due to higher temperature of the trough than that in E1T-1.

These test results provide understandings on fundamental flow and freezing behavior of melts including metallurgical interaction in the steel flow channels with a variety of melt and flow channel conditions and also offer useful information for developing analytical models to describe such behavior. Details of the initial development of the analytical model are described in Wright, etal. [1]

Table 1. Summary of CAFE test conditions and results

Test ID	UT-1	UT-2	UT-3	UT-4	E1T-1	E1T-2
Fe fraction of U melt (%)	0	0	0	0	33	33
Melting point of melt (K)	1403	1403	1403	1403	998	998
Melt temperature in crucible (K)	~1770	1783	1603	~1770	1773	1793
Melt temperature into trough (K)	~1630	~1620	1448	~1650	~1620	1643
Initial trough temperature at top (K)	363	353	873	873	673	873
at middle (K)	343	333	853	853	763	883
at bottom (K)	338	328	623	513	423	573
Trough angle (degree)	10	30	30	30	30	30
Inner diameter of discharge tube (mm)	3.2	6.4	6.4	6.4	6.4	6.4
(a)Melt mass into trough (g)	1030	1560	1490	1644	1264	1270
Post-test mass of melt debris (% of (a))						
In trough	90	18	10	2	6	12
Escaped over or through trough	10	37	0	3	22	2
In catch cup	0	45	90	95	72	86
Melt travel time to bottom of trough (s)	-	~1.6	~0.9	~1.3	~0.4	~0.4

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CAFE experiments on the flow and freezing of metal fuel and cladding melts (2) – Results, analysis, and applications –

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Postulated severe accidents in metal-fueled fast reactors involving cladding failure could include the flow of molten fuel within mostly-intact cladding, within inter-pin flow channels, along the surface of duct walls, or into structures above or below the core. Because molten uranium-bearing alloys metallurgically attack stainless steel at a rate highly dependent upon temperature, melt composition, and the potential presence of U-Fe eutectics or U-Fe intermetallic compounds, there has been considerable uncertainty in the way molten fuel could flow and freeze under such accident conditions.

The series of Core Alloy Flow and Erosion (CAFE) laboratory experiments generated melts of uranium or uranium-iron eutectic to flow within inclined stainless steel troughs to investigate the fundamental aspects of that behavior,. Details of the experiment design and a summary of the data obtained are described in Fukano, etal.,[1] and the principal components of the experiment apparatus are shown in Figure 1. Analysis of the experiment results utilizes various approaches up to and including computational fluid dynamics (CFD) modeling.

The materials, test conditions, and phenomena in the CAFE experiments are relevant to severe-accident situations but where reactor coolant is absent. Although the geometries in the experiments differ from that in a reactor, it is anticipated that models of the behavior observed in the experiments will be useful in modeling the same general phenomena in the analysis of reactor behavior.

Relevant phenomena include the simulaneously-occurring phenomena of flow down an open trough, heat transfer between flowing melt and trough, metallurgical attack of the trough (liquefaction of the trough at temperatures below its melting point), uptake of the ablated trough material into the flowing melt (thereby changing the composition and thermophysical properties of the melt), plate-out of melt onto the trough as the melt freezes, development of two-phase melt flow as low-melting zones within the inhomogeneous melt freeze, and flow disruptions due to physical changes in the surface of the trough (resulting from the attack and plate-out.)

A basic part of the analysis requires the time-varying rate of flow of melt into the top of the trough, which was not directly measured during the tests. Information from the video records and from post-test mass measurements assisted in making such estimates. Comparison of the computed and measured temperatures needed to take account of the fact that the melt did not flow perfectly down the center of the trough (where the thermocouples were located) but included some lateral motion that caused the melt to slightly "snake" up and down the walls of the trough in an irregular manner. Furthermore, the analysis is complicated by the video observation that the initial flow exhibited unexpected accelerations and decelerations, as well

as breakup into sub-rivulets, some of which were later overtaken and subsumed by faster-moving melt approaching from behind.

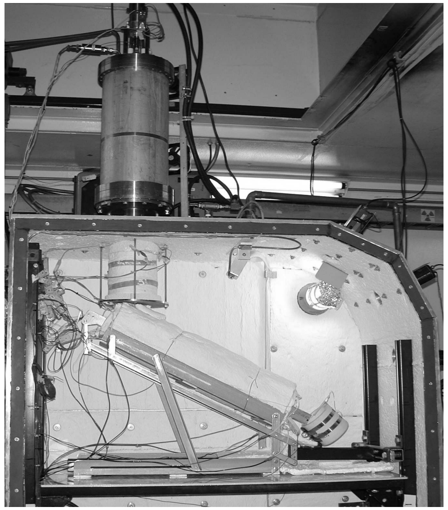

Figure 1. CAFE Experiment Apparatus

The analysis must take account of the rate at which the melt may have been interacting metallurgically with (attacking) the trough, in order to predict the degree to which such action may have been affecting the flow. The basic nature of the interaction[2] was found by analysis to involve both endothermic and exothermic regions, depending upon the melt composition. The potential for the dramatic increase (by nearly three orders of magnitude) in interaction rate known to occur at 1350 K is a key aspect of the analysis. Details of the interaction are revealed by post-test microphotographs of the interface between fuel and melt, showing the depth and nature of the attack at various locations along the troughs.

The program of detailed CFD modeling involves a progression of increasing complexity, from initial plausible models of time-varying flow of molten material down the trough, then adding heat transfer without phase change, followed, as feasible, by incrementally adding the complexities of freezing and metallurgical attack.

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Analysis of Mechanical and Thermal Consequences of Core Disruptive Accident: Approach for Current and Future SFRs

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For the 500 MWe mixed oxide fuelled Prototype Fast Breeder Reactor (PFBR), which is under construction in India, Unprotected Transient Over Power Accident (UTOPA) is found to be not energetic, i.e. it does not have potential to release the mechanical energy. However, during the pre-disassembly phase, the main vessel temperature rises to 767 K and the hot pool temperature is 960 K. Unprotected Loss of Flow Accident (LOFA) is however, energetic. During pre-disassembly phase, the temperature of main vessel remains at 683 K and the hot pool temperature rises to 853 K. By providing high safety margin, a Core Disruptive Accident (CDA) was defined as beyond design basis event which has a work potential of 100 MJ. The mechanical and thermal consequences of pre-disassembly and CDA phases are analysed in detail. The analysis includes determination of mechanical straining of main vessel and its internals, impact of sodium slug on the bottom of the top shield, sodium release to reactor containment building (RCB), design loads and analysis for core catcher. Mechanical analysis is carried out using the finite element code called 'FUSTIN' which uses Arbitrary Lagrangian Eulerian Co-ordinate system for describing fluid motions and convected co-ordinate system for modeling the geometrical non-linearity of structures. Further to overcome the limitations of the algorithm to handle the complicated core boundary movements with the presence of reactor internals, particularly, the control plug is a "two phase fluid element" is developed and implemented in the code. With this element, the core bubble boundary nodes need not be purely Lagrangian. For the thermal consequences, commercial codes are employed. For ensuring the structural integrity of decay heat removal exchangers and intermediate heat exchangers, mockup tests were conducted with $1/13^{th}$ scaled down models. Based on the numerical and experimental investigations, safety criteria specified for PFBR were respected with comfortable margin.

Based on the PFBR analysis and further through introduction of innovative shutdown systems and decay heat removal system, it is proposed to consider CDA as non-energetic for the future SFRs. However, thermal consequences would be investigated comprehensively.

Safety Related Investigations of a LFR Core with the Coupled TRACE/ERANOS System

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A core configuration of a lead-cooled fast reactor (LFR) was investigated with the coupled thermal-hydraulic/neutron kinetic system TRACE/ERANOS [1, 2]. With this system the axial power distribution of each fuel assembly (FA) was calculated taking into account thermal-hydraulic feedbacks. Due to the complexity of the core configuration (different zones of Pu enrichment, dummy assemblies and blank positions for control/shut-down rods), a coupled system seems to be the most suitable way to perform design optimization studies.

For the present investigation a new approach for the modeling of the thermo-hydraulic part was introduced. Instead of 1D parallel channels, the core is now represented by 3D component. TRACE allows to employ a 3D vessel either in cylindrical or cartesian geometry. The cartesian geometry option, which was used for the present study, provides the possibility to model the core with its FAs whereas only ¼ of the core was modeled thanks to symetry. The 3D approach allows to consider cross flow and mass flow redistribution as a consequence of the power/temperature development inside the core. These effects are of importance for several transients where 3D effects play an eminant role like loss of flow accident (LOFA) or loss of secondary heat sink.

For the presented design, the core inlet temperature is specified at 673 K and the average core outlet temperature is envisaged to be in the range of 753 K. It is obious that at certain core positions the FA outlet temperature will exceed the value of 753 K. Hence, it is important to identify these positions since the coolant temperature will effect the cladding temperature which is one of the crucial parameters in any safety related investigation. On the left side of Fig. 1, the temperature profile of the coolant is given for normal operation conditions. One can cleary see that the profile depends on the core design parameter (mainly Pu enrichment zones). For the case shown on the right side in Fig. 1 (non-symetric core feeding) the temperature on one inlet side was kept at 673 K while the other was fixed to 753 K (regular core outlet temperature). Besides the clear mixing in the lower and upper part, the channel with the highest lead temperature (as well as cladding temperature) has now moved to an other channel (hot channel positions are indicated by arrows). For the first cases the cladding temperature of 823 K was not exceeded, but during the second one the limiting temperature was exceeded considerably. The situation might be the similar for other scenarios. Hence, it is important to investigate related scenarios to evaluate the impact on the integral plant.

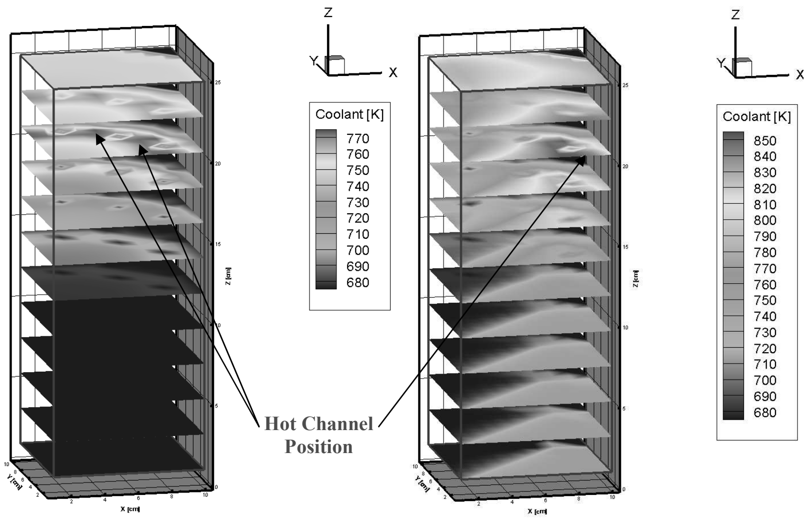

FIG. 1. Temperature distribution of the lead in the core for normal operation (left) and for non-symetric core feeding (right)

[1] MONTI, L, SCHULENBERG, T., Coupled ERANOS/TRACE System for HPLWR 3 Pass Core Analyses (Proc. Int. Conf. on Advances in Mathematics, Computational Methods and Reactor Physics. Saratoga Springs, USA, 2009)

[2] SANCHEZ ESPINOZA, V.H., JAEGER, W., TRAVLEEV, A., MONTI, L., DOERN, R., Neutronics and thermal Hydraulics Coupling Scheme for Design Improvemnts of Liquid Metal Fast Systems (Proc. 13th Int. Topical Meeting on Nuclear Reactor Thermal Hydraulics. Kanazawa City, Japan, 2009)

Computational efficiency analysis of fuel pin damage registration and fuel assembly damage location by means of a sector fuel failure detection and location system

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Fuel pin clad integrity loss detection at early accident stage is of major importance and therefore detection should be carried out timely. The solution allows to minimize consequences of the initial failure event. Another problem important for mitigating the accident consequences is to execute high-speed location of an fuel assembly with demaged pin within a reactor core configuration.

Both abovementioned problems have been solved by means of computational transport analysis of fission products ingressing after fuel pin clad integrity loss into reactor coolant flow and migrating along a primary circuit train.

The calculation runs have been carried out as implemented for the BN-600 fast reactor which is provided with an especially designed fuel pin clad integrity loss registration system (FPCILRS). Six delayed neutron detectors are mounted on outer surface of a safety reactor closure within faces of the intermediate heat exchangers (IHX) at their income window level (Fig.1).

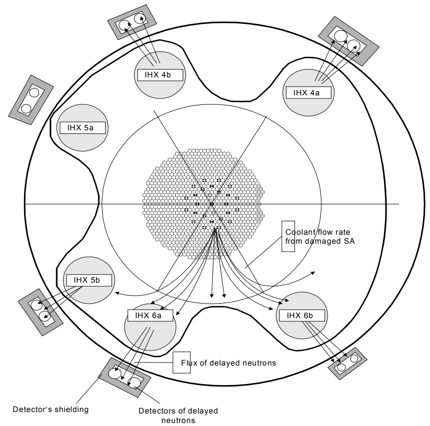

Figure 1. Location of delay neutrons detectors for reactor BN-600

When clad integrity of a fuel pin is occurring, fission products generate delayed neutrons, ingress into coolant volume, thereafter the products are captured by coolant flow and are transported toward the IHX income windows, where their presence is detected by the delayed neutron detectors.

Computational analysis have been carried out after three steps as follows:

- First step: Detailed calculation of a coolant flow velocity field at top reactor region including a reactor core part located above integrity loss point plus upper reactor chamber and intermediate heat exchangers.
- Second step: Transport calculation for the fission products ingressed into coolant flow train.
- Third step: FPCILRS detector signal estimate under established unsteady-state in-core delayed neutron source concentration fields.

The reactor design has been described under 3-D cylindrical geometry approximation. Unsteady-state in-core delayed neutron source concentration fields have been computed under two approximations (Euler approximation and La Grange approximation). First case is solution of a 3-D mass transfer equation; second case is movement path calculation for well-established velocity fields of the fission products ingressed into coolant flow. Both approaches account fission products decay in the course of their in-core transport.

For known delayed neutron source concentration fields, FPCILRS detector signal have been approximated by bulk in-core concentration field with a space weight factor which accounts delayed neutron absorption and scattering along a path from given point to a detector. The factor has been computed after neutronics computational procedures.

Initiating failure has been assumed as follows:

- major fission products ingress across fuel pin clad leakage point at steady velocity as long as 2 seconds;
- fuel pin clad integrity loss point is located at upper reactor core end level.

Parameters have been varied as follows:

- radial and azimuthal location of fuel pin clad leakage against the reactor core;
- half-time period of delayed neutron sources ingressed after fuel pin clad integrity loss.

Typical computed FPCILRS detector signal value/time dependence is presented by Figure 2.

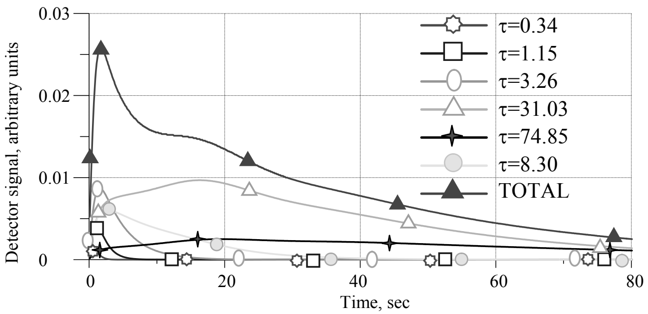

Figure 2.Total FPCILRS detector signal and contribution of 6 source groups with half-time period τ after fuel pin clad integrity loss at a peripheral fuel assembly

The calculation runs resulted in conclusions as follows

- isotopic composition of particulated fuel ingressing after loss of fuel pin clad integrity into coolant flow is major factor impacting on opportunity to detect an accident by means of the FPCILRS detectors. Transportation time for delayed neutron precursors ingressed into coolant flow from core up to FPCILRS detectors is as long as 5÷10 seconds. Therefore, precursors with commensurable half-time period will be decayed considerably

during their drift toward the detector operation range. It a reason of more pronounced contribution of the long-lived isotopes. E.g. namely group with half-time period 31.03 sec presents maximum contribution.

- accident detection efficiency is under strong impact of radial and azimuthal location of a failed fuel pin rod. There are several low sensitivity regions within horizontal cross-sections (they are displaced toward primary coolant pump); if the loss of fuel pin clad integrity is occurring at these regions, expectable detector signal could be reduced by one order of magnitude.
- FPCILRS data provides rough location of a fuel assembly with damaged pin (one could indicate reactor core sector pertaining to an IHX).
- location of fuel assembly with damaged pin could be considerably enhanced by increasing number of delayed neutron detectors (up to 2-3 detectors for each IHX)

Severe accident containment-response and source term analyses by AZORES code for a typical FBR plant

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JNES is developing severe accident analysis codes in order to apply to the probability safety analysis (PSA) for a typical fast breeder reactor (FBR). AZORES code analyzes the severe accident phenomena in the reactor containment that reactor coolant (sodium) and molten core debris are released from the primary cooling system boundary, and the discharge rate to the environment of fission products (FP). This report summarizes analysis results using the AZORES code for a PLOHS (loss of decay heat removal function) accident sequence with the actual plant system about the containment bypass scenario (CVBP) and the containment failure scenario by hydrogen deflagration or detonation. The coolant temperature of the primary system and the secondary system in the PLOHS sequence increases at the almost same temperature, and the creep damage to the reactor coolant boundary will become remarkable if coolant temperature exceeds about 1,100 K.

In the CVBP scenario, when an intermediate heat exchanger is ruptured by creep and the boundary of the secondary system is failed, the path from the primary system to environment is formed. Then, the reactor vessel (RV) is failed and sodium in the primary coolant system releases into the reactor vessel room (RV room). Sodium of high temperature which fell in the RV room damages the floor liner, and generates hydrogen by a reaction with concrete. In addition the reactor core is exposed into atmosphere and the core temperature increases with decay heat and then volatile FP and non-volatile FP are released to the environment through the secondary system from the primary system.

In the non-CVBP scenario which the intermediate heat exchanger does not fail by creep, core debris falls into the RV room after reactor vessel failure or evaporation of sodium coolant molten. FPs released from the reactor vessel are retained in the RV room, the primary system room, the containment dome and so on. The hydrogen generated by sodium-concrete reaction and debris concrete reaction moves to the reactor containment dome with sodium aerosol. Although the diffusion flame combustion of hydrogen occurs by igniting with sodium aerosol, since the flame velocity is slow, the reactor containment vessel is not damaged. However, if ignition of hydrogen is considerably delayed, the containment vessel may fail by deflagration or detonation of hydrogen although possibility is small enough.

The CVBP scenario and hydrogen burning scenario were analyzed in the AZORES code, and FPs released to environment were calculated as ratios to the initial core inventory. The release ratios of CVBP case were obtained to be $2x10^{-4}$ about the rare gas (Xe), $1x10^{-5}$ about the volatile FP (I), and $1x10^{-6}$ about the non-volatile FP (Ce). The release ratios of containment vessel failure case by hydrogen burning were obtained to be 0.82 about the rare gas (Xe), 0.06 about volatile FP (I), and 0.003 about non-volatile FP (Ce).

These analyses by the AZORES clarified quantitatively the release ratio to the environment of FP in both case of CVBP and non-CVBP for the PLOHS sequence of a typical FBR plant.

Study on Energy Release Mechanism during ULOF Initiating Phase of LMFBR

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Japan Nuclear Energy Safety Organization, JNES, has prepared safety analysis codes for liquid-metal cooled fast breeder reactor (LMFBR) as tools of technical support for licensing procedure by the national regulation authorities.

The LMFBR cores should be designed to eliminate a re-criticality potential during the postulated core disruptive accidents (CDAs). This is because the cores are not designed to operate in an optimum configuration from a reactivity point of view. Once a super-prompt criticality occurs, the core fuel will begin to melt and vaporize. The expansion of the fuel vapor drives the upper sodium slug to strike the reactor vessel head and generates a significant mechanical consequence to threaten the vessel integrity. Hence, the amount of the energy release during CDAs is the most important issue in the licensing procedure.

In this study, such a mechanical consequence generated during an Unprotected Loss-of-Flow (ULOF) initiating phase (IP) was investigated by using SAS4A[1] for the end-of-equilibrium-cycle (EOEC) core of Japanese prototype fast reactor, MONJU[2]. The reactor power during ULOF initiating phase is characterized by the superposition of negative and positive reactivities. The negative reactivity feedback is governed by fuel dispersion in the boiling subassemblies at the center region of the core. On the other hand, the positive reactivity insertion is introduced by FCIs in non-boiling subassemblies at the external region of the core. The mechanical consequence was evaluated based on the phenomenological event tree of ULOF IP as shown in Fig.1. Each heading parameter and its value were selected keeping a conservative aspect, based on the results of CABRI experiments[3]. Even in the case with the most conservative assumptions, the mechanical energy release was evaluated to be about 0MJ. This is caused by the inherent core degradation mechanism during ULOF IP as follows:

(i) The reactivity transient driving the super-prompt criticality is governed by the molten fuel and coolant interaction(FCI) at fuel pin rupture of non-boiling channels at the external region of the core. FCIs which simultaneously occur between several fuel subassemblies introduce a rapid insertion of positive void reactivity to the core. To simulate the FCIs, Cho-wright model is used and the fuel particle diameter of 100 μm is assumed in this study.

(ii) And also, the fuel failure location strongly affects the sensitivity of the 0MJ result. The fuel pin rupture takes place at the off-center of the axial location of the fissile fuel by taking account of the criterion of the fuel pellet-cladding mechanical interaction(PCMI) based on the results of the high-powered LOF-driven-TOP test of CABRI experiments, e.g. BI2. The failure at the off-center location suppresses the insertion of positive void reactivity and also introduces the insertion of negative fuel dispersion reactivity.

(iii) No mechanical energy release during the initiating phase implies to shift the recriticality potential to the afterwards. The typical core condition at the end of initiating phase is described in Fig.2, which is an example of typical calculation result. Sodium coolant is almost excluded out of the active core region. Core materials such as disrupted fuel particles and molten steel are axially divided and accumulated around the axial blanket fuel pins. Almost fuel of the core is remains the core region, involving a recriticality potential due to the fuel compaction. During the transition phase, recriticality might occur due to the multi-dimensional molten fuel motion.

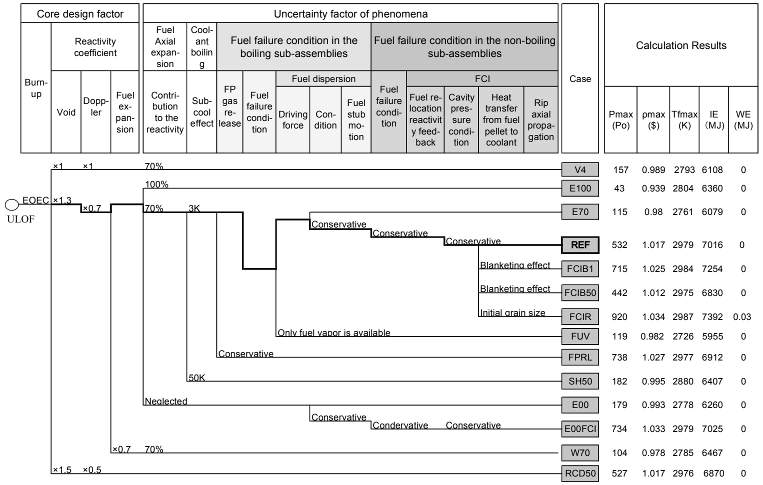

FIG.1. Evaluation of mechanical energy release during ULOF initiating phase based on the phenomenological event tree

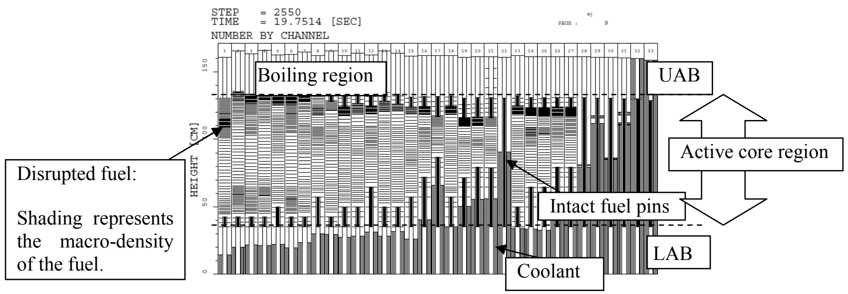

FIG.2. Core condition at the end of ULOF initiating phase

REFERENCES [1] TENTNER, A.M.,"The SAS4A LMFBR Whole Core Accident Code, "Proceeding Interational Meeting on Fast Reactor Safety, Knoxville, TN(April 1985) 989-998, [2] "Study on the Mechanical Energy Release during ULOF for FBRs Based on the Evaluation by Using SAS4A," JNES/SAE08-030, September, (2008) [in Japanese], [3] BARESCUT, J. C., et. al.," Interpretation of post failure phenomena observed during CABRI-TOP experiments, " Proceedings of an International Conference on Science and Technology of Fast Reactor Safety, Guernsey, UK (May 1986) 109-114

Plans of Verification Tests for the ACTOR Code Analyzing Fission Products Behavior in Primary Heat Transportation System of FBR

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The ACTOR is the code to analyze the transfer behavior of fission products (FPs) in the coolant sodium and cover gas, which are released from the fuel plenum or fuel pellets due to a cladding breach caused by temperatue increase during an accident (See Figure 1).

Principal anlysis models adopted in the ACTOR code; such as "FP release model", "Bubble transfer model in sodium", "FP transfer model from bubble to sodium" and "FP adsorption model" , are constructed based on the experimental results relating to each phenomenon

But, some of the analysis models are still desirable to be verified by experiments considering influence on the evaluation results.

This study is planned to carry out the necessary verifivation tests for the ACTOR code concerning the following phenomena relationg to the analysis models.

(1) Cesium transfer behavior from inert gas bubble to sodium
(2) Cover gas radiation behavior during temperature increase due to decay heat of FPs

As for (1), transfer behavior of cesium to coolant in fast reactor is different from that of LWR, because cesium is alkaline metal same as sodium. So, cesium is to be an important fission product in severe accident of fast reactor plant. The analysis model of ACTOR, transfer behavior of cesium to coolant sodium is conducted based on the results of tests carried out by using iodine as fission product, so it is necessary to carry out the tests by using cesium for verification of ACTOR analysis model.

As for (2), radiation heat transfer is not considered during cover gas temperature increase in the ACTOR analysis model, because argon applied as cover gas in fast reactor plant is monoatomic molecule. It is necessary to confirm the validity of this analysis model and to investigate a means to control cover gas temperature.

This paper will report the test plans and test apparatus of verifivation tests for the ACTOR code mentioned above.

The verification tests will be carried out by Hokkaido University.

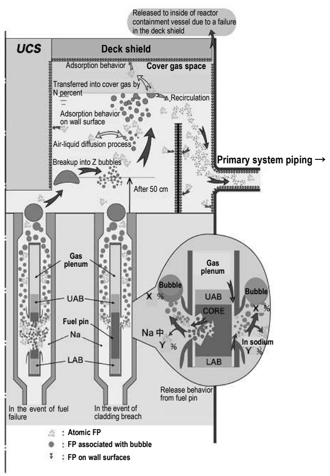

Figure 1 Phenomena analized by ACTOR

A fundamental study on criticality evaluation of damaged core under disruptive accidents of LMFBRs

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In order to evaluate the feasibility of accident management (AM) against postulated core disruptive accidents of liquid-metal fast breeder reactors (LMFBRs) with mix oxide fuel, criticality evaluations were performed with several selected states of the disruptive core using a fast reactor core analysis code ARCADIAN-FBR[1]. In the Protected Loss of Heat Sink (PLOHS) accidents, the melted core debris may penetrate the core vessel and shift underneath in about an hour from the initiation of the accidents, after being kept within the core for 30-60 minutes. At this stage, the piled core debris may reach a recritical condition according to its amount and shape. Three states of core debris were considered in the analysis: While 1) being kept within the core after a core-slumping, 2) forming a layer on the basement of the guard vessel, and 3) being piled on the concrete slab under the reactor pit. As for the last state, three models were considered as the shape of the debris: a cone, a hemisphere, and a sphere. The repose angle of the cone model was also varied as a parameter. In the first two models, the debris is posed upon a concrete slab which may function as a neutron reflector. In contrast, the debris was assumed as penetrating through the slab in the sphere model.

In the calculations, the conditions of a typical prototype reactor was assumed: The volume fraction of the debris was assumed as 27.4% of fuel, 50% of sodium, and 22.6% of stainless steel, which is that of the core in initial state and does not always correspond to realistic situation. The temperature of the debris was set to 3000K. The volatile FP elements were assumed to escape from the core. The amount of debris, i.e., the melted area fraction against the whole core was varied as a parameter from 1/6 to 1. All the critical calculations were done using a continuous energy Monte Carlo code MVP with JENDL-3.3 nuclear data library. The number of neutron histories was set to a million for each case. Regarding the volume fraction of melted area against the whole core as a parameter, 'criticality map' was drawn for each case. Consequently, the criticality was found to be very sensitive to the shape of the debris especially in the third state. Among the three shapes, the sphere revealed the maximum multiplication factor when the volume of the debris is identical. It was observed that the surface area shows good linearity with the multiplication factor regardless of the shapes (Fig. 1).

Through the present study, useful data which are necessary for criticality evaluation of the core debris under the disruptive accidents were obtained. A further investigation will follow in order to improve the modeling of the melted debris. Estimation of uncertainty due to potential sources such as temporal deformation of the debris, and uncertainty in nuclear data is included in the issues.

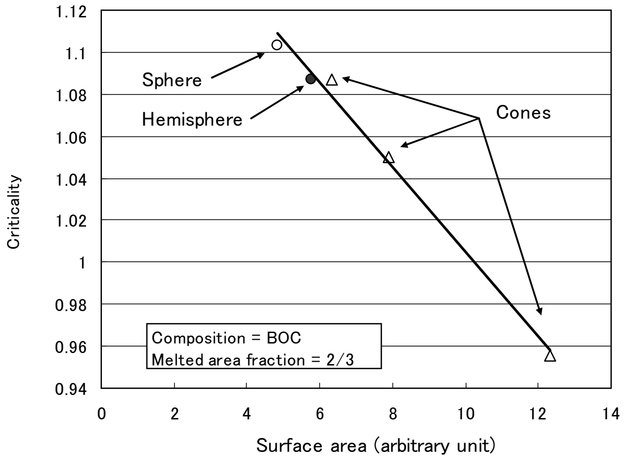

FIG. 1. Linearity of criticality as a function of surface area of debris in different shapes with an identical volume

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Reliability analysis for a cooling system of a typical FBR plant's Ex-vessel fuel storage tank (EVST) using the PSA method

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The Ex-vessel fuel storage tank (EVST) is designed to store new and spent core elements in sodium and has its cooling system. The EVST cooling system consists of three independent loops which respectively have a main cooling system, a sodium purifying system, an electromagnetic pump, an air cooler, etc.. A part of piping of the main cooling system is placed in the fuel storage tank and transfers the decay heat of the core elements to the cooling system. The heat is released to the atmosphere through the air cooler.

The cooling capacity is designed to remove the maximum decay heat of 660kW by the two loops in constant operation and to maintain a sodium temperature in the tank below 300C.

JNES has analyzed the reliability of the EVST cooling system using the probability safety analysis (PSA) methodology as a condition of two loops operation by electromagnetic pumps required. Here, the loss of the decay heat removal event is defined the situation that the sodium temperature in the tank reaches 750C, applying the limiting condition for the reactor vessel, and this initiating event can be assumed to occur in the cases of the failure of the cooling system or the loss of the power supply. This paper summarized the results of the analyzed dominant scenario of the EVST cooling system failure and the risk importance of the components by using the PSA method.

According to the results, the time lapsed for the sodium in the EVST to rise up to 750C was 80 hr with one loop cooling operation and 40 hr without the circulation cooling, and in consideration of this allowance times, the fault trees and the event trees were constructed for the component failures and recovery actions, then it was found that the component failures which immediate recovery is not expected, such as a sodium leakage from piping system, an electromagnetic pump, an air cooler, were dominant for the loss of the decay heat removal event.

These analyses could showed that the predictive information of a sodium leakage from the components should be previously grasped through the system maintenance and the oversight activities, from the viewpoint to prevent the loss of the decay heat removal event for the EVST cooling system.

Safety System Designs and Characteristics of the 4S

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The 4S (Super-Safe, Small and Simple) is a small-sized, sodium-cooled fast reactor with a reflector-controlled core. The 4S design includes features such as " No on-site refueling", "Passive safety" and " Low maintenance requirement". In this paper the safety design and safety features are described.

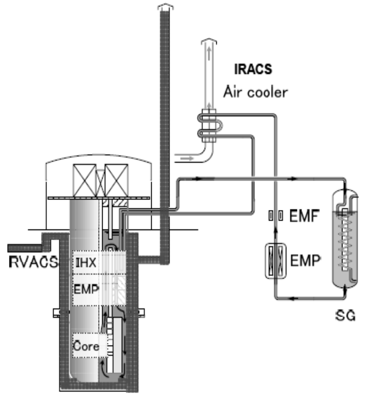
The 4S has two redundant and diverse shutdown systems.One is an annular reflector and the other is a central shutdown rod. The Residual Heat Removal System (RHRS) consists of IRACS (Intermediate Reactor Auxiliary Cooling System) and RVACS (Reactor Vessel Auxiliary Cooling System). The IRACS removes decay heat by using an air-cooler in the intermediate heat transport system, and the RVACS also removes decay heat using natural circulation of air around the reactor guard vessel. The IRACS and the RVACS are diverse systems and either can remove 100% of the core decay heat. The reactor protection system parameters monitored include neutron flux, liquid sodium level in the reactor vessel, primary outlet temperature of the intermediate heat exchanger, and the supply voltage/current of the primary EM pumps. When one or more abnormal signals are detected by the sensors for the above parameters, the scram systems are initiated. The containment system consists of a guard vessel and a top dome that surround the reactor vessel, shielding plug, and above plug equipments.

Measures to exclude the previously identified accident initiators have been applied to the 4S design. The reflector drive mechanism used has mechanical stops to restrict the potential reactivity insertion during startup . Also, the reflector drive mechanism to compensate burnup reactivity swing during steady state moves at very low speed for the same reason. The electromagnetic pump has a flow coast down power supply system to mitigate transients when the main power supply is lost. The double wall steam generator includes leak detection systems for both inner and outer tube failures to prevent a sodium-water reaction.

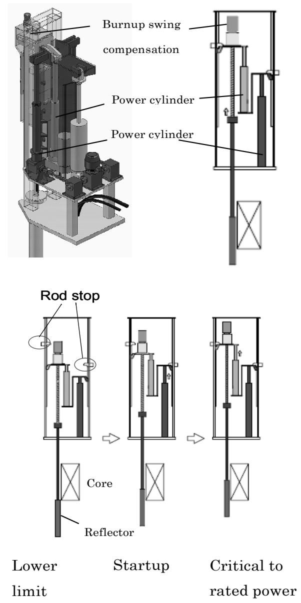
The 4S safety analysis has been carried out in accordance with the basic philosophy and contents of the Standard Review Plan (NUREG-0800), modified as necessary to allow application to a liquid metal reactor. For selecting the accident events, the failure modes for all equipment and systems were identified and categorized by performing a FMEA (Failure Modes and Effects Analysis). The 4S AOOs and DBAs were selected based on the frequency and severity of the resulting events.

In this report, 'Loss of Offsite Power' (LOSP) will be described as the representative AOO, and 'Failure of a Cavity Can' (FCC), which is a 4S-specific reactivity insertion accident, will be described as the representative DBA.

This report will describes the plant characteristics during the transients. As for the short term, the change of power and flow ratio from steady state induce a first temperature rise for the fuel and cladding. Consequently after the first small temperature peak, no longer magnitude of temperature peak will occure over the long term operation, because the decay heat is removed by natural circulation using IRACS and RVACS. These evaluations demonstrates that the safety acceptance criteria are satisfied, and the safety of the 4S is confirmed.



Schematic diagram of the residual heat removal system



Reflector drive mechanism

Evaluation of risk reduction measures on the UTOP event of the 4S

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The 4S (Super-Safe, Small and Simple) is a small-sized, liquid sodium-cooled reactor with a reflector-controlled core. The 4S has inherent safety features of the core and passive safety systems. In this paper, those risk reduction measures are evaluated based on the probabilistic risk assessment and the statistical analysis for unprotected transient of overpower (UTOP) event.

UTOP event is an anticipated transient without scram at which the core power increases by external reactivity insertion. The 4S design enables passive shut down against those event owing to the inherent safety feature such as negative reactivity feedback and the risk reduction measures such as the reflector rod stoppers, the reflector drive mechanisms at very low speed, etc.

After the metric and the criteria for an occurrence of the core damage are identified for UTOP event, the evaluation was done following the two methodologies as follows. (1) The initiators of UTOP event are investigated, and the amount of the reactivity insertion as well as the rate is selected as a factor which dominates the accident consequences. The occurrence frequencies of each initiator are plotted on the matrix of the reactivity insertion amount and the insertion rate, and a map is produced (FIG.1). Then, UTOP event are analyzed setting the reactivity insertion amount and the insertion rate as parameters. According to the results of the analysis the degree of influences on the metric is plotted on the other matrix of the reactivity insertion amount and the insertion rate, and another map is produced (FIG.1). The risk reduction measures are evaluated by those two maps mentioned above on the basis of the probabilistic risk assessment. (2) Important phenomena affecting the metric as well as the parameters associated to those phenomena are selected by some in-house experts using phenomena identification and ranking table (PIRT). Sensitivity analyses are performed and the parameters which have relatively large sensitivity is identified. Statistical analysis is performed setting those identified parameters as variables, and finally the achievability of the inherent safety features of the core and the passive safety systems are evaluated.

The effectiveness of the risk reduction measures for UTOP is evaluated by both the methodology of the probabilistic risk assessment and the statistical analysis. The risk reduction measures enable to reduce the core damage frequency to the negligibly small level.

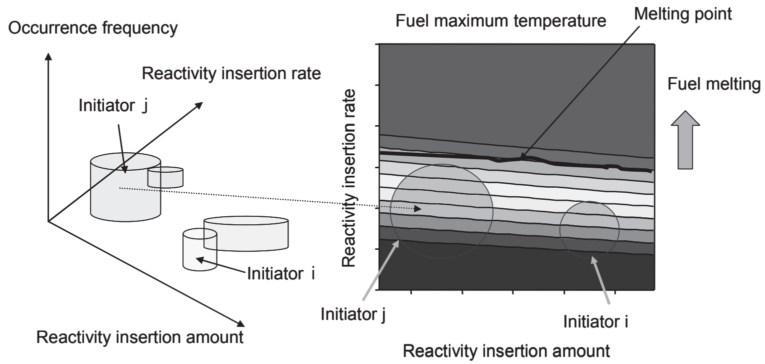

FIG.1 Conceptual maps of occurrence frequencies and the degree of influence on the metric (fuel maximum temperature)

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Elimination of severe recriticality events in the Core Disruptive Accidentof JSFR aiming at In-Vessel Retention of the core materials

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It is intended in the JSFR design to secure In-Vessel Retention of the core materials in the Core Disruptive Accident, although this accident category with extremely low probability is regarded as beyond the design-basis events. For this purpose, it is necessary to assure integrity of the primary coolant boundary against possible mechanical loading due to neutronic power burst on one hand, and against thermal loading of the core materials on the other hand.

Potential of the neutronic power burst becomes eminent in the early part of the accident. The ULOF accident is regarded as a representative of the Core Disruptive Accident, in which power-to-flow mismatch leads to short-term sodium voiding within the core thereby introducing positive reactivity insertion. This void development phase is called Initiating Phase. In the JSFR design, the maximum void reactivity is limited to about 6$ so that void reactivity insertion can be limited. Some other design parameters are also selected so that severe power burst can be prevented. With this design and application of the SAS4A code, which has been validated with the CABRI and TREAT experimental data, Initiating Phase of JSFR is judged free from energetics.

Another mechanism which could lead to a large reactivity insertion is free motion of the molten-core materials after development of a core-wide molten pool in the so-called Transition Phase. In the JSFR design, in order to prevent a large molten-pool formation, each fuel sub-assembly is equipped with an inner duct as illustrated in Fig.1, which enables early discharge of the molten core materials. Effectiveness of this design concept to prevent an extensive reactivity insertion can be confirmed if one can demonstrate that molten fuel discharges through the duct before failure of the wrapper tube in each sub-assembly. Therefore, JAEA performed the EAGLE-1 experimental program[1] under a collaboration with National Nuclear Center of Republic of Kazakhstan. With this experimental program employing a downward discharge geometry, early fuel discharge was observed confirming that the downward-discharge concept met the requirement at least. A SIMMER-III code evaluation based on the knowledge obtained from the EAGLE-1 program showed that the upward-discharge concept also fulfilled the requirement. Based on this information, present JSFR design with the upward discharge concept is understood to be effective and appropriateness of this understanding is to be confirmed with additional experiments in the on-going EAGLE-2 program.

In addition to the early fuel discharge, following two factors will enlarge the subcriticality in the reactor accident scenario. Firstly, axial blanket pellets will mix with the fissile fuel. Secondly, the backup control rods of JSFR, which are equipped with SASS(Self Actuated Shutdown System) [2], will encounter exremely high coolant temperatures at which the magnet can no more hold the rods bringing a large negative reactivity insertion.

Fuel remaining in the core at this stage is entirely solid. However, some part of the remaining core materials could not be cooled within the core region and this part will relocate downward with melting. In the JSFR design, it is intended to cool the relocating core materials with a multi-layer in-vessel debris tray placed at the bottom of the reactor vessel. It has been confirmed through an evaluation assuming a typical relocation scenario of the core materials that decay heat can be sucessfully transferred to the heat sink with the natural circulation. It is planned to clarify the core-material relocation behavior with future experiments and computer model simulation so that the integrity of the boundary against the thermal loading can be demonstrated.

Therefore, with the JSFR design and newly obtained experimental data, a perspective has been obtained to eliminate severe recriticality events in the Core Disruptive Accident. Now, it is focusing on the long-term core-material relocation phase to address the thermal loading toward the final target of In-Vessel Renention.

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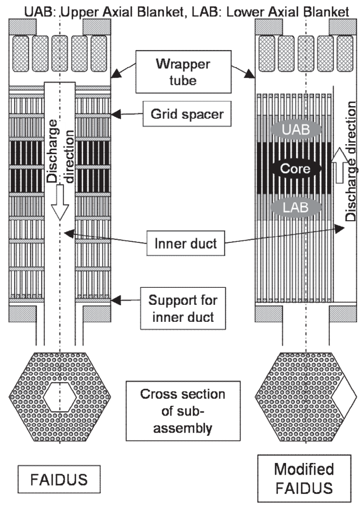

Fig.1 JSFR fuel subassembly design options with the discharge duct

Level-2 PSA for the Prototype Fast Breeder Reactor MONJU Applied to the Accident Management Review

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1. INTRODUCTION

An accident management guideline (AMG) of the prototype fast breeder reactor MONJU has been presented to Nuclear and Industry Safety Agency (NISA) of METI by Japan Atomic Energy Agency (JAEA) with an evaluation result of an effectiveness of the AMG by employing Level-1 and Level-2 PSAs. Japan Nuclear Energy Safety Organization (JNES) evaluated the three events – PLOHS, LORL and ATWS events – and scrutinized the results of the Level-2 PSA carried out by JAEA from the view point of an accident management (AM) review. Regarding ATWS events, we have carried out a qualitative evaluation of the results of JAEA's evaluation and carried out a quantitative evaluation of the containment failure frequency (CFF) in relation to Protected-Loss-of-Heat-Sink(PLOHS) and Loss-of -Reactor-Level (LORL) events.

2. ANALYSIS METHOD

Evaluation of the containment failure probability CFF has been conducted based on the results of the Level-1 PSA by employing the code system developed by JNES. We conducted a close examination of the procedure that JAEA followed to evaluate CFFs in PLOHS and LORL events. It was confirmed that JAEA's Level-2 PSA quantified the phenomenal event trees was expanded in the three processes – the plant response process, the core damage process and the containment vessel response process – based on various analytical and experimental evidence and otherwise followed much the same basic evaluation procedures employed by JNES.

As for PLOHS and LORL, quantitative evaluation of CFF was conducted according to the following procedures: Development of an event flow diagram, Development of a phenomenal event tree, Quantification of the phenomenal event tree, Evaluation of containment failure frequencies, and Evaluation of the effectiveness of the AM measures.

In the evaluation of the PLOHS and LORL events, the following analytical codes were used ; Plant dynamic characteristic analytical code (NALAP-II), Nuclear characteristics analytical system (ARCADIAN-FBR/MVP), Nuclear dynamics analysis code (APK), Structure analytical codes (ABAQUS, FINAS, etc.), Containment response and behavior analytical code (AZORES), and In-sodium radiation source behavior analytical code (ACTOR).

3. FEATURES OF THE PLOHS SCERARIO

Based on the analysis method above, the phenomenal event trees of the PLOHS and LORL were quantified, and the characteristics and incidence probabilities of the following main scenarios were evaluated.

i. The evaluation of the plant response process shows that the incidence probability of the containment vessel by-pass event (CVBP) is 0.15/PLOHS.

ii. At the end of the core damage process, coolant and fuel debris will fall into the reactor chamber to cause sodium-concrete reactions. Debris-concrete reactions will generate hydrogen. If a mixture of generated hydrogen and sodium vapor is released onto the containment vessel floor, sodium will spontaneously catch fire to initiate inducted diffusion combustion of hydrogen but

nonetheless the flame propagation rate is so small that the containment vessel will not rupture. Based on sodium-concrete reaction tests conducted by the former PNC and the HEDL testing and study concerning hydrogen/sodium jet ignition conditions, the incidence probability of the diffusion combustion was evaluated using a phenomenological relationship diagram (PRD) and the evaluation finds that most of the events will end up in induced diffusion combustion (0.97/combussion event). On the other hand, if hydrogen ignition is retarded, containment vessel failure may occur due to explosive burning or detonation of hydrogen accumulated in the containment vessel. This evaluation applied the results of the AZORES code analysis to the PRD and rated the overpressure failure of the containment vessel at 0.007/event.

iii. In light of the above observations, the probability of the loss of containment vessel capabilities before the preparation of the AM measures was rated at 0.19/PLOHS. Judging from the results of a Level-1 PSA (the PLOHS incidence frequency: 4.0E-8/reactor year), the total CFF of PLOHS was rated at 7.5E-9/reactor year.

iv. In the PLOHS event, the major cntributors to the CFF are CVBP (79%) and a containment vessel isolation failure (21%). Both events account for almost all of the PLOHS/CFF, whereas a hydrogen burning-caused overpressure failure of the containment vessel accounts for no more than 0.1%.

4. CHRACTERISTICS OF THE LEVEL-2 PSA

As a result, the total CFF before the preparation of the AM measures was rated at 9.2E-9/reactor year (CDF at 2.7E-7/reactor year), so the author has confirmed that these numerical values are well below the power reactor performance goal indicator values (CDF: 10-4/year or so; CFF: 10-5/year or so) even before the preparation of the AM measures (See Fig.1). In addition, CFF after the preparation of the AM measures was rated at 8.3E-9/reactor year (CDF at 1.6E-7/reactor year), so the author has confirmed quantitatively that the Phases I and II AM measures proposed by JAEA would help reduce CFFs.

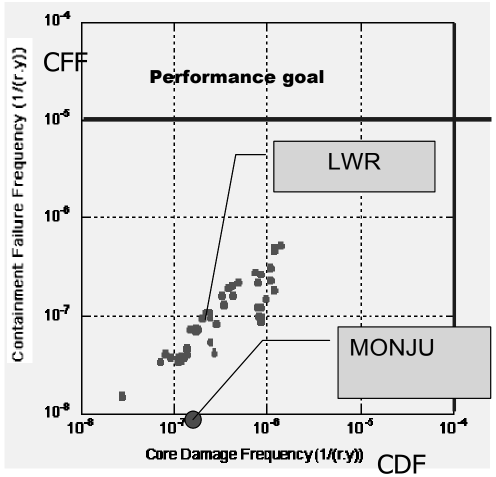

Fig. 1 Characteristic of CFF and CDF for MONJU

Validation of Two-Phase Flow Model in the RELAP5/3D Code for Steam-Generator Blow Down Analysis Using a Test Data of MONJU

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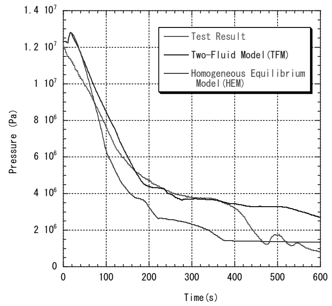
JNES (Japan Nuclear Energy Safety Organization) is supporting the NISA (Nuclear and Industry Safety Agency)'s regulation activity to restart a proto-type FBR (sodium cooled fast breeder reactor) plant, MONJU, first time after the sodium leak accident. The activities contain reconfirming the structural integrity of SG (steam-generator) to a high temperature condition formed in the heat transfer tube bundle by the sodium-water reaction in the case of heat-transfer tube failure accident. To prevent damage of sorounding heat transfer tubes from the ductility and/or creep due to the high temperature, fast leak detection and prompt water/steam blow down from SG is neccesary. The blow down decreases the high pressure in the heat-transfer tube and cools the tube-walls by the fast flow rate.

This paper focuses on the validation of the two-phase flow model for SG blow down analysis using RELAP5/3D[1], which is a one-dimensional thermal-hydraulic analysis code. RELAP5/3D can be applied to simulate the situation of SG where water/steam two-phase flows in a tube-bundle and the bundle is surrounded by sodium flow. The user chooses a two-phase flow model among prepared models that include the non-homogeneous equilibrium two-fluid model (TFM) and the homogeneous equilibrium model (HEM). We tried to use both models in analyzing the 40% power-load turbine trip test of MONJU[2].

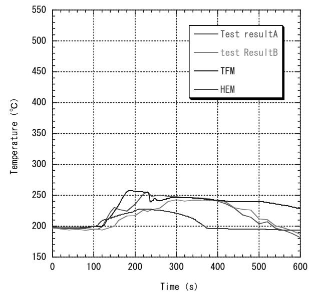
Firstly, TFM was adopted. In the analysis the water/steam flow pass was modeled by six water supply chambers connecting a multiple heat transfer tubes expressed by one channel, respectively. The analysis was performed until 600 seconds after the turbine trip and the results are depicted in Fig.1a (the outlet steam pressure of EV (evaporator)) and in Fig.1b (the temperature of the water supply chamber)[3][4][5]. It was found that TFM predicted well the physical behaviors. The results showed that the void of steam generated by the depressurization boiling moves upward in the down-comer tubes accompanied the enthalpy tranfer to the supply water resulting in the temperature increase in the water supply chambers. The pressure change of a "shoulder" like shape between about 300s to 400s drawn in Fig.1a is explained by the mass balance between the steam mass generated in the down-comer tubes and the steam mass blown from the steam/water system. Thus, the analysis made choosing TFM to the flow behvior in the heat transfer tubes of SG and using RELAP5/3D well explained what ocurred in the actual FBR system.

Secondly, HEM was applied to the same turbine trip test usig RELAP5/3D[6]. The configuration of the system was simulated by seven channels for the tube bundles and one channel for the water supply chambers. As seen in Fig.1, HEM did not show a good agreement with the test data and the calculated pressure is lower than that of TFM. The water supply chamber temperature by the HEM analysis is also lower than that of TFM analysis due to the decreased saturation temperature corresponding to the lower pressure (Fig.1b).

From the discussing above, it has been concluded that the RELAP5/3D code is applicable more to the analysis of SG blow down test of MONJU by adopting TFM for the two-phase flow in the steam/water system.



a. EV outlet steam pressure



b Temperature of water supply chamber

Fig.1 Comparison of analysis results with MONJU test data to the turbine trip test at a 40% power-load operation

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A study on the sensitivity analysis for the safety feature of KALIMER600 with MARS-LMR

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The Korea Atomic Energy Research Institute (KAERI) has been developing fast reactor design technologies since 1997 under a National Nuclear R&D Program. The goals of the project are to develop fast reactor design technologies for enhanced safety, an efficient utilization of uranium resources and a reduction of a high level waste volume. In accord with that consideration, the conceptual design of KALIMER600 is developed.

The safety of KALIMER600 is mainly dominated by the inherent reactivity feedback mechanisms with the metallic fuel and a uniquely designed passive decay heat removal system. The most important safety systems of the KALIMER600 is the PDRC loops composed of decay heat exchanger (DHX), air heat exchanger (AHX), and conneting pipings. When a transient accompanies a pump trip and occurs the increase of hot pool level and overflow of coolant into the shell side of DHX where the heat removal rate rapidly increases with beginning of the overflow, the heat removal through the PDRC loops is going to be balanced with the core heat generation rate to maintain the reactor condition within the safety limit.

In this study, the sensitivity analysis for the safety feature of KALIMER600 has been performed for a transient overpower (TOP) and a loss flow (LOF) events which are two most important DBEs in KALIMER600 using MARS-LMR code. MARS code which has been developed by coupling the RELAP and COBRA-TF in Korea Atomic Energy Research Institute (KAERI) has been improved to use the code in the area of safety analysis of liquid metal reactor. (1) Sodium property table including dynamic properties, such as, conductivity and viscosity, is generated to fit for the MARS code. (2) The heat transfer correlations for the liquid metal are implemented in the code. (3) The models describing the flow resistance by wire-wrap spacer in the core of LMR are applied.[1] MARS-LMR code shows a good agreement with the experimental data conducted in the EBR-II plant and the appropriateness of the models related to liquid metal reactor.

For the sensitivity analysis, some design variables are applied to be conservative. The effect of uncertainties is evaluated on the control rod worth, Doppler reactivity coefficient, and the sodium void worth. Sensitivity studies have also been performed to find the most conservative condition of initial operating variables for the reactor power, primary flow rate and reactor outlet temperature. [2]

The results of sensitivity analysis provide the comparison of the fuel, clad, and the coolant temperatures with the safety limit temperatures. Therefore safety margins can be calculated related to the fuel damage or the threat to its structural integrity during the transients for the considered DBEs.

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Safety Analysis of P-DEMO, a Pool-type Lead Bismuth Fast Reactor

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By utilizing important safety features of PEACER (Proliferation-resistant, Environmental-friendly, Accident-tolerant, Continual, and Economical Reactor), a pool-type naturally-circulated experimental transmutation fast reactor, P-DEMO reactor, is being developed to demonstrate the transmutation capability for pyrochemically recycled trans-uranic (TRU) elements and long-living fission products. The thermal power of P-DEMO is 100 MW and its initial core is loaded with U-Zr metallic fuel rods in HT-9 cladding. The DEMO is cooled during normal operation by the natural circulation of lead-bismuth eutectic (LBE) coolant which is made possible by employing large square open channels and adequate elevation difference between the core and steam generators.

In order to comply with safety requirements in P-DEMO, key criteria at accident condition have to be considered. When any accidents are occurring in un-protected condition, fuel temperature does not exceed the melting temperature, cooling ability has to maintain and there is no outflow of the radiation referring to USA 10CFR100 value.

Unprotected Loss Of Heat Sink (ULOHS) and Unprotected Transient OverPower (UTOP) were simulated using by MARS-LBE 3.11 code, which is Multi-dimensional Analysis Reactor System (MARS) code is developed by coupling with the RELAP-5 code and the COBRA-TF code. All accident analyses were conducted with conservative assuming as no emergency shutdown. Final purpose of this analysis is that core melting incident will occur.

Remakable things of these accidents is that Reactor Vessel Air Cooling System (RVACS) was simulated in system code. Especially, In ULOHS, it shoud be operated and have a function of heat removal in order to satisfy of heat removal in reactor for long term situation.

Figure 1 is nodalization schmatics of MARS-LBE modeling in P-DEMO reactor.

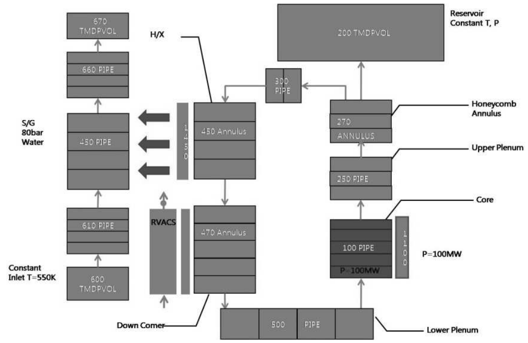

Figure 1. Nodalization Schematic of P-DEMO

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Inherent and Passive Safety Characteristics of Modular Fast Reactor, SVBR-100 with Lead-Bismuth Coolant

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The safety level of currently operating nuclear power plants (NPP) assures realization of safety requirements for social acceptance. For the 20 years from the moment of happening the Chernobyl accident no significant accidents have ocurred.

Along with this, multiple increase of the reactor park expected in the current century will require to rise the level of safety in order to keep the value of the risk of happening the severe accident at an acceptable level.

Improvement of safety of the NPPs with traditional type reactors is only possible due to increase of the number of safety systems and protection barriers that will cause deterioration of their technical and economical characteristics.

To avoid this, we should begin to implement in the nuclear power the innovative type reactors, which possess high-developed inherent self-protection and passive safety properties allowing to eliminate the conflict between safety and economic requirements.

Namely, SVBR-100 that is a small power fast neutron modular reactor (~ 100 MWe) with heavy liquid metal coolant – lead-bismuth eutectic alloy – belongs to that type of reactors.

Natural properties of lead-bismuth coolant (LBC): high boiling point (~ 1670 ºC), chemical inertness while contacting with water and air, and physical features of fast reactor make it possible to construct the reactor installation (RI), in which the principle of inherent self-protection against the certain severest accidents have been realized to the maximal possible extent i.e. the causes of their occurrence have been eliminated.

In particular:

- the high boiling point (~ 1670 ºC) eliminates the necessity to maintain high pressure in the primary circuit. Moreover, the reliability of heat transfer from the core and safety is heightened due to lack of a heat removal crisis phenomenon;

- LBC reacts very slightly with water and air. Progress of accidental processes caused by tightness failure in the primary circuit and inter-circuit leaks in the steam-generator (SG) occurs without release of hydrogen and any exothermic reactions. Therefore, the likelihood of chemical explosions and fires due to internal reasons is virtually eliminated.

The inherent safety is mostly determined by potential energy stored in coolant. The values of this energy are as follows: ~ 20 GJ/m^3, ~ 10 GJ/m^3, ~ 1 GJ/m^3 for water, sodium and lead-bismuth coolants correspondingly.

Thus, the potential energy margin of the LBC cooled RIs is minimal.

At this, the safety systems do not include the elements, in which actuation can be blocked in an event of their failure or impacts of human factors:

- removal of heat decay when there is no heat removal via the SG is provided passively by LBC natural circulation in the primary circuit. This is realized by transferring heat via the reactor monoblock vessel to water in the tank of the passive heat removal system, in which the monoblock is installed, and further due to water boiling, with steam removal to the atmosphere. (This represents a grace period of about four days long without exceeding the allowed temperature limits);

- in an event of rupture of several tubes or termination of operation of the gas system's condenser, localization of the SG leak is provided also passively at increasing steam pressure in the gas system over 1 MPa due to break-up of the preserve membrane and discharge of the steam into the bubbling device;

- the rods of the additional emergency protection system, which have been mounted in the "dry" channels, operate passively due to gravity while increasing the LBC temperature over a dangerous value owing to existing fusible locks made of the alloy with a corresponding melting temperature.

The results of computation analysis of such postulated events as insertion of positive reactivity of 0,25 \$, blacking out the NPP during two days, and 50 % plugging of the coolant pass-through section at the core inlet have revealed that maximal temperature of fuel elements claddings that determines their operability does not exceed the allowed limiting value of 650 °C. Radioactivity release via vent stack in the case of the full depressurization of gas system is not exceed of the permissible level.

Radiation shielding and radiation safety in the pool-type reactor SVBR-100

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The key feature of heavy lead-bismuth eutectic as the reactor coolant is it chemical inertness to water. It allows using two-circuit heat removal layout without intermediate circuit as well as integrated into one vessel primary circuit equipment design. Such equipment arrangement in combination with the inherent safety features can lead to improved nuclear power plant economics.

Nevertheless, intention to develop compact reactor facilities with integrated arrangement of the core, steam-generator and primary circuit pumps is related to solution of some radiation safety questions. First of all to these questions should be associated:

- access to equipment, placed on the reactor head and in the reactor room;
- radiation shielding of reactor vessel;
- radiation shielding of steam-generator and minimization of secondary circuit coolant activation;
- tritium transfer etc.

3-D computation of neutron- and gamma-fields in SVBR-100 reactor facilities and nearest arrangement were performed.

Boron carbide blocks with lead-bismuth coolant and steel are the main materials of invessel SVBR-100 radiation shielding in radial direction, vibropacked boron carbide with steel plates are used in safety plug under the core in axial direction.

Full neutron flux in vertical cross-section through the steam-generator of SVBR-100 reactor facilities is presented on the figure below. Significant heterogeneity of invessel materials composition leads to complicated distribution of neutron- and gamma-fields. Some efforts should be made to optimize operational reactor and nuclear power plant radiation safety features, including review of maintenance and repair procedures as well as decommissioning concept. It should be mentioned, that neutron spectrum in different elements of invessel area differs significantly. That fact significantly influences on steam radioactivity, which is formed by two factors:

- impurities activation;
- short time activity of N^{16}.

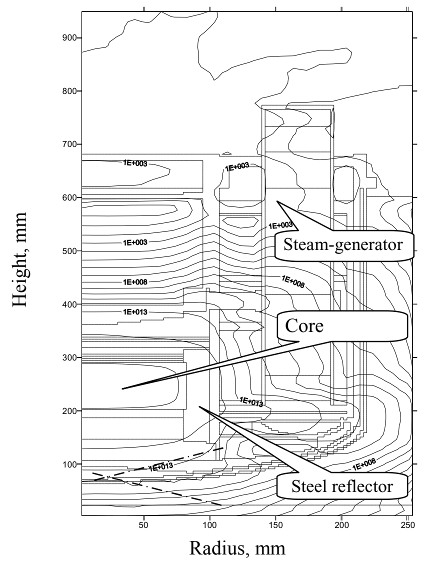

Figure *Full neutron flux in vertical cross-section through the steam-generator of SVBR-100 reactor facilities*

The main criteria selected for definition of radiation shielding are presented in the paper. The main results of radiation fields on the reactor head, steam radioactivity and reactor vessel activation computation are presented. Possibility to achieve established radiation criteria in compact design is demonstrated.

POSTERS OF SESSION 4:

Fast reactor structural materials: achievements and new challenges

Core Materials Development and Testing for the Advanced Fuel Cycle Initiative

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The Advanced Fuel Cycle Initiative is investigating methods of burning minor actinides in a transmutation fuel. To achieve this goal, the fast reactor core materials (cladding and duct) must be able to withstand very high doses (>200 dpa design goal) while in contact with the coolant and the fuel. Thus, these materials must withstand radiation effects that promote low temperature embrittlement, high temperature helium embrittlement, swelling, accelerated creep, corrosion with the coolant, and chemical interaction with the fuel (FCCI). Research is underway that includes determining radiation effects in ferritic/martensitic steels at doses up to 200 dpa, testing and development of liners and coatings to prevent/reduce FCCI, and developing advanced alloys with improved irradiation resistance.

To develop and qualify materials to a total fluence greater than 200 dpa requires development of advanced alloys and long term irradiations in fast reactors to test these alloys. Important data required on these advanced alloys include but are not limited to tensile, fracture toughness and creep data after irradiation to doses greater than 200 dpa at irradiation temperatures of 350-600°C. Test specimens of ferritic/martensitc alloys (T91/HT-9) previously irradiated in the FFTF reactor up tp 210dpa at a temperature range of 350-700°C are avialable for mechanical testing and structural analysis in the near future. This includes analysis of a duct made of HT-9 after irradiation to a total dose of 155 dpa at temperatures from 410 to 470°C with lower dose material covering irradiation temperatures from 370 to 510°C. Figure 1 shows a schematic of this duct. Compact tension, charpy and tensile specimens have been EDM'ed from this duct and mechanical testing as well as SANS and Mossbauer spectroscopy are currently being performed.

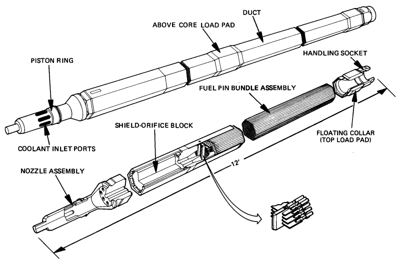

Figure 1 Schematic of ACO-3 duct irradiated in FFTF to a total dose of 155 dpa.

Large scale mechanical testing delivers direct engineering data but these tests are only possible if enough sample material and required hot cell capabilities are available. Small scale materials testing methods in addition to large scale materials testing allow one to gain more insight on the same specimen and probe directly the areas of interest which are not accessible otherwise (small welds, sheared areas, areas with different microstructures, etc.). However, in order to use small scale testing techniques and probe materials changes using these methods, the relationship between data measured at the different scales needs to be investigated. In order to establish a research based relationship between small scale and large scale materials testing several different mechanical testing techniques were conducted on the same specimen irradiated in the STIP rradiation program up to a dose of 13 dpa. Micro hardness testing and micro compression testing on focused ion beam (FIB) manufactured pillars were performed on remaining parts of HT-9 tensile test specimens tested and irradiated at PSI in Switzerland. It is shown that the yield strength measured by tensile testing, micro compression testing and micro hardness testing all show the same trend as shown in figure 2.

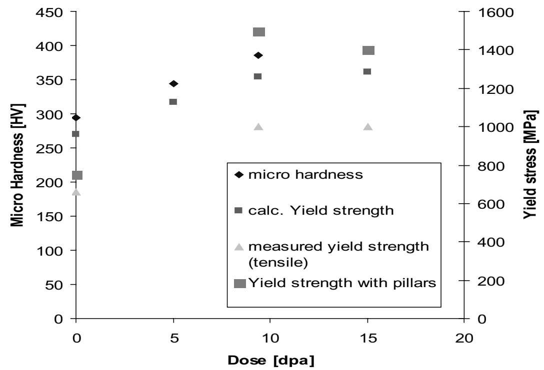

Figure 2: Results of microhardness and micro pillar measurements performed at room temperature and tensile test results measured at irradiation temperature on HT-9 after STIP irradiation.

In addition, the new small scale sample preparation techniques allow one to manufacture local electrode atom probe samples from active materials to investigate eventually local changes in chemical composition on the samples after irradiation.

Approaches to validation of fast reactor lifetime extension

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As compared with the other reactors, the main feature of fast neutron sodium-cooled reactor operation is the effect of increased temperatures (up to 550-600 ^{o}C) and intensive fast neutron flux (up to ~$2{\cdot}10^{21}$ n/cm^2·year of energy E>0.1 MeV) on structural materials. Under these conditions, the basic mechanisms damaging fast reactor component material are creep, fatigue and their interaction as well. Under intensive neutron flux, the austenitic steel used as reactor structural material is embrittled. Except thermo-mechanical load, the additional loading factor for fast reactor components is non-uniform material swelling due to the effect of high-dose irradiation. This results in considerable deformation of reactor components that can violate their normal operation in the ultimate case. It is necessary to note that most of main fast reactor components are difficult of access for non-destructive inspection in order to detect defects. Therefore, in case of reactor service lifetime prolongation, it is necessary to take account of availability of process and operation defects in these components.

In view of above-mentioned, to validate prolongation of BN-600 reactor service life up to 45 years, procedural and material-study activities were performed to develop procedures and methods of strength and lifetime analysis for structural components with defects under the effect of high temperatures and intensive irradiation, as well as to obtain service characteristics of fast reactor structural materials in view of their degradation under the effect of high temperature during more than $2{\cdot}10^5$ hours and intensive neutron irradiation.

Within procedural tasks, the following main procedures and methods have been developed:
1. Definition of design dimensions and shapes of postulated defect.
2. Formulation of constitutive equations of thermal-viscoelastic-plastic deformation of structural material in view of swelling to analyze stress-strain state (SSS) of structural components under thermal-mechanical load in view of neutron irradiation.
3. Analysis of the ultimate state "Crack initiation under cyclic loading due to fatigue mechanism".
4. Analysis of the ultimate state "Crack initiation during long-term static and cyclic loading under creep and fatigue".
5. Analysis of defect growth under cyclic and long-term loading.
6. Definition of critical defect dimensions.
7. Analysis of the ultimate state "Achievement of ultimate deformation of structural component".

For effective application of developed procedures, additional material-studies of basic structural material (steel Cr18-Ni9) were performed. Previously used standard dependencies (with safety factors) for fatigue, creep-rupture strength, etc. were revised and new ones were obtained.

The above-mentioned procedural developments and material studies are the basis of standard guideline document "Procedure for strength analysis of basic components of fast neutron sodium-cooled reactors ", which was licensed by Russian nuclear regulatory bodies and put in force in 2007.

The provisions of this standard guideline document are used when validating operability of "critical" irreplaceable components of BN-600 reactor within 45 years of operation. List of "critical" irreplaceable components of BN-600 reactor that determine its lifetime is made on the basis of the following criteria:

- impossibility to replace and repair;

- inaccessibility for technical condition inspection and check;

- maximum values of main damaging factors, first of all, irradiation and temperature and cyclic loads effect as well;

- influence on safety.

Cyclic and long-term damageability in BN-600 reactor irreplaceable components estimated as per standard fatigue and creep-rupture curves according to SSS analysis results show that crack initiation is possible in some structural components within 45 years of operation.

The subsequent analyses of possible growth of cracks initiated during operation and those developing as a result of process defects in welds show that these cracks do not reach their critical sizes during 45 years of operation of BN-600 reactor.

As per the conditions of neutron irradiation, considerable radiation swelling is predicted for the core restriction shell (fluence up to ~10^{23} n/cm^2 of energy E>0.1 MeV). The results of analysis of core restriction shell deformation due to non-uniform radiation-thermal effect by volume show that it will not result in operability loss of associated equipment (core FAs, loading-unloading elevators and refuelling mechanism) during 45 years of operation.

Thus, according to the results of investigations, the operability of BN-600 reactor irreplaceable components will be ensured during 45 years of Beloyarsk NPP power unit-3 operation.

Now, 45 year service life of BN-800 reactor is under validation. The core restriction shell that is under intensive irradiation, remains in this reactor, however, high-radiation resistant material (steel Cr16-Ni11-Mo3) is used for this shell.

As for BN-1200 reactor design, intensive irradiation of the core restriction shell and its swelling are eliminated. However, this design provides considerable increase of sodium temperature at core inlet (410 ^{o}C instead of 354 ^{o}C in BN-800 and 365 ^{o}C in BN-600) and vessel components respectively. It requires comprehensive validation of operability and more wide application of steel Cr16-Ni11-Mo3. It is anticipated to provide BN-1200 reactor operation during 60 years.

Development of microscale mechanical testing methods for assessing radiation damage in cladding steels

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This paper describes efforts to develop microscale methods for mechanical testing of radiation damaged metals. The objective of this program is to simulate fast neutron damage in cladding steels using high energy, heavy ions. While these ion beams can create large levels of radiation damage, e.g. 100 displacements per atom, they do so over a limited depth from the surface of the sample, on the order of several microns.

Recent work by Hosemann et al. introduced the concept of applying the micropillar compression approach to irradiated metals to assess their mechanical properties. [1] Building on this work, the current paper examines the fundamental issues of accuracy and precision of the micropillar approach by using copper single crystals with several levels of irradiation damage and many pillars per condition. The result is that we can accurately model and assess the changes in the mechanical behavior of the copper as a function of radiation damage.

Copper single crystals of <110> and <111> orientations were irradiated using 20MeV copper ions at nominally room temperature until a damage levels of 50dpa and 100dpa had been achieved. Micropillars of 5μm in diameter and 10μm in height were fabricated from these samples using a 30keV gallium ion beam on a FEI dual-beam focused ion beam (FIB) machine. Compression testing was performed in feedback displacement control using a Hysitron Performech Triboscope nanoindentation machine with a diamond, flat punch of 25μm in diameter.

As shown in Figure 1, there are two distinct bands of stress strain curves; distinguishing the control samples (no irradiation) from the irradiated (100dpa) samples. Even, so there is no distinct difference in initial yield point between the two populations with many, visible displacement jumps discernable for both conditions. There is a marked increase in the work hardening rate for the irradiated material. The level of scatter in the data appears to be greater for the irradiated material.

The SEM micrographs in Figure 2 explain these observations and underscore the need for careful development of the micropillar approach prior to application to complex microstructures. In Figure 2A, slip bands are observed throughout the volume of the micropillar as would be expected. These distinct slip bands mostly likely correspond to the observed jumps in strain (displacement) in Figure 1. In Figure 2B, slip bands are only observed in the lower half of the micropillar. The upper half of the pillar has been hardened by irradiation so that no slip is possible under the current loading conditions. The lower half, however, has several, discrete slip bands as observed in the control pillar.

We are currently performing micropillar compression tests on a series of pillars of different sizes to force the deformation into the irradiated zone nearer to the surface. We will discuss the optimization of this approach and its application to cladding steels.

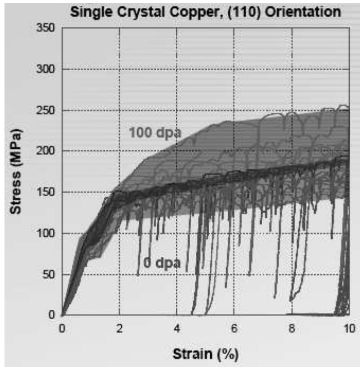

FIG1. Stress-strain curves measured by micropillar compression. The solid lines are from individual curves, while the solid shading shows the extent of the data for a given damage level, (red=100dpa, blue=0dpa).

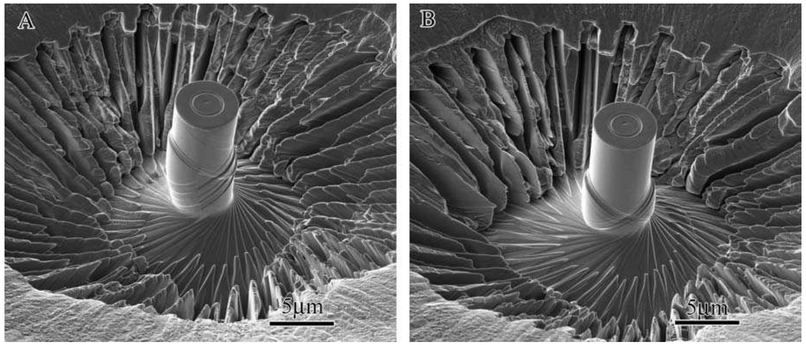

FIG 2. SEM micrographs from two, tested micropillars. A.) from control and B.) from material irradiated to 100dpa.

Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United Stated Department of Energy (DOE) under contract DE-AC0494AL85000.

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Development of Long-lived Control Rods for the Fast Reactor

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Sodium bonded control rods are developed for fast reactor operation. Its irradiation test was done in fast experimental reactor JOYO. This control rod is irradiated around 100×10^{20}cap/cm^3 at B_4C pellet as maximum burn up. For protecting Absorber-Cladding Mechanical Interaction (ACMI), shroud tube is loaded between B_4C pellet and cladding tube in irradiated sodium bonded type control rod[1]. And these gap areas are filled with coolant sodium. After irradiation, there was no remarkable change of cladding tubes. At high burn up area, there are some cracks with shroud tube in cladding tube. The cracks of B_4C pellets in shroud tube are increased as increasing of burn up. From these results, if cracks of shroud tubes are occurred by ACMI which between B_4C pellets and shroud tube, there are almost no effect of diameter increasing of cladding tube. So shroud tube has an relocation inhibitory effect on B_4C pellets for cladding tube safety around 100×10^{20}cap/cm^3 burn up, as shown in figure 1.

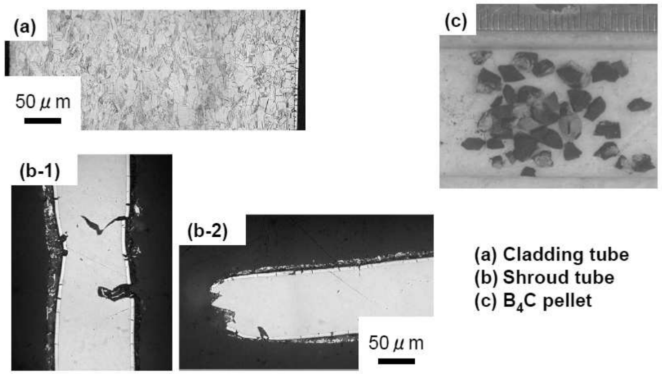

FIG.1. appearance of materials after the irradiation (Burn up; 100×10^{20}cap/cm^3)

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Development of Welding Technique for Double Wall Tube

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A helical-coil-type double wall tube filled with a wire mesh layer, is proposed for a steam generator of the new-type small reactor 4S. The 4S, Super-Safe, Small and Simple is a small-sized sodium cooled fast reactor[1]. This double wall tube consists of inner tube, outer tube and a wire mesh layer between inner and outer tubes shown as Fig.1[2]. Adopting this wire mesh layer, a crack which is occuerd at an inner or an outer tube could not propagate into the other tubes. Moreover, helium gas is filled in the wire mesh layer. If a crack occurred at an outer tube penetrates the outer wall, the gas will leak into the liquid sodium. Detecting the helium in liquid sodium, the penetration of an outer tube can be detected. On the other hand, when a crack occurred at an inner tube penetrates the inner wall, the water/steam leaks into helium.It is also able to detect the leak by monitoring the moisture in helium.

It is necessary to make a tube-to-tube joint of an independent inner-to-inner and outer-to-outer way without bonding inner and outer walls. So we have carried out to develop the welding technique that satisfy this requirement. A keyhole type laser welding technique was selected and a small laser welding head, which can weld from the inside the inner tube, was developed. Using this welding head, welding conditions, such as welding speed, laser power and the shape of the groove were surveyed. As a result of the test, the welding condition, which did not make the inner and outer tubes bonding, was obtained. From the cross-sectional observation, there were no non-adequate phase change. Tensile strength is satisfied the specification for the base metal.

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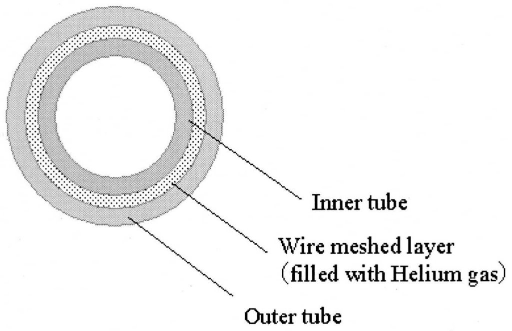

Fig.1 Schematic cross section of double wall tube

Development of Flexible Neutron Shielding Resin as an Additional Shielding Material

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In a present study to improve heatproof of the neutron shielding resin *, the new flexible heat resistant resin has been developed for the future fast reactor.

The concrete is used to shield the neutron around the reactor on a large scale. As for the surrounding of the reactor, the effects of the streaming in the plumbing are involved the radiation shielding design. The shielding methods like the panel and the sheet is devised as one of the more easily technique. The water for the neutron shielding is generally used. However, it is difficult to use the water on the structure around the plumbing and the collimator of the diagnostics for the fast reactor. As one additional shielding material, polyethylene is widely used, but heatproof temperature of general polyethylene is about 120 ºC. The flexible resin with the heat resistance applicable to ~220 ºC has been newly developed as a neutron shielding material with flexibility in shape. The developed resin has been improved the epoxy-based resin (HB) in the previous work [1]. The developed resin will apply to prevent the effects of the neutron streaming and to control the movement of vibrated pipe as the seal material around the plumbing in the near future fast reactor and innovative fission reactor.

The developed resin is made of polymer resin that has improved heatproof and Colemanite. Colemanite is the natural rock that contains boron. The resin is shown in Fig.1. The density of the resin is ~1.4 g/cm^3 and the composition is hydrogen, carbon, nitrogen, calcium, manganese, iron, aluminum, boron, silicon, sodium, chlorine and oxygen.

Neutron attenuation experiments of the resin were carried out using ^{252}Cf neutron sources for 0 ~ 20 cm thickness the resin slab (40cmx40cmx5cm). The neutron attenuation was measured by the neutron rem counter. The neutron penetration characteristics were estimated by 3D Monte Carlo Code (MCNP-5) using the continuous energy cross section data sets based on the JENDL3.3. The 3D calculation results by the based library on the JENDL-3.3 at 20 ºC agree with the experimental results. The neutron attenuation of the experimental results and the analysis results of the shielding materials are shown in Fig.2. The shielding performance of the developed resin is almost same as that of polyethylene.

The excellent flexible heat resistant resin for neutron shielding was demonstrated. This fabrication technology can be applied also to the sheet of gamma rays shielding resin for maintenance of the fast reactor.

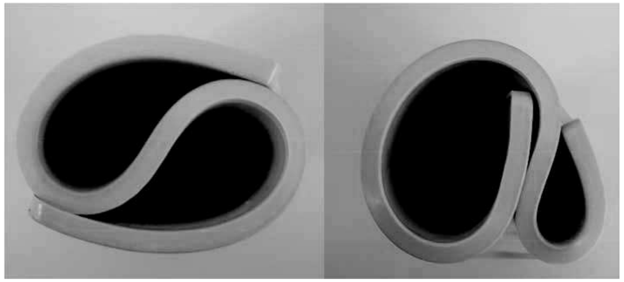

FIG. 1. Developed flexible neutron shielding resin

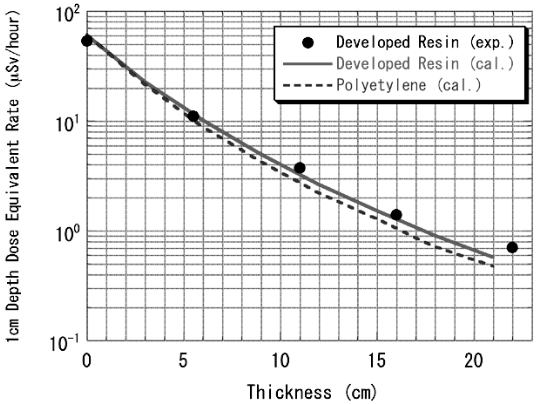

FIG. 2. Neutron dose attenuation of the shielding materials based on neutron attenuation experiments using ^{252}Cf source

REFFERENCE

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* Present study includes the results of "Research and Development for High Heat Resistant of Gel-type Neutron Shielding Resin" entrusted to Japan Atomic Energy Agency by MEXT.

Swelling and irradiation creep of three Russian austenitic steels neutron irradiated in a wide range of doses and temperatures

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Austenitic stainless steels continue to be the main material for manufacturing fuel pin cladding of fast reactors. Alternative steels such as ferritic-martensitic steels or high-nickel alloys, yet have not found wide application.

For calculation of stress-strain state of fuel pins in fast reactor and for determination of their life-time it is necessary to know the initial properties of fuel and structural materials and their change under irradiation. The void swelling and irradiation creep of cladding steels are the most important factors determining the behavior of fuel pins under irradiation. Obtaining data from reactor experiments on swelling and irradiation creep encounters a lot of difficulties. The most widespread and relatively cheap method is the irradiation of gas pressurized tubes. Measurement of diameters of irradiated tubes allows to obtain data on strains caused by both the swelling and irradiation creep.

In BN-350 fast reactor (Kazakhstan) nine experimental sub-assemblies were irradiated with pressurized creep tubes made of austenitic stainless steels EI-847 (16Cr-15Ni-3Mo-Nb), EP-172 (16Cr-15Ni-3Mo-Nb-B) and ChS-68 (16Cr-15Ni-2Mo-2Mn-Ti-V-B) in the solution-treated and cold-worked conditions. Hoop stresses in irradiated samples were determined by initial gas (argon) pressure and by irradiation temperature and varied from 0 to 280 MPa. An unique feature of the BN-350 reactor is the low inlet temperature of sodium coolant that has allowed to obtain data on swelling and irradiation creep at irradiation temperatures ranging from 330°C to 700°C. Irradiation doses were determined by the location of samples in the reactor core and duration of sub-assembly exposure and fall in the range from 20 to 96 dpa. In parallel with the irradiation, the thermal creep of the steels was investigated at approximately same temperatures and exposure times.

The temperature dependences of the volume changes in steels EI-847, EP-172 and ChS-68, obtained by measuring the diameter of creep tubes with zero pressure are complicated demonstrating two or even three maxima at temperatures of 400-410°C, 470-475°C and 625-675°C. At low irradiation temperatures (up to 520°C) the slope of the linear dependence of irradiation creep strain of these steels on hoop stress changes at a hoop stress in the 100-150 MPa range. In high-temperature irradiation conditions the irradiation component of in-reactor creep appears rather significant. The tests conducted at temperatures in the 630-700°C range, have shown, that the creep strains of irradiated samples were noticeably higher than of thermally aged samples.

R&D of ODS steels for fuel pin claddings of fast neutron reactors

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Increasing of fuel burn-up in fuel pins of fast neutron reactors is one of the main problems, which determines technical and economical prospective of fast neutron reactor's progress. At the same time, the working capacity of fuel pins at temperatures ranging from 370 to 720 ºC and achieving of maximum damage dose 150 d.p.a. and more must be provided.

One of the most important characteristics of material for fuel pin claddings (along with resistance to radiation swelling and embrittlement) is resistance to creep at high exploitation temperatures (500-700 ºC). Resistance to creep of high-strength aging steels is determined by stability of hardened by particles steel matrix at increased temperatures and loadings. Usually particles of intermetallic phases or disperse carbides, applying for purposes of strengthening of structural reactor materials, begin to coagulate at temperatures more than 700 ^{0}C. Therefore, at last years more intention is attended to ferritic-martensitic steels, received by method of powder nanotechnology, strengthened by disperse (3-5 nm) oxides of yttrium (ODS steels - oxide dispersion strengthened steels), which, in contrast to carbides, nitrides or intermetallic phases, don't coagulate and don't dissolve in matrix at temperatures up to 1300 ºC. Such steels retain high radiation resistance resided to ferritic-martensitic steels and show excellent mechanical properties at temperatures up to 750 ºC.

Since 2005 in Institute of Inorganic Materials the works over development of technology of receiving the ODS steels have been started. In particular, the wide spectrum of works over R&D of ODS steel on the base of steel EP450 (Fe-13Cr-2Mo-Nb-V-B-0,12C) has been carried out. At present time steel EP450 is a state material for wrapper tubes of reactors BN-600 and BOR-60.

In present work results of investigations of structure and phase composition of steel EP450 ODS on all stages of receiving are shown. Characteristics of long-term strength and thermal creep of plated and tube specimens of steel EP450 ODS in comparison with matrix steel EP450 are presented.

By method of electron microscopy it's detected that structure if steel EP450 ODS in initial state consists of ferritic grains, inside of which the particles of Y-Ti-O with mean size 7 nm are uniformly distributed.

It's shown that creep rate of steel EP450 ODS at temperatures 650 and 700 ºC and loadings 140 and 120 MPa correspondingly is on two orders of magnitude less than creep rate of steel EP450.

Initial comparative results over corrosion resistance in sodium flow and results of imitating corrosion interaction with products of fuel fission for steels EP450 and EP450 ODS are presented.

Technology of joint hermetization of fuel pins with EP450 ODS steel claddings by method of pressurized resistance welding is tested. Structure and properties of welded joints are investigated. As a result of carried out works the principle possibility of receiving the qualitative welded joints is shown.

Further prospects of development of works over ODS steels of different classes and compositions for innovative fast neutron reactors with different coolants are reviewed.

Construction materials for molten salt reactor: design and tests under e-irradiation

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Chemically aggressive high-temperature liquid fuels and/or coolants are considered in various prospective designs of fast and epithermal nuclear systems like Lead-Cooled Fast Reactor (LFR), Molten Salt Reactor (MSR), or Supercritical Water-Cooled Reactor (SCWR). The corrosion resistance of structural materials and, specifically, the long-term combined (n,γ)-irradiation impacts on their composition and mechanical properties at interfaces with corrosion agents becomes a key issue of material selection currently aggravated with the lack of full-scale reactor irradiation tests.

In this paper, we summarize the NSC KIPT developed methodology [1] and present the selected results of simulation experiments [2] on corrosion behaviour of MSR candidate materials, the KIPT designed Hastelloy™ type Ni-based alloys A and B that differ only by the presence (in alloy B) of minor dopants of Nb and Y.

The specimens of alloys A and B embedded into the NaF–ZrF_4 melt (660°C) were irradiated by 10 MeV e-beam at the KIPT located Electron Irradiation Test Facility (EITF). It was designed [2] to provide an entirely controllable irradiation environment — the target temperature, e-beam spectrum and irradiation load of melt and specimens.

The latter issue (prediction of dose and damage rates in EITF and their scalability to the target MSR irradiation environment) has been resolved by means of extensive Monte Carlo (MC) computer simulation of EITF (e,γ)-experiment and MSR (n,γ)-environment using the same CERN Geant4 Toolkit based MC code RaT 3.0 with the G4NDL3.8 neutron data library. Similar to MCNPX, this in-house developed code is capable of consistent MC modelling of electromagnetic and hadronic (*incl*. neutrons and fission fragments) interactions peculiar to EITF and MSR irradiation of fuel and structural materials.

The MC simulation results show that though (e,γ)-beams are not well suitable to simulate bulk effects of reactor neutron damage, they are feasible to activate corrosion at interfaces since it is mainly affected by nuclear heating scaled by the specific deposited energy E_{dep} (or absorbed dose) that speeds up chemical reactions and enhances diffusion. The nuclear heating rates obtained in the EITF (e,γ)-beam experiment are fairly consistent with those expected within various scenarios of construction materials irradiation in MSRs by neutrons, γ-quanta and, optionally, by fission fragments of molten salt based fuel mixes.

Owing to the specific design of the EITF target container assembly (CA), the energies E_{dep} deposited at various interfaces of alloy with melt differed by ≈51 times ranging from 121 up to 6192 eV per atom at the same 700 hrs long e-irradiation. The measurements of e-irradiation

† corresponding author

enhanced corrosion rates of 0.5wt%Nb and 0.05wt%Y doped alloy B (see figure 1) has revealed precisely the same ratio that is the evidence of the proportionality of an irradiation effect to an absorbed dose. No such a simple scaling was observed for the reference (undoped) alloy A that is characterised by substantially lower threshold of e-irradiation impact on corrosion resistance.

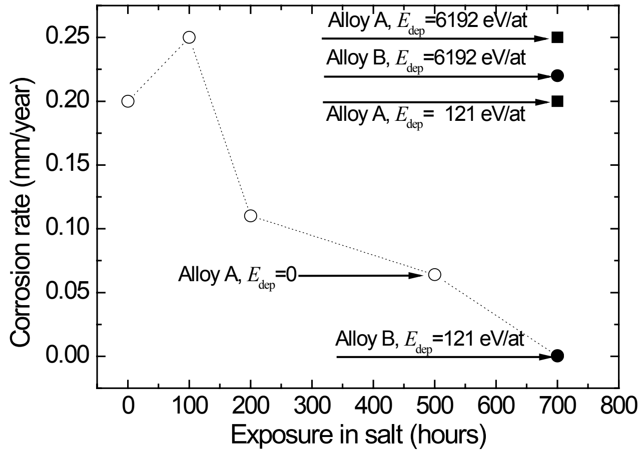

FIG. 1. Corrosion rates of Hastelloy™ types Ni-Mo-Cr alloys: open markers – unirradiated specimens; bold markers – specimens irradiated at EITF to different specific absorbed energies E_{dep}.

Our MC calculations argue that the presence of these minor dopants has no noticeable effect on the EITF e-beam interaction with targets as well as on the nuclear responses of alloys under consideration. Thus we discuss possible explanations of the substantial impact of minor dopants on corrosion behaviour from the point of view of the theoretical consideration of corrosion kinetics in liquid metal (see [1], pp. 114–119) and fluoride melts.

We also announce the planned extensions of the NSC KIPT EITF experimental setup directed toward the simulation experiments using convection loops. They are intended for LFR and SCWR related applications to provide the investigation of irradiation impact on corrosion resistance of topical construction materials.

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Performance of the BN-600 reactor fuel pins with claddings made of austenitic steels EI-847, EP-172 and ChS-68 at high radiation damage levels

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Development of austenitic stainless steels for fuel pin cladding in fast reactors, capable to provide their reliable and economic operation, is one of the most important problems of reactor materials science. Intensive works aimed to increase the fuel burn-up in the BN-600 reactor were conducted for many years under the leadership of VNIINM. These works included the development of new cladding steels. A set of experimental subassemblies with fuel pin cladding fabricated from the new steels has been produced and irradiated in the BN-600 reactor. Characteristics of the subassemblies and main irradiation conditions are shown in the Table. Post-irradiation examinations of a part of fuel pins from these subassemblies (7-9 fuel pins) have been conducted in the IPPE hot laboratory

SA #	Cladding material	Wrapper material	Max. burn-up, % h.a..	Max. linear power, kW/m	Max. dose, dpa
P-34	EI-847, ST	16Cr-11Ni-3Mo, TMT	6.21	53.3	35.4
C-1028	EI-847, CW,	16Cr-11Ni-3Mo, CW	6.9	42.9	55.5
C-1027	EI-847, CW,	16Cr-11Ni-3Mo, CW	8.93	36.1	71.2
EC-1	EP-172, CW,	EP-172, CW.	6.9	43.3	50
M-115	EP-172, CW	EP-450	7.8	38.4	60.9
S-112	EP-172, CW	EP-450	10.5	40.5	86.2
B-163	EP-172, CW	EP-450	11.6	39.8	83.3
C-11	ChS-68, CW	EP-450	7.4	45.1	61.3
C-65	ChS-68, CW	EP-450	9.1	52.0	71.4
C-63	ChS-68, CW	EP-450	11.5	49.0	87.5

As a result of post-irradiation examination of fuel pins which reached the maximum burn-up of 11.6 % h.a., it was established, that the largest degradation of operational properties of fuel pin claddings is observed in the region of the maximum diameter increase and is reveled as a total

embrittlement of the cladding material and the appearance of cracks of a substantial depth on the cladding internal surface.

Processes resulting in the deterioration of fuel-pin cladding properties (embrittlement, formation of microcracks) are directly connected with swelling and/or with the radiation-induced segregation which occursin the swelling temperature range and due to action of forces causing the swelling.

The interrelation of processes of corrosion cracking and swelling of fuel pin cladding is considered. The effect of the stresses arising due to gradient of swelling in the cladding wall appears as most important. The level of stresses is also determined by the temperature dependence of swelling of steels used for fabricating the fuel pin cladding. After high dose irradiation there is a rather high level of residual stresses in fuel pin cladding that leads to failure of fuel pins during manipulations with them in hot cells or during cutting samples from the fuel pin cladding.

Among cold-worked steels investigated, the steel ChS-68 has minimum swelling. The swelling is higher in the steel EP-172 CW and especially in the steel EI-847 CW. The swelling temperature dependence in the steel ChS-68 CW is smoother than in EP-172 CW, that reduces the level of stresses in fuel pin cladding made of the steel ChS-68 CW.

Materials Testing Aspects of Fuel Elements Development for Lead-Bismuth Cooled Fast Reactor SVBR - 100

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Specific operating conditions of SVBR - 100 reactors developed in Russia as modular sources of the electric power for regional needs put forward a complex of requirements to structural and fuel materials of fuel elements. Necessity of maintenance, first of all, of high thermal stability, a low level of radiation creep, long structure-phase stability and, especially, long corrosion resistance of cladding materials in Pb-Bi coolant to such requirements.

At the first stages of the SVBR reactor technology learning it is supposed to use the fuel on the basis of uranium dioxide with the subsequent transition to dense fuel within the limits of developed strategy of a closed NFC. In view of requirements on duration of fuel lifetime in the SVBR core, special demands on gas evolution, heat conductivity, crack resistance characteristics are also made to fuel.

At present, the base structural materials for SVBR fuel elements are the compound alloyed 12% chromium-silicon ferritic-martensitic little swelling steels. Investigations carried out in IPPE for the last years have shown that long corrosion resistance of such steels in the Pb-Bi coolant is provided with formation in the coolant on a surface of the fuel element cladding the protective barriers on the basis of the compound alloyed oxide with the Me_3O_4 spinel structure. The spinel chemical compound varies permanently, moreover, during the tests spinel is enriched with chromium and silicon. Reached to the beginning of 2009 a resource of steel corrosion tests in non-isothermal circulating stands at the temperature of 600°C has exceeded 35 000 hours. Any corrosive-erosive damages of the fuel cladding pipes were not observed. According to electron-microscopic investigations, the cladding steel after tests has the structure-phase composition close to initial one, and after 35 000 hours of tests at 600°C has not practically undergone to processes of ageing. For the present time test proceeds, and the prospective integrated resource of corrosion stand tests will make 60 000 hours to 2012.

By present time the high radiation resistance of steel under neutron irradiation is confirmed by tests of the steel samples in the fast reactors BR-10, BOR-60, BN-350 and BN-600. The maximum reached damaging dozes amount to 60 dpa at irradiation temperatures of 340 – 670°C.

It is supposed to use tablets of the modified uranium dioxide as the fuel in the SVBR reactor fuel elements. Such fuel developed in IPPE with application of new technologies allows to fabricate the fuel tablets with the increased heat conductivity that will allow to reduce a temperature gradient in the fuel core and to decrease the thermomechanical interaction of the fuel with the cladding in the transient regimes.

Adding of the microadditions to the fuel allows to produce the tablets with the grain size of 20-40 microns that can provide the gas evolution reduction in the fuel element. These technologies allow to produce the modified oxide fuel both on the dry and on the wet scheme.

In the report the basic directions of the materials testing researches for the nearest years are presented. These ones will allow to confirm experimentally the declared characteristics of the efficiency of the core fuel elements of the fast reactors with the Pb-Bi coolant and also will allow to perform verification of the developed strength and thermomechanical fuel element computation codes.

POSTERS OF SESSION 5:

Fast reactor fuel cycles

Possibility of Reprocessing of SNF WWER and BN in Compressed Freon HFC-134a

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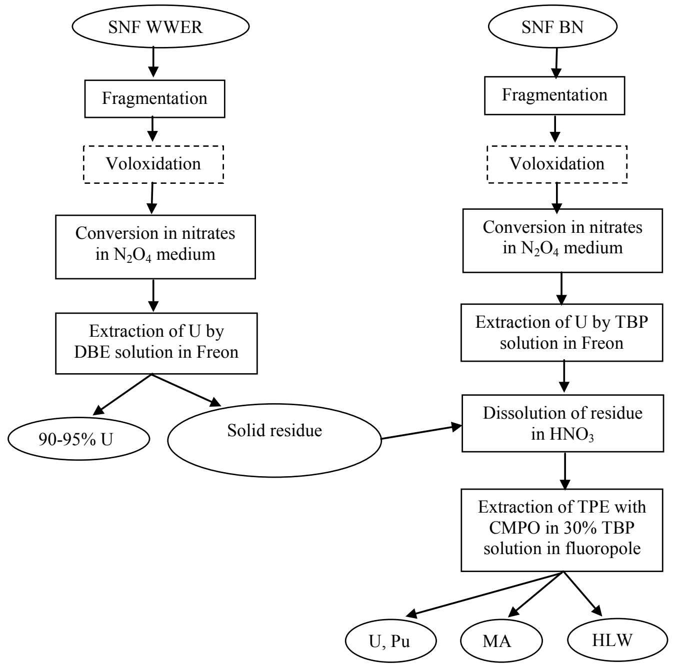
Reprocessing of spent nuclear fuel (SNF) formed at the nuclear power plant is the most promising procedure to decrease the volume of radioactive waste for long-term storage. "The strategy and main lines of the development of the nuclear power industry in Russia in the XXI century" envisages the development of power industry using thermal (TR) and fast (FR) reactors. It is obvious that the reprocessing of SNF from these reactors within a single plan will strongly decrease both capital and current investments owing to use of the uniform (common) infrastructure. At present only the hydrometallurgical procedures are developed for combined reprocessing of SNF TR and FR, which suggest utilization of the large volumes of aqueous solutions. The environmental safety of such radiochemical plants is provided by the procedures based on concentration of the aqueous solutions with subsequent preparation of solid forms of radioactive waste suitable for prolonged storage of disposal. At present, the economical effectiveness and environmental safety are the key requirements to radiochemical technologies. These requirements focus the attention on the non-aqueous procedures. In this paper we analyze the approach based on both non-aqueous and hydrometallurgical procedures for combined reprocessing of SNF TR and BR.

The combination of the conversion of oxide SNF in nitrates in the nitrogen dioxide medium and extraction of the target component using tri-n-butyl phosphate (TBP) solutions in Freons is considered as a main procedure for SNF TR reprocessing. SNF TR is first fragmented and then voloxidized and converted in nitrates. Then, using solutions of dibutyl ether (DBE) in Freon HFC-134a nearly 90-95% of uranium can be recovered from the melt, which further treatment is not analyzed in this paper. As a result, the solid residue (~10% to the initial SNF weight) is obtained, whose composition corresponds to SNF FR and ,thus, can be reprocessed with it[1].

SNF FR after fragmentation and voloxidation can be combined with the residue after SNF TR reprocessing mentioned above and then converted in nitrates. Uranium and plutonium from the resulting melt can be recovered with solutions of TBP in Freon HFC-134a and then used for preparing MOX fuel. The residue obtained can be either solidified as high-level waste (HLW) or it can be treated to remove light actinides (LA) for utilization in FR in the form of AMOX fuel[2].

The cycle of our studies showed the possibility of combined reprocessing of SNF WWER and BN using low-temperature low-water procedures. This approach will allow to use the infrastructure of the full-scale radiochemical plant for SNF TR reprocessing for smaller scale reprocessing of SNF BR and also to attain the following advantages as compared to the common hydrometallurgical and well-known "dry" procedures of SNF reprocessing:

– sharp decrease in the volume of secondary radioactive waste;
– absence of organic solvents;
– rather low temperature (60-80°C) of the process.



Scheme of combined treatment of SNF TR and BR

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Development of hybrid reprocessing technology with solvent extraction and pyro-chemical electrolysis

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Toshiba has been proposing a new fuel cycle concept for a transition period from LWR to FR. This concept has better economical process of the LWR spent fuel reprocessing than conventional process and the proliferation resistance for FR cycle of plutonium with minor actinides.

Toshiba has been developing a new Toshiba Hybrid Process technology with solvent extraction and pyro-chemical electrolysis of spent fuel reprocessing for a transition period from LWR to FR. The Tsohiba Hybrid Process combines the advanced solvent extraction process of the LWR spent fuel in nitric acid to recover pure uranium and the pyro electrolysis in molten salts to recover impure plutonium with minor actinides(Fig.1). High pure uranium is used for LWR fuel and impure plutonium with minor actinides for metallic FR fuel. The pyro-chemical process for the FR fuel recycle system is based on the research of electrorefining process in molten salts since 1988 in cooperation with CRIEPI. The new ToshibaHybrid Process can reduce the burden of the high-level waste disposal and the amount of the secondary waste from the spent fuel reprocessing plants.

Electrolytic reduction test using LWR spent fuel and oxalate precipitation test were carried out to confirm the feasibility of the Toshiba Hybrid process. The purpose of electrolytic reduction test was to investigate a uranium recovery and the oxalate precipitation test was to evaluate the recovery yield of plutonium with minor actinides. The results suggested that the purity of recovered uranium (U) and the recovery yield of plutonium with minor actinides (Pu+MA) could be achieved the target value (U purity : 99.97%, Pu+MA recovery yield:99.9%).

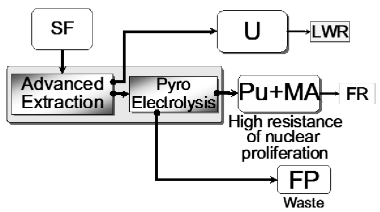

Fig. 1 Toshiba Hybrid reprocessing based on advanced solvent extraction and pyroelectrolysis.

Pyrochemical Cleaning of Final Wastes into Low and Intermediate Level Waste: PyroGreen

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The management of spent nuclear wastes are one of important issues for sustainable utilization of nuclear power. These days agreements for fuel cycle is that spent nuclear fuels should be recycled to minimize final waste volume and its toxicity. Korea Atomic Energy Research Institute (KAERI) has been proposing a national back-end fuel cycle strategy, called KIEP-21, that encompass all the requirements of the advanced nuclear fuel cycle such as reduction of waste volume, toxicity and high level waste (HLW) heat load [1-2]. In addition to KAERI's proposal, the concept of partitioning and waste disposal for back-end fuel cycle is being developed to make all the final wastes from Nuclear Power Plants (NPPs) into Low and Intermediate Level Waste (LILW), named "PyroGreen" by Nuclear Transmutation Energy Research Center of Korea (NUTRECK).

Performance assessments of disposal concepts and partitioning process are necessary to evaluate the possiblity of LILW disposal. A parameter to represent the performance of partitioning process is decontamination factor (DF) which is defined as the ratio of total mass of nuclides into the partitioning process to the mass that escapes the partitioning process and get into the waste stream. The required DF for LILW was calculated applying near-surface disposal concept and glass waste form in previous study. It is necessary to obtain the DF of 10^6 for plutonium in pyroprocessing to meet US 10 CFR 61.55 Class C regulation [3]. It is impossible to achieve such high DF with current pyroprocess methods.

Approaches of PyroGreen concept is composed of two parts, decreasing the required DF and increasing the DF of pyrochemical partitioning process to make final wastes into LILW. Deep geological disposal concept and ceremic waste forms (CWF) are introduced to decrease the required decontamination factor for LILW disposal. The risk by human intrusion scenario can be extremely lowered by choosing deep geological disposal concept (~200 meters underground) instead of near-surface disposal concept (~20 meters underground). The migration rate of radioactive isotopes from waste forms can be decreased by using CWF due to its smaller leach rate than glass waste forms. By utilizing deep geological disposal and CWF, the required DF can be reduced to 3.7×10^4.

To increase the DF of pyrochemical partitioning process, multi-stage hull electrorefining and residual actinide recovery (RAR) process named "PyroRedsox" is additioned to KAERI's pyroprocess. PyroRedsox is a new process concept combining binary(LiCl-KCl)/ternary(LiCl-KCl-LiF) molten salt purification and multi-stage counter-current selective chlorination to increase the partitioning process. To reduce the radioactivity of solid waste like hull to a clearance level, multi-stage hull electrorefinig is suggested. It is already reported that radioactivity of zirconium can be successfully reduced only by 3-stages of zirconium electrorefining in the LiCl-KCl-LiF salt [4]. PyroRedsox is introducted to purify the contaminated molten salt and to recover residual actinide which should be recycled by electrochemical method. Ternary salt(LiCl-KCl-LiF(10wt%)) from hull electrorefining and binary salt(LiCl-KCl) from electrorefining/electrowinning are contaminated with actinides and lanthanides. At first, all the actinides and lanthanides in molten salts are reduced to liquid bismuth metal cathode using glassy graphite anode. Then, lanthanides can be selectively oxidized into clean LiCl-KCl salt with minimum amount of actinides by adding the oxidant ($BiCl_2$). After the exchange reaction, lanthanides can be removed from molten salt as an oxide form by exposing the air. All the remaining actinides in bismuth metal is also oxidized into pure molten salt by exchange reaction with oxidant and recyled to electrowinning process. The preliminary study about mass balance of PyroRedsox shows that the DF of 5.0×10^4 can be achieved by additioning proposed processes. The schematic of PyroGreen concept is shown in FIG. 1.

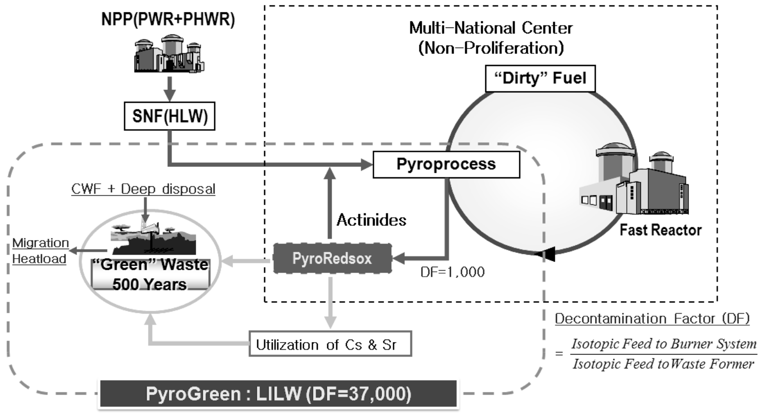

FIG. 1. Schematic of PyroGreen concept

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Development of high-temperature transport technologies for liquid Cd cathode of pyro-reprocessing

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Pyro-reprocessing is currently being developed in many countries for features; an intrinsic proliferation resistance to handle Pu with other actinides in any steps of the process and a recovery of long-lived transuranium elements for transmutation in the fast reactor without addition of further treatment. Pyro-reprocessing mainly entails electrorefining, cathode processing, injection casting, and salt treatment. Electrorefining is carried out to dissolve spent fuel and to recover actinide on the cathode. U and U-Pu-Cd alloy are deposited on the cathodes; these deposits are entrained with the salt and cadmium, respectively. These deposits are then distilled in cathode processing to separate salt and cadmium from the deposits. Feasibility of pyro-reprocessing has almost been convinced through many laboratory scale experiments. Hence development of the engineering technology basis of pyrometallurgical reprocessing is a key issue for industrialization. Development of transport technologies for high temperature fluids (molten salt and liquid cadmium) is one of crucial technologies basis, however there was a very little transport studies of the high temperature fluids. As for molten LiCl-KCl eutectic salt at approximately 773 K, we have already reported the successful results of transport using gravity and a centrifugal pump [1]. Also, the transport of liquid pure Cd was examined as follows by several transport methods[2]. The transport of liquid Cd using gravity was controlled by adjusting the valve. The liquid Cd was transported by a suction pump against a 0.93 m head and the transport amount of Cd was well controlled with the Cd amount and the position of the suction tube. The transportation of liquid Cd at approximately 700 K could be controlled at a rate of 0.5 to 2.5 dm^3/min against a 1.6 m head using a centrifugal pump.

On the basis of these high-temperature fluid transport experiments, an engineering-scale electrorefiner was newly designed and fabricated in an argon glove box as shown in Fig.1.[3] This electrorefiner has a high-temperature fluid transport system for liquid Cd cathode instead of the conventional cathode in small-scale tests. This transport system worked for transporting the Cd alloy and Cd, as shown in Fig.2. Cd cathode deposit was transported to the vacuum tank from Cd cathode crucible by the suction pump, and was transported to the Cd alloy buffer tank from the vacuum tank by the gravitation, and then was transported to the distillation crucible from the Cd alloy buffer tank by the centrifugal pump. The cathode processor was distilled for separating with actinide and Cd, and Cd vapor was condensed to the condenser. The pure Cd was transported from the condenser to the Cd supply tank by the gravitation, and the pure Cd was supplied from the Cd supply tank to the cathode crucible by the gravitation. Consequently, the handling time between the electrorefiner and the cathode processor can shorten without increasing and decreasing the temperature of the deposits by introducing this system.

The electrorefiner was tested the transport of Cd cathode deposit by using Gd as simulated actinides. The liquid Cd was transported by a suction pump against a 0.93 m head and the transport amount of Cd-Gd alloy was well controlled with the Cd amount and the position of the suction tube. The transportation of Cd-Gd alloy at approximately 735 K could be controlled at a rate of 0.01 to 0.4 dm^3/min against a 1.6 m head using a centrifugal pump. The transport of Cd -Gd alloy from the Cd cathode crucible to the distillation crucible was good material balance of 97 %. The handling time between the electrorefiner and the cathode processor was shorten within 1 hour.

Fig.1 The electrorefiner with high temperature fluid transport system.

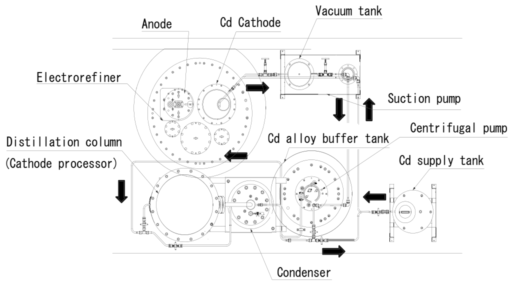

Fig.2 The transport system diagram for Cd cathode deposit of the electrorefiner.

ACKNOWLEDGMENTS

This study is the result of the "development of engineering technology basis for electrometallurgical pyroprocess equipment" entrusted to the Central Research Institute of Electric Power Industry (CRIEPI) by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT).

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Handling Technology of Low Decontaminated TRU Fuel for the Simplified Pelletizing Method Fuel Fabrication System

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Japan Atomic Energy Agency has been conducting the Fast Reactor Cycle Technology Development (FaCT) project. The combination of the sodium-cooled fast reactor with oxide fuel, the advanced aqueous reprocessing and the simplified pelletizing fuel fabrication has been developed principally because it was the most promising concept for commercialization. The advanced aqueous reprocessing, consisted of simplified low decontaminated extraction process and minor actinides (MAs) recovery process, allows the residual of fission products(FPs).[1] This low decontaminated process brings benefit such as cost reduction for reprocessing, proliferation resistance, etc. However, new development issues are given to the fuel fabrication system because source material is the low decontaminated TRU fuel produced from such reprocessing process. ***Figure 1*** shows the relations of R&D issues caused by three features of low decontaminated TRU fuel. As for fuel pelletizing, the fuel composition has multiple elements of MAs and FPs; target composision in the FaCT project is 5 wt.% of MAs of and 0.2 wt.% of FPs at maximum. Process behaviour of such material system will be more complex comparing with ordinary UO_2 and PuO_2 system that affects pellet manufacturing process and fuel irradiation performance. The feasibility was almost verified by lab-scale fabrication tests and JOYO irradiation tests of MOX pellets containing Np, Am and simulated FP. As for equipment system, measures against high radio-activity and high heat generation by such fuel will be tough development issues because a realized fuel fabrication plant has capacity of 200 tHM a year. R&D deployment and current status in the FaCT project will be reported focussing on these area.

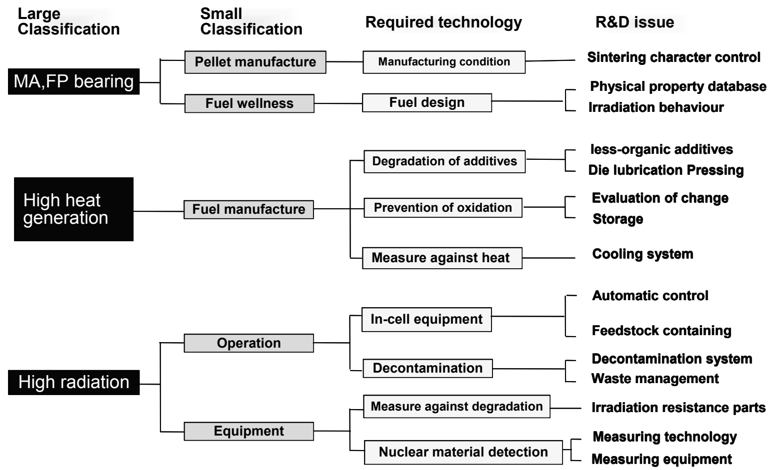

Figure 1 R&D Issues Specific on Low-DF TRU Fuel Fabrication

1. Development of remote maintenance system in a hot cell

In a fuel fabrication plant for the low decontaminated TRU fuel, main process equipment are required to have grater remote operability and far grater remote maintenance performance than conventional glovebox type equipment, because equipment are installed in hot cells. In particular, it is essential to establish repairing system in the hot cell, because the equipment consist of machines required to have high precision and operators can not maintain these machines directly. We proposed basic concept for repairing and maintenance system consisted of three stages; (a) replace the out ordered module in the main process cell, (b) decontaminate and roughly disassemble the module in the maintenance cell and (c) refurbish the module using the globe box in the maintenance room. To confirm feasibility of this concept, representatives of in-cell equipment, a pressing machine, a pellet inspection equipment and some powder analysers have been investigated by cold mock-up examination. A pressing machine is favorable for testing feasibility of modurarized equipment because the process uses various machining. A pellet inspection equipment and powder analysers are favorable for developing how to maintain such precise and sensitive equipment. Modularized pressing machine was designed conceptually consisting 30 modules with 70 kinds of maintenance terms. In order to assist operation on replacing modules, a robot arm manipulator is introduced. Also, a small size robot arm for handling a pellet is developing as well. Cold mock-up tests using modularized equipment and handling equipment will be completd by the end of JFY 2009.

2. Development of heat removal system

High heat generation by decay heat of MAs causes much undesirable effects to fuel quality, such as, degradation of organic additives, re-oxidation of fuel. Development of heat removal system is necessary for realizing mass-production plant, even though the simplified pelletizing process brings some advantage on this issue (We have adopted less inorganic additives process ; binder-less granulation and die wall lubrication pressing process. This adoption can solve the problem of potential evaporation of the additives caused by higher temperature). The measure will be chosen from the followings by considering particular process equipment ; (a) disperse source fuel, (b) improve function of standing to cool, (c) prevent oxidation by surrounding inert gas, and (d) forced cooling operation etc. Among above maesures, (a) is the easiest to be adopted. For examaple, temperature rising on stand-alone pellet is neglisible to cause undesirable effects under heat generation condition of 20 W/kgHM. However, at the fuel pin bundle assembling equipment, the heat generation reaches 2.6 kW per one subassembly, so in this reguard, forced cooling operation to remove heat is essential. The fuel pins are handled horizontally and are assembled up to a bundle in this equipment as same as an equipment for a conventional MOX fuel assembly. Cooling down should be performed by air spraying at right angle to the bare pin bundle which flow channel geometry is complicated by lots number of pins and wrapping wire. Considering such complicated system, cold mock-up test have been performed to develop the heat removal system and the evaluation tool. Obtained data shows such cooling sysytem is feasible though various tests are necessary to be done.

Large storage for the pellet and the new fuel assembly are designed at the realized plant because sufficient buffer is needed to keep plant operation running smoothly. For these storage, cooling down by air blow is preferable. At the plant of 200tHM/y scale, rate of the air blow was estimated to be 110,000 m^3/hr approximately. The additional cost of this countermeasure is seemed low because the total ventilation rate of main process building is already 710,000 m^3/hr even if there is no additional air blow for cooling down.

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Development of spent salt treatment technology by zeolite column system

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In the pyrometallurgical reprocessing of metallic fuel, the spent electrorefiner salt containing fission product (FP) elements should be purified and reused. For this purpose, we are developing a salt treatment process by using selective absorption of FPs on zeolite 4A. The absorption behaviors of FP elements in molten LiCl-KCl salt have been studied by batch type absorption tests [1]. However, salt treatment process by using "Column type method", in which molten salt flows through columns filled with zeolite, is preferable in the practical process in the view points of high decontamination of FPs in a single step and effective treatment of the salt in shorter processing time.

For obtaining the basic data of the column system such as flow property and ion-exchange performance while high temperature molten salt is passing through the column, an experimental apparatus equipped with a fraction collector was developed. By using this apparatus, following results were obtained.

(1) The relationship between velocity of molten salt passing through the columns filled with zeolite 4A powder and argon gas pressure to push the molten salt go through the columns was measured by using columns of 1cm in inner diameter and 10cm or 30cm in length. As shown in Fig. 1, the average flow velocity increased in proportional to the gas pressure and decreased inversely proportional to the column length. Additionally, the relationship between velocity and gas pressure was quite close to that of obtained by using water instead of molten salt, whose kinematic viscosities are quite similar [2].

(2) The absorption behavior of cesium, which was used as a representative of FP elements, on zeolite 4A in the columns was measured by taking the exhausted molten salt samples by using the fraction collector. As shown in Fig.2, it was revealed that the decontamination factor of cesium was highest at the beginning of the salt flow and its value decreased with the increase of amount of the passed molten salt.

From these results, feasibility of the zeolite column system for purification and recycle of the spent molten salt was confirmed.

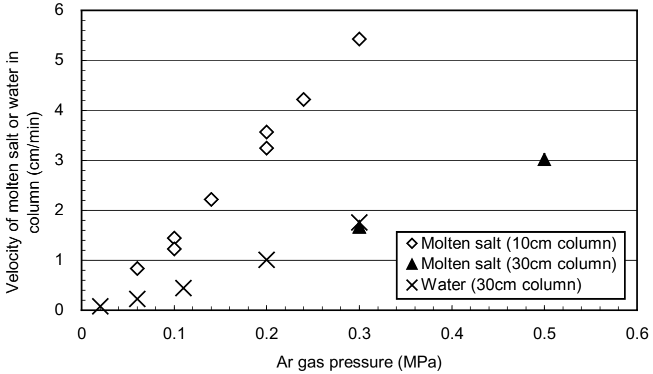

FIG.1. Relationship between velocity of molten salt or water passed through column with zeolite 4A powder and argon gas pressure. Velocity of molten salt or water was calculated taking volume occupied by zeolite in columns into account.

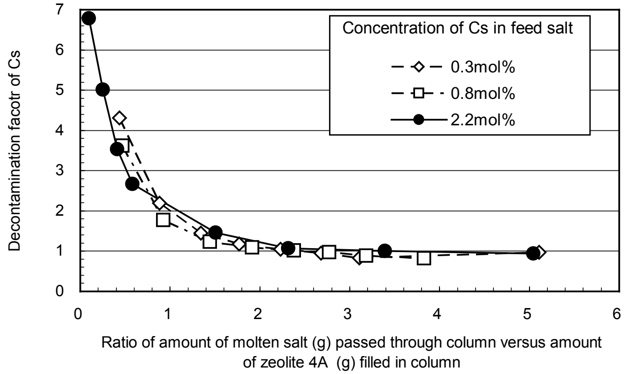

FIG.2. Decontamination factor of cesium in molten salt passed through zeolite column filled with zeolite 4A powder.

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Burnup behavior of FBR fuels sourced in uranium and plutonium recycled in PWRs and its influence on fuel cycle economy

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Considering the strategic security of uranium resources, authors investigated the effective use of fuel recycling in the current light water reactor system (1.1GWe-class PWRs of standard type), where we supposed the uranium fuel reformed by re-enrichment of the recovery uranium from the PWR spent fuels and the MOX fuel produced with the recovery plutonium as main fissile, and examined three types of MOX fuels prepared by making choice of these matrixes among (A) the recovery uranium, (B) the depleted uranium as the waste from natural uranium enrichment, and (C) the depleted one sourced from recovery uranium re-enrichment. Calculations were carried out of the multi-component uranium isotope separation employing the ideal centrifuge-cascade and of the reactor core burnup analysis using the comprehensive neutronics computation system SRAC-2006 [1]. Burnup analysis was performed under the following conditions: The concentration of ^{235}U in the depleted natural uranium is 0.2 mol%. This conditional mass-effect on isotope separation affects the other isotope concentrations in the centrifuge cascade. The integrated burnup of uranium and full-MOX fuels is 45 GWd/t-HM during 3 cycles in one batch burning. The spent original-uranium-fuels are cooled for 10 years before these reprocessing and then fabrication of reformed uranium or MOX fuels. In the reprocessing, the plutonium is recovered as a 50-50 mixture with the spent uranium oxide and the remaining uranium oxide is recovered in isolated form.

The result of burnup analysis shows that the uranium recovered from the spent original fuel contains ^{235}U enriched about 20 % more than that of the natural uranium and also ^{236}U transformed from ^{235}U capturing neutrons during fuel burning. The constituent ^{236}U behaves as a neutron absorber. Hence, the reformed uranium fuel containing ^{236}U enriched additionally in the re-enrichment process requires the fissile ^{235}U concentrated 1.154 times more than that in the original fuel, in order to established the same burnup performance, mainly because of poisonous influence of ^{236}U. The problem of ^{236}U is, rather than its neutron absorptivity, its transformation to ^{237}Np having a large cross-section area of neutron capture. The MOX fuels manufactured with the matrixes sourced in the recovery uranium, however, can compensate the neutron toxicity of ^{236}U with ^{235}U enriched in the matrixes while contributing to core burnup. So, these MOX fuels require a decrease of plutonium, compared with the MOX fuel made with the depleted natural uranium. The quantitative fuel reproduction in the first recycle was estimated as follows: In the case of A-type matrix, the MOX fuels for 13.7 PWRs and the reformed uranium fuels for 12.4 PWRs can be obtained from the conventional spent fuels of 100 PWRs. In the case of B-type matrix, the MOX fuels for 13.2 PWRs and the reformed fuels for 14.2 PWRs can be manufactured from the same spent fuels. In the case of C-type matrix, the MOX fuels and the reformed uranium fuels can be provided for 12.7 and 14.2 PWRs, respectively. Therefore, the fuel recycle efficiencies in the PWR system are 26.1 %, 27.4 % and 26.9 % for the cases A, B and C, respectively. (partially reported in 4th RRTD Int. Workshop for Asian Nucl. Prospect (1st Asian Nuclear Prospect Workshop), October 19-21, 2008 Kobe, Japan.)

The multi-recycle in the light water reactor (LWR) fuel system, however, will be involved by that ^{236}U in reformed fuels increases more and more with recycle times. The spent-fuel resources are recommended to be utilized in the fast breeder reactor (FBR) fuel cycle. Thus, we have been studying the effective use of uranium and plutonium resources in the FBR cycle cooperating with the LWR system. In this work, a FBR of practical type (1.5 GWe-class) was conceptually designed in the SRAC numerical system, referring to a commercial-type FBR model proposed by Japan Atomic Energy Agency and The Japan Atomic Power Co. [2]. Its main design parameters are listed in Table 1. The FBR designed in this work has the burning core consisting of two regions for power smoothing: inner and outer cores where the fueling is of lower and higher plutonium concentrations. A trial calculation verified that the present FBR demonstrates the burnup performance similar to the reference FBR.

The burnup behavior of FBR fuels was analyzed which were sourced in the original uranium spent-fuel, the reformed uranium one and the full-MOX ones with various uranium matrixes. And also, the behavior of blanket materials was investigated which were individually of the natural uranium, the depleted natural uranium, the recovery uranium from the original spent-fuel, and the reused uranium recovered from the reformed uranium one or the MOX ones. In comparison with the fuel using the matrix made of the depleted natural uranium, a type of fuel using ^{236}U-riched matrix exhibits hardly declining burnup behavior. The breeder containing the fissile ^{235}U enriched reduces the necessary content of plutonium fissile in the core fuel, of course. More detailed results of analytical investigations are to be reported in this conference. On the basis of analytical results, we have estimated the economical strategy of uranium and plutonium utilization in the FBR cycle following the LWR system.

Table 1. Conceptual design parameters of a practical model FBR in this work.

Power output	3570 MWt	Fuel rod cladding	φ 10.4 mm × t 0.71 mm
Inlet/outlet temps.	668/328 K	Cladding material	SS-316
Operation cycle mode	4 cycles/batch	Triangular pin pitch	11.44 mm
Cycle period	800 days	Pins in a fuel assembly	271
Core (+ blanket)	φ 550 cm × 140 cm	Triangular assem. pitch	196.0 mm
Burning core	φ 512 cm × 100 cm	Core assems. (in./out.)	558 (279 / 279)
Blanket layer thickness	t 20 cm (upper/lower)	Blanket assemblies	96 (annulus blanket)
Spent fuel burnup rates (a standard case)	90 GWd/t (core av.) 150 GWd/t (inner core)	Fuel smear density	MOX: 82 %TD, UO_2: 90 %TD
Pu-content in fuel (a standard case)	18.3 wt% (inner core) 20.9 wt% (outer core)	Breeding ratio (a standard case)	1.1

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The Way to Achieve Sustainable Nuclear Energy Fuel Cycle

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This paper presents how to deploy our advanced reprocessing system from the current reprocessing system to sustainable fast reactor cycle.

Toshiba is developing advanced reprocessing process based on pyro-chemical technology that can be applied to nuclear fuel cycle from the introduction phase through the era of fast reactors, and aim to use it as the basis to achieve the next generation nuclear fuel cycle (Fig.1)

As a first step, we propose the Aqua-Pyro Process whereby MA is recovered from the high-level radioactive liquidous waste (HLLW) generated by operation of the Rokkasho Reprocessing Plant, using an electrolysis technology cultivated by pyro-chemical reprocessing.[1] Next, Toshiba Hybrid Reprocessing Process will be applied which is composed of the Aqua-Pyro Process combined with the aqueous reporcessing porcess, achieving a reprocessing process that makes it difficult to separate pure plutonium and that reduces burden of final disposal of HLLW significantly. This process realizes for light water reactor fuel reporcessing with a high-level proliferation resistance. Subsequently, pyro-chemical reprocessing system by electrolysis technology will be applied to reporcess fast reactor fuel.

This approach enables smooth transition from the coexistence period of light water reactors and fast reactors through to fast reactor era.

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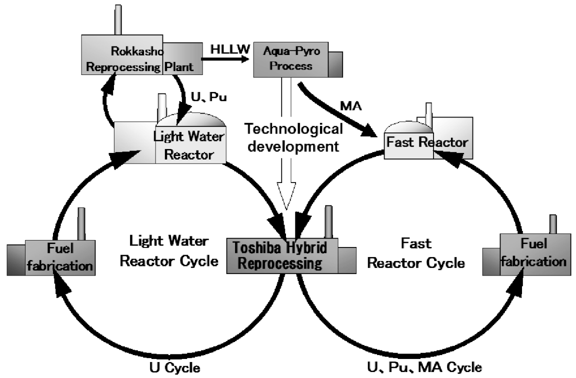

FIG. 1 The concept of next-generation nuclear fuel cycle

TRU management by fast reactor toward sustainability and flexibility

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This paper presents transuranics (TRU) management by fast reactor changing TRU conversion ratio (CR). A closed nuclear fuel cycle consisting of Advanced Recycling Reactor (ARR) and Consolidated Recycling Facility (CRF) is investigated so as to consume TRU from used nuclear fuel from light water reactor (LWR) by sodium cooled fast reactor (SFR) and reduce the amount of TRU and the decay heat of nuclear waste disposed in repository. On the other hands, self-sustainability is one of strong points in SFR and it is preferable and possible that the ARR is able to change the conversion ratio.

The design targets are set as as follows

- TRU CR will be 0.5-0.6 in TRU burning core of the ARR and the breeding ratio of TRU self-supply core will be 1.0 or more. And these characteristics will maintain after multi recycling. The TRU CR is defined by INL as follows.[1]

  TRU CR. = (RHM – RTRU)/RHM (1)

  RHM: mass consumption of HM between beginning of life and end of life, RTRU: mass consumption of TRU between beginning of life and end of life.

  TRU burning capability is defined as follows.

  TRU burning capability = (TRU-B – TRU-P)/EL-P (2)

  TRU-B: burnt TRU (kg), TRU-P: produced TRU (kg), EL-P: produced electricity (TWeh)

- The outer demension of subassembys are the same between the the TRU burning core and the TRU self-supply core, taking the replacement of core into account.

It is assumed as follows.

- The ARR is a 500 MWe of SFR (thermal power:1180MW), with oxide fuel and burnup of 150 GWd/t.
- The TRU composition of fresh fuel is Np-237 5.3%/Pu-238 3.1%/Pu-239 45.7%/Pu-240 21.3%/Pu-241 7.2%/Pu-242 3.9%/Am-241 3.9%//Am-243 2.1%/Cm-244 0.8% (wt.).

According to the nuclear calculation, it is found that the TRU burning core can consume TRU and the TRU self-supply has an appropriate breeding characteristics, as shown in TABLE I. There is no doubt that the provided ARR can change the TRU cosuming and breeding characteristics from TRU CR. of 0.5-0.6 to B.R. of 1.0.

TABLE I

TRU consumption in TRU burning core of ARR and Breeding Ratio in TRU self-supply core

(kg/TWeh)

	TRU burning core		TRU self-supply core	
Recycling No.	1	10	1	10
B.R.	-	-	1.03	1.08
TRU CR.	0.56	0.51	-	-
TRU burning capability (kg/TWeh)	48	53	<0	<0
(breakdown)				
Pu	41	42	-6	-9
Puf	44	26	-2	-7
Am	6	3	5	0
Np	2	8	1	2

Recycling No.1: All of TRU are from LWR, Recycling No.10: Cycle after 9 recycling

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Fuel cycle for reactor SVBR-100

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Modular fast reactor with lead-bismuth heavy liquid-metal coolant of 100 MWe (SVBR-100) belongs to the IVth Generation reactors and must operate in a closed nuclear fuel cycle (NFC) without consumption of natural uranium. Launch of fast reactors (FR) used to be considered straight at mixed uranium-plutonium fuel. However, for currently existing costs of natural uranium and service on its enrichment that launch of FRs is not economically effective.

This is conditioned by the fact that the volume of reprocessing the spent nuclear fuel (SNF) of thermal reactors (TR) per ton of plutonium, which determines the cost of constructing and operating the corresponding enterprise, will be very high because of low content of plutonium in the TRs' SNF. Economical effectiveness of FRs will be reduced as the enterprise on reprocessing the TRs' SNF should be constructed prior to implementation of FRs in nuclear power (NP). Moreover, the pace of implementation of FRs in NP will be constrained by the quantity of the TRs' SNF.

With reference to modular RIs SVBR-100 purposed for application first of all in regional and small power, the report has validated an alternative strategy of implementation of those FRs in NP. That strategy could be more economically efficient, unless considerable escalation of costs for natural uranium happens. This is conditioned by the fact that in the nearest future use of mastered oxide uranium fuel and operating in the opened fuel cycle with postponed reprocessing will be the most economically efficient.

Changeover to the mixed uranium-plutonium fuel and closed nuclear fuel cycle (NFC) will be economically efficient in an event of increase of natural uranium costs when the expenditures for construction of enterprises on reprocessing the SNF, re-fabrication of new fuel with plutonium and their operating are becoming less than the corresponding costs of natural uranium, service on its enrichment, expenditures on manufacturing the fresh uranium fuel and long temporary storage of SNF.

As FRs using uranium fuel and operating in the opened NFC consume much more natural uranium in comparison with thermal reactors (TR), and at the expected high paces of NP development the cheap resources of natural uranium will be exhausted prior to the middle of the century that will cause increase in the uranium cost, the period of FRs operating in the opened NFC should be maximally reduced.

The expenditure caused by changeover to the closed NFC will be less, if plutonium extracted from the own SNF of uranium loads is used in fabrication of first MOX fuel loads.

If the oxide uranium fuel is used, by the end of the lifetime a comparatively high breeding ratio (BR) of reactor SVBR-100 provides a sufficiently high content of plutonium in the SNF that may be used in the next fuel lifetimes while organizing the closed fuel cycle.

Moreover, the own SNF of starting loads from oxide uranium fuel contains large quantity of unburned uranium-235 that is expedient to use for forming the load for the next lifetime.

That approach to organization of fuel cycles with full reprocessing of the own SNF will reduce considerably the integral consumption of natural uranium, and thus it will make the NPPs based on SVBR-100 type RIs quite competitive with those based on RIs with TRs.

The report also reveals that in the closed NFC instead of pile uranium, the TRs' SNF (of both WWER and RBMK without separating of uranium, plutonium, minor actinides and fission products (FP) can be used (utilized) as makeup fuel similarly to the DUPIC-technology for reactors CANDU.

CAPABILITIES OF THE BREST REACTORS AND THEIR FUEL CYCLES IN DEVELOPMENT OF NUCLEAR POWER BASED ON FAST REACTORS

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Summary

The inexhaustibility of fuel resources when using fast reactors (FR) in closed fuel cycles and the absence of natural Pu has formed a notion of the two-phase nuclear power evolution process. At phase 1 nuclear power relies on thermal reactors (TR) with U-fuel in an open cycle, while at phase 2 the Pu accumulated in the TR SNF as the TR initial load fuel, gives rise to a broad-scale development of nuclear power based on FRs, which later on embark on a self-evolution path through Pu breeding (BR>1), thus providing Pu not just for themselves but also for the introduction of new FRs. However, erroneous is a more than half-a-century long and still existing opinion that the development scale and pace of FR-based nuclear power are confined by the quantities of plutonium accumulated in the spent nuclear fuel of thermal reactors (TR SNF) and the plutonium breeding rate in FRs. In reality, FRs can be deployed not just based on Pu but also based on a Pu mixture with enriched U and even solely on enriched U with further conversion to U-Pu fuel in the process of U-235 burnup and Pu-239 generation. As far as the cost of natural U and its separation work is concerned, this is 4 to 5 times as profitable way to do than to generate in the TR SNF the Pu needed to start FRs.

Deployment of a BREST lead-cooled FR based on the enriched uranium nitride is considered as an example. It is shown that the reactor switches to operation on (UN-PuN) fuel in three to four five-year fuel cycles, the fuel burnup-induced reactivity change not exceeding β_{eff} even during the transition period. It is shown that, with nuclear power developing in conditions of limited resources of economically affordable natural uranium, the selection of the proper evolution scenario with allocating some of the natural uranium resources intended for TRs to the deployment of the BREST-type FRs helps more than double by the end of the 21 century the anticipated total capacity of FR-based nuclear power as compared to these started on Pu from the TR SNF. The feasibility of BREST deployment lifts constraints on the scale of FR-based nuclear power development with no plutonium breeding thanks to uranium blankets, which runs counter to nonproliferation requirement.

The BREST reactors satisfy in the best way possible to all requirements with respect to innovative reactors of future nuclear power. Safety of these is achieved largely through inherent natural properties of the fuel, the coolant and other reactor components, which do not require expensive engineered systems, this harmonizing safety and economic efficiency. When operated in a closed fuel cycle using electrochemical reprocessing of fuel with no separation of Pu and MA from the fuel mixture, at CBR~1 these fully provide themselves with plutonium, while at the same time excluding production of weapon-grade Pu. Return of Pu, MA and long-lived fission products to the reactor for burning and transmutation offers a solution to the RW management and final disposal problem without disturbing the natural radiation balance of the Earth.

STUDY ON TRANSITION SCENARIO FROM THERMAL REACTOR TO FAST REACTOR IN JAPAN

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JAEA has been promoting the Fast Reactor Cycle Technology Development Project (FaCT) which aims at suggesting promising fast reactor (FR) cycle concepts. To realize sustainable nuclear power utilization and stable global society in 21st century, development of the adequate transition scenarios from LWR cycle to FR cycle is important. This report clarify the feature of the typical transition scenarios, focusing on operating method of the second reprocessing plant after the Rokkasho plant, and recycling mode of minor actinide (MA) recovered from spent fuels. It is assumed that the Japanese nuclear power generation capacity is constant at 58GWe after 2030, and evaluation period is 2000-2150. FRs are deployed commercially at 2050. This evaluation was performed by using nuclear fuel cycle material balance analysis code “FAMILY-21”.

Typical scenarios regarding operating method of the second reprocessing plant which we took in this analysis are as follows,

Scenario 1) Mixed reprocessing of LWR UO_2 spent fuels and LWR MOX spent fuels with mixture ratio of 5:5

Scenario 2) Mixed reprocessing of LWR UO_2 spent fuels and LWR MOX spent fuels with mixture ratio of 8.5:1.5

Scenario 3) Mixed reprocessing of LWR MOX spent fuels and FR spent fuels

As to recycling mode of MA in FR cycle, we assume some parameter cases, namely with or without MA recycling, upper limit of MA concentration in new FR fuels, only Np recycling, Am/Cm delayed recycling and so on.

As a result, the second reprocessing plant with annual capacity of 1,200 tons for LWR spent fuels is necessary in all scenarios. In addition, other reprocessing plants with total 600 tons/year capacity are necessary for FR spent fuels after FR deployment. Reprocessing plant capacities are shown in Fig. 1 and nuclear power generation capacity of scenario 1 is shown in Fig. 2. In the case of recycle of MA recovered from not only LWR spent fuels but also FR spent fuels, the accumulative amount of MA transferred into high level radioactive waste (HLW) at 2150 decreases by approximately one fourth compared with no MA recycling case.

The maximum amounts of MA interim storage are presumed to be about 30 tons in upper limit of MA concentration 5% and 4% cases, and 70 tons in 3% case. While the Am/Cm delayed recycling have some challenges regarding MA storage, it is effective in reducing heat generation in FR new fuels fabrication.

In conclusion, the adequate transition scenarios from LWR cycle to FR cycle with MA recycling are expected to contribute toward sustaining the utilization of nuclear power from the viewpoints of environment and energy security in Japan.

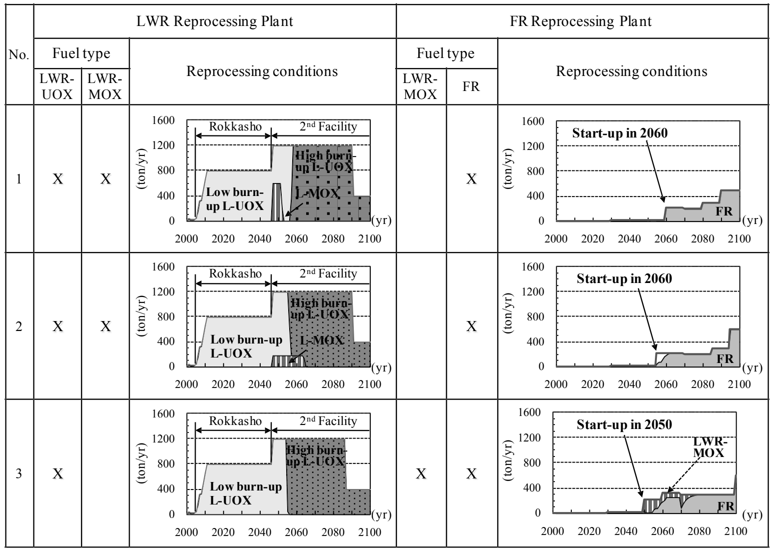

FIG. 1. Reprocessing plant capacity for LWR and FR spent fuels

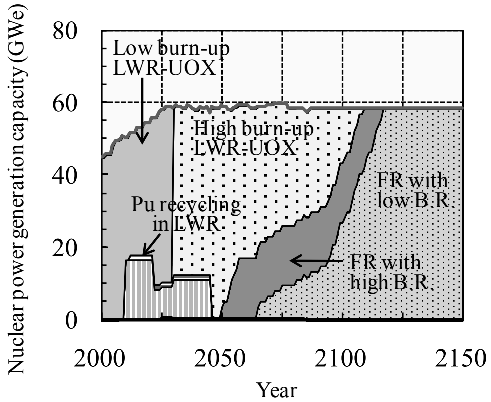

FIG. 2. Nuclear power generation capacity of scenario 1

Fuel cycle investigation for the flexible deployment of FBR

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According to the Nuclear Energy Policy Framework, Japan will deploy fast breeder reactors (FBR) from around 2050 by the replacement of light water reactors (LWR) after their 60 years life [1]. Plutonium needed to deploy FBR should be supplied from the LWR spent fuel (SF) reprocessing. As the LWR were not constructed with constant speed, the deployment rate of FBR and annual Pu amount necessary for FBR will not be constant. The LWR reprocessing facility must have the capacity to supply Pu of the highest necessary amount and it needs the Pu balance adjustment to correspond the FBR deployment rate changes. The fuel cycle systems were investigated for the flexible deployment of future FBR.

The flexible fuel cycle system was proposed and compared with reference (ordinarily considered) system for the various FBR deployment scenarios. Figure 1 shows the both systems. The reference system consists of full LWR reprocessing to supply FBR fresh fuel (FF) from LWR SF and full FBR reprocessing to recycle FBR fuels in FBR cycle. The FFCI system consists uranium removal (a part of LWR reprocessing) to separate most U from LWR SF, and full FBR reprocessing to recover Pu from U removal residue (recycle material) and recycle FBR fuels [2].

In case of FBR deployment rate changes (delay), the reference system will store FBR FF (Pu product) with continuing the LWR reprocessing or store LWR SF with partially stopping the LWR reprocessing. The storage of Pu product needs to overcome the problems of lower proliferation resistance for relatively pure Pu and Am-241 accumulation from Pu-241. The storage of LWR SF needs to overcome the problems of lower capacity factor for LWR reprocessing and SF accumulation. On the other hand, the FFCI system will store recycle material (RM) with continuing the U removal. The RM is the residual material after most U removal from LWR SF, and contains Pu, fission products (FP), minor actinides (MA) and small amount of U. The issue for FFCI is the innovative technology development for the RM storage system.

The characteristics of the FFCI system were basically investigated and compared with the reference system from the viewpoint of flexibility, economy, proliferation resistance as well as compatibility of the RM storage system. The investigations revealed that the FFCI system could flexibly respond to the various FBR deployment scenarios, reduce the fuel cycle cost

about 30% lower than the reference system and have high proliferation resistance for RM compared with Pu product.

As the RM has similar (slightly higher) heat density with vitrified high-level waste, the authors aimed to apply its vault storage facility already realized in Japan and analyzed the heat removal and criticality safety properties. Simulated RM oxide powders were prepared from nitric acid solutions of U and other stable elements to get the heat conductivity values. The values were found to depend on the powder particle size and the heat conduction analyses clarified that the temperatures at important points of the RM storage facility could be kept under the limited ones.

The criticality safety was analysed by the same method for re-criticality for severe accident of FBR under the conditions of coolant loss, core melting and heavy elements (actinides) sedimentation at the bottom. Preliminary analyses clarified that the RM storage facility was also safe for criticality even in the severe accident case. Thus the FFCI system was proved to have good abilities for the flexible FBR deployment.

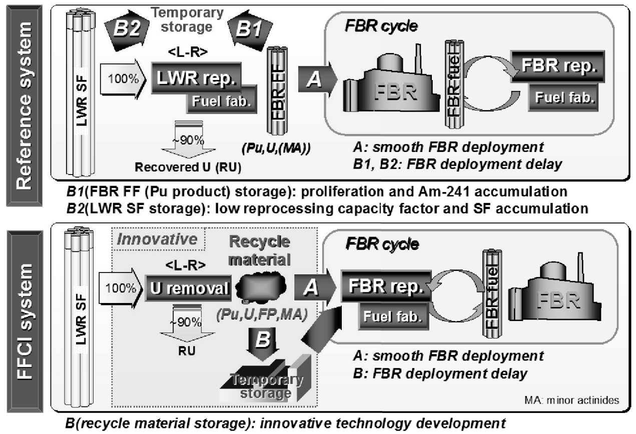

Fig. 1 Fuel cycle systems for the deployment of FBR

*This study includes the result of "Research and Development of Flexible Fuel Cycle for the Smooth Introduction of FBR" entrusted to Hitachi-GE Nuclear Energy, Ltd. by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT).

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Fast Breeder Reactor Core Concept Consistent with Fuel Cycle System during the Transition Period from LWR to FBR Cycles in Japan

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During the transition period from the light water reactor (LWR) cycle to a fast breeder reactor (FBR) cycle, Pu needed to start up the FBRs will be provided from the next reprocessing facility which will be constructed around 2045. From the viewpoint of economy for this next reprocessing facility, the FBR core with lower Pu inventory may be suitable. In the feasibility study for commercialization of the FBR cycle [1], two types of sodium-cooled MOX-fueled FBR core concepts were investigated. The first one is the compact-type core whose diameter and fuel inventory are relatively small. The second one is the high conversion ratio-type core whose effective burnup is relatively larger due to the increased ratio of the MOX fuel to the whole core including the blanket fuel. In this paper, a design study of the FBR core concept which is consistent with the fuel cycle system during the transition period was carried out based on the compact-type core. This core was chosen because it has the potential to reduce the capacity of the next reprocessing facility.

In previous studies, many FBR core characteristics have usually been evaluated for their equlibrium state using the multi-recycling fuel composition. However, Pu from LWR spent fuel should be used to start up the FBR at least for the initial core and the succeeding first few cycles in the transition period. The FBR core being loaded with Pu from LWR spent fuel trends to increase burnup reactivity compared to a core loaded with Pu from the FBR multi-recycling fuel composition. The reason is that the isotopic fraction of Pu-241 in LWR spent fuel is larger than that in the FBR multi-recycling spent fuel. The increased burnup reactivity may reduce the cycle length of the FBR core under the same control rod design conditions for an equilibrium cycle core. This characteristic is particulary remarkable in the compact-type core.

We chose loading of minor actinides (MAs) into the FBR MOX fuel as the countermeasure to the increased burnup reactivity from the viewpoint of utilizing nuclear characteristics of the MAs. The maximum MA content in the MOX fuel was set as 5% reflecting irradiation test

results obtained in the experimental fast reactor Joyo [2]. MAs recovered from LWR spent fuel which provide Pu to start up FBRs are loaded into the initial loading fuel assemblies and exchanged fuel assemblies during some cycles until equilibrium. The average MA content of the initial loading fuel was assumed to be 3%, and that of the exchange fuel was assumed to be 5%. The core performance including burnup characteristics and reactivity coefficient were also evaluated, and it was confirmed that the transient core from the initial loading until the equilibrium cycle for loaded Pu from LWR spent fuels could maintain performance resembling that of the FBR multi-recycling core.

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Transition scenarios of nuclear power development in Russia

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The main system-scale problems of the present day nuclear power (NP) are well-known: inefficient use of natural uranium and accumulation of spent nuclear fuel (SNF). A path to the solution of these problems might be seen as the changeover of NP to the technologies of closed fuel cycle with fast breeder reactors. The results obtained to date in the area of fast reactor development ensure the technological preparedness for Russia, to begin the realization of such changeover of NP during the period until 2030. In this aspect the main question is an optimal strategy of the NP development in the transitional phase that could minimize the investment-related risks. When determining the priorities, it is necessary to take into account the levels of mastering achieved in different technologies and the risks related to the non-achievement of the strategic long-term objective set – creation of the large-scale nuclear power by the middle of the century.

System requirements to the large-scale nuclear power of century XXI have been referred to, with NP structure defined. It is suggested that NP of the second half of this century will include the fast reactors and advanced thermal reactors united in one closed fuel cycle. A solution of the non-proliferation problem could be in the creation of international centers for management of SNF and production of artificial fuel for exported reactors - advanced thermal reactors or fast reactor without breeding blankets.

Two scenarios of the deployment of technological base for the large-scale NP have been considered for Russia during the transitional period until 2030.

In the frame of the first scenario it is suggested that before 2030, the key elements of technological base of the large-scale NP will be created relying on the technologies of fast reactors and closed fuel cycle that have been tested, namely:

- Construction and commissioning of BN-800.
- Development of a advanced sodium cooled fast breeder reactor (BN-K), with improved economic indicators based on the experience with BN-600 and BN-800.

- Construction of a small series of such reactors.

A complex solution of the problem of the VVER SNF accumulation and fuel supply for the small series of BN-K reactors is suggested at the same phase via the establishment of a nuclear fuel center; this center would incorporate entities balanced in terms of plutonium flows:

- Industrial production for radiochemical reprocessing of SNF from VVER (and that of BN-K, if necessary) based on the improved technology of water chemistry.
- Industrial production for manufacturing of MOX-fuel for fast sodium reactors from uranium and separated plutonium.

The non-proliferation regime for VVER export in the transitional stage is ensured by providing integrated services of fuel cycle for the exported reactors – procurement of fresh fuel and SNF return to Russia.

The scenario under consideration also includes the development and pilot demonstration of innovative technologies in the area of fast reactors and a closed fuel cycle (CFC):

- Pilot-level demonstration of lead-bismuth fast reactor LBFR-100 (SVBR-100) technology for regional low- and medium- power NPPs.
- R&D and demonstration of lead cooled fast reactor technology.
- R&D and pilot-level demonstration of advanced CFC technologies including nitride or metallic fuel for fast reactors and dry methods of SNF reprocessing.

If the demonstration of these technologies with the improved safety performance and economic characteristics is a success, they are expected to be able to gradually take their rightful place in a large-scale nuclear power engineering of the 21st century.

Within the framework of the second scenario it is suggested that the decision to embark on the development of large-scale nuclear power engineering technological base should be postponed until the results of pilot demonstration of the innovative technologies have been obtained.

The paper presents the analysis of technological, ecological and financial risks of implementation in Russia for the scenarios being compared.

Joint Processing of SNF MOX FR and SNF RBMK

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Processing of MOX spent nuclear fuel (SNF) of the fast reactors (FR) is a key step for closing of the nuclear fuel cycle. Owing to the high burning out and short cooling time, the SNF FR reprocessing is rather difficult problem due to high concentration of the fissile nuclides and high content of the fission and, primarily first of all, long-lived elements. Moreover, the amount of SNF formed per 1 GWt capacity at the fast reactors is smaller by a factor of 2-3 in comparable with LWR. As a result, the SNF volume for reprocessing will be relatively small, especially at the stage of the FR start-up, which, in turn, will decrease equipment loading and increase the reprocessing cost.

These parameters can be improved using combined treatment of both SNF MOX FR and SNF RBMK at the radiochemical plant for processing of SNF WWER-1000 (or their analog PWR). At mixing of SNF of the thermal reactors (RBMK) with burning out of ~ 20 GWt·day/т U with 7-10 % SNF MOX BN-800 (fast reactor) with burning out up to 90 GWt·day/тU we will obtain the material with the average content of fission elements and sum of the fissile nuclides comparable with that in SNF WWER-1000 with burning out ~ 50 GWt·day/тU. This approach provides simultaneous solution of several problems, which will appear at processing of only SNF MOX FR:

- The mixture of SNF RBMK and SNF MOX FR can be reprocessed at the common radiochemical plant for reprocessing SNF WWER-1000 (or SNF PWR) without deterioration of the nuclear safety owing to high content of the fissile radionuclides in MOX FR and to the increase in the volume of the wastes to be disposed.
- In the course of treatment of the SNF RBMK and SNF MOX FR mixture it is advisable to recover plutonium for its reuse in the fast reactors, whereas uranium with low content of uranium-235 (less than 0.4%) can be disposed.
- The cost of SNF MOX FR would be decreased to the cost of treatment of the thermal reactor SNF owing to the output increase, which would allow to reach the economical parameters of the closed nuclear fuel cycle comparable with that of the thermal reactors even for 3-4 reactors of the BN-1800 (commercial) type.

Fast reactor core design considerations from proliferation resistance aspects

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Sodium-cooled fast reactor core design considerations are made to imrove the proliferation resistance by focusing the plutonium generated in the UO_2 blanket in the frame of the Fast Reactor Cycle Technology Development (FaCT) project which has been in progress by Japanese government and the private sectors.

Three 1500MWe cores are configured: The first one is a conventional type core (FaCT reference core) [1], in which the driver core is surrounded by the usual axial and radial UO_2 blankets; In the second one, the radial blanket is replaced by the neutron reflector or the low-enriched fuel; In the third one, the driver core is surrounded by axial and radial MA-doped UO_2 fuel instead of the UO_2 blanket.

In the present study, the two provisional criteria are adopted in regards to the proliferation resistance of the plutonium generated in the UO_2 blanket, which require the minimum fraction of the particular isotopes within the generated plutonium: 18% for Pu-240 [2] or 6% for Pu-238 [3]. Core neutronics analysis methods are as follows: The adjusted 70-group cross section set ADJ2000R is used based on the JENDL-3.2 nuclear data library; Diffusion calculations are performed in RZ core geometry.

Neutronics analyses are performed to see the effect of the core configuration on isotopic compositions of the plutonium generated in the UO_2 blanket, low-enriched fuel and minor actinide-doped fuel as well as the general core neutronics characteristics including the mass balance in the fast reactor recycling systems.

In the radial blanket-free core, since the integrated-type core fuel pin includes both the driver core fuel and the UO_2 axial blanket, the discharged UO_2 axial blanket could be reprocessed together with the driver core fuel. Therefore, the proliferation resistance criteria can be met for the whole reprocessed plutonium. General core neutronics characteristics are found to be similar to the FaCT reference core with a breeding ratio of 1.1. Moreover, in the core having the radial low-enriched fuel zone, it is shown that the plutonium enrichment of >3% in the radial low-enriched fuel satisfies the criteria (Pu-240 fraction > 18%) (see FIG. 1).

In the core having axial and radial MA-doped UO_2 fuel, it is shown that the Am-241 doping fraction of >2% in the UO_2 fuel satisfies the criteria (Pu-238 fraction > 6%).

In summary, a significant improvement is obtained in the proliferation resistance of the plutonium generated in the UO_2 blanket. Although high proliferation resistance is kept by the conventional security measures, the appropriate design and treatments of the UO_2 blanket as shown in the present core design study might help to streamline the security process.

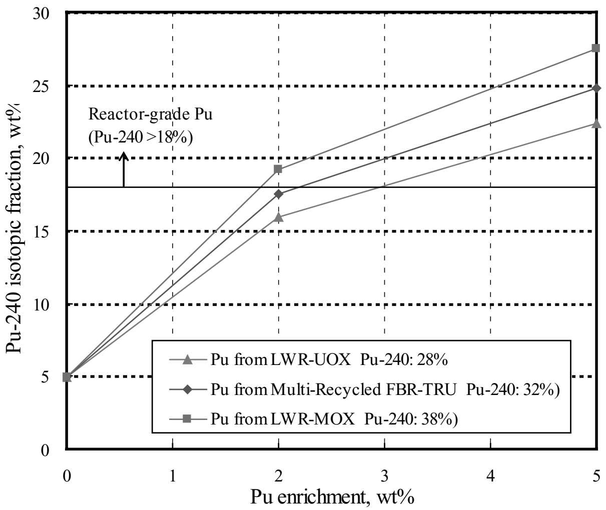

FIG. 1 Pu-240 isotopic fraction in the discharged radial low-enriched fuel as a function of Pu enrichment

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Core Performance and Isotopic Plutonium Vector Analysis in MA doped FaCT FBR

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The objective of the present study is to evaluate the core performance and plutonium vector composition of large power reactor based on the FaCT FBR as reference[1]. Driver core divided into two core region as inner and outer core regions and blanket regions (radial and axial regions) are adopted as converter region for plutonium production. MOX fuel is loaded into driver fuel for different fissile contents of plutonium and depleted uranium is fueled in the blanket region. Different Minor Actinide (MA) doping contents (MA contents : 1-3% and Np-237 3%) are loaded into the blanket region. Several parametric surveys have been performed for optimizing the protected plutonium production [2] as well as the analysis of its effect to the reactor performances. The composition of MA is based on the spent fuel of PWR type of 33 GWd/t irradiated fuel with 3.3% enriched U-235 and 3 years of cooling time [3]. The contenct of MA compositions are mainly Np-237 (56.9%), Am-241(26.3%), Am-243(13.6%), and Cm-244(3%).

Core performance analysis on 3 dimensional X-Y-Z FBR core configuration has been performed which is based on the core optimization calculation of SRAC-CITATION code and nuclear data library of JENDL-3.3. The cycle length of 800 days is used based on the FaCT FBR design by adopting once batch system. This system has been performed to evaluate its effect to the plutonium vector composition and reactor performances.

Plutonium vector analysis is performed based on the plutonium proliferation criterion of Kessler criterion which adopted plutonium vector of Pu-238 production for denatured plutonium (Kessler, 2006)[4] and Pellaud criterion as plutonium characterization criterion of Pu-240 [5].

Doping MA is effective to reduce excess reactivity as well as reducing criticality condition. At the begining of operation until a certain of time, the blanket regions with loading MA becomes the converter regions that produce more plutonium through neutron absorption. In addition, those converted plutonium can be used as additional fissile material to compensate the loss of fissile in the core regions. Additional MA in the blanket region, it may give some effects to criticality coefficient which related to void coolant and doppler coefficients. This study will be performed in the future work regarding safety aspect of criticality pefromance.

In relation to the plutonium vector composition, adopting MA as doping material is also effective to increase the Pu-238 vector composition since some operation time until the end of operation. Pu-240 vector composition is slightly reduced by increasing MA doping and it reaches to a slightly constant value at certain operatin time as shown in **Fig. 1**. Isotopic compositions of Pu-238 and Pu-240 are sensitive to the core operation time. The compositions of Pu-238 decreases and Pu-240 increases with increasing operation time.

Based on the plutonium composition at the end of cycle, to reach Kessler's proposal (Pu-238, 6%-8%) for plutonium denaturing by Pu-238, MA doping rate of about 1.2%-1.5% are required. However, the investigated results show lower Pu-240 vector composition than the Pellaud's proposal (Pu-240, 18%) for fuel grade isotopic plutonium composition as shwon in **Fig. 1.**

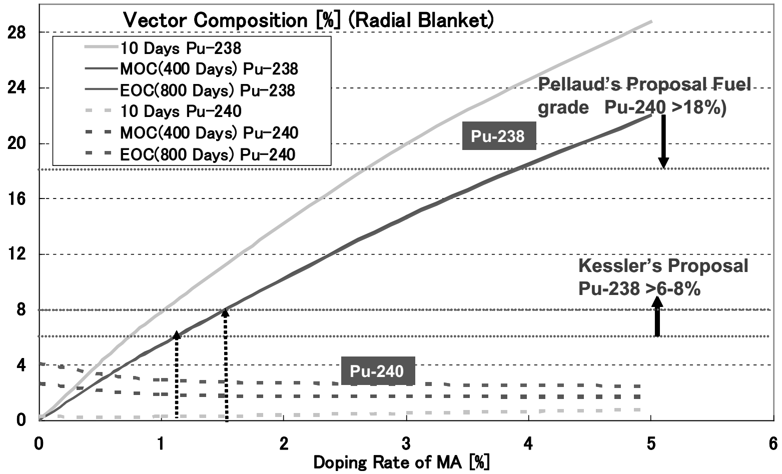

Fig. 1 Isotopic plutonium vector of Pu-238 and Pu-240 as a function of MA doping ratio.

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Protected Plutonium Production (P^3)by Transmutation of Minor Actinides for Peace and Sustainable Prosperity

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"Protected Plutonium Production (P^3)" has been proposed to enhance the proliferation resistance of plutonium by the transmutation of Minor Actinides (MAs). Doping the small amount of MAs such as ^{237}Np or ^{241}Am with large neutron capture cross-section into the uranium fuel to enhance the production of ^{238}Pu or ^{242}Pu, which have high spontaneous fission neutron source or also high decay heat to makes the process of the nuclear weapon manufacture and maintenance technologically difficult, can be effective for improving the isotopic barrier of proliferation resistance of the plutonium in thermal reactors. [1],[2],[3],[4]

Super weapon grade plutonium could be produced in the blanket of a conventional FBR. However, by increasing the ^{238}Pu or ^{242}Pu ratio in the total plutonium by MAs doping into the fresh blanket, the protected plutonium with high proliferation-resistance can be bred. [5]

A new evaluation function, "***attractiveness"***, defined as a ratio of potential of fission yield to the technological difficulties of nuclear explosive device, has been proposed to evaluate the proliferation resistance of Pu based on the nuclear material property for Plutonium Categorization.[6],[7] The new evaluation function of *attractiveness* is applied for assessing the existing plutonium criteria as summarized in the following,

(a) weapon grade plutonium[8]
(b) plutonium with 30% fraction of ^{240}Pu[8]
(c) plutonium with 6% fraction of ^{238}Pu[9]
(d) plutonium exempt from safeguards[10]

Since both proliferation resistant plutonium compositions (b) and (c) give almost the same value of ***attractiveness***, plutonium is categorized by following well accepted terminology, ***weapon grade***, ***usable***, ***practically unusable*** and ***exempt*** as shown in Fig. 1[7]. It is concluded based on the new evaluation function ***"Attractiveness"*** that P^3 mechanism by the transmutation of MA is very effective to improve the proliferation resistance of plutonium.

In the conference, the fundamentals of P^3 mechanism by transmutation of MA, and the comparison of the "***attractiveness"*** of the Pu produced in advanced reactors based on P^3 mechanism and in the conventional reactors will be presented.

ACKNOWLEDGEMENT

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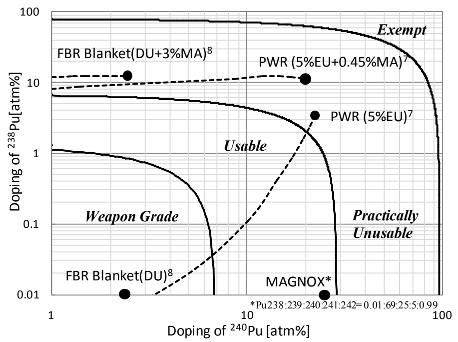

Fig.1 Example of plutonium categorization based on the present methodology [7]

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Proliferation resistance of plutonium based on decay heat

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The proliferation resistance of plutonium can be enhanced by increasing the decay heat of plutonium. For example, it can be enhanced by increasing the fraction of ^{238}Pu, which has large decay heat, produced by the transmutation of Minor Actinides (*P^3: Protected Plutonium Production*) [1, 2]. In the present paper, the proliferation resistance of plutonium is discussed based on the decay heat of plutonium with simple device model [3].

The simple device model consists of a plutonium sphere and 4 spherical shells, a natural uranium reflector, an aluminum layer, a chemical explosives layer, and an outer steel casing [3]. For various plutonium isotopic compositions, the subcritical mass of a plutonium sphere (k_{eff} = 0.98) with a spherical natural uranium reflector was calculated by MCNP4C code coupled with JENDL3.3 nuclear data libraries. The temperature profile of simple device model was calculated with 1-dimensional approach. The proliferation resistance of plutonium was assessed with comparing the inner temperature of chemical explosives layer and its limiting temperature, e.g. melting point or self-explosion point.
The inner temperature of chemical explosives layer increases with increasing of the ^{238}Pu content and exceeds the limiting temperature with the high ^{238}Pu content. Furthermore, the increasing rate of the inner temperature of chemical explosives becomes larger as ^{240}Pu or ^{242}Pu content increases. This is because of the effect of ^{240}Pu or ^{242}Pu on the subcritical mass of a plutonium sphere. The subcritical mass becomes larger and the total heat generation of a plutonium sphere increases. Therefore, it is important to consider the ^{240}Pu or ^{242}Pu content when the proliferation resistance of plutonium is discussed based on the decay heat.

The present methodology was applied to discuss the effect of P^3 by the transmutation of Minor Actinides in Fast Breeder Reactor Blanket [1, 2]. It is comfirmed that in the Fast Breeder Reactor Blanket based on P^3 mechanism, the even plutonium isotope contents increase by the transmutation of MA, and the proliferation resistance of plutonium is strongly enhanced.

The present paper describes the part of essence of the results reffered from the paper [4].

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Enhancement of Physical Protection Measures and Observation on Future JAEA's Measures Reflecting INFCIRC/225/Rev.5 (Draft) under Consideration at IAEA

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Abstract:

The revised Nuclear Reactor Regulation Law is aimed at strengthening of the physical protection taking into account of INFCIRC/225/Rev.4 (Corrected) requirements by focusing on introducing confidentiality of secret information concerning the protection of the specific nuclear fuel material, physical protection inspection, and Design Basis Threat (DBT) following latest recommendations by International Atomic Energy Agency (IAEA), enforced in December, 2005.

Japan Atomic Energy Agency (JAEA) possesses many facilities using Plutonium that fulfill the requirements of Category I and applicable DBT. In order to respond to the revised law, JAEA has conducted (a) development of information management manual for the reinforcement of the physical protection information control, (b) evaluation of the implementation of physical protection measures based on the DBT presented by the competent authority, and (c) enhancement of physical protection measures based on the evaluation result. Moreover, the inspection of physical protection measures and the verification of compliance with the physical protection regulation by competent authority were conducted for each JAEA facility, and the additional corresponding measures to reflect an inspection result were conducted.

In this paper, we brief the requirements for the facility using Plutonium, an overview of the instruction by competent authority, an overview of corresponding measures by JAEA. Furthermore, we describe an image of a future physical protection system of JAEA's plutonium facilities influenced by the enhancement of physical protection measures following the recommendation document for the physical protection of nuclear material and nuclear facilities (INFCIRC/225/Rev.5 (Draft)) currently under consideration by IAEA.

POSTERS OF SESSION 6:

Fast reactor analysis: basic data, experiments and advanced simulation

Comparisons of cross section sensitivity coefficients in a small fast reactor

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Various benchmarking cores have been analyzed with the 4S nuclear design methodology [(1),(2)]. Through those analyses, data base for sensitivity and uncertainty (S&U) analysis for the 4S code designs have been accumulated. For example, C-to-E values have been obtained for the important characteristics, such as criticality, reflector worth, sodium void worth, reaction rates and absorber material worth.

Series of analyses for physics mockup cores, which were leakage-dominant, showed that the Sn-transport calculations are much superior to the conventional diffusion method in maintaining bias factors reasonably small and consistent among various characteristics from the view point of prediction accuracy.

Cross section sensitivity coefficients for major nuclear characteristics are useful measures to quantify similarity between the target core and the physics mockup/benchmarking cores in the neutronic-view points. Evaluations for the consistencies among C/E-values obtained under the different experimental situations are able to provide indications to proper uncertainties associated with extrapolation of biases obtained from the physics-mockup/benchmark core analyses.

Currently, diffusion-based generalized perturbation code SAGEP, which is originally developed at Osaka-university, has been used in fast reactor field to calculate cross section sensitivity coefficients for various reactivity worth and reaction rates. But, as mentioned above, utilization of transport-based generalized perturbation is much desirable to keep cross section sensitivity coefficients consistent and to avoid unnecessary method errors in application of S&U analyses for the designs. This paper compares the generalized forward/adjoint flux distributions and sensitivity coefficients for several leakage-dominant reactivity characteristics, such as sodium void worth, obtained by the Sn-base generalized perturbation code and conventional diffusion-base one. Applicability of those S&U analyses is also discussed for the 4S core model.

Figures 1 and 2 exemplify comparisons among values in the sensitivity coefficients of uranium-238 cross sections for fission reaction rate near core center and sodium void worth for the whole core region for a modeled 4S-like core (ternary alloy

metallic fuel version) configuration in RZ geometry. The calculations were done with 7-group S8/diffusion theory in individual manners. Calculated characteristics are listed below for examples.

Characteristics/ Methods	S8-transport (T)	Diffusion (D)	Ratio (T/D)
Reaction Rate near core center U235(n,f)/Pu239(n,f)	0.894	0.890	1.004
Sodium void worth in core (%delta-k/kk')	-0.614	-1.122	0.55

Sensitivity coefficients are defined as the following conventional way.
Sensitivity coefficient= (delta-R/R)/(delta-x/x), where R expresses a integral characteristics and x corresponds to a cross section value.

Through those comparisons, it revealed that the S8-based cross section sensitivity coefficients were generally larger as much as doubled in the absolute basis than those calculated by diffusion-based calculations in a modeled 4S core for leakage dominant sodium void worth. However, group-wise sensitivity components showed more complicated manners obtained through the sensitivity profiles of several U-238 cross sections shown in Fig.1. On the other hand, sensitivity coefficients for the reaction rate ratio showed quite close similarity between the calculation methods as shown in Fig2.

Those results support that application of the new Sn-transport-based sensitivity analysis tools can be effective to determine appropriate uncertainties consistent with applying the C-to-E data base accumulated by the 4S nuclear design methodology.

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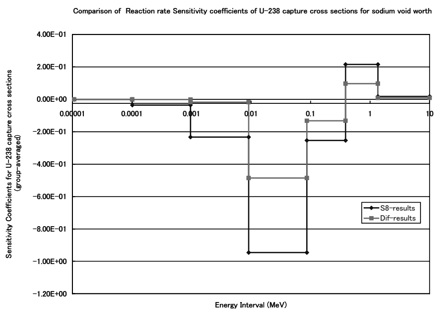
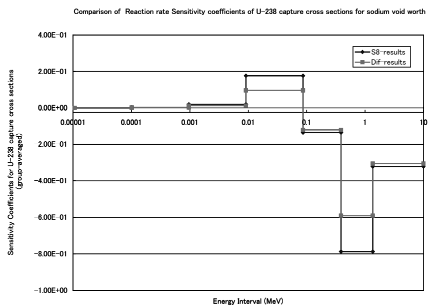
Sodium void worth vs. U238 cross sections

(1) U238 capture cross sections



(2) U238 transport cross sections



(3) U238 scattering cross sections

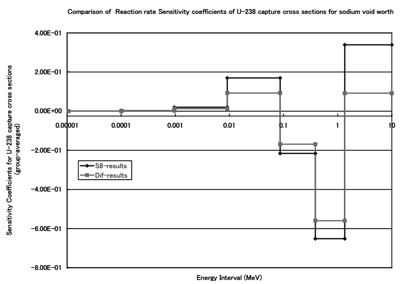
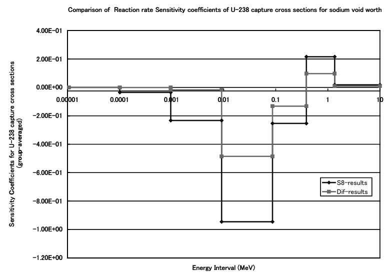


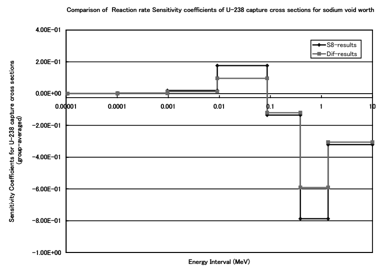
Fig.1 Comparisons of sensitivity coefficients to core sodium void worth for several U238 cross sections

Reaction Rate Ratio U235-fission/Pu239-fission near core center vs. U238 cross sections

(1) U238 capture cross sections



(2) U238 transport cross sections



(3) U238 scattering cross sections

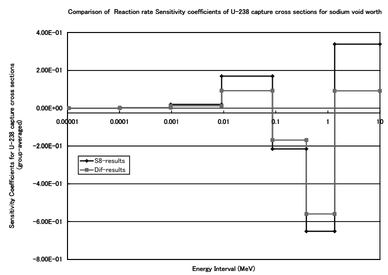


Fig.2 Comparisons of sensitivity coefficients to reaction rate ratio (f5/F9) for several U238 cross sections

Development of Adjoint Method of Characteristics Code for Fast Reactor

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In order to obtain reference with the consideration of the heterogeneous geometry for the fast reactor core calculation, a new MOC (Method of Characteristics) code which can calculate both forward and adjoint fluxes has been developed in this study.

For the safety analysis of the fast reactor core, we need many data, such as reactivity coefficient, effective neutron ratio, sensitivity coefficient etc. To evaluate these data, perturbation calculation is useful. To evaluate the imperceptible variation accurately by perturbation caluculation, the evaluation of adjoint flux as well as forward flux is necessary in the same system.

Integro-differential transport equation of the forward flux can be written as equation (1), and that of adjoint flux can be written as equation (2). Equation (3), which is generally used for MOC, can be obtained from equation (1). In this study, equation (4) is obtained from equation (2) in the same way. By the implementation of equation (4) to the existing MOC code, the adjoint flux can be obtained in MOC calculation [1][2].

$$\frac{d\Phi^g(\mathbf{r}+s\mathbf{\Omega}_l,\mathbf{\Omega}_l)}{ds}+\Sigma_t^g(\mathbf{r}+s\mathbf{\Omega}_l)\Phi^g(\mathbf{r}+s\mathbf{\Omega}_l,\mathbf{\Omega}_l)=q^g(\mathbf{r}+s\mathbf{\Omega}_l) \quad (1)$$

$$\frac{d\Phi^{*g}(\mathbf{r}-s\mathbf{\Omega}_l,\mathbf{\Omega}_l)}{ds}+\Sigma_t^g(\mathbf{r}-s\mathbf{\Omega}_l)\Phi^{*g}(\mathbf{r}-s\mathbf{\Omega}_l,\mathbf{\Omega}_l)=q^{*g}(\mathbf{r}-s\mathbf{\Omega}_l) \quad (2)$$

$$\Phi^g(\mathbf{r}+t\mathbf{\Omega}_l,\mathbf{\Omega}_l)=\exp\left[-\tau^g(\mathbf{r},\mathbf{r}+t\mathbf{\Omega}_l)\right]\times\left(\Phi^g(\mathbf{r},\mathbf{\Omega}_l)+\int_0^t ds\, q^g(\mathbf{r}+s\mathbf{\Omega}_l)\exp\left[\tau^g(\mathbf{r},\mathbf{r}+s\mathbf{\Omega}_l)\right]\right) \quad (3)$$

$$\Phi^{*g}(\mathbf{r}-t\mathbf{\Omega}_l,\mathbf{\Omega}_l)=\exp\left[-\tau^g(\mathbf{r}-t\mathbf{\Omega}_l,\mathbf{r})\right]\times\left(\Phi^{*g}(\mathbf{r},\mathbf{\Omega}_l)+\int_0^t dsq^{*g}(\mathbf{r}-s\mathbf{\Omega}_l)\exp\left[\tau^g(\mathbf{r}-s\mathbf{\Omega}_l,\mathbf{r})\right]\right) \quad (4)$$

In order to validate the above method, calculation was performed for the cell of fast reactor with 7 group cross section. MOC calculation conditions are the same between forward calculation and adjoint flux calcutation.(track spacing is between 0.010 and 0.009 cm, azimuthal angle division is 64 per 360 degrees, polar angle division is 3 per 90 degrees)

The results are shown in table 1. The eigenvalue of adjoint flux calculation was compared with that of forward flux calculation. The difference between the two is 1.4E-6. The CPU time is almost the same. This result shows the validity of the above method.

The result of whole core calculation of fast reactor will be described In the full paper.

	Eigenvalue	cputime(sec)
Forward	1.500510	101.5
Adjoint	1.500524	110.6

Table1.Result of both adjoint flux calculation and forward flux calculation

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Performance and reactivity coefficient analysis of large TRU burning fast reactors

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In pursuit of size of economy, a core design study on higher powered TRU burning fast reactor has been performed. The selected core power levels subject to the core design development were 1500, 3000, and 4500 MWt. During the core design development, a set of the same design ground rules were applied simultaneously regardless of the core power levels, which enables comparisons in a consistent basis.

In addition to the different core power levels, a variation in the TRU conversion ratio was also exercized at each power level. As a result, there became available a set of nine different core performance parameters and reactivity coefficients.

The trending analysis for the developed cores having similar TRU conversion ratios showed that the core performance at a higher power level is as good as those of a lower powered one and a TRU transmutation rate is proportional to the core power level. As for the TRU loading, 1500 MWt is shown to be the lowest power level above which the relative neutron leakage compared with the neutron production becomes saturated. This means that a core power lower than 1500 MWt has a TRU loading penalty due to more neutron leakage.

As the power level increases at a similar TRU conversion ratio, notable trends are as follows: a similar Doppler coefficient, a less negative axial expansion coefficient and a more negative radial expansion coefficient. Therefore, the power level dependent total reactivity change is negligible once the sodium void worth and the TRU enrichment are kept constant.

Upon the conversion ratio change at each power level, a core with a reduced conversion ratio has a less negative Doppler coefficient, a more negative axial expansion coefficient, a more negative radial radial expansion coefficient and a less positive sodium density coefficent.

As an effort to synthesize all the reactivity coefficients dependent on the core power level and TRU conversion ratio, a set of simple equations describing reactivity coefficient behavior were derived, based on the six factor formula. The necessary one group cross sections are obtained from the base case core neutronics calculation for each power level and each TRU conversion ratio. Since these equations inherently break down each reactivity coefficient into their reactivity components, a further insight into the reactivity coefficients could be obtained.

In conclusion, it is feasible to design a TRU burner core to accommodate a wide range of conversion ratios. The TRU consumption rate can be made to be proportional to the core power level without any significant adverse effect in the core performance parameters and reactivity coefficients at higher power levels. Similar reactivity coefficients observed even in different power levels are explained by an application of simple formulae.

Innovative Fast Reactors: impact of fuel composition on reactivity coefficients

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Innovative fast reactors will have to comply with the major objectives as stated, e.g. within the Generation-IV initiative, namely sustainability and waste minimization, keeping their safety characteristics at least as good as Generation-III evolutionary reactors.
As a consequence, a specific feature of most innovative fuels envisaged for future fast reactors is the potential high content of Minor Actinides (MA: Np, Am, Cm etc). This is the case whatever the mission assigned to a fast reactor: to be a breeder, an "isogenerator" or a TRU burner, i.e. fast reactors with very different conversion ratios (CRs), and whatever the strategy adopted for the spent fuel processing [1]. The MA content can vary in a wide range, according to the core design characteristics and the fuel type. In practice, the MA content can vary from a few percent (e.g. <10%) in a core with a CR $\geq$ 1 and with the homogeneous recycle of not-separated TRU, aiming to a MA inventory stabilisation in the fuel cycle, to ~20-30% in the case of the fuel of a burner fast reactor (CR<1) with a Pu/MA ratio ~1. It is well known that the introduction of such significant amounts of MA can have an impact on the core reactivity coefficients (e.g. coolant void and Doppler coefficients) and, in the case of cores with a very low CR value, on the delayed neutron fraction. Several recent studies have been devoted to quantify these effects (see e.g. [2]). However, it is important to use a flexible and powerful physics tool, in order to understand the physics features of the calculated effects, to predict trends and potential limitations during preliminary design phases. This flexible and powerful physics tool is provided by the Equivalent Perturbation Theory (EGPT, [3]), as implemented in the ERANOS code system [4]. In particular, the effect of the variation of isotope j associated to its cross section type k, is given by:

$$S^{\Delta\rho}_{j,k} = \frac{\partial(\Delta\rho)}{\partial\Sigma_{j,k}}\cdot\frac{\Sigma_{j,k}}{\Delta\rho} = \left\{\frac{1}{I_f^p}\left\langle \underline{\Phi}^*_p, \Sigma_{j,k}\underline{\Phi}_p\right\rangle - \frac{1}{I_f}\left\langle \underline{\Phi}^*, \Sigma_{j,k}\underline{\Phi}\right\rangle\right\}$$

where $\underline{\Phi}^*$, $\underline{\Phi}$, $\underline{\Phi}^*_p$, $\underline{\Phi}_p$ are the real and adjoint fluxes of the reference and perturbed system respectively and I_f and I_f^p are normalisation factors. This method allows then analysing by isotope, reaction type and energy region the effect of introducing MA in the core on reactivity coefficients as coolant void and Doppler reactivity coefficients.
The specific cross section characteristics of the different MA isotopes, and in particular their η ($=\nu\sigma_f/\sigma_a$) energy behaviour, will have an impact on the neutron importance energy shape and, consequently, there will be a potentially significant impact on reactivity coefficients mainly on those reactivity coefficients very sensitive to the neutron importance energy shape, like the void reactivity coefficient in liquid metal cooled FRs.
A typical example is given in Table I. The values are sensitivity coefficients, integrated over energy as in equation (1), relative to the Na-void reactivity coefficient of a burner fast reactor, with a MA/Pu ratio equal approximately ~0.1 and CR~0.25 [5]. The reference void reactivity coefficient is~1.0% Δk/k in diffusion theory and 1.7%Δk/k in transport theory. This is already a noticeable effect that indicates the need to avoid calculation approximations that can lead to not-conservative estimations of important reactivity effects (in this case there is an underestimation by a factor of ~2 of a positive reactivity effect). As far as the sensitivity coefficients of Table I, the addition of "fertile" isotopes results in general in an increase of the positive void coefficient value and the addition of fissile isotopes in a decrease of the reactivity effect. A detailed sensitivity analysis shows that the global effect of each isotope (here indicated under "SUM") can be interpreted as the combination of competing effects:

- The increase of capture at high (e.g. E>100 keV) energy results in a flattening of the high energy slope of the adjoint flux (responsible for the positive reactivity Na-void effect). On the contrary, the capture increase at low (e.g. E<100 keV), results in a flattening of the low energy slope of the adjoint flux (responsible for a negative contribution to the same coefficient. The sum of the two effects results in a decrease of the Na reactivity coefficient, due to the predominance of low energy effects.

- The opposite effects are observed for the high energy and, respectively the increase at high energy of the fission cross section will result in an increase of the energy gradient of the adjoint flux at high energy (and then of the associated positive reactivity effect), that is counterbalanced, in the case of fissile isotopes, by a corresponding effect of opposite sign at low energy. This means that "fertile" isotopes with threshold fission cross sections will induce only the increase of the positive reactivity effect at high energy, while the predominance of low energy effects for fissile isotopes give a global decrease of the positive reactivity.
- Apart from actinide content and associated data uncertainties, a very significant effect is due to the elastic cross section of Fe-56. Any variation, both due to composition variation or due to data uncertainties, has a very significant impact on the Na void reactivity. For example, a 10% increase of the Fe-56 content, will give rise to a 13% increase of the positive Na void effect.
- Finally, the increase of the inelastic component of each isotope will result systematically in the reduction of the high energy flux and then in a reduced weight of the high energy positive contribution to the reactivity coefficient value. Fe-56, Na-23 and U-238 are the main contributors to this effect.

The full paper will give a detailed analysis of these phenomena for both Na-void and Doppler reactivity coefficients in a variety of different FRs with different core characteristics.
It will also be shown how to apply this type of analysis for optimizing core design features.

Table I: Sensitivity coefficients

ISOTOPE	CAPTURE	ELASTIC	INELASTIC	NU	FISSION	SUM
Am-241	16,76	0,11	-1,39	8,92	5,75	30,14
Am-242m	2,77	0,06	-0,95	-44,01	-33,50	-75,63
Am-243	13,27	0,08	-1,46	5,79	3,71	21,39
Cm-242	0,30	0,00	-0,06	0,81	0,54	1,60
Cm-244	4,58	0,04	-0,49	10,26	6,85	21,25
Cm-245	0,72	0,02	-0,17	-10,54	-8,28	-18,25
Cm-246	0,42	0,01	-0,04	0,95	0,65	2,00
Fe-56	17,13	143,75	-30,53	0,00	0,00	130,40
Na-23	10,05	-1,94	63,89	0,00	0,00	71,99
Np-237	13,39	0,07	-0,93	8,66	5,07	26,26
Pu-238	6,86	0,24	-0,42	-0,27	-2,09	4,31
Pu-239	40,03	0,57	-3,18	-125,68	-95,80	-184,05
Pu-240	37,40	0,78	-4,56	82,76	50,75	167,14
Pu-241	6,06	0,23	-0,78	-73,82	-54,12	-122,43
Pu-242	11,07	0,28	-1,36	24,90	15,53	50,42
U-238	38,33	1,93	-15,12	11,44	5,63	42,26

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Reactor Physics and Safety Aspects of Metal Fuelled FBR

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For faster growth of nuclear power in India, it is essential to shift to the use of metal-fuels in fast breeder reactors (FBR), which gives a higher breeding ratio (BR) and lower fuel doubling time (DT). Two-dimensional diffusion calculations had been performed to investigate the physics parameters of metal (U–Pu–Zr) fuelled FBR cores as a function of reactor parameters like reactor power, and zirconium content in the fuel. A binary metal case with Zr liner is a candidate option as it gives higher BR. However metal cores have higher positive Na void reactivity effect in comparison to MOX core design. Methods to reduce the sodium void reactivity, had been considered which tends to reduce BR. Still, the higher Na void reactivity effect is perceived to be a safety concern under loss of flow in a pool type FBR. Unprotected loss of flow accident (ULOFA) for 500 MWe U-Pu-10%Zr metal fueled sodium cooled reactors is analysed and compared with that of the 500 MWe (U-Pu) MOX Prototype Fast Breeder Reactor (under construction in Kalpakkam, India). A flow halving time of 8*s* for primary pumps is considered for the two cases. It is well known that under ULOFA, the oxide fueled reactor shows early onset of sodium boiling and fuel slumping leading to near prompt criticality and entry into the disassembly phase in less than 100 s. However, ULOFA is found to be benign for the metal fueled reactor as the sodium coolant boiling is delayed beyond 1500 s. This behaviour is due to higher negative fuel expansion feedbacks and lower fuel temperatures (close to that of coolant). The reactor becomes subcritical and the principal heat source after 1200 s is the decay power. An accurate modeling of decay power is important when dealing with the transient at long times. If heat removal systems are designed to reliably remove decay heat, after ULOFA, the metal fuelled reactor reaches a safe subcritical state at a elevated Na temperature. The summary results are shown in Table.1. The paper brings out that safety of high BR metal core is not compromised under ULOFA despite its higher Na void reactivity effect compared to oxide core.

Table 1: Summary Results of ULOFA for PFBR and FBR-M

	PFBR	FBR-M (Zr 10%)
Reactor power at 80 s	761 MWt	224 MWt
Net reactivity at 80 s	0.153 $	- 0.271 $
Net voiding fraction at 80 s	0.331	0.0
Na voiding initiation	26 s	1600 s
Fuel slumping initiation	78 s	No fuel slumping
Reactor power at 1800 s	--	17.6 MWt +
Net reactivity at 1800 s	--	- 0.587 $
Net voiding fraction at 1800 s	--	0.036

+ decay power

Compatibility of Iodides with Stainless Steels of Cladding for LLFP Transmutation

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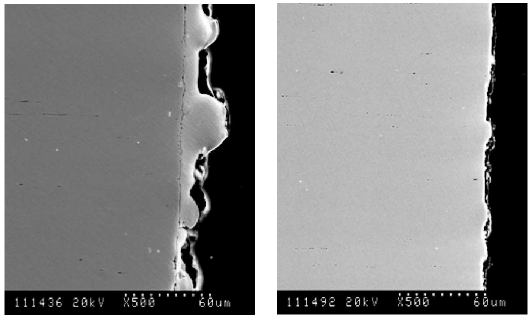
Compatibility between cladding material of a fast reactor and iodides of LLFP transmutation targets was investigated because it was the most important issue for transmutation of LLFP in fast reactors. Several stainless steels such as SUS316, PNC-FMS and 9Cr-ODS, were kept at 873K for 3000h maximum in contact with iodides such as BaI_2, CuI, MgI_2, YI_3 and RbI which were selected as the candidate compounds for transmutation target [1, 2].

Observation of the appearances and the surface of stainless steels after the experiment indicated the stainless steels in contact with BaI_2 had good compatibilities. On the other hand, it was clarified that the stainless steels in contact with CuI and YI_3 had such problems as chemical interactions and pit corrosion as shown in Fig.1. The stainless steels in contact with MgI_2 also were observed pit corrosions after the experiment for 3000h. Though traces of degradation had not be observed at the region of stainless steels in contact with RbI directly, the region in contact with gaseous phase was corroded hardly. The same behavior was observed in the stainless steels in contact with YI_3 too. It is necessary to pay attentions to gaseous - solid phase interaction as same as solid-solid phase interaction to evaluate the compatibilities of stainless steels with iodides.

Furthermore, oxidation layers appeared in the surface of stainless steels in contact with BaI_2 powder. The reason of oxidation layers was found out to be the contained H_2O in the BaI_2 powder by thermal analysis. The forms of oxidation layer were different between SUS316 and 9Cr-ODS by their crystal structures. The oxidation layer of 9Cr-ODS having BCC structure was larger than that of SUS316 having FCC structure. The oxidation layer had bigger hardness than the matrix of stainless steels. Then sensitivity of cracking at oxidation layer is predicted to be higher than that of the matrix. Oxidation layer is not good for stability of cladding material. An impurity was thought to affect severely degradation of stainless steels even if stainless steel had good compatibility with BaI_2. It is important to control the impurities during the fabrication of targets for iodine transmutation.



(a) CuI (b) YI_3

Fig.1 The surfaces of SUS316 contacted with iodides at 873K for 1000h

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Experimental and analytical study of failed fuel detection and location system in JSFR

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A conceptual design study of an advanced large-sized (1500MWe class) sodium-cooled fast reactor (named JSFR) is in progress in the "Fast Reactor Cycle Technology Development (FaCT)" project in Japan. JSFR adopts a Selector-Valve method for the failed fuel detection and location (FFDL) system. A Selector-Valve FFDL system identifies the failed fuel subassembly by sampling outlet sodium of each fuel subassembly. One of the JSFR design features is an upper internal structure (UIS) with a radial slit, in which an arm of fuel handling machine can move and access the fuel assemblies under the UIS. This UIS can simplify a fuel handling system, and can downscale the reactor vessel diameter. Thus, JSFR cannot place the sampling ports right above the fuel subassemblies located under the slit. So, it is necessary that the sampling ports will be set around the UIS slit so as to catch the sodium flow from the fuel subassemblies located under the slit. To demonstrate sampling performance of under-slit subassemblies, water experiments and numerical analyses have been conducted.

Before water experiments, a numerical analysis was carried out to decide sampling locations in the water experiment. This study employed the commercial computational fluid dynamics code, STAR-CD. From the analyses results, the high concentration flow from the fuel assemblies reaches at the edge of the baffle plate (BP) and the fuel handling machine (FHM) plug. Therefore, we decided to locate the sampling ports at the edge of BP, on the side of FHM plug and on the undersurface of FHM plug in water experiment.

The water experiment was carried out to evaluate Selector-Valve FFDL performance around the slit UIS. The test apparatus is a 1/5-scale-model of UIS slit. In this experiment, salt water is used as dummy fission products. The targeted concentration at the sampling ports (relative concentration normalized the concentration at the outlet of fuel assembly) is selected 2% from requirements of the signal-to-noise ratio and the measurement time in the operation of JSFR. Water experiments were conducted for all fuel subassemblies located under the UIS slit. From the experimental results, the high concentration over the targeted value was detected at the sampling port in all experiment cases. And, identifications of a under-slit failed fuel subassembly is thought to be capable by comparing concentrations at sampling positions. In addition, the numerical analysis showed that it could reproduce the overall trend of concentration distribution of experimental results.

In a future study, we would evaluate the detection performance of failed fuel detection and location system in JSFR conditions based on these results.

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Thermohydraulics of sodium-cooled-reactors

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Thermohydraulic analysis of reactor facilities (RF) is the most important component of a complex of the interconnected problems on substantiation reactor parameters relative to reactor physics, thermomechanics, durability theory etc.

At the IPPE the complex of experimental liquid metal experimental facilities was created, as well as methods for modeling, measurement techniques, instrumentations, methods of numerical calculation and codes. Thermohydraulic investigations for RF with sodium cooling carried out in Russia for more than 50-year period, as a rule, are complexes of experimental and numerical researches.

During this time wide data on velocity and temperature distributions, hydraulic resistance and heat transfer, initial thermal section in the channel of complex form and rod bundle, interchannel mixing for different variants of fuel element spacing were obtained. These data have been under lied to both engineering techniques of thermohydraulic calculations of fuel subassemblies, to development of subchannel codes and model of «porous media». The experimental data obtained with the use of fuel subassembly (SA) models on velocity and temperature distributions for different geometry of peripheral area of SA, for the deformed bundles of fuel elements, variable power distribution in cross-section section and along FSA, heat exchange through cover FSA etc., and also results of numerical investigations with code MIF were a reliable substantiation of temperature regimes of fast reactor core.

Results of experiments on the model of tube bundle of intermediate heat exchanger (IHX) "sodium-sodium" of BN-600 with complex character of coolant circulation in the inter-tube space, where fields of velocity and local heat transfer taking into account an influence of the width of entrance window obtained in wide range of parameters were measured, became a basis for development and verification of 3D codes PROTVA and UGRA for thermohydraulic analysis of heat exchangers. The analysis of heat exchanger efficiency taking into account non-uniform distribution of the coolant on tubes and within inter-tube space was carried out, and calculations with reference to IHX of BN-600 have shown that it is possible to apply its design for BN-800.

With use of the data on heat transfer in steam generator channels obtained in experiments design procedures were developed as for preliminary and design calculations, as well as for substantiation of thermohydraulic characteristics of steam generators "sodium-water". Relative numerical codes were developed, with considerable attention given to estimation of heat transfer and durability conditions in the areas of depositions and plugging of generator channels pipes.

Last years semiempirical methods of calculation of coolant flow in such complex systems as channels with large roughness in the form of cross fins have been developed, mutual influence of consistently located local hydroresistance have been investigated, the results of experiments on collector hydrodynamics have been analyzed and generalized. But general attention has been given to thermohydraulic researches of non-nominal, transient and accident

regimes, namely to modeling of sodium natural circulation in reactor tank in case of decay heat removal for different systems of emergency shut-down cooling, to research of influence of blockages in cross section of SA, development of sodium boiling in case of under design accident in fast reactors. Continuation of these investigations with reference to safety of fast reactors of new generation is rather actual.

In light of existence of cyclic thermal stresses in structure elements the following researches are of large practical significance: temperature pulsations in difficult flowing parts of fast reactors with integrated configuration; process of temperature pulsation transfer to solid walls, creation of numerical models and codes.

The important issue defining working capacity of various elements of steam generator "sodium-water" is the coordination of thermomechanical deformations and decrease in thermomechanical stresses due to temperature distribution. It should be to have analyzed sections of boiling and concentration of impurity in flowing water, as well as washing off time in regulations a water-chemical regime due to existence of specific deposition belt in the evaporator.

Necessity of development of modern methods and codes for calculation of local turbulent characteristics of complex flows of liquid metal coolants in channels and great volumes (mixing chambers) taking into account large-scale vortical flows, an influence of stratification of the coolant has reached crisis point.

Calculation of the thermohydraulic parameters of a fast neutron reactor with account of inter-fuel-assembly space influence

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Calculation accuracy enhancement for validating thermohydraulic parameters of a fast neutron reactor is an important applied issue because finally the aspect could impact on reliability and performance of the reactor type. Besides, refining the calculation runs could provide more plausible design conservatism level. Common-type fast neutron reactor design is featured by an inter-fuel-assembly space (IFAS) or inter-fuel-assembly-gap available within the reactor core. Actually, it means that two simultaneous in-core coolant flow are extant: main one is flowing inside of a fuel assembly; second one is penetrating through inter-fuel-assembly-gap (inter-fuel-assembly space) and contributing several percents of the total coolant flow of the reactor. During their propagation, both coolant flows are under mutual steady heat exchange which is carried out across the fuel assembly shroud walls. This interaction results in some impact on fuel pin temperature; besides, mixing of main coolant flow from the fuel assenbly heads with coolant flow from inter-fuel-assembly space could rsult in several influence on coolant temperature distribution within the upper reactor core chamber.

As for methodological point of the view, the IFAS effect has been calculated after numerical IPPE procedures (namely 3-D single-phase computational code GRIF and velocity field calculation code SACTA for in-core analysis and more detailed review). Major parameters are 3-D simulated.

It was stated that IFAS flow patterns have several nonuniformity and 3-D nature; therefore some impact on fuel assembly temperature is occurring.

The processes should be accounted and analyzed.

Polynomial Regression with Derivative Information in Nuclear Reactor Uncertainty Quantification

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We introduce a novel technique of uncertainty quantification using polynomial regression with derivative information (PRD). Our applied task is to assess the effect of the uncertainty in values of physical parameters on the performance of the model of a sodium-cooled fast reactor. The inputs and outputs of the model can be chosen arbitrarily, the result is an estimate of a range of values, statistical distribution, and sensitivities of the output. Achieving a more precise assessment has immediate significance in the area of nuclear reactor design and control, where greater engineering precision results in increased safety and preservation of resources.

We use the following uncertainty structure, consistent with the geometry of the model and the available information on the input data. The uncertainty in the output is attributed to the experimental errors in material properties. Each considered error is then expressed as a polynomial expansion in terms of the model state; the coefficients of the expansion are stochastic variables, used as the uncertainty quantifiers. The goal is to construct the approximation of the output as a function of uncertainty quantifiers.

Traditionally, the influence of the uncertainty in the inputs on the outputs is described either by linear approximations using first-order derivative information and disregarding the nonlinear effects; or by sampling that requires many runs of the model (Monte Carlo is the most basic approach, see [1] for a primer of sampling methods). We estimate the uncertainty of the target functional due to input data uncertainties using a hybrid method, as an expansion in the multivariable polynomial basis, constructed by collocation based on a sample of input values. An important part of the approach is the use of derivative information: we augment the collocation at each sample point by also fitting the derivative of the polynomial expression to the derivative of the target functional computed, effectively reducing the number of required samples by the factor equal to the number of variables. The computational expense may be further reduced by using a basis with high-order polynomials only in the most important quantifiers. Their importance is assessed using a combination of derivative and statistical information.

The resulting approximation efficiently reproduces both the distribution and the major global sensitivity effects of the target functional [2]. The uncertainty range estimation is an order of magnitude better in comparison with the linear approximation (linear sensitivity) approach [3]. The computational cost for obtaining the estimation is reduced with the use of derivative information, resulting in computational advantage in comparison with Monte-Carlo sampling.

The performance of the proposed method was tested using a simplified 3-dimensional model of a fuel assembly. We have chosen the maximal fuel centerline temperature as the output of the model. The uncertainty of this output was calculated by taking into account the uncertainties in heat conductivities and neutron cross sections, described by 38 quantifiers. Table 1 presents a comparison of a typical performance of PRD with that of the linear sensitivity methods and the random sampling (with a sufficiently large sample size).

The following metrics are used in the table: sample size, or the number of evaluations of the model required to construct an approximation; range and standard deviation of the output function values; marginal significance, or the percentage of the total variance in the output attributed to variance in individual uncertainty quantifiers. The PRD estimates all the metrics more accurately than the linear sensitivity method, and more efficiently than Monte Carlo method since significantly fewer sample points are needed. Note that while finding complete gradients requires additional computation, the overhead is bounded, in practice, by at most 100%; [4] suggests a theoretical bound of 500%. The overall cost of PRD depends only on the number of variables of high importance, and the degree of nonlinearity associated with them.

Additional questions addressed in our work include application of the method to time-dependent models, and an improved selection of the collocation sample points.

	Random sampling	Linear approximation	PRD
Sample size	100	1	10
Range of output (K)	2237 – 2460	2227 – 2450	2237 – 2459
Standard deviation of output (K)	59.05	59.12	58.96
Significance of individual components (%) Metric shown for 2nd , 0th and 1st order terms in the expansion of fuel thermal conductivity uncertainty	60.97 38.76 0.05	68.44 31.47 0.09	58.93 40.90 0.17

TABLE 1. Performance of the polynomial approximating model for maximal fuel centerline temperature; marginal significance of the most important components.

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Development of Multigroup Cross Section Generation Code MC^2-3 for Fast Reactor Analysis

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Under the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program of U.S. DOE, an integrated, advanced neutronics code system that allows the high fidelity description of nuclear reactors and simplifies the multi-step design process is being developed. As part of this effort, an advanced multigroup cross section generation code named MC^2-3 is being developed for fast reactor applications by improving the resonance self-shielding and spectrum calculation methods of MC^2-2 [1] and integrating these improved methods with the two-dimensional method of characteristics solver of the high-fidelity transport code UNIC [2].

In order to enhance accuracy and efficiency, various improvements have been made on the ETOE-2/MC^2-2 code system that has been successfully used for decades in generating multigroup cross sections for fast reactor analysis. First of all, in order to eliminate the limitations of the current generalized resonance integral method for resolved resonance self-shielding, a new approach based on the numerical integration of pointwise resonance cross sections with the narrow resonance approximation has been introduced. The continuous slowing-down method used for the spectrum calculation in the resolved resonance energy range has been replaced by the consistent P_1 method with extended transport approximation up to P_9 in order to minimize the approximations involved in light elements (in particular, hydrogen) treatment and to simplify the program structure. The consistent P_1 equations are sloved in ultra-fine (~2,100) group level; the hyper-fine (>50,000) group spectrum calculation in the resolved resonance range can optionally be invoked to enhance the resolved resonance self-shielding accuacy. A new capability of handling anisotropic inelastic scattering matrices has also been added. These improved methods and one-dimensional transport calculation capability have been integrated into a new cell code MC^2-3 with a modern programming structure.

The new cell code has been incorporated into the neutron transport code UNIC to generate the multigroup cross sections consistently with the material and temperature distributions used in transport calculations. The cell calculations provide the multigroup cross sections for two-dimensional lattice or whole-core transport calculations, depending on the desirable level of approximation appropriate for the computational resources and analysis goals of the user.

Initial verification tests of the MC^2-3 code and its ENDF/B-VII.0 libraries have been performed using various fast critical experiments including eight LANL critical assemblies, ZPR-6/6A ZPR-6/7, ZPPR-15, and ZPPR-21 critical experiments. The resulting effective multiplication factors have shown very good agreements with MCNP5 and VIM Monte Carlo solutions: within 150 pcm for the LANL critical experiments; within 40 pcm for ZPR-6 assembly 6A and 7; within 250 pcm for six configurations of ZPPR-21 and three configurations of ZPPR-15A.

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Nuclear Calculation Methodology and Development of 3-D Transport Nuclear Design Code

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This paper presents nuclear calculation methodology and the experience in Mitsubishi Heavy Industries Ltd. (MHI) and the development of 3-D nuclear design code. MHI, merging Mitsubishi Atomic Power Industries Inc. in 1995, has worked for the core design of the irradiation cores (Mk-II, Mk-III) of JOYO and the core of Monju. Recently MHI has been conducting core designs of Japan Sodium cooled Fast Reactor (JSFR) and a next demonstration fast sodium cooled reactor with Japan Atomic Energy Agency (JAEA).

In 1970s, the generation code of effective cross sections and 1D-diffusion caculation code, ODD (ODDBURN: revised ODD for burn-up calculation), 2D-diffusion caculation code, 2DBURN (2D-XY, 2D-RZ), TRIANGLE (2D-triangle mesh), and HANYO (a code system consisting of module codes in 2D-XY and 2D-RZ geometries) were developed and used in the works for JOYO and Monju. These codes were confirmed to be valid from comparisons with measured values in critical experiments and comparisons with the other calculation codes.

After then, 3D-triangle-Z diffusion calculation code, TRISTAN and 3D-discrete ordinates transport calculation code, ENSEMBLE have been developed for the purpose to improve the accuracy of methodology toward cost reduction and high performance, keeping the safety of fast reactor.

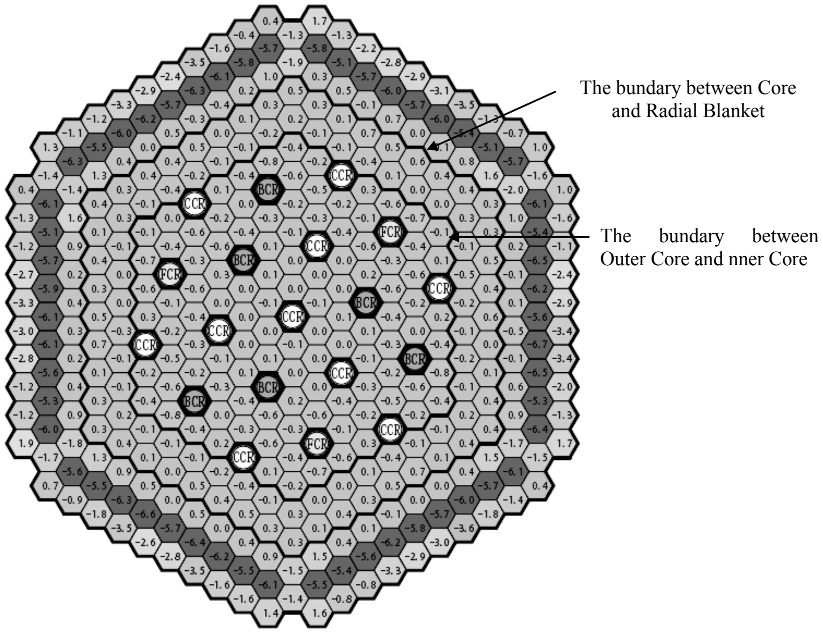
TRISTAN can calculate multiplication factor, power distribution, burning and adjoint flux with multi-group cross sections. Using calculted real flux and adjoint flux, the Doppler coefficents, the material reactivity, and the sodium void reactivity can be calculated by the perturbation theory. The feature of this code is that a calculation point is vertex of triangle mesh. This code has been used in the design study of JSFR, in the conceptual design of the next demonstration fast reactor of Japan, and also in the analysis of the core performance test of Monju.

ENSEMBLE-TRIZ, one of ENSEMBLE codes for the 3D-triangle-Z geometry[1, 2], are being developed to be able to calculate any nuclear characteristcs required for the design, as well as TRISTAN. It can be concluded tentatively that ENSEMBLE-TRIZ is valid from the following limited data.

- The difference of eigen values is less than 0.0002 between ENESMBLE-TRIZ and the average of Monte Carlo codes in a small fast flux core, which a cell pitch of sub-assembly of 129.9 mm, axial mesh of 50 mm, and four group constants were given by Prof. Takeda. This benchmark problem is included in bencmak calculations of OECD/NEA[3].
- The discrepancy of the maximum linear heat rate of the assemblies between ENSEMBLE-TRIZ and TRISTAN is less than 0.8% in the acitive core of Monju, and a large transport effect in the blanket region is calculated as shown in Fig. 1. Since the large transport

effect would be identical to the siginificant difference between measured values and calculated values by the diffusion code, ENSEMBLE-TRIZ will be able to better the precision of calculated power distribution in sodium cooled fast reactor.

Current used CPU is 2.0 GHz of Opteron and platform is PC-Linax for ENSEMBLE-TRIZ and TRISTAN. The most important future work is considered to be an accumulation of data for verification and validation, which is necessary for licensing.



Value (%): (ENSEMBLE-TRIZ/TRISTAN)-1

Fig.1 Comparison of Maximum Linear Heat Rate between ENSEMBLE-TRIZ and TRISTAN

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Nuclear Energy Advance Modeling and Simulation Program- Fuels Integrated Performance and Safety Code Program – A Multi- Scale Approach to Modeling and Simulations

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Abstract

The increased use of nuclear energy in the nations energy portfolio has been suggested recently by various social, economical and political organizations. Several options for the extension of nuclear energy being considered are; 1- Life Extension of Current Nuclear Reactors (operations at high burn ups), 2-Advanced New Generation Reactors (Gen III systems), 3- Generation IV Nuclear Energy Systems (particularly Next Generation Nuclear Plant (NGNP) concentrating on high temperature applications), and Advance Fuel Cycle Initiatives (AFCI) (fast reactor and advanced transmutation fuels). These new technology concepts will require new types of fuels (except the first option that may require more understanding of fuel behavior than development or minor modifications of fuels), and the new fuels have be developed and qualified.

In the Nuclear Energy Advanced Modeling and Simulation (NEAMS) fuels Integrated Performance and Safety Code (IPSC) program we initially focus to the multi-scale modeling and simulation of new fuel types that AFCI Transmutation Fuel Campaign (TFC) program is developing. TFC is a natural customer of the NEMAS fuels IPSC project and a strong interaction and integration between the campaign and IPSC must be implemented. The program plan in terms of approach is general enough to be applicable to other fuel types of the future nuclear technology solutions. Requirements, however, may need to be updated for fuels not considered by TFC, depending upon the new physics findings.

The advanced fuels of interest to AFCI programs are more complex than the traditional fuels previously and currently used in existing reactors. It is clear that using a traditional, heavily empirical approach to develop and qualify fuels over the entire range of variables pertinent to AFCI on a timely basis with available funds would be very challenging and costly, if not impossible. As a result, AFCI TFC has launched an advanced modeling and simulation campaign to revolutionize fuel development. NEAMS is a natural extension of the TFC M&S program with a consideration of a larger scope. Therefore, the most of TFC M&S roadmap goals and requirements remained same in this program plan. The M&S approach is predicated upon transferring the recent advances in computational sciences and computer technologies into the fuel development enterprise. The NEAMS fuels IPSC project will couple computational science with recent advances in fundamental understanding of physical phenomena through *ab initio* modeling and targeted experimentation to leapfrog, many fuel-development activities. Realizing

the full benefits of this approach will likely take some time. However, it is important that the developmental activities for modeling and simulation are tightly coupled with the activities of the AFCI campaigns to maximize feedback effects and accelerate both the experimental and analytical elements of the programs towards a common objective. The close integration of modeling and simulation, and experimental activities and the verification, validation, uncertainty quantification and licensing program, also enables science-based predictions subjected to rigorous verification and validation and explicit quantification of uncertainties embedded within these predictions.

The primary objective of the NEAMS fuels IPSC project is to deliver a coupled, three-dimensional, predictive computational tool for nuclear fuel pins and assemblies, applicable to both existing and future reactor fuel design, fabrication and for both normal and abnormal operating conditions. The validated tools can be used for lifetime extension, development of more informed safety margins, and the design of future fuels for potential new nuclear systems. It is important to re-emphasize that the NEAM fuels IPSCs objective of a successful modeling and simulation program is not a luxury but an indispensable necessity for the future of the nuclear energy. Without NEAMS fuels IPSC capabilities the use of extended nuclear energy in the nation's energy portfolio will take an unacceptably long time with an increased cost. In this paper we discuss the multi-scale approach we are taking to develop fuels IPSC.

Atomistic modeling of the U-Zr system

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Simulation of the behaviour of multicomponent alloy systems under high temperature and neutron flux conditions is of interest to recent activities related to science based advanced modelling and simulation of nuclear fuels. In particular, multicomponent U-Z, U-Pu-Zr, and U-TRU-Zr fuels are of interest to advanced fast reactor systems. Issues related to the performance of those fuels, such as constituent redistribution and fuel cladding chemical interaction, are the focus of some of the advanced simulation activities. The activities aim at replacing empirical relations that describe those phenomena with mechanistic models that are based on fundamental understanding at different scales ranging from the atomistic level to the macroscopic level.

Presented in this paper are atomistic level simulations of the two phenomena in relation to the U-Zr based fuels. The simulation is based on the BFS method [1] which is a quantum approximate method that allows for straightforward atom-by-atom analysis of multicomponent systems without limitations on the number of elements considered. The method, based on perturbation theory and the equivalent crystal theory, is founded on the assumption that the energy of a system can be computed as a sum of individual contributions from each atom in the system. In turn, such contributions can be partitioned between strain (structural) and chemical (composition) effects via corresponding virtual processes that simulate the actual process of alloy formation by changes in the electron density in the overlap region between a given atom and its environment. Due to this formulation the method is free of limitations on the number and type of elements. It provides equal accuracy for simple binary systems or multicomponent systems allowing for the simulation of complex systems such as those resulting from the interaction between a multicomponent fuel and its cladding.

The methodology has been applied previously to modelling and simulations of the interaction of UMo-based fuels with Al cladding in low enrichment test reactors [2]. Here, the methodology is applied to the U-Zr fuel system. First, the fundamentals of the methodology are discussed. Its first principles-based parameterization of the U-Zr system is presented, in addition to validation and verification of the parameters via comparison to the experimental phase diagram and properties of the U-Zr solid solution. For example, using results from large scale simulations, the concentration dependence of the lattice parameter as a function of temperature is estimated, as shown in Fig. 1. Other properties used for validation include the coefficient of thermal expansion. Fig. 2 shows preliminary results for Fe interdiffusion (red spheres) in U-10 wt%Zr (blue-yellow) with increasing temperature. Similar results are also presented for the U-Pu-Zr system for the characteristic range of concentrations considered for such fuels. This provides insight into the changes in composition and concentration associated with this interdiffusion phenomenon that is fundamental to the understanding of fuel cladding chemical interaction in this fuel type.

Finally, the well-known constituent redistribution due to temperature gradient in the fuel is dealt with by means of a finite temperature BFS-based algorithm. The algorithm is designed to determine the concentration profiles for arbitrary changes in temperature distribution in the fuel. Validation of these results will be done with available experimental data of concentrations profiles in U-Zr and U-Zr-Pu fuels.

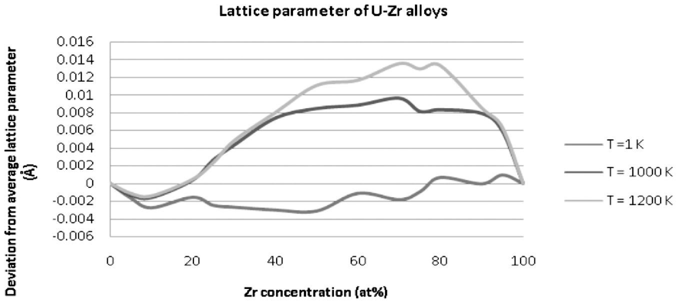

Fig. 1 Concentration dependence of the U-Zr lattice parameter (as deviation from the average lattice parameter) at different temperatures.

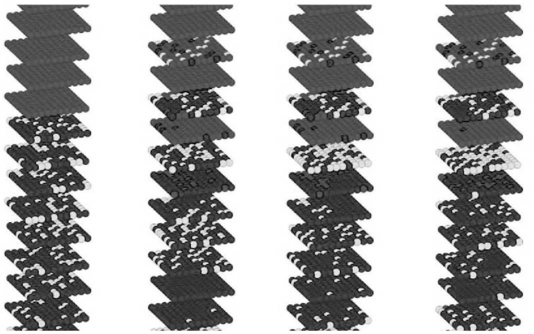

Fig. 2: Fe interdiffusion into U-10wt%Zr. Fe, U and Zr atoms are denoted with red, blue and yellow spheres, respectively. The section of the computational cell shown corresponds to the interface between Fe and the U-Zr fuel, denoting the formation of an interaction layer with increasing temperature.

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Design modeling of fuel particles for high-burnup in pebble-bed fast reactors

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The thermomechanical and neutron lifetime of different fuel particle designs is assessed by applying a new performance modeling platform comprised of an analytical stress code and finite element engineering hydrocode. Our investigation is based on fuel for fast reactors with the goal of high-burnup to provide minimal waste disposal. Fuel designs are considered based on variations of the standard Modular Pebble-Bed Reactor (MPBR) design in which the spherical fuel kernel contained by a three-layer coating system comprised of inner Pyrolytic Carbon (IPyC), Silicon Carbide (SiC), and outer PyC (OPyC). The neutronics calculations used in our investigation are based on a new fusion-fission engine concept called LIFE (Laser Inertial Confinement Fusion-Fission Energy) [1,2].

Particle stresses are calculated accounting for the interplay between mechanisms such as irradiation-induced swelling and creep, thermal expansion, anisotropic elastic effects, and layer asphericity. In addition, mechanisms such as corrosion and void coalescence are considered in order to avoid failure of the particles by way of layer cracking and leakage of the fission products, or other pathways. Our design investigation involves a parametric study of layer materials with respect to their thermal conductivity, irradiation resistance, constitutive and other properties and layer thickness to develop a fuel particle design with optimized resistance to failure mechanisms for the desired operating conditions.

A key component of the modeling platform is the capability to examine the time and space evolution of all mechanisms affecting performance which are often neglected for the conditions at low burn-up levels. Specifically, temperature variation as a function of depth into the layers generates stresses and also affects the amount of swelling, particularly at high fluence. Moreover, irradiation temperature cycling has been identified as a source of additional time-varying stresses that can lead to cracking and fatigue [3]. In the LIFE concept, thermal cycling of the particle will also take place due to a microsecond-pulsed laser source. Figure 1 shows finite element hydrocode simulation results for (a) tangential and (b) radial stresses in IPyC, SiC, and OPyC layers of TRISO particle over two thermal cycles of amplitude ~7 K at 50 kHz and to show the cyclical stress behavior. In Fig. 4(a), the average tangential stress is 10.5, 20.5, and 2.0 MPa in the IPyC, SiC, and OPyC layers, respectively. All of the stresses are positive indicating tension but are well below the threshold strengths beyond which the layers crack. The amplitude of the tangential stress during the cycling is 1.5, 3.0, and 1.5 MPa in the IPyC, SiC, and OPyC layers, respectively, and do not contribute significantly to variation of the layer stresses. In Fig. 4(b), the average radial stress is -20, -41, and -4 MPa in the IPyC, SiC, and OPyC layers, respectively. The negative values indicate compression as a result of the thermal expansion in the layers. The amplitude of the radial stress during the cycling is 2, 7, and 4 MPa in the IPyC, SiC, and OPyC layers, respectively.

This work performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract DE-AC52-07NA27344.

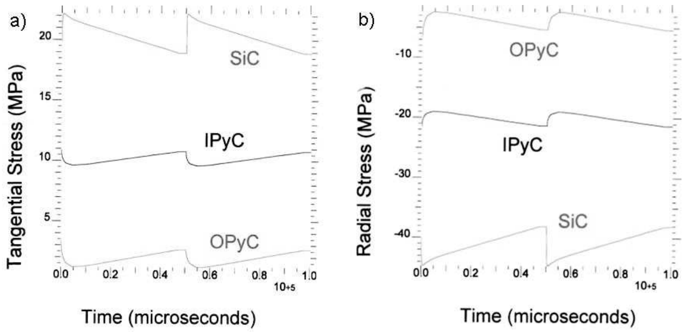

FIG 1: a) Tangential and b) radial stresses in IPyC, SiC, and OPyC layers of TRISO particle during thermal cycling.

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Numerical Modeling of Gaseous Fission Product Transport at the Meso-scale: A Multi-physics Mechanical Response of Fuel Pin Swelling

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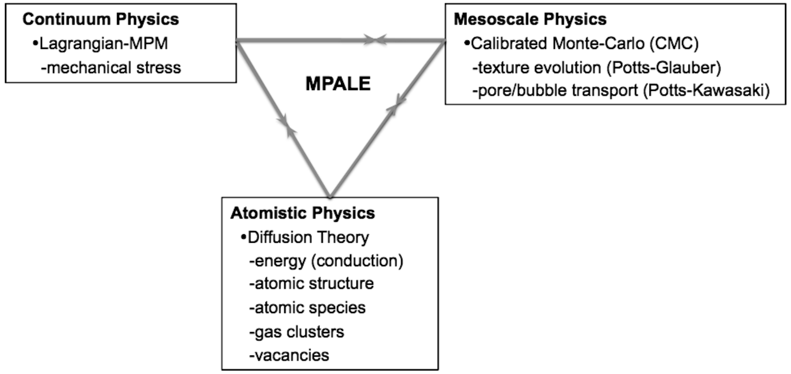
Meso-scale simulation of microstructural evolution in fast reactor fuels offers an intermediate perspective between atomistic simulation and macro-scale continuum models. The polycrystalline nature of the fuel and clad is modeled explicitly with heat transfer, deformation, and fracture treated within a grain-level based continuum framework. Grain boundaries are idealized as mobile, sharp dividing surfaces that move in response to mechanical, thermal, misorientation, inclination and bulk defect driving forces. Vacancies and dislocations are accounted for using evolving density fields, and fission product bubbles are tracked as they nucleate, move and coalesce. The mechanical response of the microcture is treated with a dislocation-based plasticity in which the effects of microstructure length scales are captured by a non-local evaluation of material hardness using deformation gradients. This simulation strategy permits the study of thermal transients on pin microstructure and the clad response, a setting in which there is currently a dearth of experimental data.

To facilitate meso-scale fuels modeling, we have developed a new hybrid paradigm which couples a time calibrated Monte Carlo (cMC) strategy for modeling grain boundary motion and bubble transport at the meso-scale with a deterministic continuum mechanics method for the mechanical stresses. Atomic level physics (e.g. heat transfer, vacancy and atomic specie transport) are modeled using diffusion theory with diffusivities as functions of both temperature and mechanical strain. This paradigm is illustrated in the following figure.



MPALE Code Physics Coupling

A particle-and-cell algorithm, the material point method (MPM), is used to compute the continuum level thermo-mechanical state. This particle based continuum method was chosen to facilitate efficient coupling with the kinetic based cMC strategy. In this cMC strategy, the probability for realizing a given fluctuation based evolution event is based on the potential for localized free energy reduction. The free energy, in turn, is taken to be a function of stress, misorientation-dependent grain boundary energy, and a set of defect densities. The cMC strategy is an extension of the Potts MC Model where each MC step now has a physical time associated with it. This is a requirement for time accurate transient simulations. A standard control volume method is used compute the atomistic physics assuming diffusion theory; the appropriate diffusivities are functions of both temperature and mechanical strain and are obtained using Density Functional Theory (DFT) and random-walk processes. This 3-D hybrid code (MPALE) uses a time splitting algorithm since the time constants for the physics processes allow the individual physics model to be solved in a decoupled manner. The particle-and-cell nature of the MPM algorithm is useful for this paridgm since the same mesh is used for both MPM and the diffusion operator and the same particles are used for both the MPM and the cMC operator. That is, the entire spatial domain is acted on by the MPM continuum mechanics, the cMC model, and the diffusion models; this is not a multi-length scale paridgm which uses upscaling to communicate information. Finally, because our intent is to study the coupled response of both the fuel and clad during a transient event, MPALE has been designed for efficient large-scale, parallel computing environments.

The unique capabilities of MPALE, that is, directly computing grain scale transport phenomena in a framework where the mechanical strain of the fuel can be directly obtained, allows the sensitivity of individual models and coefficients to be assessed. For example, four

diffusivities and one thermal conductivity coefficient are needed for the atomistic physics models in MPALE. Standard stastistical tools can be used to obtain latin hypercube response surfaces so that the sensitivity of the clad strain response from each of these coefficients can be determined. Therefore, the dominant models and inputs for determining the clad strain during a transient event can be ranked to determine if further analysis is needed.

We will describe MPALE with an emphasis on the transport of bubbles within a polycrystalline environment wherein bubbles move in response to inhomogeneities in thermal, deformation and defect fields. We will also present initial results for the radial clad strain for a 3D fuel pin scale simulation during a transient event. The domain of this simulation will be ¼ of the cross-seciton of a representative fast reactor fuel pin with an approximate axial length of 50 grains. Finally, we will present our status for obtaining a completely coupled meso-scale simulation capability for the simulation of nuclear fuel pin and clad interaction.

Calculation capability of NETFLOW++ code for natural circulation in sodium cooled fast reactor

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The present paper describes the simulation of the natural circulation in the secondary heat transport system (HTS) after an intentional plant trip of the experimental fast reactor 'Joyo' at 140MWt power using the plant dynamics analysis code NETFLOW++. This code is an integrated network code to calculate the nuclear steam supply system (NSSS) and the balance of the plant (BOP), i.e., turbine/feedwater system developed by the author[1]. Up to now, the code has been validated using transient data of the experimental sodium facility PLANDTL, experimental fast reactor 'Joyo' and the prototype fast breeder reactor 'Monju'. These validations are steps to evaluate the natural circulation of a large-scale fast breeder reactor because it is important that the code can simulate not only a large plant but also a small plant or apparatus in order to have versatility. Therefore, the former validation results are introduced to show the degree of agreement. In order to consolidate the applicability of the code to the evaluation of the natural circulation, the present simulation was selected.

Natural circulation tests were conducted at the experimental fast reactor 'Joyo' for the two reactor cores. The latest natural circulation test was conducted from the rated power of the Mark-II irradiation core at 100 MWt, and the NETFLOW++ code was validated using the measured data. At this occasion, the importance of the inter-subassembly heat transfer was recognized by the author [2] in order to predict the core outlet temperature although the whole plant behavior was little effected. The 'Monju' reactor conducted some natural circulation tests in the primary and secondary heat tranport systems utilizing the heat of primary pump operation. The NETFLOW++ code was validated using these data. Although these were not the complete natural circulation expected in the power plant, these results were important because basic factors relating the natural circulation such as the buoyancy force, the static head produced by cooling, pressure losses and so forth were included. An intentional scram test at 140 MWt at 'Joyo' was conducted to check the function fo the heat transport system after the modification in order to increase neutron flux. This result was summarized by Kawahara[3] in an unclassified report and calculated by Takamatsu et al.[4] using the Mimir-N2 code for exclusive use.

When the reactor was tripped intentionally, the primary motor was tripped and driven by a pony motor instead. The flow rate in the primary HTS leveled off at approximately 15% of the rated flow rate. Meanwhile, the secondary HTS was circulated by the natural circulation after the secondary pump trip. This event is simulated using a calculation model. The reactor core is divided into 10 subassembly groups from the center to the peripheral. All major components in two loops are taken into account.

At time zero, the reactor power is tripped in the calculation and decreased along the decay heat characteristic. At the same time, the pumps in the primary and secondary HTSs in the computer code are tripped and run down based on the coast-down characteristics given by inertias. After a certain period, the primary pumps are operated with the low speed in the same manner as the test using a pump model which can trace the given flow rate

automatically. While, the pumps in the secondary HTS are not operated after the trip. Comparison between the measured result and calculation result are shown in Fig. 1. The DHX in the figure stands for a dump heat exchanger. The word DHX is used in 'Joyo' instead of the air cooler. Trends with closed symbols and solid lines stand for test result. The open symbols stand for the calculated results. In this case, the same shaped symbol means the same parameter as in the test result. The agreement is the same order as the Mimir-N2 exclusive code for 'Joyo'. As a result, it raise expectations that the NETFLOW++ code is firmly applicable to the natural circulation analysis of sodium-cooled fast reactors with the scale of the prototype reactor 'Monju'.

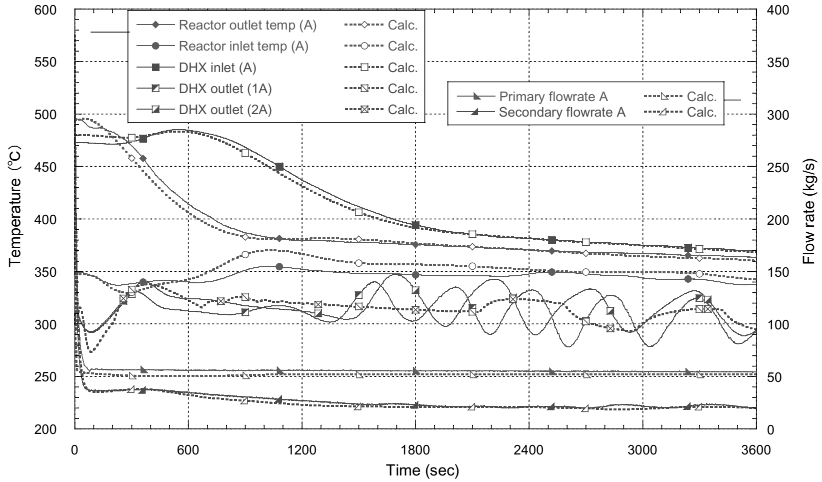

Fig. 1 Comparison of plant transients in the manual plant trip event at 'Joyo' between measured and calculated results.

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CFD Anaylsis of Natural Convective Non-Darcy Flow in Porous Medium

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In this study, the fluid dynamics in porous media is treated as a non-Darcy flow condition that the inertial resistance and the viscous resistance must be considered. According to Niven [1], the porosity and particle size determine the flow condition, whether it is Darcy or non-Darcy flow. From this, Ergun [2] developed the well-known equation of the relation between momentum loss and the geometrical parameters. When Darcy equation is introduced, the permeability, Forchheimer's coefficient *b*, and momentum loss can be expressed. Based on these, the governing equations of mass, momentum and energy were implemented in ANSYS CFX-11.0.

The physical domain is a square rectangular enclosure. The mesh was generated with hexagonal mesh and included 50 X 50 nodes, uniformly. In transient analysis, zero velocity and the reference temperature was applied as an initial value. The convergence criterion was below 10^{-5} for all averaged residual RMS value. The temperature and velocity were calculated. For validation, the porous media with constant temperature on side wall and with the heat generating inside was modeled. The results of numerical analysis for validation of the case of only constant side wall temperature were compared with the previous results of Nithiarasu et al. [3]. Most of the data agreed with the data of Nithiarasu et al. [3] and reasonable trends were found. For the case of the porous media with heat generating inside as well as the constant temperature on side wall, the results showed a good agreement with the previous works(Table 1 and Table 2). The influence of Darcy number was examined and we found that the natural circulation can be depressed more by the denser porous media (Fig. 1). For deeper level of validation, transient analysis was also conducted. The steady state was attained at about 1 or 2 minutes as the natural circulation became stable.

To describe the flow behavior, previous works applied the Darcy flow assumption. However, non-Darcy flow is also important and must be considered. By conducting Scale Analysis, Nusselt number correlation in terms of Rayleigh number can be derived.

$$Nu = \frac{C_1 (Ra - A)^{1/2}}{1 - C_2 (Ra - A)^{-1/6}}$$

Where $Nu = \frac{Sd^2}{2k\Delta T}$, $A = \frac{\rho \alpha K^{1/2} d^2 C_F}{\mu \delta^3}$ and $\delta_H = \alpha \left(\frac{C_F K^{1/2}}{g \beta K \Delta T} \right)^{1/2}$. C_1 and C_2 are constants of order 1 and could be determined empirically or analytically.

Based on Ergun [2] equation, CFD methodology was developed to understand the natural convection within the porous media for, both Darcy flow and non-Darcy flow condition. Furthermore, the physical model and numerical method was validated showing good

agreements with the previous works. To suggest a theoretical base for non-Darcy flow case, Nusselt number correlation was derived by the scale analysis.

As an extension of this study, the further analysis on applications, such as core catcher in LMFBR will be added to complement the study.

Table 1 Constant side wall temperature

Ra	Da	Porosity	Nu (present)	Nu [1]
10^7	10^{-6}	0.4	1.09	1.08
10^8	10^{-6}	0.4	3.01	3
10^9	10^{-6}	0.4	11.06	11.57
10^5	10^{-4}	0.4	1.1	1.08
10^6	10^{-4}	0.4	2.81	3
10^7	10^{-4}	0.4	8.84	9.63
10^3	10^{-2}	0.4	1.19	1.08
10^4	10^{-2}	0.4	1.49	2.3
10^5	10^{-2}	0.4	3.61	5.58

Table 2 Internal heating and constant temperature of side wall

Ra	Ra_I	Da	Porosity	Nu (present)	Nu [1]
10^5	10^3	10^{-2}	0.4	3.61	2.84
10^5	10^3	10^{-4}	0.4	1.06	1.02
10^5	10^3	10^{-6}	0.4	1.02	0.954
10^5	10^5	10^{-2}	0.4	3.28	2.4
10^5	10^5	10^{-4}	0.4	0.6	0.57
10^5	10^5	10^{-6}	0.4	0.55	0.51
10^5	10^7	10^{-2}	0.4	-42.42	-44.1
10^5	10^7	10^{-4}	0.4	-46.10	-46.4
10^5	10^7	10^{-6}	0.4	-47.89	-48.4

* I stands for internal

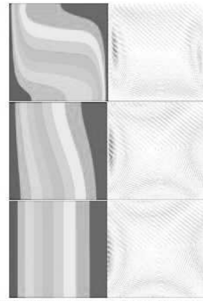

Fig.1 Internal heating and constant side wall temperature (Darcy number)

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Validation of plant dynamics analysis code Super-COPD by MONJU startup tests

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The plant dynamics analysis code COPD was validated by the experimental fast reactor JOYO, the 50MWt class steam generator (SG) experiments and the in-pile experiments for MONJU components[1]. The flow network models and the control systems of the MONJU plant in the COPD code, even those of the main components, were divided into simple calculation modules. These modular programs have been developed as the Super-COPD code and made it possible to construct the flow networks, the control systems and main components of arbitral liquid metal fast reactors by connecting these modules [2]. However, it is necessary to validate the constructed flow networks in each component, because the flow paths and the heat transport routes change by the plant operational conditions and affect the whole plant dynamics.

In this study, flow network models of the upper plenum of MONJU reactor vessel (RV) and those of the primary inlet plenum of the intermediate heat exchanger (IHX) were modified correctly by using the startup test conditions [3,4]. Then the whole system from RV to turbines was calculated by Super-COPD and the plant dynamics were validated by the measured data. The evaluated cases were the plant trip transient from 40% rated operational condition (Case 1), the natural circulation in the primary loops (Case 2) and that in the secondary loops (Case 3), which were both caused by the released heat from the primary pump operation.

The calculated results of Case 1 and the measured data are shown in Fig. 1. In this transient, the primary and secondary main pump operations were switched from the flow coast down to the pony motor operations and the air cooler (AC) operations in the secondary loops were switched from the SG operations. The RV outlet sodium temperatures agreed well with each other until 3600 s. The calculated inlet sodium temperature were approximately 20ºC higher larger than the experiments from approximately 400 s to 1300 s. The outlet temperature of IHX in secondary heat transport system had similar discrepancies in the similar time interval. The calculated AC outlet temperatures were higher than measured temperatures from approximately 1200 s to 2400 s. These differences at the AC outlet did not affect those at the RV inlet because the former was caused before the latter. The heat transfer coefficients of the IHX tubes were validated by the 50MWt class SG experiments. Hence these discrepancies were mainly caused by the complete mixing model of the IHX primary outlet plenum. However, these temperatures had good agreements in another time and the other whole plant dynamics were reasonably simulated the experimental ones by Super-COPD.

We also calculated and compared in Cases 2 and 3. The flow networks in the RV upper plenum were constructed by only two complete mixing models in the inner and the outer barrel in these cases, since the temperature in the upper plenum does not change so quickly in these natural circulation conditions. The flow rates of the primary loops and the RV outlet

and inlet temperatures of Case 1 agreed well with those of the experimental data. The flow rates of the secondary loops and the AC outlet and inlet temperatures also agreed well with those of the experimental data.

From these results, it was estimated the flow network models of the present Super-COPD were constructed reasonably in the whole system of MONJU in these conditions. The 75% and 100% rated power operational data are also scheduled to be applied to the validations of Super-COPD in near future and its numerical models are also advanced to apply to the designing demonstration fast reactor.

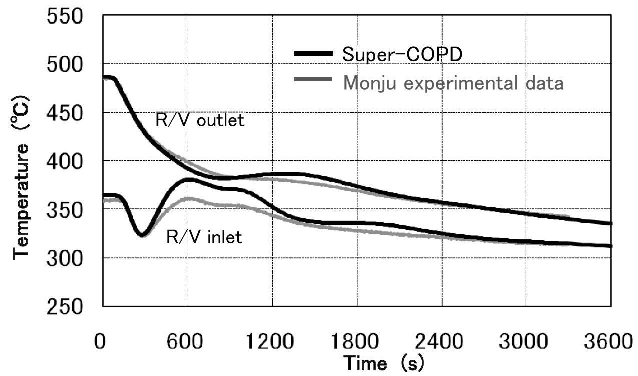

Fig.1. Inlet and outlet sodium temperature (A-loop) of Monju reactor vessel in plant trip transient from 40% rated power condition.

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Monte Carlo Simulation of BN-600 LMFR Hybrid Core

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The safe operation of a large fast reactor requires accurate estimation of power produced in different parts of the reactor core and blanket. MCNPX code was used to develop a model to simulate and study the whole core of a prototype LMFR hybrid core; the BN-600 [1,2].

In this model, the core is composed of eight radial zones (typical code model layout is illustrated in Figure (1-a)) the first two inner zones are low enrichment zones (LEZ), followed by a medium enrichment zone (MEZ). In the forth zone is the mixed oxide zone (MOX) composed of $(U,Pu)O_2$ fuel subassemblies, then the outer high enrichment fuel zone (HEZ). The rest of the core are two zones of steel shielding assemblies (SSA) and an outer radial reflector to enclose the whole core. There is also 19 shim and control rods (SHR), and 6 scram rods (SCR). The model also take into account the axial variation in geometry and composition, this is accomplished by dividing the core axially into eight different zones with a definite thickness and composition. Partial insertion of control assembly which distorts the reactor flux and fission rates distribution are simulated using the three dimensional model of the reactor core. The spectrum of neutron flux is divided into 23 energy groups.

Through this work several parameters are analyzed including criticality, axial and radial power distributions at different zones of the reactor core and burnup analysis in a typical operating conditions of the reactor core.

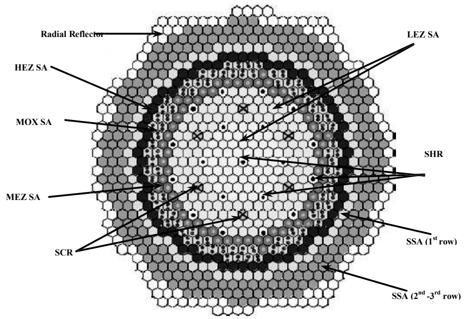
F4 tally was used to calculate the flux distribution in the core and FM4 card was used to calculate the power distribution which is normalized to a total power of 1470 Mw. The energy release per fission was fixed to 200 Mev, as suggested in the BN-600 benchmark details [3].

The temperature variation inside every cell (assembly) were considered by using the 'TMP' card. All fuel cells are at a uniform temperature 1500 K and all structural and coolant isotopes are at a uniform temperature 600 K, and in our model we assign a cross section library at which neutron cross section are processed at 1200 K (The nearest one to 1500 K) for fuel isotopes U235 , U238 ,Pu239 which consider the Doppler resonance on the cross section.

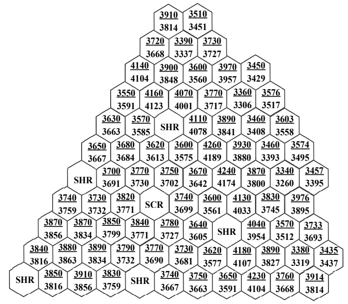
Most of isotopes cross section were taken from the ENDF/B VI library and the name of library which consider Doppler broadening at 1200 K is " endf62mt".

The core was divided to 25 zones for depletion (burn-up) calculations, as there is 25 different materials that contains fissionable isotopes. We compromise between the number of burnup zones and the program run time.

Fission Products are modeled as MCNPX computer code generates a groups of fission products automatically and its concentrations are varied at each time step.



(a) MCNPX Model Layout for BN-600 Fast Reactor Hybrid Core



(b) Power Distribution (Kw) at BOC (upper values reresent our model compared to the lower values previously calculated [2]

Figure (1) MCNPX Model Layout and Burnup Radial power Distribution

Results from this model were compared with published results of benchmark problems [3] and found acceptable. The control rod worth calculated using the present model, at the beginning of cycle, is 0.065479 compared to 0.0661 calculated using diffusion method and 0.0652 using transport method.
Figure (1-b) illustrates the radial power distribution (kW) at BOC (upper values are for the present model) compared to benchmark reults (lower values) [2]. The difference between the two did not exceed 2.6 % in all cases.

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Deterministic analysis of the Encapsulated Nuclear Heat Source by the European transport code ERANOS

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A complete characterization of the Encapsulated Nuclear Heat Source (ENHS) [1,2], a 125 MW_{th} Lead-Bismuth cooled fast reactor, has been carried out by means of the European deterministic transport code ERANOS [3], in order to provide a complementary analysis of the system with respect to the neutronic design already performed by MCNP. The interest in a comparison between Monte Carlo results with deterministic ones is mainly in a more direct and physically meaningful study (giving rise f.i. to the possibilities of perturbative calculations for a detailed sensitivity analysis) of the neutronic design of the system, also with respect to the evaluation of the conceptual idea in a broad "reactor phase space", looking for the effectiveness of the ENHS project with respect to Generation-IV specimens.

Aiming at the proliferation resistence, the ENHS concept foresees the core to be self-sustaining along a 20 years long operation cycle: the main design parameter, the pitch-to-diameter (P/D) ratio, is therefore to be defined in order to provide a Breeding Gain (BG) which compensates almost exactly the reactivity loss due to Fission Products (FPs) build up, maintaining, as far as possible, a costant $k_{eff} = 1$. No refueling nor external intervention is thus needed during the overall cycle: as a matter of fact, after the 20 years operation, the whole nuclear battery is removed and substituted by the installation agency, preventing any access to the nuclear island.

The overall dependence of the system upon the P/D ratio has been investigated by sensitivity analyses to what concerns the criticality swing during operation, for instance in case of thermal expansion of the support diagrid (the pins diameter remains unaltered, while the pitch is changed). A linear coefficient providing the reactivity change as a function of the pitch variation, around the "optimal" design point, has been also evaluated, providing a useful parameter for reactor design in case of system rearrangement.

Heterogeneous cell calculations have been performed for the core region by means of the European Cell COde (ECCO) [4], using the JEFF3.1 nuclear data library. A fine energy discretization (1968 groups) has been used to resolve the main effect of resonances: the resulting library has been then condensated, to produce a 33 energy groups, P1 dataset. The choice for such approximations is suggested by the previous experience gained within the study of other similar systems (i.e.: ELSY [5]): any refinement in energy, angle or space was found to modify the results within few percent, a gain that does not justify the associated increase of computational load.

The reactor calculation has been brought on a cylindrical model of the system discretized according to the finite differences scheme; the angular dependence has been resolved in S4 approximation. The core has been segmented in 30 Burn Up (BU) regions to take into account the differential evolution of the fuel, and the criticality investigated along the 20 years cycle. According to the evolution of the fuel with BU, the results have been also checked performing a more detailed calculation where the microscopic cross sections have been recomputed every step, to take into account any spectral effect.

The criticality analysis, shown in FIG 1, confirms the choice of the 1.36 P/D ratio as the optimal one: the corresponding core configuration (1.16 m equivalent outer radius) is made critical by a Pu enrichment (thus a U complement) which implies the required BG. It is to be noticed that the strong initial deviation of the reactivity swing from the expected exponential one is mainly due to an excessive time step for the initial burn up transition, excess that is somehow compensated in the successive step.

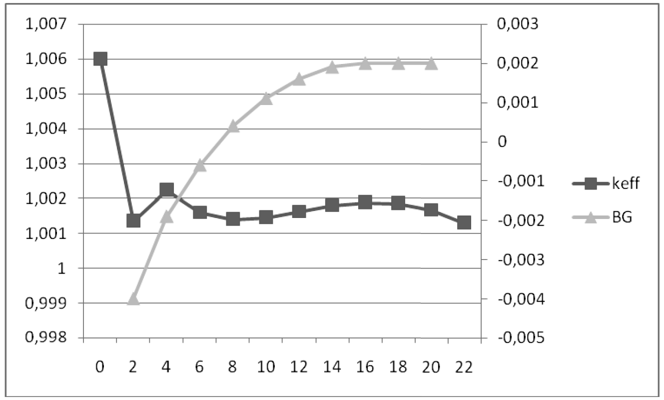

FIG. 1. Criticality swing during the ENHS cycle and associated Breeding Gain (BG).

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Monju core physics test analysis with various nuclear data libraries

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JAEA has been re-analyzing Monju core physics tests to validate the JAEA's neutronics calculation system to be used in the next Monju core physics tests. Precedent results presented in PHYSOR2008[1] have demonstrated the validity of the system based on the basic physical parameters, such as criticality, control rod worth, isothermal temperature coefficient, and power coefficient. This paper is a continuation of the validation study focusing on the other parameters, such as fixed absorber reactivity worth, fuel sub-assembly reactivity worth, coolant reactivity worth, burnup coefficient, and reaction rate.

The fixed absorber reactivity worth is a reactivity induced by the replacement of a blanket sub-assembly to a fixed absorber sub-assembly. The fuel sub-assembly reactivity worth is a reactivity induced by the replacement of a fuel sub-assembly to a non-fissile dummy sub-assembly. The coolant reactivity worth is a reactivity induced by the replacement of a non-fissile dummy sub-assembly containing sodium to that containing helium. The reaction rate data include the reaction rate ratio of ^{238}U capture to ^{239}Pu fission.

Each of the data is useful to check the calculation system in a particular aspect. For example, the first two data are suitable to check the calculation accuracy of a blanket region and a fuel sub-assembly, respectively.

The parameters are simulated using the JAEA's neutronics calculation system with various nuclear date libraries, JENDL-3.2 [2], JENDL-3.3[3], JENDL/AC-2008[4], JEFF-3.1[5], and ENDF/B-VII [6]. A continuous energy Monte Carlo calculation code, MVP[7], is employed to check calculation methods.

Figure 1 shows an example of the C/E (Calculation over Experiment) values. The C/E values are within experimental errors for the fixed absorber reactivity worth and the fuel sub-assembly reactivity worth. Those for the burnup reactivity coefficient are around the experimental error and show a tendency of overestimation. About the comparison with the Monte Carlo calculations, good agreement is also confirmed in general, but discrepancy is observed on the fuel sub-assembly reactivity in the 7th row.

Further investigation is ongoing to improve the accuracy of the system. Obtained knowledge and experience will be reflected to the next physics test analyses.

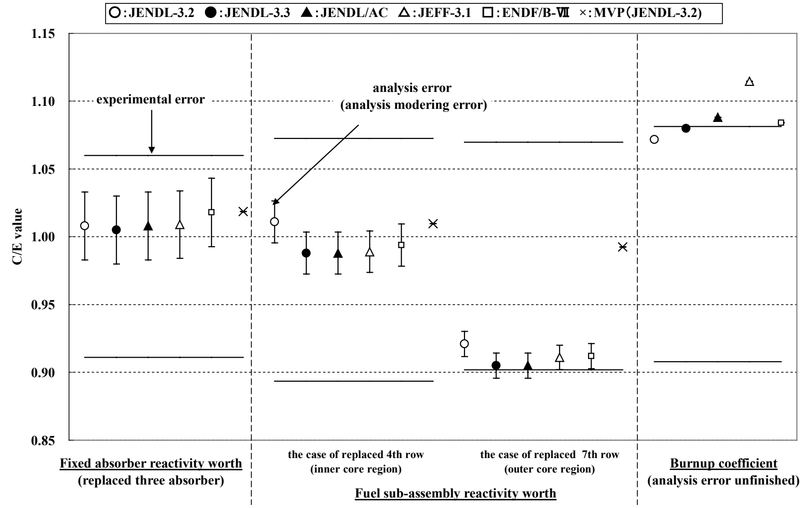

Fig.1. The analysis result of the fixed absorber reactivity worth, the fuel sub-assembly reactivity worth and the burnup coefficient (MVP results are tentative).

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Calculation of spatial harmonics in fast breeder reactor Monju

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Even in fast neutron reactor system, the core exhibits significant spatial decoupling with an increase in the size. In such spatially decoupled core, flux distribution is very sensitive to any perturbation. This high sensitivity results in the failure of the one-point-kinetic model, the large reactivity interference effect between control rods, the instability of spatial power distribution and so on. These spatial effects have been actually experienced in the Super Phoenix reactor [1] and the Zero-Power Physics Reactor (ZPPR) [2].

As a measure of the spatial decoupling, the λ-mode eigenvalue separation has been frequently employed. The degree of the decoupling increases with decreasing the eigenvalue separation. The separation between the fundamental eigenvalue λ_0 and the n-th higher harmonic eigenvalue λ_n is defined as $(E.S.)_n = 1/\lambda_n - 1/\lambda_0$. The main purposes of this study are to calculate the eigenvalue separation of the prototype fast breeder reactor Monju and to evaluate the spatial decoupling.

The λ-mode eigenvalue problem in two-dimensional (hexagonal), 6-energy group and diffusion model was solved to obtain the eigenvalues and eigenfunctions. The mesh spacing of 1 mesh/1 hexagon(fuel assembly) was employed. The power method, to which the removal technique of lower orthogonal modes was attached, was used to solve the eigenvalue problem and its adjoint one. For comparison, the calculation for the experimental fast reactor Joyo was also carried out.

In both Monju and Joyo, the harmonics from the first to the fourth higher mode are azimuthal spatial modes and the fifth is a radial spatial mode. The first and the second modes are degenerate. These eigenvalue separations of Monju and Joyo are 11.0 and 44.7 [%Δk/k], respectively. The third and the fourth modes are also degenerate, and the eigenvalue separations are 27.5 and 109.2 [%Δk/k], respectively . The fifth harmonic eigenvalue separations are 38.9 and 126.7 [%Δk/k].

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Figure 1 shows a comparison of first harmonic eigenvalue separations $(E.S.)_1$ of Monju and Joyo reactors with those of several critical assemblies [3]. The ZPPR-9 assembly is an approximately 650 MWe-size, two-enrichment-zone, homegeneous core. The ZPPR-18A and -19A are 1000MWe-size homogeneous cores. The ZPPR-13A and -13C assemblies are approximately 700 MWe-size, radially heterogeneous core, which had a large central blanket and two internal blanket rings. The above homogeneous cores were intermediately decoupled, while ZPPR-13C was most decoupled assembly in large fast reactor assemblies ever built at the ZPPR. The ZPR III-2A and -31 are small size homogeneous assemblies.

The eigenvalue separation of Monju is much larger than those of ZPPR heterogeneous assemblies where spatial decoupling effects have significantly observed. And the separation of Monju is sufficiently larger than those of ZPPR homegeneous assemblies which are intermediately decoupled. In Monju, therefore, it can be expected that spatial effects should be slight. This figure also shows that the eigenvalue separation of homogeneous assembly is inversely proportional to the squares of equivalent core diameter. On the other hand, the relation fails for heterogeneous large cores.

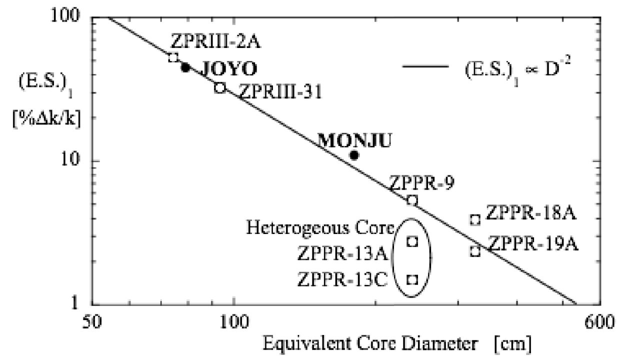

FIG. 1. Comparison of first harmonic eigenvalue separation of Monju and Joyo with that of several critical assemblies.

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Calculation and test of core flowrate distribution of CEFR

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China Experimental Fast Reactor (CEFR) is a pool type sodium-cooled fast reactor. It is now under commissioning stage.

The reactor core is composed of 81 fuel subassemblies, 8 control rods, 1 neutron source, hundreds of stainless steel subassemblies as radial reflector and shielding. There are 127 subassemblies cooled by forced convection. According to the requirements of thermal hydraulic design and safety analysis, it is necessary to implement calculation and test of whole core flowrate distribution in reactor commissioning stage.

A series of pressure drop tests for fuel subassemblies, control rods and some reactor vessel components were performed in a test rig with water instead of sodium in 2005, some empirical formulas for these components were obtained. Based on these formulas and data, a steady-state thermal hydraulic analysis code CEFR-DAEMON coupled with 2 main pumps hydraulic characteristics parameters was developed to calculate and estimate pumps, reactor core and bypass flowrate in different conditions. The curves in figure 1 show primary circuit and pumps hydraulic characteristics. By this code, the whole core flowrate distribution was obtained. The results indicated that the flowrate varied no more than••±5% from the design value in the 127 subassemblies cooled by forced convection in operation state. And it was found that the structure size played a most important role and the power distribution had a little effect to the flow distribution coefficient.

In CEFR commissioning stage, a flowrate distribution test for a mock-up core was performed with a permanent-magnet sodium flowmeter of range 5kg/s. The numerical results of code CEFR-DAEMON showed good agreement with these test data. The test results also proved that pressure drop in lower and upper plenum was very small. Detailed test content, method and results were discussed in this paper. And another test for a real core will be implemented later in CEFR.

After validated by experimental data, the code CEFR-DAEMON could be applied to calculate and estimate core and bypass flow distribution in reactor operation state. The revised edition of this code is also capable of calculating core flowrate in some other steady and transient states such as one loop operation and main pipe double-ended breaks.

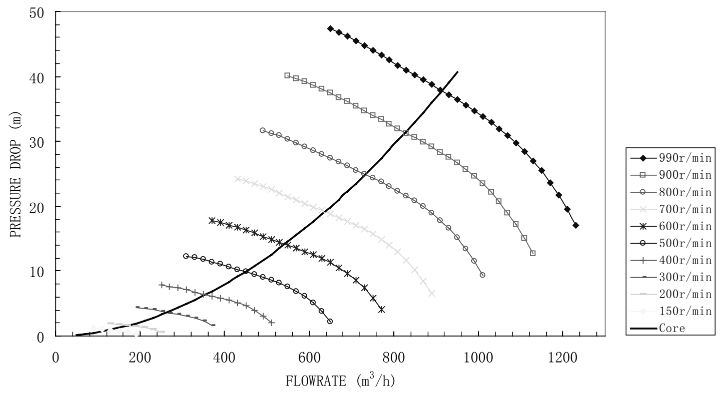

FIG.1. Hydraulic characteristics of main pumps and primary circuit of CEFR

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Evaluation on natural circulation behavior of the 4S by integrated analytical models

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The 4S (Super-Safe, Small and Simple) is a small-sized, liquid sodium-cooled reactor with a reflector-controlled core.

After the reactor trip, 4S removes decay heat by the natural circulation. The spatial temperature distribution in the Reactor Vessel affects the natural circulation behavior.

The object of this work is to obtain the asymmetric flow behavior in the primary system,which is difficult to evaluate by the flow network model. Therefore, Toshiba made three dimensional models of Computational Fluid Dynamics (CFD), which were integrating the primary coolant system and the intermediate coolant system and RVACS and IRACS. Each system models the equipment shape in detail, and the analysis condition is defined according to the system design.

Figure 1 shows the analytical model of the primary system and the intermediate system.

The primary cooling system is in the Reactor Vessel, and the coolant from the core flow up into IHX and down into primary EM pump and Radial Shield, and flow again into the core.The IHX and the primary EM pump are annular structures, so, the coolant flow path of these structures is symmetric.The intermediate cooling systems in single loop and consist SG, Intermediate EM pump, Air Cooler, and pipings.

The 4S Residual Heat Removal System (RHRS) consists of the IRACS (Intermediate Reactor Auxiliary Cooling System), which removes decay heat by using an air-cooler in the Intermediate heat transport system (IHX), and the RVACS (Reactor Vessel Auxiliary Cooling System), which removes decay heat using natural circulation of air around the reactor guard vessel. The IRACS and RVACS are diverse systems and either can remove 100% of the core decay heat.

In the case of reactor trips, the primary and intermediate EM pump stops, and the primary and intermediate flow shifts to the natural circulation. Decay heat is removed by RVACS through the surface of the reactor vessel, and is removed by the IRACS through IHX. The air of RVACS and IRACS flows by the natural circulation. All systems are driven by the natural

Figure 2 shows the velocity and temperature distribution of IHX primary coolant in steady state. Because the intermediate coolant enters in reactor vessel by single pipe, the flow velocity of IHX intermediate coolant becomes not uniform. However, flow velocity becomes almost uniform on the upstream side because there are current plates in the intermediate system IHX entrance and heat exchange part. So, the velocity distribution of the IHX primary coolant becomes almost uniform and the temperature becomes symmetry as shown in Figure 2.

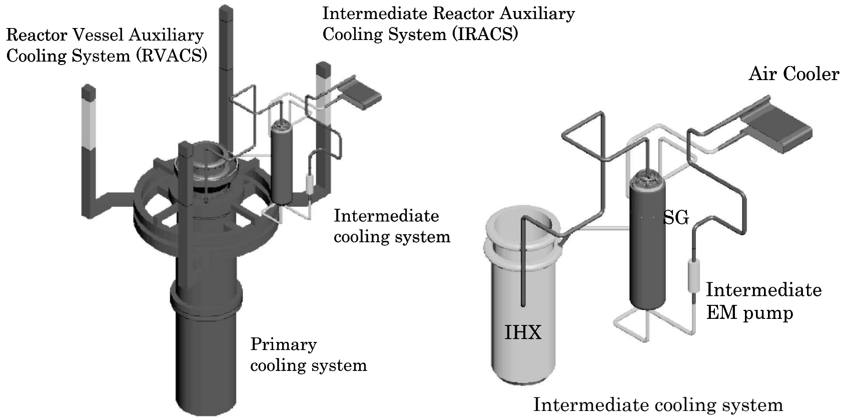

Figure 1 Analytical model

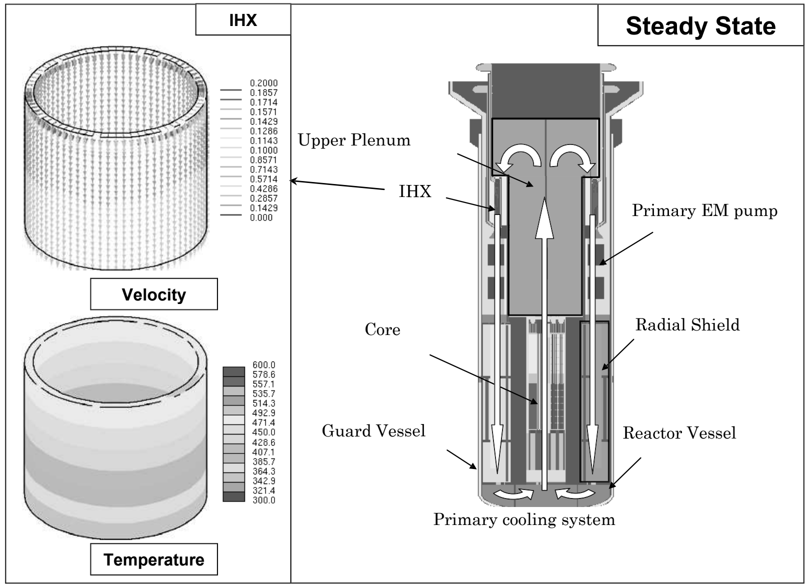

Figure 2 Velocity and temperature distribution of IHX primary coolant.

Development of computational method for predicting vortex cavitation in the reactor vessel of JSFR

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The conceptual design of Japan sodium-cooled fast reactor (JSFR) is being conducted in a Fast Reactor Cycle Technology Development (FaCT) project [1]. One of specific design in JSFR with economic competitiveness is to compact the reactor vessel. The compact reactor vessel with significantly increased primary flow velocity might cause a vortex cavitation, which should be avoided from the viewpoint of the reactor structural integrity, near the inlet of the hot-leg piping. With the increased speed, cavitation is likely to occur due to the pressure reduction at the vortex center in the accelerating flow.

The purposes of the present study are to develop a computational method for predicting vortex cavitation of JSFR, based on the theory of vortex stretching [2], and to evaluate the influence of the major parameters related to pressure reduction.

First, we developed a method to calculate static pressure reduction at vortex centers, utilizing the computational results obtained by a commercial computational fluid dynamics (CFD) code, Star-CD (CDAJ Co., Ltd.), in a combination with a commercial visualization code, FIELDVIEW (VINAS Co., Ltd.). In this study, we calculated static pressure reduction accompanied by the vortex cavitation, focusing on vortices near the inlet of the hot-leg piping. The calculation procedure is as follows. Pressure reduction ΔP is proportional to velocity gradient α and circulation Γ squared [2]. As major parameters for calculating the velocity gradient α, we selected the degree of a polynomial equation approximating the vortex center line position. As major parameters for calculating circulation Γ, we selected the angular devision [2], the second invariant of velocity gradient tensor [3] and the ratio of vorticity to vortex center vorticity. The values of these parameters were qualitatively obtained through examining their effects. In addition, we evaluated the influences of two methods to identify the vortex center, which utilize standard functions in FIELDVIEW and two integration methods for calculating circulation.

Next, we analyzed the influence of the major parameters related to the pressure reduction on the major physical quantities of vortex cavitation for a 1/10 scale water flow test for the upper plenum of the JSFR reactor vessel, which was slightly different from the current design [4]. The vortex center lines with static pressure contour obtained by the developed method are shown in Fig.1. These vortices are created by the three dimensional velocity field in complex

structures. The predicted cavitation occurrence under the hot-leg piping was in good agreement with the test result.

Based on this examination, we tried to apply this method to predicting vortex cavitation at the inlet of the piping of the 1/11 scale water flow test, which recently reflected the current structural arrangement in the reactor design.

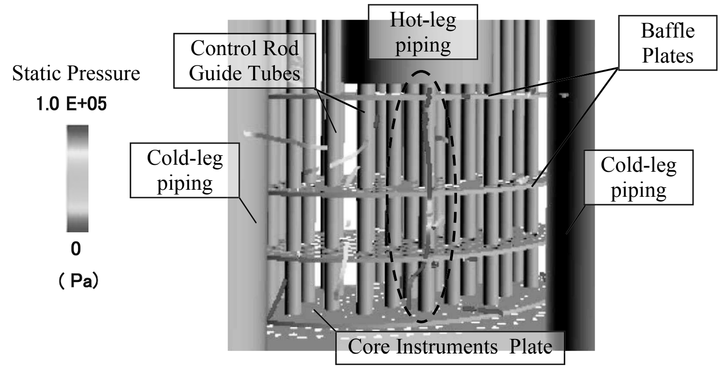

Fig.1 Vortex core lines with static pressure contour

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Thermal-Hydraulic Calculation for Simplified Fuel Assembly of Super Fast Reactor Using Two-Fluid Model Analysis Code ACE-3D

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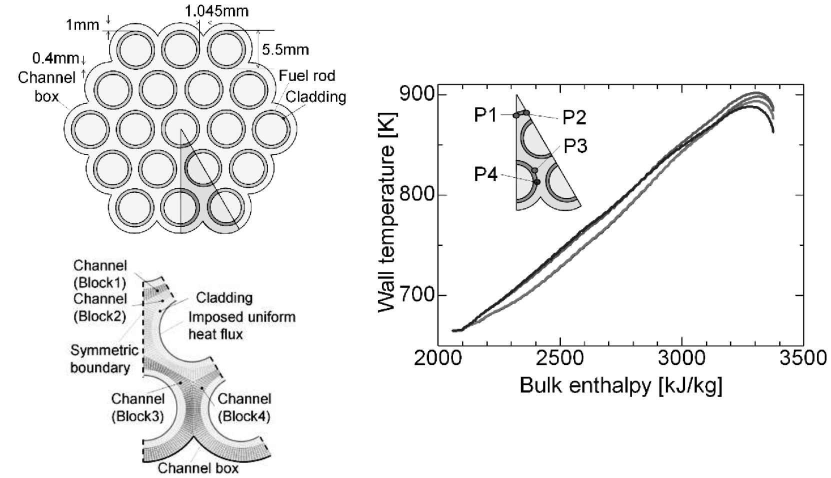
To evaluate thermal hydraulic characteristics of a fuel assembly of supercritical water-cooled fast reactor (Super Fast Reactor), a simplified fuel assembly was analyzed with a three-dimensional two-fluid model analysis code ACE-3D which has been enhanced by Japan Atomic Energy Agency. In the ACE-3D code, the two-phase flow turbulent model based on the k-ε model were adopted.

The analytical geometry simulates a 19-rod fuel assembly, which is a simplified geometry of the 271-rod fuel assembly and includes all three kinds of different subchannel types; (1): adjoining to the channel box, (2): next to type (1), and (3): located inside types (1) and (2). Figure 1(a) shows the cross sectional view of the simplified fuel bundle and mesh division. The rod diameter is 5.5 mm, the gap width between the rods 1.045 mm and the axial length 2.0 m which are the same as the design parameters of the Super Fast Reactor. In this calculation, grid type spacers which are designed to be installed are neglected. One-twelfth of the area with deep color in Fig. 1(a) is adopted as the computational domain taking advantage of symmetry. As the boundary conditions, mass velocity is set to be 2406 kg/m^{2}s, inlet enthalpy 2091 kJ/kg, and power per rod 27kW, which are the same as the steady state condition of the Super Fast Reactor. Cross-sectional local power distribution in the fuel assembly is set to be flat.

Figure 1(b) shows calculated rod surface temperature profiles at four locations in the fuel assembly. Rod surface temperatures take peak values near the top of the rods. Maximum clad surface temperature (MCST) is observed at the position facing to the narrowest gap on the center rod (P2 in Fig.1(b)) near the outlet and the value is 902 K (629°C). It was confirmed that the predicted MCST satisfies a thermal design criteria to ensure fuel and cladding integrity: the MCST should be less than 650°C [1]. In the future, coolability of a real-size fuel assembly of the Super Fast Reactor will be assessed.

Present study is the result of "Research and Development of the Super Fast Reactor" entrusted to The University of Tokyo by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT). This research was conducted using a supercomputer of Japan Atomic Energy Agency.



(a) Cross section and mesh division (b) temperature profile

FIG.1. Calculated rod surface temperature profile in simplified fuel bundle

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POSTERS OF SESSION 7:

Advanced fast reactor fuels

Numerical Simulation of Fuel Microstructural Evolution in a Thermal Gradient

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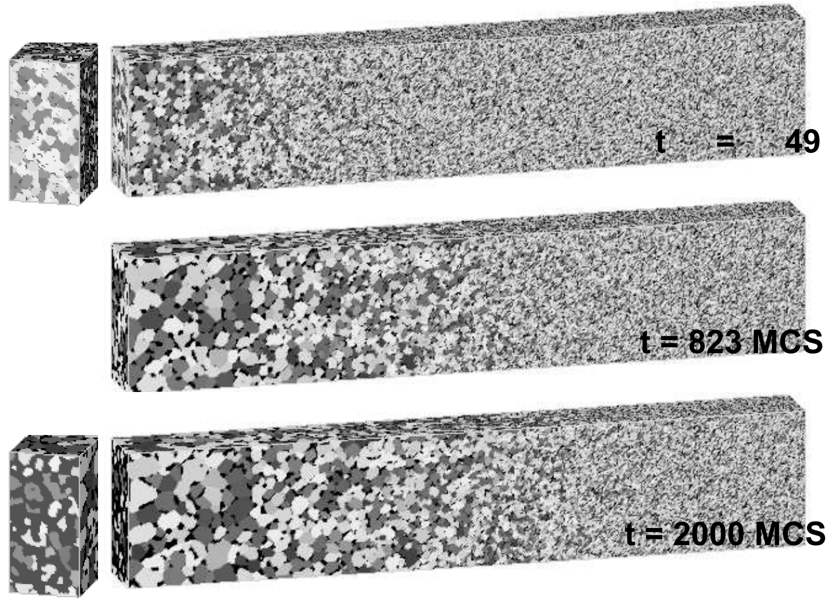
A key feature of virtually all nuclear fuels, including fast reactor fuels, is the large temperature gradient that fuel pellets experience during normal operation. These large temperature gradients drive a number of microstructure evolution processes that are unique to the nuclear fuels, particularly, to fast reactor fuels. Among them is the non-homogeneous diffusive processes. In this work, we focus on grain growth coupled to void and gas bubble migration in such a large temperature gradient. A discrete statitstical-mechanical model, the Potts kinetic Monte Carlo model will be used to numerically simulate microstructural evolution at the mesoscale. The Potts model has been developed over two decades to study many microstructural evolution processes including normal and abnormal grain growth, Ostwald ripening, grain growth in the presence of mobile and immobile pinning phase and sintering.[i] The processes of interest in this work are grain growth and bubble migration in large temperature gradients. Previous work has developed 2D[ii] and 3D[iii] grain growth in temperature gradients as well as grain growth coupled with void migration in a 2D polycrystalline[iv] and 3D idealized polycrystalline[v]. In this work,we extend the the 3D grain growth and bubble migration to the general case of a 3D polycrystalline material such as that of nuclear fuels.

Potts kinetic Monte Carlo Model

The model used here to simulate the microstructural evolution of a full 3D digitized microstructure has been described in detail elsewhere[2-5]. Curvature driven grain growth and bubble migration by surface surface in a temperature gradient has been simulated by varying the grain boundary mobility and surface diffusivity as a function of temperature. A heat of transport term as described in the diffusion literature[vi] has been added to the simulate the net transport of the gas towards the high-temperature side.

The resulting microstructural evolution of grain growth and bubble migration is shown in FIG 1. The colored features are grains; the black features are bubbles. The starting microstructure is a homogeneous polycrystalline material with small isolated bubbles evenly distributed in the fuel; in case of fresh fuel the bubbles are pores due to incomplete densification. A temperature gradient is applied across the material with the high-temperature at the left side and the low-temperature at the right side. As the results show, both the grains and bubbles coarsen at a much higher rate at the high-temperature side due to higher bubble diffusivity and grain boundary mobility at the high-temperatures. The rate controlling process is the coarsening of bubbles, which pin the grain. As the bubble coarsen by coalescence, they unpin the grains locally and allow a burst of grain growth. Simultaneously, there is a net flux of the bubbles towards the high-temperature side as shown in the side view at the left side. Early in the simulation at time t = 49 MCS, the bubbles are homogeneously distributed through out the

FIG 1: Simulation of grain growth and bubble migration in a temperature gradient.



sample. However, at time t = 2000 MCS, the bubbles are diffusing toward the high-temperature side. The bubbles have become large and connected and grains are poking into the bubbles.

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Development of a Probabilistic Design Method for Fast Reactor Fuel Rods

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The increase of linear power and burn-up during reactor operation is considered as one measure to ensure the utility of fast reactors in the future; for this the application of annular oxide fuel is under consideration because of their availability for both high power and high burn-up. The CEPTAR code was developed as a design code for annular oxide fuels and results of verifications using the results of various experiments indicated that this code could accurately evaluate the fuel performance of annular oxide fuels during irradiation in fast reactors [1], [2]. In addition, in order to improve fast reactor performance, optimization of the design margins is required. Consequently, a probabilistic method for fuel rod design has been considered, and the BORNFREE code, which includes the fuel design code as deterministic models to compute the fuel rod behavior and computes the statistical responses of several performance parameters concerned with fuel rod integrity by taking into account the uncertainty of each design parameter, has been developed [3], [4]. The probability that the performance parameters exceed the respective criterion and the design margins can be quantitatively estimated by using BORNFREE. CEPTAR was applied in the deterministics models in BORNFREE, and this system is called BORNFREE-CEPTAR [5].

Under the current deterministic method to evaluate fuel rod performances, various kinds of uncertainties for design parameters, i.e. dimensions, material and fuel properties, operation conditions, and analysis models, are established under the worst-case assumption, as shown in Fig. 1. As the evaluation conditions for design parameters are established under conditions to give a pessimistic result, a very conservative margin for reactor power and too rigid specifications of fuel rods are required. This pessimistic margin followed by costly overdesign must be optimized to improve upon fast reactor performance and to decrease the rod fabrication costs. In the probabilistic design method, the margins included in the deterministic design method can be quantified as a probability density function (PDF) for each performance parameter, and optimization for this can be done. Furthermore, the probabilistic sensitivity analysis for fabrication tolerance allows the impact on fuel rod performance to be estimated quantitatively and also the fabrication tolerance to be optimized reasonably.

The improvement of reactor performance for application of the probabilistic design method was studied through probabilistic estimations with the BORNFREE-CEPTAR system. Each design parameter was established as a PDF based on its statistical characteristics, and the statistical responses of performance parameters were estimated with the Monte Carlo method. The results in this study indicated that the conservative design margins and too rigid specifications as the result of the current deterministic design method could be reasonably improved by applying the probabilistic design method to the fuel rod design.

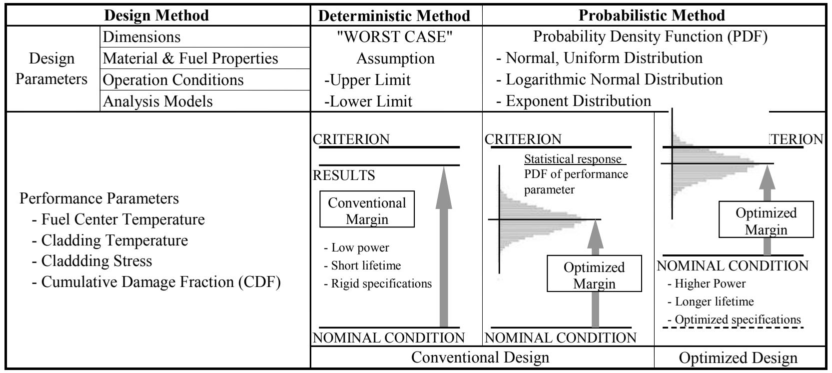

Fig. 1 The concept of deterministic and probabilistic design methods

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Inert matrix fuel concept for the rapid incineration of minor actinides harmonious with a fast reactor cycle system

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Recovery and recycling of minor actinides (MAs) in fast reactors (FRs) are key technology for the successful realization of the FR cycle system, which can lead to a reduced environmental burden and a sustainable energy supply. The most promising candidate fuel for such FRs is considered a mixed oxide (MOX) fuel doped with 1 ~ 5 wt% of MAs [1]. On the other hand, the MAs should be incinerated as soon as possible in the introductory phase of FRs since there exists a large amount of MAs, which result from the long time operation of light water reactor.

Inert matrix fuels with a high content of MAs are considered as one promising option for the rapid incineration of MAs, and a FR cycle concept that incorporates these fuels has been proposed as a high performance optional device [2]. These fuels are a composite of MA oxide host phase and an inert matrix. Table 1 shows the fundamental specifications of the inert matrix fuels investigated. MgO, Mo and Si_3N_4 are being investigated as promising inert matrices. These inert matrices has been selected so that the fuel performance could be effectively enhanced, *e.g.* higher thermal conductivity, higher chemical/physical stability, and so on. In addition, the important point for the selection of inert matrices was adaptability to the main FR cycle technology, especially fabrication process, since the inert matrix fuel would be deployed in the introductory phase of the main FR cycle. This paper describes a part of our challenges towards establishment of the fast reactor cycle that incorporates the inert matrix fuel..

Fabrication technology is one of the most important techonology, and it should be adaptable to that of the presently established ones as described above. In addition, the fabrication process should be simple so that the remote operation can be applied, the same as for the case of Am doped MOX fabrication. Therefore, a practical fabrication processes for the respective inert matrix fuels were designed based on traditional powder metallurgy for the standard fuel, i.e., ball-milling, uni-axial pressing and pressureless sintering. For MgO-based fuel in this study, the fabrication process was optimized by the investigation of fabrication properties and sintering behavior. The MgO-based fuel with good characteristics, i.e. having no defects, a

high density and a homogeneous dispersion of host phase, was successfully fabricated by a simple powder metallurgical technique [3]. For Mo-based fuel and Si_3N_4-based fuel, the additives were optimized in order to achieve high density of pellets in relatively low sintering temperature. The Mo-based fuel and Si_3N_4 based fuel with a high density were successfully fabricated in low sintering temperature [4,5].

Characterizations of these fuels were now underway. Regarding the MAs oxide host phase, phase relation and oxygen potential of $(Pu,Am)O_{2-x}$ which was assumed to be contained in MgO and Si_3N_4 matrix fuel were experimentally investigated. These results show oxygen non-stoichiometry should be optimized for this system [6].

As related technology, burn-up characteristics of a FR core loaded with the present inert matrix fuels were also analyzed, mainly in terms of core criticality. The effective transmutation of MA and flattened power distribution could be achieved by a both inner/outer core region loading of the inert matrix fuels [7].

Table 1. Fundamental specification of the inert matrix fuels investigated.

		MgO-based	Si_3N_4-based	Mo-based
Fuel	Shape	Sintered columnar pellet		
	Density	> 90 % of theoretical		
Host phase	Chemical form	$(Pu,Am)O_{2-x}$		$(Am,U,Th)O_2$
	Weight fraction	0.50	0.40	0.50 – 0.90
	Morphologic shape	Sphere, >φ100μm/ Particle, <φ10μm	Grain boundary phase	Particle, <φ10μm
Inert matrix phase	Chemical form	MgO	Si_3N_4	Mo (recovered from spent fuel)

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Suitability of a thermal design method for FBR oxide fuel rods

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The integrity of the oxide fuel rod during its lifetime should be ensured in oxide fuel design for fast breeder reactor (FBR) use. In order to ensure the fuel rod integrity, the fuel performance values, i.e. fuel centerline temperature, cladding temperature, cladding stress, cumulative damage fraction, etc., should satisfy their respective design requirements. The criterion for thermal design in Japan requires that the oxide fuel should never melt even at transient conditions during the irradiation.

Fig. 1 shows the concept of thermal design for FBR oxide fuel in Japan. The criterion for thermal design is determined by the melting point of oxide fuel. In order to decide the criterion, several factors must be taken into account including the melting point dependence on Pu and Am concentrations, the Pu-isotope composition of raw powder, the fabrication tolerance of the O/M ratio and Pu concentration, and the radial redistributions of Pu, Am and the O/M ratio during irradiation. A transient factor and a safety factor decided from the uncertainties, i.e. fuel and material properties, fabrication tolerances, operation conditions, and analysis models, are taken into account and the maximum fuel temperature is evaluated from the fuel centerline temperature computed with a fuel design code, which is verified with various irradiation tests.

To study suitability of the thermal design method, the B14 irradiation test with 4 fuel rods was carried out in the experimental fast reactor “JOYO”. The pellet-cladding gap width and O/M ratio of oxide fuels to be used in the B14 irradiation test were specified as experiment parameters. In addition, by taking into account the actual design conditions for FBR oxide fuel, the conditions in the B14 irradiation test, i.e. the linear power and the cladding temperature, were investigated, and the irradiation was carried out to encompass the hottest conditions in the actual design conditions. The oxide fuel pellets for the B14 irradiation test were fabricated from the raw PuO_2 powder with americium concentration of ~ 2.4 wt%Am/HM, which was close to the maximum concentration in the conventional FBR design.

No indication of fuel melting was observed in the non-destructive examinations after the B14 irradiation test. The thermal computations to evaluate the fuel temperature during irradiation were done with the thermal analysis code “DIRAD” [1], and ceramography samples for destructive examinations were obtained from the axial position where the fuel centerline temperature during irradiation reached the maximum and the local linear power exceeded the maximum power in the conventional FBR design. Early ceramography results related to the maximum fuel centerline temperature, the influence of pellet-cladding gap width and O/M ratio on the fuel restructuring were obtained, but no evidence of fuel melting was seen. Consequently, the margin to the criterion in the thermal design would be suitable and the fuel melting would be prevented under the conditions designated in the conventional FBR design.

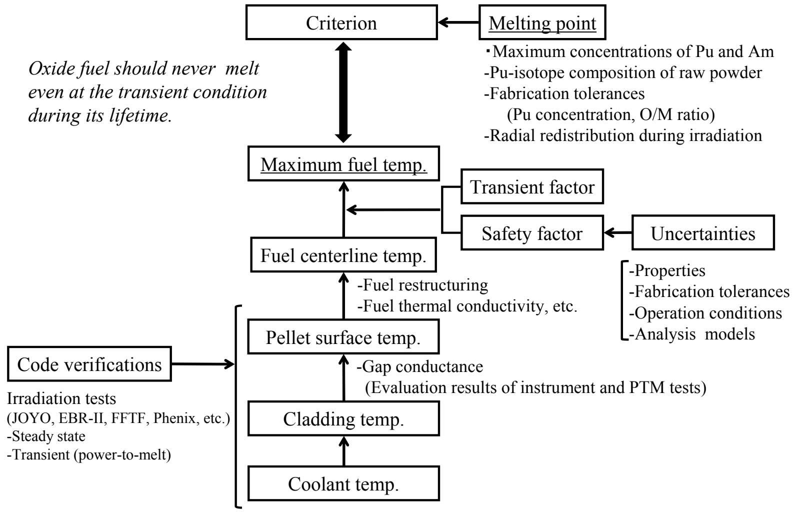

Fig.1 The concept of thermal design for FBR oxide fuel

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Studies of interaction of GEN-IV advanced fuels with metallic coolant (Na, Pb) in operating conditions.

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The main challenge for the development of innovative fast reactor fuels, envisioned in the international GEN-IV program, comes from the necessity to incorporate the minor actinides (Np, Am, Cm) into the fuels up to significant concentrations. Though various types of fuels, e.g. nitrides, carbides, metals, are potential candidates, oxide fuels are still currently considered as reference for Sodium-cooled Fast Reactors for which worldwide industries have already accumulated substantial experience in terms of fabrication, reactor operation, reprocessing and risk assessment. Though extremely rare during normal operation conditions, the fuel can hypothetically come into contact with the sodium if a breach occurs in a defective cladding. The work already performed in the past [1][2] showed that sodium can react with $(U,Pu)O_2$ fuels to form sodium urano-plutonate compounds which have a lower density and thermal conductivity than the fuel, leading to a local temperature and stress increase in the cladding and potentially enlarging the original breach up to a pin failure. The compatibility of a fuel with the reactor coolant is a basic criterion, which must be fulfilled for its deployment in a reactor system. This is a safety relevant issue, which requires no loss of fuel into the coolant in the event of a fuel pin breach. Incorporation of minor actinides will introduce a more complex fuel–coolant chemistry for which virtually no data are available. We are currently investigating the implications for safety when minor actinides are introduced in the fuels up to a significant concentration and come into contact with the metallic coolant.
The study of the interactions of AmO_{2-x}, and $(Pu,Am)O_{2-x}$ compounds (historically embedded in inert matrices Mo, MgO) with sodium and lead for different compositions and O/An ratio (An = Pu, Am) have been initiated. Tests were performed at 550°C, 648°C in sodium and 800°C in lead. X-ray radiographs and X-ray diffraction experiments were performed prior to the heat treatments. Compositions and stoechiometry (when known) are listed in table 1. These fuels are composites of Mo and MgO and are foreseen as targets for deployment in FR systems (Futurix irradiation–EUROTRANS Programme) [3][4].

Table 1: composition of compounds used for Pb and Na compatibility tests. Question marks are the samples to be investigated by X-ray diffraction to characterize the final products.

	COMPATIBILITY TESTS			
		Pb	**Na**	**Inert matrix**
1	$AmO_{1.62}$		Yes	MgO
2	$AmO_{1.82}$		?	MgO
3	$AmO_{2.00}$		?	MgO
4	$(Pu_{0.756},Am_{0.244})O_{2-x}$	Yes		Mo
5	$(Pu_{0.8},Am_{0.2})O_{2-x}$		Yes	Mo
6	$(Pu_{0.5},Am_{0.5})O_{1.73}$		?	MgO
7	$(Pu_{0.2},Am_{0.8})O_{1.88}$		?	MgO

X-ray radiographies showed that all molybdenum based CERMET fuels were intact after the lead and sodium compatibility tests. No interaction between lead nor sodium and the fuel has been observed

after 50 hours in direct contact with lead at 800°C. The visual aspect of the pellets after the tests were identical to the starting material (sample 4,5).
For the MgO based CERCER fuels, X-ray radiographies showed that only sample number 1 remained intact after the sodium compatibility tests. Pellets 2, 3, 6 and 7 were completely transformed into powder (see Fig 1). The reaction products of samples 6 and 7 were recently examined by X-ray diffraction to characterize the final products formed. Surprisingly the X-ray diffraction pattern of sample 6 (Fig 2) showed only two phases, identified as $(Pu,Am)_{O2-x}$ + MgO. No evidence for interaction with sodium was found whereas $Na_3(An)O_4$ compounds were expected. It demonstrates that for this composition and stoechiometry the actinide material does not react with Na. The cause of the collapse of the pellet remains unclear but investigation is still ongoing (e.g. mechanical strains).

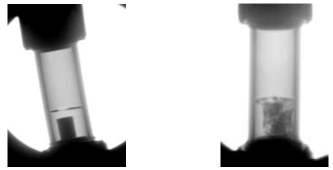

Figure 1: Na compatibility test on sample 6. $(Pu_{0.5}Am_{0.5})O_{1.73}$ + MgO. Left (before treatment), right (after treatment at 648°C)

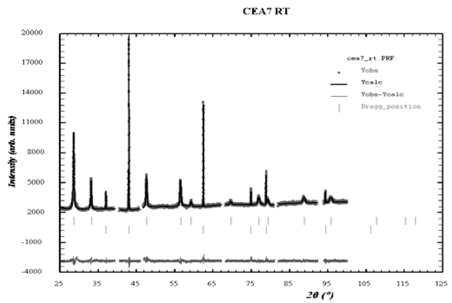

Figure 2: X-ray diffraction pattern of sample 6 after Na compatibility test. Only $(PuAm)O_{2-x}$ and MgO phases are present. No interaction with Na could be found despite the fact that the pellet collapsed. The first row of green stick bars refers to the Bragg reflections of $(Pu,Am)O_{2-x}$ whereas the second row corresponds to MgO

Future investigations will concentrate on synthesis of ternary and quaternary compounds in the Na-U-An-O systems (An=Pu,Np,Am), their characterization followed by the experimental study of their phase relations with excess sodium under various conditions. In conjunction with phase diagram investigations, thermodynamic studies will be performed on the materials and the experimental data will be compared to the results obtained by theoretical approach This presentation will describe the most recent experiments and discuss the results and the possible implications for MA-bearing fuel for SFR.

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[4] A. Jankowiak, F. Jorion , L. Donnet , NSE, 2008, 160,5, 378-384 , Synthesis and characterization of $(Pu_{0.5}Am_{0.5})_{2-x}$ MgO CERCER

Progress of demonstration experiment on irradiation of vibro-packed MOX Fuel Assemblies in the BN-600 reactor

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According to the Concept of Russian Federation MINATOM and DOE of USA, plutonium to be released as a result of nuclear weapon dismantling is to be used in nuclear power engineering - in the form of MOX fuel in fast or thermal reactors.

The scenarios of Russian weapon-grade plutonium disposal provide for its application as a MOX fuel in the hybrid core of BN-600 (BNPP) and in the BN-800 reactor under construction. The following procedures developed at JSC "SSC RIAR" can be taken as basic ones:

- pyroelectrochemical granulation of uranium-plutonium oxides resulting in granulated MOX fuel production
- vibropacking of granulated fuel directly in the fuel pin cladding, fig.1, [1].

Experience in vibro-packed fuel tests in the BOR-60, BN-350 and BN-600 reactors showed that vibro-packed MOX fuel had acceptable service life parameters even at super high burnup (about 30% h.a.).

The demonstration experiment has been conducted since 2004 within the framework of international cooperation between Russian and Japanese organizations: RIAR, IPPE, OKBM, BNPP, MEXT, JAEA, and PESCO. The goal of the experiment was to validate the possibility of vibro-packed MOX fuel assembly (FA) application for weapon-grade plutonium disposal in a fast reactor.

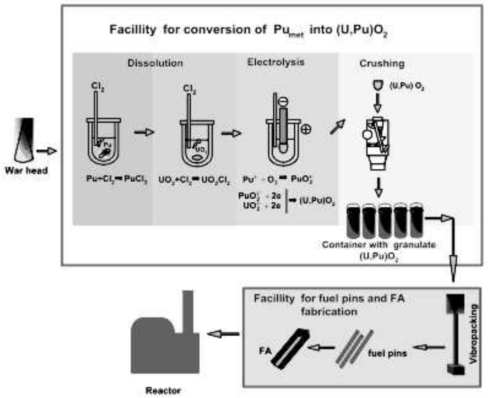

Fig.1. Flowsheet of pyroelectrochemical conversion of weapon grade plutonium into fuel for fast reactors

Under the program of the demonstration experiment, RIAR conducted pyroelectrochemical plutonium conversion into granulated MOX-fuel. The granulated fuel was used for manufacturing fuel pins by vibropacking and the fuel pins were assembled into 21 experimental fuel assemblies (EFAs) to be tested in BN-600.

OKBM developed the EFA design. Besides, OKBM prepared the detailed program of tests and substantiated irradiation parameters in cooperation with IPPE, and also provided licensing of irradiation in the BN-600 reactor in cooperation with BNPP.

BNPP provided irradiation of EFAs. All EFAs were irradiated during the specified life in accordance with the program of tests. The maximum liner heat rate of fuel pins in the FA were in the range of 39.5-45.3 kW/m, the maximum cladding temperature was within 652-703 °C taking into consideration uncertainty of parameters. The maximum duration of EFA irradiation was 569 eff. days. The peak achieved burnup was 10.6 % h.a., and damage dose was 80.9 dpa. No violations of safe operation limits were observed during irradiation in the BN-600 reactor.

After irradiation was completed, BNPP carried out primary investigation of the FA, and then RIAR performed postirradiation examinations- both nondestructive and destructive.

Oxidation behavior and sintering property of MOX powder obtained by microwave heating direct denitraition

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The simplified pelletizing process has been developed to demonstrate performance of Japan’s sodium-cooled fast reactors [1,2]. In the process, MOX pellets are prepared from MOX powder obtained by the micro wave direct heating denitation. Oxidation of the powder is caused by its high decay heat in powdering process. In this work, oxidation behavior of the MOX powder was investigated, and MOX pellets were prepared from oxidized powders.

Oxidation rate and oxide phases of the MOX powders was analyzed by thermo gravimetry and powder x-ray diffractmeter, respectively. The oxidation reaction of raw powders took place in two steps, the M_4O_9 phase precipitates in the first oxidation step, and the M_3O_8 phase precipitates in the second oxidation step, which were dominant in O/M ranges of less than and more than the O/M ratio of 2.28, respectively. The oxidation rates of MOX powder were evaluated by isothermal oxidation examinations.

The MOX powders were heated at 423K and 873K in the air and then were pressed and sintered into the pellets. The XRD patterns of the oxidized powders and the sintering pellets were shown in Fig. 1(a)-(d). The pattern of M_4O_9 phase was observed in the powder with O/M = 2.33 which was heat-treated at 423K as shown in Fig. 1(a). On the other hand, the extra peaks of the M_3O_8 phase were observed in the powder with O/M = 2.44 which was heat-treated at 873K in Fig. 1(b). The patterns of both sintered pellets in Fig. 1(c) and (d)were identified as a MO_2 single phase.

Many cracks were observed in the sintered pellets prepared from the oxidized powder with M_3O_8 phase. It was considered that the cracks were caused by the phase transformation from orthorhombic M_3O_8 phase to cubic MO_2 phase. The analysis results of the oxidation rate showed that the MOX powder was needed to keep in the temperature range of less than about 500 K and to avoid the precipitation of M_3O_8 phase for obtaining good appearance pellets.

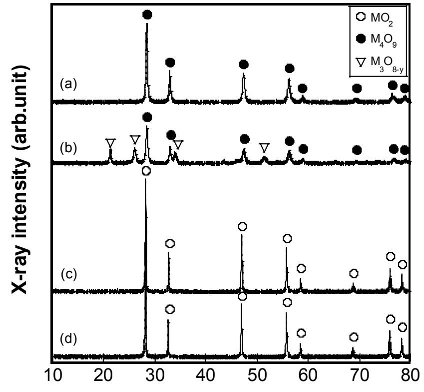

FIG. 1. XRD patterns of (a) powder with O/M=2.33, (b) powder with O/M=2.44, (c) pellet prepared from powder(a) and (d) pellet prepared from powder(b).

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Oxygen potentials of (MA,Pu,U)O_{2-x}

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Oxygen potential of MOX fuel is an important parameter for adjusting oxygen-to-metal (O/M) ratio in the fabrication process and for evaluating fuel-cladding chemical interaction (FCCI). In this work, oxygen potentials of Am-containing MOX were measured by thermogravimetry, and the effect of the MA (minor actinide) on the MOX fuel was evaluated.

$(Am_{0.024}Pu_{0.311}U_{0.665})O_{2-x}$ was prepared by a mechanical blending method from $(Am,Pu)O_2$ and UO_2 powders. After mixing with a ball mill, the powder was pressed and sintered at 1973 K for 3 h in mixed gas (Ar and 5% H_2) with added moisture. The homogeneity of sample was analyzed with X-ray diffractometer (XRD) and electron probe micro analyzer (EPMA).

The initial weight of sample for the thermal gravimetry measurements was about 200mg. The measurements were carried out at 1473, 1573 and 1623 K. The weight changes correspond to the changes of O/M ratio, and the changes were driven by the oxygen partial pressure (P_{O2}) which was adjusted by the H_2O/H_2 ratio in the annealing atmosphere. The P_{O2} in the atmosphere was measured using stabilized zirconia oxygen sensors. The oxygen potentials, $\Delta\bar{G}_{O2}$, were calculated from the P_{O2} and temperature using the following equation,

$$\Delta\bar{G}_{O2} = RT\ln P_{O2.}$$

The oxygen potentials of $(Am_{0.024}Pu_{0.311}U_{0.665})O_{2-x}$, $(Am_{0.02}Np_{0.02}Pu_{0.30}U_{0.66})O_{2-x}$ [1], $(Pu_{0.2}U_{0.8})O_{2-x}$ [1] and $(Pu_{0.3}U_{0.7})O_{2-x}$ [2] at 1623 K are shown in Fig.1 as a function of the O/M ratio. Systematic data with little scattering are obtained in this work. The oxygen potentials of $(Pu_{0.2}U_{0.8})O_{2-x}$ are lower than those of $(Pu_{0.3}U_{0.7})O_{2-x}$ and data of $(Am_{0.02}Np_{0.02}Pu_{0.3}U_{0.66})O_{2-x}$ are slightly higher than those of $(Pu_{0.3}U_{0.7})O_{2-x}$. $(Am_{0.024}Pu_{0.31}U_{0.65})O_{2-x}$ data are also slightly higher than those of $(Pu_{0.3}U_{0.7})O_{2-x}$. The difference of oxygen potentials between MOX and MA-containing MOX was small. From this work, therefore, it may be concluded that the effect of small MA addition in the range of about 2% on the fuel properties of MOX is not significant.

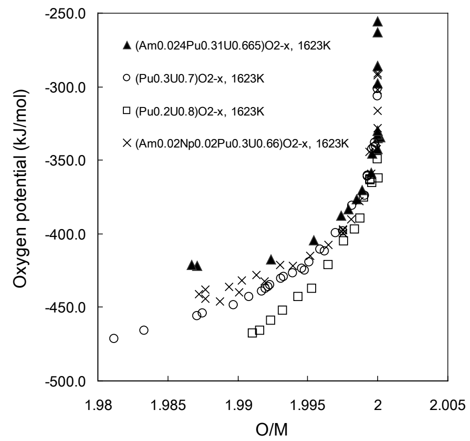

FIG.1 Oxygen potentials of MOX and MA-containing MOX at 1623K as a function of O/M ratio.

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Behavior of (Pu,Si) oxide formed from impurity Si in MOX pellet

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A uranium and plutonium mixed oxide (MOX) fuel containing minor actinides (MAs) has been developed for fast reactors. To reserch an irradiation behavior of MOX fuel containing MAs, an irradiation examination was carried out. In post-irradiation examination, the evidence of MOX-SiO_2 reaction around a central void was detected and a precipitation of Si-compound was observed along the grain boundary[1]. To evaluate the behavior of impurity Si, phase state in MOX-SiO_2 system and PuO_2-SiO_2 system were investigated.

Samples were prepared by mixing powders of 30%Pu-MOX and SiO_2 in molar ratio of 3:1 and mixing powders of PuO_2 and SiO_2 in molar ratios of 3:1, 3:2 and 3:3. In the MOX-SiO_2 examination, samples were annealed at 1973K, 2273K and 2673K as a function of oxygen-to-metal (O/M) ratio. In the PuO_2-SiO_2 examination, samples were annealed at 1623-1993K as a function of O/M ratio. After annealing, these samples were analyzed by ceramography, electron probe micro analyses (EPMA) and x-ray diffraction (XRD) methods.

In the examination in MOX-SiO_2 system, (Pu, Si) compounds were observed along the grain boundaries in the annealed samples, which seemed to be liquid phase at the annealing temperature. The compound tended to form more observably with decreasing O/M ratio and with increasing annealing temperature. Fig 1(a) shows the XRD pattern of the sample, and some extra peaks were observed in addition to peaks from MOX. These compounds were not observed in the grain interior and the MOX matrix was not affected significantly by Si impurity.

Phase state was also investigated in PuO_2-SiO_2 system. Two kinds of compounds $Pu_{4.67}Si_3O_{13}$ and $Pu_2Si_2O_7$ were observed in the annealed PuO_2-SiO_2 samples. The phase of $Pu_{4.67}Si_3O_{13}$ was precipitated in wide composition range as compared with another one. The single phase of their compounds were obtained and measured by XRD. (See Fig.1 (b) and (c)) The extra peaks in the pattern of MOX-SiO_2 sample corresponded to ones from $Pu_{4.67}Si_3O_{13}$ compounds.

In summary, (Pu,Si) compound which was obtained in MOX-SiO_2 system was identified as $Pu_{4.67}Si_3O_{13}$. The effect of impurity Si on the fuel properties was evaluated.

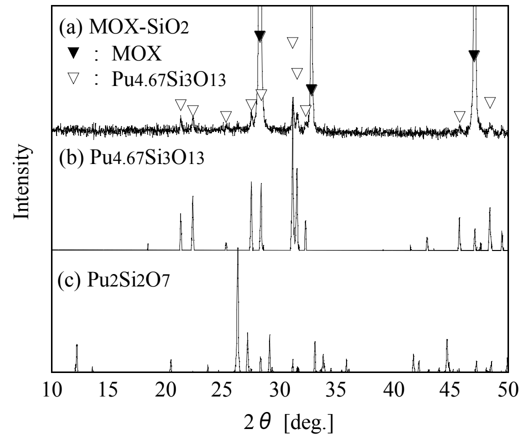

FIG. 1. XRD patterns of the sample of MOX-SiO_2, $Pu_{4.67}Si_3O_{13}$ and $Pu_2Si_2O_7$

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The influence of Pu-content on thermal conductivities of (U, Pu)O_2 solid solution

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Plutonium and uranium mixed oxide (MOX) fuels containing Pu from 20 to 35 % in heavy metal have been developed for fast reactors. During the fast reactor operation, a large thermal gradient from the fuel center to the fuel periphery is generated. The redistributions of Pu and U in fuel pellets used in fast reactors are caused by this thermal gradient, and it is estimated that Pu-content at the pellet center increases to about 40 % [1]. Consequently, thermal conductivity of nuclear fuel is a one of the most important physical properties for fuel design and performance analysis of fuel rods.

Stoichiometric MOX specimens containing Pu from 20 to 40 % were prepared by conventional powder technology and the Pu-content dependency on thermal conductivity was investigated. Thermal conductivity (λ) was obtained from thermal diffusivity (α), heat capacity (C_p) and density (ρ) by the following equation, $\lambda = \alpha \cdot C_p \cdot \rho$. The thermal diffusivities of specimens were measured at temperatures from 980 to 1860 K with a laser flash apparatus. The heat capacities of specimens were estimated from the heat capacities of UO_2 and PuO_2 by using Kopp's law. The densities of specimens were measured by the immersion method.

The results of thermal conductivity measurements of these specimens showed hardly influence from Pu-content on thermal conductivity of MOX in the range from 20 to 40 % (Fig.1) and these thermal conductivities were only slightly lower than that of UO_2. Duriez et al. [2] showed that the Pu-content did not affect the thermal conductivity when the difference of Pu-content was small (about 10 %). Some researchers [3-5] have studied the thermal conductivities of U oxide and MOX, and some levels of Pu-content dependencies on thermal conductivity were reported. But the influences of Pu-content shown in each report were significantly different because the discrepancy in the experimental results was large when they were compared comprehensively.

The Pu-content dependency on thermal conductivity of stoichiometric MOX specimens was investigated. It was found that hardly influence from Pu-content on thermal conductivity of MOX was observed in the range from 20 to 40 % and the dependency trend in the experimental results was confirmed by using theoretical analysis.

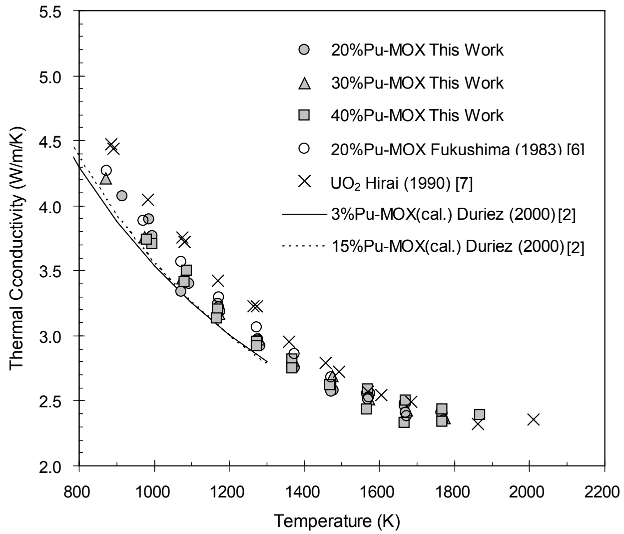

FIG. 1. Comparison of temperature dependency on thermal conductivities of (U, Pu)O_2. The thermal conductivity results were normalized to those on 100 % of the theoretical density.

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Property changes and thermal recovery in self- irradiated MOX

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In plutonium compounds, self-irradiation induces lattice defects with storage time. Properties such as the lattice parameter and thermal conductivity are changed by accumulation of lattice defects, and the changed properties are recovered by annealing. In this study, the effects of self-irradiation and annealing on properties of mixed oxides (MOX) were investigated by using MOX pellets which had undergone long-term storage in the ambient atomosphere. The maximum storage period was 32 years. Plutonium contents in the MOX samples were 20-48 %. Properties investigated were lattice parameters, densities, microstructures and thermal conductivities.

The lattice parameter increased with amount of self-irradiation and became saturated after increasing 0.29%. The expansion of lattice parameter was formulated as a function of the self-irradiation amount:

$$\Delta a / a_0 = 2.9 \times 10^{-3} \times (1 - \exp(-12200\lambda' t)).$$

Here, λ' is the average of decay constants of isotopes, and t is the storage time. The 0.29% expansion of lattice parameter corresponded approximately to a 1% density decrease. The expanded lattice parameter was recovered to those of as-fabricated pellet by annealing at 1473K for 2h because of recoveries of lattice defects. The thermal recovery were analyzed from a relationship between recovery rate and annealing time. From the results, the recovery rates of lattice parameter had three stages, which occurred in the temperature range of below 673 K, 673 – 1073 K and over 1073 K.

The changes of densities due to annealing were measured at 1073, 1473K, 1673K and 1873K for 2h. The change of densities indicated that up to 1473K densities increased more with annealing temperature. The highest densities were found after annealing at 1473K, and the densities varied within the range of 1% corresponding to density decrease calculated from the expanded lattice parameter. The density decreased with annealing temperature above 1473K. After annealing at 1873K, the densities decreased about 1%. It seemed that the interstitial helium atoms migrated to pores due to the annealing, and the pores were expanded by the helium gas. Ronchi and Hiernaut [1] reported that 1073K had been reported as the starting temperature of helium gas release. However, as Fig. 1 shows, no significant microstructure changes of annealed samples could be observed.

Regarding to the thermal conductivity of the annealed MOX samples, three recovery stages were seen, similar to the lattice parameter recovery. After annealing at 1000K, thermal

conductivity was recovered to 80% of the value of the as-fabricated pellet. Thermal conductivity was almost completely recovered by annealing at 1473 K for 30 min.

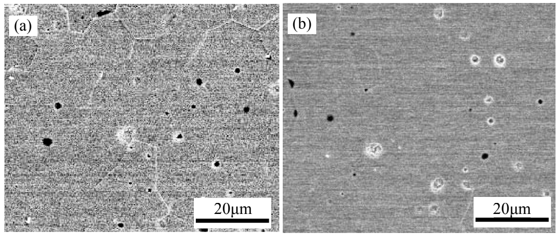

FIG. 1. Microstructures of samples: after storage (a) and after annealing at 1473K (b).

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Evaluation of diffusion behavior of actinide dioxide by molecular dynamics simulation

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Minor actinides (MA: Np, Am, Cm) are key elements in a nuclear fuel cycle because some of them have long half-lives and high radiotoxicity. MA-containing fuels are, therefore, anticipated to be burned as advanced nuclear fuels in fast reactors or transmutation systems in order to reduce the environmental burden and to efficiently use the repository. For the safety management of fast reactor fuels, it is very important to deeply underastand physico-chemical properties of MA-MOX (MA-containing mixed oxide).

Diffusion behaviors of atoms in nuclear fuels are related to various phenomena such as redistribution of actinides and oxygen, distribution of FPs, generation and annihilation of defects induced by irradiation. In order to systematically investigate diffusion behaviors, there are some difficulties in experimental studies especially for actinides. The purpose of this study is, therefore, to evaluate diffusion bihaviors of actinides as well as oxygen by molecular dynamics (MD) simulation.

In this study, we used the MXDORTO program which was developed by Kawamura [1]. And the Born–Mayer–Huggins potential with the partially ionic model was applied to each ion pair in simulated crystals. This potential function is given by,

$$U_{ij}(r_{ij}) = \frac{z_i z_j e^2}{r_{ij}} + f_0(b_i + b_j)\exp\left(\frac{a_i + a_j - r_{ij}}{b_i + b_j}\right) - \frac{c_i c_j}{r_{ij}^{\ 6}} . \qquad (1)$$

The parameters (*a*, *b*, and *c*) for oxygen ion were given by Inaba [2] and the parameters for actinides were determined by Arima [3]. MD calculations were performed for fluorite systems, i.e. UO_2, PuO_2, AmO_2, $(U,Pu)O_2$ and $(U,Am)O_2$, consisting of 500 cations and 1000 anions, in NPT ensemble from 2800 to 4800 K. In addition to actinide dioxides without defects, MD calculations were performed for that with 1-50 Shottky defects especially for activation of cation diffusion.

Figures 1 (a) and (b) show the temperature dependence of self-diffusion coeffeicient of Am and O in AmO_2, respectively. The diffusion coefficients of Am and O increase with an increase of temperature. As inceasing Shottky defects, the diffusion coefficient of Am increases up to melting point, whereas that of O increases with Shottky defects at low temeratures and almost independent on Shottky defects at high temperatures.

MD results obtained from other dioxides and detail discussion on diffusion mechanism of cation and oxygen will be presented in FR09.

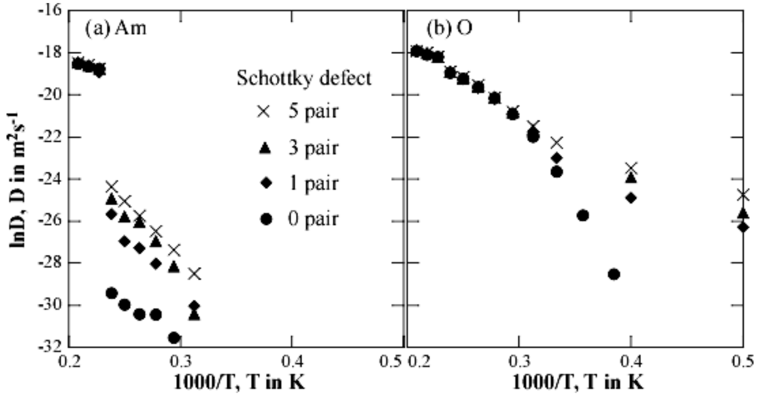

Fig.1. Diffusion coeffients of (a) Am and (b) O in AmO_2.

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Granulation Technology for the Simplified Pellet Fabrication Process

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Japan Atomic Energy Agency (JAEA) has been developing the simplified pellet fabrication process as an innovative technology for the fuel cycle of the first reactor (FR) [1]. This process is composed of 3 stages, (1) to adjust the concentration of Pu in the Pu/U mixed nitrate solution, (2) to convert the Pu/U nitrate solution to the MOX powder, and (3) to fabricate the MOX fuel pellet. An innovative granulation technology for the stage (2) is ongoing with the intent of obtaining satisfactory flowability of the MOX powder produced via de-nitration conversion by microwave heating (MH), calcination, reduction, and crushing. In order to prevent the scatter of the fine MOX powder generated in the stage (2), it is preferable that all operations included in this stage, the de-nitration conversion by MH, the calcination, the reduction, crushing, and the granulation are carried out in a common vessel made of SiN. In this research, a specially designed agitation granulator was employed in order to examine the feasibility of the granulation in a common vessel, utilizing WO_3 and UO_2 powders. Both examinations were successful, and the performance of the particles produced, size and flowability, were checked.

1. Specification of agitation granulator

Figure 1 shows the construction of agitation granulator which was specially designed, being used in this experiment. It is composed of the powder container made of SiN whose inner diameter is 200 mm, and 40 mm in depth. The upper vessel is the lid coupling with the powder container, making it easy to attach and detach the powder container. The total volume of the powder container and the upper vessel is 3.5 L. This machine has 2 types of rotary blades, mixing blade and auxiliary blade. The former advances the granulation, whereas the latter suppresses the over granulation by chopping larger particles. Both blades are individually fixed by different shaft and rotate in different speed. The powder container is available updown shift. To make the blades contact with the powder, the powder container is shifted up and attached to the upper vessel when granulation runs.

2. Results of granulation experiments

The granulation experiments using the agitation granulator were carried out employing WO_3 powder as well as UO_2 powder. The binder was water. The yield of moderate size (100-1000 μm) of WO_3 particles exhibited the highest 43 % at the conditions of 500 rpm of mixing blade, 13 % of moisture, and 1 minute running. The Carr's flowability index marked 82 which greatly overs the ordinary target 60.

On the other hand, in case of UO_2 particles, the yield reached 98 % at the conditions of 500 rpm, 13 % of moisture, and 2 minutes running. The flowability index exceeded 90 which greatly overs the ordinary target 60.

3. Experiments adjusting size distribution of UO_2 granules

Experiments to adjust the size distribution of UO_2 granules were carried out using the particle standardizer (Nebulasizer, Nara Machinery Co.,Ltd). The particle standardizer receives coarse UO_2 granules, shaves them through crashing by rotating pins, resulting in particles of appropriate size. It was found through this experiment that slow rotation of the pins is preferable to obtain a high flowability.

4. UO_2 pellet fabrication experiment

The UO_2 pellets were fabricated employing the granules obtained in the experiment (3). The theoretical density of the sintered pellets exceeded 95 %.

5. Conclusions

From these experimental results, it was demonstrated that the agitation granulator works well and produces the UO_2 granules of the appropriate size in high yield. The scaling up of the machine is underway.

(a) Situation that container is detached *(b) Situation that container is attached*

Figure 1 Photographs of specially designed agitation granulator

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Development of innovative system and technology on MOX fuel production for FBR

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Japan Atomic Energy Agency (JAEA) has been engaging the development of MOX fuel to be burned in the FBR in order to efficiently use the limited amount of natural resources. The fundamental technology needed to this near reaching purpose was accomplished through previous two projects, so called Phase 2, and FaCT [1]. Urgently needed subject is its scaling up from experimental scale to industry scale and giving a guarantee on economical profit on the MOX fuel production. It is eagerly required to surely accomplish this requirement until 2015.

In this paper, the system of MOX fuel production based on a microwave heating (MH), simplification of traditional processes piled up through many years' experiments, and update approach for upgrading of simplification of our production system are described. Future plan for promoting this strategy is also introduced.

1. Existing system of pellet fabrication

Figure 1 demonstrates the entire system of MOX pellet fabrication; (a) is the current system, (b) an innovative apparatus newly considered, being a forceful candidate for simplifying the system, (c) sectional side view of press for pressing MOX particles into a die, and (d) hollow type MOX pellet obtained through the press and sintering.

In case of current system, Pu/U mixed nitrate solution and Uranium nitrate solution are individually de-nitrated by MH, resulting in MOX powder and UO_2 powder. The adjustment of final ratio of PuO_2/UO_2 is carried out by mixing MOX powder and UO_2 powder so as to satisfy the requested value offered from pellet specifications. After that the powder is pressed into a die, sintered, being fabricated a pellet. In these processes, diffusion of MOX dust is the biggest problem which occurs in the route of powder transfer. In addition, the number of process is so many, complicated, and blocking the reduction of production cost. Thus, the prevention of powder diffusion, rationalization of production processes, improvement of economic efficiency, and so on are the subject of present urgency.

2. Rationalization of system and future target

Fig.1 (b) shows a unique apparatus combining three processing (de-nitration, crushing and granulation) within one globe box (GB) by employing only one tray for common use. Considerable stages in Fig.1 (a) can be reduced by means of this new apparatus. We are now attacking the subject to scale up of this new method so as to satisfy the high speed, mass production of MOX powder and particles.

3. Elucidation of de-nitration mechanism

According to our experiments [2], if a high power microwave is applied to the Uranium nitrate solution to advance the high speed de-nitration, spouting occurred. Due to the spouting, Uranium nitrate solution overflowed from the tray. In order to prevent the overflow, fundamental research on the mechanism of spouting which is generated by the microwave heating (MH) is ongoing by employing water. Up to present, the specific effect of the microwave on liquid heating has been clarified.

4. Introduction of computer simulation technology

The computer simulation on the electromagnetic distribution inside a high power microwave oven is advancing from the point of view to improve the heating efficiency. The computer simulation on behavior of particles which was developed in the ceramics industry is applied to our MOX powder field to improve the yield constant by optimization of production conditions.

In this paper, we will totally report our update results and conclusions.

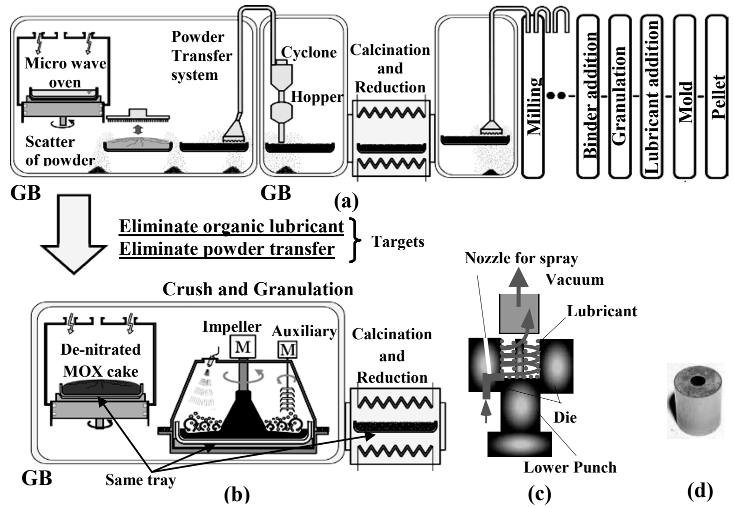

Fig.1 System and innovative apparatus for MOX fuel fabrication;
(a) current system, (b) innovative apparatus,
(c) press of MOX particles into die, (d) product of hollowed pellet.

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Fabrication of MOX Fuel elements for irradiation in Fast Breeder Test Reactor (FBTR)

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Advanced Fuel Fabrication Facility (AFFF), Bhabha Atomic Research Centre, Tarapur is fabricating Uranium – Plutonium Mixed Oxide Fuel (MOX) for different types of reactors. Recently an experimental fuel subassembly of 37 pins has been fabricated for irradiation in Fast Breeder Test Reactor at Kalpakkam near Chennai. MOX fuel pins containing 45% PUO_2 have also been also made for the hybrid core of FBTR.

The experimental sub-assembly for irradiation testing in FBTR cosists of 37 short length PFBR MOX fuel elements. The composition of the fuel was (0.71 U – 0.29 Pu) O_2 with U_{233} O_2 content of 53.5% of total UO_2. Uranium enriched with U^{233} was used to stimulate the heat flux of PFBR in FBTR neutron spectrum. MOX fuel pellets were made by powder metallurgy process consisting of pre-compaction, granulation, final compaction and sintering at high temperature. Initially U_3^{233} O_8 / U^{233} O_3 powder was subjected to heat treatment. MOX powder were mixed, milled, pre-compacted and granulated. The final compaction was done using a multistation rotary press with suitable tooling for making annular MOX pellets. The technology for making annular pellets was developed for this purpose. The pellets were sintered at reducing atmosphere at 1650^0 C for 4 hours to obtain acceptable quality pellets. Over sized pellets were centrelessly ground.without using a liquid coolant.

The acceptable pellets were degassed before encapsulation. MOX fuel stack, UO_2 insulation pellets, plenum spring and spring support were loaded in bottom endplug welded clad tube. The end plug welding was carried out by TIG welding technique. The welded elements after inspection were wire wrapped. During the fabrication of pins for experimental subassembly, technology was developed and conditions were optimized for making annular pellets, TIG welding of D9 tubes with SS 316 end plugs and wire wrapping. Quality control procedures and process control procedures at different stages of fabrication were developed. The experimental fuel pin assembly is being irradiated at FBTR and has seen a burn-up of 82,000 MWd/t. The assembly has been designed for operating at a peak linear heat rating of 450 watt/cm and a target peak burn-up of 1,00,000 MWd/t. During the fabrication, shielding was provided around the material handling containers and glove boxe panels to minimize personal exposure since the fuel contains U^{232} in ppm quantities.

The hybrid core of FBTR consists of Mixed Carbide (MC) sub-assemblies containing (0.70 PU – 0.30 Uranium) C pellets and MOX fuel sub-assemblies containing (0.44 PU – 0.56 U) O_2. Studies were made to fabricate fuel containing higher percentage of Plutonium and the conditions were established.. Separate studies have been conducted about the compatibility of fuel with the Sodium coolant and reported elsewhere. Advanced Fuel Fabrication Facility has fabricated short length MOX fuel pins containing 44% PUO_2 for making a number of assemblies. The process flow sheet , fabrication and quanlity control procedures were developed . A few MOX sub-assemblies are at present undergoing irradiation at FBTR.

This paper describes the development of flowsheet for making annular MOX fuel pellets containing plutonium and U^{233} , the technology for welding of D-9 clad tubes, wire wrapping and inspection. The paper also highlights the innovative techniques developed for the fabrication and quality control of the MOX fuel produced for the experimental irradiation at FBTR.

Research and development of fast reactor metal fuel by CRIEPI

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Considering great potential of U-Pu-Zr metal fuel as fast reactor fuel for low-cost and secure electricity supply in the future, Central Research Institute of Electric Power Industry (CRIEPI) has been conducting research and development of metal fuel since 1986 to understand irradiation behavior up to high burnup and fuel slug casting performance and to evaluate the applicability of metal fuel to commercial fast reactors. For these purposes, the research and development of metal fuel by CRIEPI covers irradiation performance analyses up to high burnup, experimental assessments on fuel-cladding compatibility, injection casting tests with U-Zr and U-Pu-Zr alloys, an irradiation test of U-Pu-Zr fuel, and an irradiation test of U-Pu-Zr fuel containing minor actinides (MA), as follows.

Irradiation behavior models for metal fuel were devised after the survey of physical properties and irradiation test data of U-base fuel alloys. The models were combined into an irradiation behavior analysis code, ALFUS. The analyses of reported irradiation test data with the ALFUS code showed that at higher burnup than ~15 at.% the accumulation of solid fission products may cause fuel-cladding mechanical interaction (FCMI) in the metal fuel of ~75% fuel smear density.

Above a certain threshold temperature, a liquid phase forms in the reaction zone at the interface between the U-Pu-Zr fuel alloy and Fe-base cladding. Since the liquid phase formation enhances the rate of cladding wastage, the peak cladding temperature of metal fuel should be limited below the threshold temperature. To understand the liquefaction phenomena and determine the liquefaction condition, isothermal reaction-diffusion tests were conducted using the diffusion couples such as U-Zr/Fe, U-Zr-Ce/Fe, U-Zr/Fe-Cr, U-Pu-Zr/Fe and U-Pu/Fe. The results of the series of tests indicated that the liquid phase forms above 923 K in the case that the Pu/(U+Pu) ratio in the U-Pu-Zr fuel alloy is lower than 0.25. Phase diagrams of U-Zr-Fe, U-Pu-Zr and U-Pu-Zr-Fe systems were determined from metallographic examinations and/or thermodynamic assessments and used for identification of the phases in the reaction zones in those diffusion couples.

As a result of an engineering-scale U-Zr injection casting test, more than 500 U-10 wt.%Zr slugs of allowable quality were produced and optimum casting parameters for U-Zr injection casting were understood. An injection casting simulation code, ICAST, was developed in collaboration with Kobe Steel, Ltd. to examine the casting performance. The analysis of the U-Zr injection casting with ICAST revealed that coating inside the silica mold was essential to obtain sufficiently long fuel slugs.

An irradiation test of metal fuel in the experimental fast reactor JOYO has been planned in collaboration with Japan Atomic Energy Agency (JAEA) to demonstrate the capability to attain high burnup together with high peak cladding temperature. The targets of the test are (1) to confirm no liquid phase formation at the interface between the fuel slug and cladding at 923 K, (2) to obtain the data of cladding wastage by rare earth fission products (FCCI: fuel-

cladding chemical interaction) at the peak cladding inner temperature of > 873 K, (3) to obtain FCMI data at the peak burnup of >15 at.% or equivalent, and (4) to acquire the first irradiation experience of the metal fuel made in Japan. The methodology of linear-power-to-melting was elaborated for the design of test fuel pins. The thermodynamic assessment of U-Pu-Zr system was useful in preparing the U-Pu-Zr solidus temperature correlation for metal fuel pin design. The U-Pu-Zr fuel slugs for the test fuel pins were successfully manufactured on the basis of the experience of U-Zr slug casting.

CRIEPI has been conducting research and development on the metal fuel containing the minor actinides (MA) since 1986. Based on the results of characterization and property measurements of MA-containing alloys, U-19Pu-10Zr fuel alloys containing up to 5 wt.% MAs were fabricated in collaboration with Joint Research Centre, Institute for Transuranium Elements (JRC-ITU). These fuel alloys were irradiated in the Phenix reactor with the support of the Commissariat à l'Énergie Atomique (CEA), and attained the maximum burnup 10 at.%. Post-irradiation examinations of the irradiated fuel alloys are now in progress in JRC-ITU.

In addition to the above activities, interdiffusivity in U-Zr alloys and diffusivity of Ce in U-Zr-Ce alloys were measured in collaboration with Japan Atomic Energy Research Institute (presently, JAEA). The measured data will be used in the future for modeling of migrations of fuel constituents and fission products.

Further development of the metal fuel applicable to commercial fast reactors in Japan will require promotion of the irradiation tests under practical conditions and the facilities for test fuel fabrication. Fundamental research and development will also be essential, such as FCCI modeling, measurements of physical properties and irradiation behavior analyses.

Development of Metallic Fuels for Indian Fast Breeder Reactors

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Abstract

The neutronic performance of metal fuel based on binary U-Pu alloy or ternary U-Pu-Zr alloys are better than conventional uranium plutonium mixed oxide or high density carbide ceramic fuel. The growing energy demand in India needs faster growth of nuclear power and warrants introduction of fast reactors based on metallic fuels in future [1]. Physics calculation [2] showed that fast reactor based on metallic fuels results in higher breeding ratio and lower doubling time compare to mixed oxide or carbide fuels. Moreover inclusion of pyro-processing of the fuel in the fuel cycle is expected to make metal fuel option more economical.

As part of metal fuel development programme for future FBR's in India, capsule irradiation of metal fuel based on sodium bonded U-Pu-Zr alloy and metal (Zircaloy) bonded binary U-Pu (Pu ~ 15 %) alloy are being actively pursued. For this purpose two design concepts have been proposed : one based on sodium bonded ternary alloy fuel of U-Pu-Zr (2-10 wt%) in modified T91 cladding material and the other is U-Pu binary alloy mechanically bonded to modified T91 cladding material with 'Zircaloy ' as a liner between the fuel alloy and the clad. The Zircaloy liner act as a barrier in reducing the fuel clad chemical interaction. It also helps in transfer of heat from the fuel to the clad. The smear density of metal bonded pin will be between 70% - 85% and that for sodium bonded pin will be ~ 70%. In metal bonded fuel pin design two/four semi-circular grooves of diameter ~1.0 mm, will be provided in diametrically opposite directions in the fuel cross section to accommodate fuel swelling. A comparison has been made on the relative merits and demerits of these two fuel pin designs. The material for the axial blanket will be 'U' or U-Zr (Zr upto 10wt %) alloy based on the results of the out-of-pile thermal cycling behavior and irradiation performance.

In the present investigation out-of-pile experiments have been carried out to address some of the issues of the metallic fuel and the proposed design of the pin. Measurement of axial and radial expansion of the fuel, axial blanket, Zircaloy liner and T91 cladding tubes under isothermal and cyclic heating condition were carried out in a dilatometer to study thermal stability and integrity of these components under simulated reactor condition. The eutectic reaction temperature between T91 and Uranium were estimated by dilatometry, differential thermal analysis and high temperature microscopy. Diffusion couple experiments were also carried out between U/Zr/T91 and U/T91 by isothermal annealing of the couples between 550^{o}C to 750^{o}C for times up to 1500hrs. to find out the extent of chemical interaction. These studies were supported by metallographic examination, micro hardness measurement, XRD, SEM/EDAX and EPMA. High temperature (ambient to 900^{o}C) hardness measurements were also carried out on the U/U-Zr, T91 cladding material and Zircaloy liner to estimate their plasticity. The data will predict the extent of mechanical interaction between these components when they are exposed to thermal

and irradiation induced swelling stresses. Similar set of out-of-pile experiments have been planned and are being pursued for Uranium metal containing 2wt%, 6wt% and 10wt% Zr. The results of these studies are expected shortly and will also be presented. The temperature distribution across the cross-section of the fuel pin under the reactor operating condition have been estimated to predict the limiting linear heat rating (LHR) of this fuel design so that there will not be any centre melting of the fuel and the fuel-clad interface will not reach the eutectic reaction temperature. The paper highlights the results of these studies and attempts to analyze them in the light of performance. The outcome of all these studies as such has been useful to the fuel designer in optimizing the design features and predicting the in-reactor fuel behavior.

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U-Pu-Zr metalic fuel core and fuel concept for SFR with 550deg.C core outlet temperature

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The primary interest of SFR design core and fuel study in JAEA is to achieve 550°C of core outlet temperature. The U-Pu-Zr metal fuel is an atractive fuel of SFR, which realizes the superior neutronic characteristic of the core due to its high content of heavy metal nuclide with appropriate irradiation experience of Zr alloy metal fuel. It is well understood that U-Pu-Zr base metal fuel has its major drawback as steel cladding temperature limiting feature due to fuel cladding compatibility concerns. Its cladding inner surface maximum temperature is to be limited as 650°C to avoid the liquid phase formation in the fuel during steady state operation due to the inter-diffusion of elements (atoms) in the cladding and fuel. It is inevitable for the core thermal hydraulic design to give cladding maximun temperature in the core higher than core outlet temperature. A typical example of oxide fuel core design has 550°C of core outlet temperature with 700°C of cladding maximum temperature with uncertainties to be considered in the engineering design. Therefore, a core and fuel design with 550°C of core outlet temperature and with 650°C of cladding maximum temperature is siginificant challenge of SFR metallic fuel core study.

Such core has been designed by significantly reducing the metalic fuel pin power variation during the irradiation. The core concept has single Pu enrichment which acheves local nuetronic conversion rataio as close as 1.0, which leads to stable fuel pin power distribution in the core and stable fuel pin power history of each fuel pin. An example of 1,785MWt core which cooresponds to 750MWe is as follows:

Fuel pin diameter : 7.5 to 8.5 mm
Core fuel column length : 900 tp 1000mm
Pu enrichment : 12 %
Zr content 6 to 10wt.%
Fuel smeared density : 70 to 75 %TD
Reactor operation cycle length : 24-26 months
Burnup reactivity swing : 0.5 % del k/kk'
Breeding ratio : 1.0 (without blanket)
Fuel average burnup : 90 GWd/t

The results of core thermal hydraulic design study show that 550°C of core outlet temperature is achievable with 650°C of cladding maximum temperature. Fuel design feasibility has been also evaluated by analytical design work based on fuel design collerations of various properties and behaviors such as cladding internal corrosion due to metalic fuel – cladding chemical interaction, fuel pin gas plenum volume change due to fuel swelling and claddign creep rupture strength.

The feasibility of 650°C of cladding maximum temperature was discussed based on the out-of-pile metallic fuel-cladding compatibility tests. Further experimental evaluation is planed

as metalic fuel pin irradiation test in Joyo of JAEA O-arai R&D center. Test fuel pins have been designed based on design models and collerations which are resonably concervative to be applied to Japanese licensing procedure. They include metalic fuel properties such as melting temperature, thermal conductivity and creep deformation and fuel related behavors such as fuel column elongation and cladding inner surface westage during high temperature transients.

The irradiation test will start with six metallic fuel pins with PNC-FMS cladding tube, which is high strength freeitic-martensitic steel cladding. The target burnups are 3 at.%, 8 at.% and over 10 at.%. Fuel smeared density and peak cladding temperature are parameters. The ODS cladding tubes are expected to be applied to the extended phase of this experiment. Figrue 1 shows a schimatic view of the test fuel pin. The first result will be obtained after a few operatin cycle of Joyo of irradiation, that will be followed by long term steady state irradiation to obtain high burnup data. The irradiation test results will show the feasibility of 650°C of cladding maximum temperature without metalic fuel-cladding compatibility problem and will give a technical background of metalic fuled SFR core concept with 550°C of core outlet temperature.

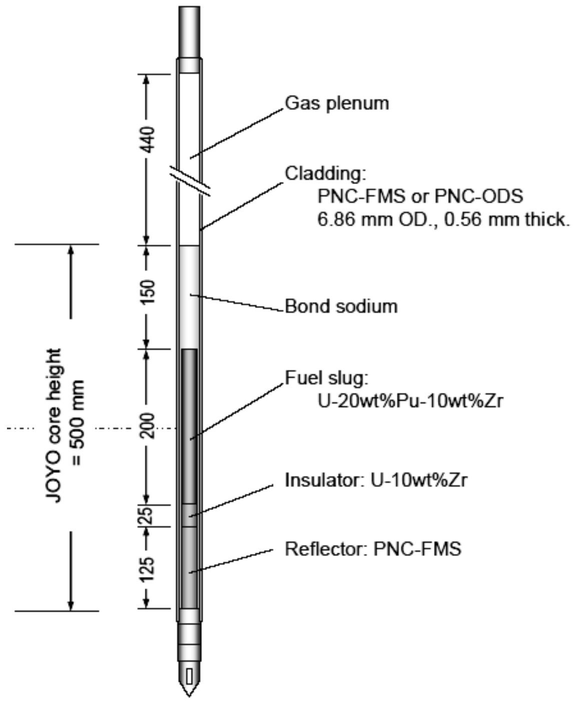

Fig. 2 Metalic fuel pin for Joyo irradiation test

Fuel Design Evaluation of the 4S

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Typical fast reactor fuel residence time is less than 5 years. Extending the life of the fuel beyond this range is an important feature of the battery type (non-refueling) fast reactors. This type of reactors requires operation without refueling for long periods of time between 15 to 30 years, after which the full core is removed from the reactor and sent to disposal or reprocessing with the possibility of replacement with new core. Extending the fuel residence time to such long periods requires an advanced fuel design that mitigates the effects of extended fuel element exposure to different severe conditions which are present in fast reactor environment. Those conditions include high temperature, large irradiation doses, possible interactions between fuel and cladding, and sodium corrosion of the outer cladding surface. One such advanced fuel design is the metallic U-10Zr fuel design proposed for use in the sodium cooled Super-Safe Small and Simple (4S) reactor [1] which is expected to operate without refueling for 30 years.

This paper describes the 4S advanced fuel design and evaluation of its expected performance over a 30 years life time. The first implication of the required long fuel life, from the neutronics prospective, is that the fuel has to be longer and wider than a typical fast reactor fuel. Beginning of life fuel length is 2.5 m compared to typical heights of up to about 1.5 m or less (PRISM design [2] is about 1.34 m while EBR-II and FFTF heights were about 1/3 m and 1 m, respectively). Pin diameter is 14 mm compared to a range between about 4-7 mm for other reactors. Mitigation of the fuel cladding thermal creep due to gas pressure over the long irradiation period is achieved through the use of a plenum region on top of the fuel slug (ratio of plenum to fuel volume is 1.3) and operating at peak cladding temperature that maintain low thermal creep rate (hot channel peak cladding temperature is 609 ^{o}C). Low plenum pressure combined with low average fuel burnup (less than 5 at%) reduce the stresses on the cladding and reduce corresponding creep strains. Lower burnup combined with the use of appropriate fuel smeared density (allowing for enough fuel-cladding gap) eliminate possible concerns regarding fuel cladding mechanical interactions (FCMI). Meanwhile fuel cladding chemical interaction (FCCI) is reduced significantly given the reduction in its driving forces for this particular design. Those driving forces include temperature gradient over the fuel cross section, fission products accumulation, contact period between the fuel and cladding and contact temperature. As shown in Fig.1, the fuel-cladding contact period is reduced at the top of the fuel where cladding temperature is the highest and fuel burnup (that is, available lanthanide fission products) is the lowest. In addition, thermal properties of U-10Zr alloy, low linear power and wider fuel cross section reduce the temperature gradient across the fuel to less than 100 ^{o}C. In addition, the cladding thickness is about double the typical cladding thickness used in EBR-II and FFTF, allowing for further mitigation of FCCI effects. Concern regarding cladding corrosion due to long exposure period between the sodium coolant and cladding outer surface are reduced by controlling the oxygen content in the sodium to levels that are know to limit this phenomenon (e.g., oxygen limits used in EBR-II, where some of the blanket fuel elements remained in the core for the full reactor life time of about 30 years without showing corrosion related problems). Finally, issues related to constituents' redistribution in metallic U-Zr fuel are eliminated based on analysis of the phenomenon for the 4S fuel where it is shown that the redistribution is not expected to take place under the 4S operating conditions.

Design criteria are established for evaluation of the 4S fuel performance, which are based on past experiences from different fast reactor programs. The LIFE-METAL [3] fuel performance code was used to evaluate the performance of the 4S fuel design given those design criteria. The performance evaluation shows the design meets those pre-set design criteria. Sensitivity analysis was performed to look at the fuel performance at conditions that are beyond expected operating conditions, that is higher cladding temperature and fuel swelling behaviour that increases FCCI. The sensitivity study shows that none of the design criteria are violated under those extreme conditions.

The paper includes discussion of the experimental metallic fuel database and extrapolation of the database to the 4S fuel characteristics and operating conditions. Also included are discussions related to licensing of the 4S reactor in the U.S., and interactions with the U.S. Nuclear Regulatory Commission on issues related to this fuel design and previous fast reactor designs in the U.S.

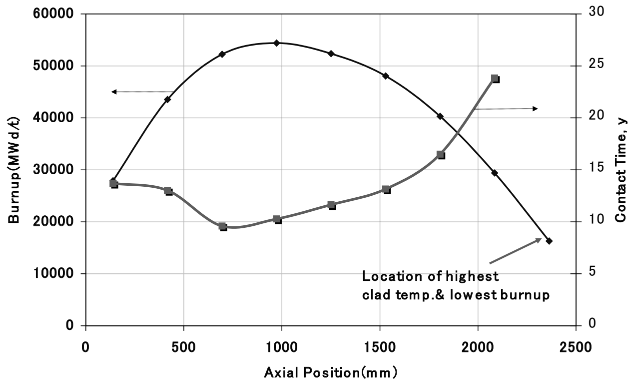

Fig 1. Axial Variations in Burnup and Contact Time between the Fuel and Cladding at Different Axial Locations

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OXIDE-METAL CORES - STAGE OF CONVERSION TO THE METAL FUEL CORE FOR THE FAST REACTORS OF THE BN-TYPE

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Radiation-thermal properties of uranium-containing metal fuel were studied, that influence reliability parameters of blanket and fuel elements for BN reactors. Specifically, the following was investigated: macroeffects of radiation growth of metal fuel column in steel claddings; macroeffects of radiation swelling of uranium and U alloys in free states and steel claddings; and also macroeffects of physical chemical interaction between uranium/uranium-plutonium alloys and steel claddings.

Dependences of these macroeffects were established on types and mass content of doped additives (0,003…40 wt.% Me, MeO…), on irradiation temperature (Tт=100..900^{o}C), burnup factor (B ≤ 10,4% h.a.), burnup rate ($1\cdot 10^{13}$ $^{fissions.}/_{cm^3\cdot sec}$ ≤ ω ≤ $12\cdot10^{13}$ $^{fissions.}/_{cm^3\cdot sec}$), design strength of steel claddings (δ/d = 0,03…0,11), cladding temperature under irradiation (T_{clad}^{max} ≤750^{o}C) and postirradiation emergency overheatings $T_т^{max} = T_{clad}^{max}$ ≤ 900^{o}C), smear fuel density in fuel mockups and full-sized elements (γ_{eff}=12…18 $^{g\ h.a.}/_{cm^3}$), on types and thickness of antidiffusion protective layers on the border between metal fuel column and steel cladding (nonmetal and metal layers of 5...40μm in thickness).

The established dependences were used for development and fabrication of experimental metal uranium fuel pins that were intended for operational conditions and parameters in heterogeneous oxide-metal cores of various types: BFAH (by FA heterogenization), IFAH (intra FA heterogenization), IFPH (intra fuel pin heterogenization).

Main features of the fuel pins:

- claddings- made of austenitic steel EI-847 of standard type-dimensions: d×δ= 6,0×0,3mm; 6,9×0,4 (0,5)mm;14,5×0,45mm;
- filling of the fuel pins - He;
- fuel columns - alloy-free uranium (depleted, natural, enriched, with additives of up to 8%wt plutonium)
- smear density of fuel - 12…18 $^{g\ h.a.}/_{cm^3}$, design similarity to the standard oxide fuel pins applied in the BOR-60, BN-350, BN-600 reactors, etc.

The fuel pins incorporated in the experimental FAs were irradiated in different zones of the BOR-60 and BN-350 reactors (the core, radial blanket, axial blankets). The experimental FAs are similar to the standard FA in geometric and operational parameters. Thus, the original requirement was met of interchangeability of standard and experimental fuel pins and FAs in the functioning and being developed BN-reactors. In total about 50 full-sized FAs of various types and purposes were tested in BOR-60 and BN-350 under standard operational conditions.

Maximum admissible fluence of fast neutrons on claddings made of steel EI-847 ($1,1\cdot10^{23}$n ⁄cm^2 ⇒ ~ 55 dpa) was total limitation for irradiation period of standard and experimental fuel pins.
It follows from postirradiation examinations, that peculiar macroeffects of radiation growth and radiation swelling of alloy-free metal fuel, and also its physical-chemical interaction with steel claddings still allow this fuel to be used in heterogeneous cores of the BN-type: BFAH, IFAH, IFPH

The obtained results were used to validate possibilities of stepwise (according to the selected plan) replacement of standard oxide fuel pins by similar fuel pins with metal uranium of high smear density (γ_{eff}≥ 13 $^{g\ h.a.}/_{cm^3}$) for conversion to heterogeneous oxide-metal core of the selected type (BFAH, IFAH, IFPH).

BN-reactors with such cores are characterized by internal breeding ratio BR=1 and burnup of oxide fuel pins may be increased by about 25%.

Fuel pins containing columns either of U-8% Pu alloy (simulators of the maximum plutonium accumulation in feertile fuel pins) or U-15 % Pu (experimental fuel pins) were developed, fabricated and tested in the BOR-60 reactor. Postirradiation examinations of these fuel pins show radiation-thermal effects, which are similar to those in uranium fuel pins.

Calculation investigations found no significant differences in power generation and reliability of $UPuO_2$ fuel pins (γ_{eff} = 9 g ⁄ cm^3 ⇒ 8 $^{g\ h.a.}/_{cm^3}$), doped U-Pu-10 Zr metal fuel pins (γ_{eff} = 11,8 g ⁄ cm^3 ⇒ 10,7 $^{g\ h.a.}/_{cm^3}$) and alloy-free U-Pu fuel pins (γ_{eff} = 13 $^{g\ h.a.}/_{cm^3}$), that are similar in geometrical and operational parameters.

The obtained results demonstrate possibility of evolution conversion from BN-reactors with oxide fuel to BN reactors with heterogeneous oxide-metal core of the selected types (BFAH, IFAH, IFPH) and in further to BN reactors with metal fuel.

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Evolution of fission-gas bubble size distributions during high temperature irradiation of uranium-alloy fuel

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Of the various irradiation performance issues of nuclear fuels, the behavior of the noble fission gases, Xe and Kr, is one of the main issues that determine the useful in-reactor life or fissile burnup capability of a particular fuel type and design, especially for the next generation of fast reactors. These gases, being largely insoluble in all forms of fissile material, tend to precipitate into gas bubbles and are the main cause of fuel swelling. The magnitude of swelling and the capability of fission gas to escape from the fuel interior depend primarily on the gas bubble morphology that develops during irradiation. In addition to the main operating parameters, i.e., temperature and fission rate, many minor variables determine fission gas behavior. The physical form of the fuel, fabrication variables, and impurity levels may all affect the evolution of the fission gas bubble morphology.

Swelling of γ-U is predominantly due to the growth of fission gas bubbles. Its fission gas behavior is characterized by high mobility at the relatively high temperatures where it exists as the equilibrium uranium phase. Because of high diffusivities, fission gas bubble swelling rates of metallic U are high even at the relatively low burnup levels. Some degree of swelling rate reduction can be achieved by alloying with elements that have a high degree of solubility in the γ–phase e.g. Mo, Zr and Nb. However, high-burnup performance can only be achieved by releasing the fission gas from the internal porosity.

Analytical solutions to coupled rate equations for the nucleation and growth of intragranular fission-gas bubbles is used to characterize fission gas bubble development in U-Mo, U-Zr, U-Nb, U-Pu-Zr, and U-Pu-Mo alloy fuel at relatively high temperatures (i.e. in the thermal-gamma regime). The goal of the formulation is to avoid a coupled set of nonlinear equations that can only be solved numerically, using instead a simplified, physically reasonable hypothesis that makes the analytical solutions viable [1]. The gas-induced swelling rate is then assessed by calculating the evolution of the bubble population with burn-up and subsequently the amounts of gas in bubbles and lattice sites. Uncertain physical parameters, such as diffusion coefficients, are determined by comparing the calculated bubble-size distributions with measured bubble size and density data. The key premise here is that the basic swelling mechanism is similar in the various uranium alloys in the γ–phase, albeit with different property values.

Let $n(r)dr$ be the number of bubbles per unit volume with radii in the range r to $r+dr$. Growth by gas atom collection via diffusion removes bubbles from this size range, but these are replaced by the simultaneous growth of smaller bubbles. In addition, $n(r)dr$ is affected by bubble-bubble coalescence. A differential growth rate between bubbles of different size leads to a net rate of increase in the concentration of bubbles. This behavior is expressed by

$$\frac{dn(r)}{dt}dr = -\frac{d}{dr}\left[n(r)\frac{dr}{dt}\right]dr - 4\pi\left(r_b + r\right)\left(D_b + D_b^{n(r)}\right)c_b n(r)\,dr \qquad (1)$$

The terms on the right hand side of Eq. (1) describe growth by gas-atom diffusion and bubble coalescence, respectively. Analytical solutions are made tractable by considering bubble coalescence as the interaction between $n(r)$ and the mean values of the bubble distribution, i.e. c_b bubbles having radius r_b, determined within the context of mean value theory. The rate of growth due to gas-atom precipitation and bubble coalescence is controlled by D_g and D_b, the gas-atom and gas-bubble diffusion coefficients, respectively. Equation (1) must be solved subject to the relevant boundary condition. In general, this boundary condition concerns the rate at which bubbles are formed at their nucleation size r_0. The rate of bubble nucleation is provided by a multi-atom nucleation mechanism.

Figure 1 shows the calculated distribution for U-8Mo irradiated at 823-873K to 4% U burnup compared with the data [2]. This fuel experienced high end-of-life constraint. The calculations were made using a self-diffusion coefficient for U diffusion in U-Mo given by

$$D_{\mathrm{vol}}^{\mathrm{U\text{-}Mo}} = 2x10^{-3} e^{-34000/kT} cm^2 s^{-1} \qquad (2)$$

This value for the diffusion coefficient is in reasonable agreement with experimental values [3].

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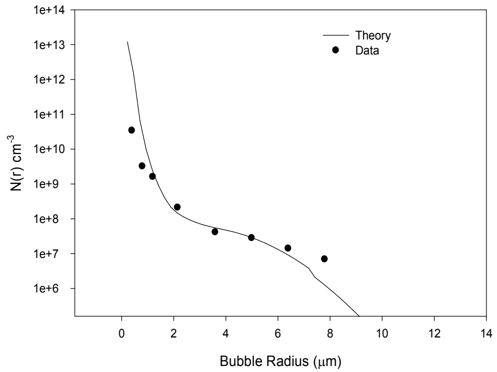

Figure 1 Calculated distribution for U-8Mo irradiated at 823-873K to 4% U burnup compared with the data [2]. This fuel experienced high end-of-life constraint.

An assessment of the Use of U-Pu-Mo Fuel in Fast Reactors

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The use of metallic U-Pu-Mo fuel as an alternative to U-Pu-Zr fuel in fast reactors offers several advantages. This paper presents the results of an assessment study that explores its possible use as an alternative fuel in advanced fast reactors.

A major advantage of the proposed alloy over the ternary U-Pu-Zr alloy stems from its presence as a single phase in the typical range of fuel operation temperatures. Migration of fuel constituents in U-19wt.%Pu-10wt.%Zr (U-19Pu-10Zr) fast reactor metallic fuel, which causes the formation of radial zones having heterogeneous properties, is a characteristic phenomenon that affects its behavior during irradiation. This phenomenon occurs because U-19Pu-10Zr extends over four phase fields in the temperature range 500 – 750 ^{o}C. Two of the phase fields are two-phase mixtures and one comprises three phases (see Fig.1(a)). The solubility of the solute atoms in the solvent changes over the phase fields, so more precipitation occurs in a phase field than the adjoining phase field. This results in a different solute concentration in the continuous phase, leading to constituents diffusion. In contrast, U-22Pu-10Mo alloy has only one phase in the temperature range 550 – 950 ^{o}C, that is, the γ-phase field (see Fig.1(b)), eliminating zone formation. In this single phase, constituent migration is driven only by the thermal gradient, which is smaller than that by a chemical potential difference. By inference, migration of minor actinides (MA) and lanthanides (LA) in the radial temperature gradient across the fuel is lowered, which may have a significant effect on the potential interaction between the fuel and the cladding.

The thermal properties of U-19Pu-10Zr and U-19Pu-10Mo are mostly similar. Some of the advantages for the U-19Pu-10Mo are its higher thermal conductivity and lower thermal expansion compared to U-19Pu-10Zr. In addition, U-19Pu-10Mo has a slightly higher melting point than U-19Pu-10Zr. Both alloys have similar heat capacities. Comparison is made between temperature distributions in both alloys based on their thermal properties.

From EBR-II test results and early fuel development tests, U-Pu-Mo alloy appears to have a slight disadvantage over U-Pu-Zr in terms of cladding compatibility. However, there is no data from a side-by-side test allowing a fair comparison. The better cladding compatibility of the Zr alloy with Fe-base cladding was believed to be fortuitously due to Zr accumulation at the cladding inner surface: This Zr accumulation at the fuel cladding inner surface increases the fuel-cladding eutectic point. In this paper, by comparing heats of formation of binary alloys between fuel constituents and cladding constituents, using the Miedema model, interaction between fuel and cladding for both alloys is predicted.

U-Pu-Mo alloy showed acceptable fuel swelling during irradiation tests performed during the 1950s and 1960s [3,4]. These data are analogous to recent fuel irradiation data that showed excellent fuel performance due mostly to the stable swelling of γ-phase of U-10Mo (see for example Ref.5 and references therein). Exploration of the swelling characteristics of the alloy is presented here using recently developed model for swelling in U-Mo alloys.

The apparent drawback in replacing Zr with Mo in the fuel alloy is that Mo has a higher capture cross section than Zr in the fast spectrum. However, results of neutronics calculations that are presented here for a typical burner fast reactor shows that this disadvantage has small effect on reactor design. More importantly, because of its higher density, the Mo alloy allows for a more compact core for the same power, which is an extra advantage from the core designer standpoint.

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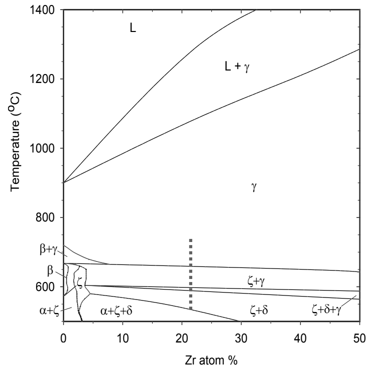

Fig. 1(a) Pseudo-binary phase diagram of (U-19wt.%Pu)-10wt.%Zr [1].

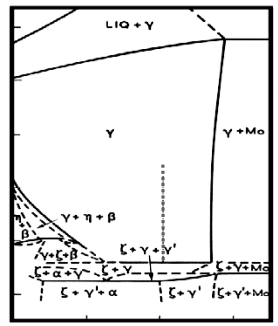

Fig.1(b) Pseudo-binary phase diagram of (U-22wt.%Pu)-10wt.%Mo [2].

On the oxidation of (U,Pu)C Literature survey, experimental and kinetic aspects, practical issues

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Although the UCx fuel oxidation behaviour is far well-known, data on the oxidation of (U,Pu)C fuel are scarce and not always relevant and reliable (Matzke, 1986). Actually, U-Pu-C (M-C, M = (U,Pu)) and M-O are complex ternary systems not fully determined. The average oxidation reaction is described as MCx + O2 -> MOx + C/COx. The reactant carbide is commonly assumed to be a two-phase mixture of MC (90%wt) and M2C3 (10%wt), without any free carbon. The oxidation product is made of two oxides (non adherent, dispersed and pulverulent), namely MO2+x and M3O8 (high temperature phase), which ratio depending on the oxidation temperature, on the Pu content in the reactant, and on the oxidising gas. Assuming the complete oxidation of carbon, the weight gain of the oxidation is between 8 and 12%wt according to the composition of the oxides (MO2 or M3O8). The higher is the oxidation temperature, the less the carbon content in the oxides (Benedict, 1979). Regarding thermodynamic considerations, the equilibrium oxygen pressure is probably very low, actually not commensurable, and the oxidation is energetically very favourable ($\Delta_r G_{Ox}$~-1300 kJ/mol^{-1} (Lewis, 1977; Iyer, 1990)). This induces a highly exothermic behaviour of this oxidation. Sintered carbide samples follow a linear oxidation behaviour in isothermal condition in air or oxygen (Novoselov, 1982) as well as in carbon dioxide (Matzke, 1986). The oxidation rate apparently depends, in a pseudo affine relationship, on the oxygen content, and increases slightly with temperature (Iyer, 1990) in different oxygen mixtures, or more roughly in carbon dioxide. The mechanisms of oxidation are neither detailed nor discussed even if comparisons can sometimes be made with the oxidation behaviour of UCx. In carbon dioxide, the weight gain curves shows extrema (gain-loss transition) whose amplitudes rise with the temperature between 700 and 900°C (Matzke, 1986). A quite similar behaviour is described on the oxidation of UC2 (Nawada, 1989) with a weight gain maximal value far exceeding the theoretical target. The weight loss observed is probably a consequence of the oxidation of carbon released by the oxidation process or residual carbon in excess in as-manufactured samples (Nawada, 1989). It is worth keep in mind that the graphite oxidation is widely influenced by the presence of uranium or plutonoium oxides (Sampath, 1988). The overweight is supposed to be due to the trapped oxidation gas product, namely CO2, which can be released more or less roughly (Nawada, 1989).

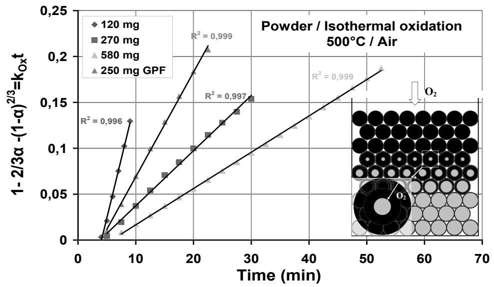
Dealing with practical issues, in usual glove-box working conditions, the oxidation is thermodynamically allowed but kinetically nearly impossible. Lewis (Lewis, 1977) provides an interesting study on the oxidation behaviour of carbides, simulating different kind of oxygen incursion in a glove-box, carrying out DSC experiments. The main results are that a powder bed can ignite (exposed area) at room temperature in a specific brutal oxygen incursion, without any self-sustaining process, and that it is necessary to heat samples over 150°C (powder) or 300°C (sintered solid) to observe quantitative oxidation. To answer to the safety questions related to the pyrophoricity, many authors propose to measure this temperature mistakenly called onset of oxidation (regarding thermodynamic aspects). Benedict (Benedict, 1979), show the feature of an extinguishing reaction, activated only with increasing temperature and oxygen content in the gas. Neither self-sustaining nor igniting projections have been observed. Dealing with reprocessing issues, development of vibrating devices are needed to expedite/activate or reach the completeness of oxidation (Iyer, 1990; Novoselov, 1982). Nevertheless, carrying batches of several hundreds of grams, attention will have to be paid to the exothermic features of oxidation and its runaway ability. Specific manufacturing steps, like milling or machining, create fresh reactive surfaces able to be oxidised leading to a huge increase in the oxygen content in the manufacturing product. This undergoing oxidation is unfortunately difficult to prevent.

The results previously presented are limited and not always reliable or relevant. Actually, whatever is the physical nature of (U,Pu)C, oxidation mechanisms are not known (no kinetic study available) even if comparisons do exist with the UC oxidation behaviour. Among other items missing in the

literature, one can obviously regret the lack of description of the system and its effect and the development of practical issues.

In order to get a better understanding of the reaction, we study the influence of various parameters on the oxidation on mixed plutonium-uranium (U=80%, Pu=20%) carbides composed of (U,Pu)C and $(U,Pu)_2C_3$. The tests have been done on scrap powder samples (S_{BET}= 1,7 m²/g, ~50% oxides/50% carbides) and on a carbothermic solid (100% carbides, approximately 50% of theoritical density, 6 g/cm^3). Isothermal (500, 600 and 700°C) as well as anisothermal (300°C/h or 600°C/h) oxidation treatments have been performed out in TGA and DSC analysis, with 0,1% to 20% oxygen contents (in argon). The average weight gain represents 5% for the powder, and 11.8% for the solid. The structural characterization shows that, in both case, the product is a powder which is composed of two oxide phases $(U,Pu)O_2$ and $(U,Pu)_3O_8$. Their fractions depend on the temperature. For isothermal conditions, we observed a parabolic oxidation behaviour for the powder samples, with an overshoot at high temperature, while the solid samples obey a linear kinetic with an overweight gain at low temperature. The temperature does not dramatically change the oxidation behaviour. On the opposite, the oxygen content and the mass of samples have a great influence on the oxidation rate. For anisothermal conditions, the accelerating oxidation temperatures lie between ~ 150°C for the powder and ~300°C for the solid. Spontaneous and uncontrolled oxidations have never been observed. The self-heating rate of the reacting powder sample (ATG tests) is slight, reaching only 4 or 8°C respectively in 10 and 20%v O2. The oxidation of powder bed do not have the ability to self sustain.

A set of basic assumptions have been made to model the thermogravimetric results, using analytical kinetic models. The powder oxidation process is obviously a diffusion limiting step (Cf. figure). Considering first that the oxidation rate does not depend dramatically on the temperature, and, second, the observed self-extinguishing ability of the burning powder, it can be concluded that the gaseous oxygen plays a key role. Thus, the oxidation process is limited on the one hand by the oxygen diffusion through the powder bed (oxide converted carbide) and within the oxide layer surrounding individual grains.



The present paper reports and comments available and experimental data on carbide oxidation, and give further details about mechanisms. As a conclusion, a few practical issues are proposed, dealing with safety questions (pyrophoricity), and the needs to develop the dry oxidation conversion process.

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U-Pu-Zr Metal Fuel Fabrication for Irradiation Test at JOYO

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An irradiation test of U-Pu-Zr metal fuel was planned in the experimental fast reactor JOYO to confirm its applicability to commercial fast reactors [1]. The fuel pin fabrication for the irradiation test required the preparation of U-Pu alloy as one of raw materials and the development of casting technology for U-Pu-Zr fuel slug.

The U-Pu alloy ingots were prepared from U-Pu mixed oxides (MOX) by meas of the following two methods. In the first method, MOX was converted to nitrides carbothermically, and then dissolved in molten LiCl-KCl eutectic salt by the reaction with $CdCl_2$. Both of U and Pu were recovered from the molten salt into liquid Cd by electro-refining or reductive extraction and then U-Pu alloy was obtained by distillation of Cd. The second method was electro-chemical reduction of MOX (U:Pu = 4:6) in LiCl-Li_2O melt. LiCl accompanying the product of U-Pu alloy was distilled in vacuum. At the same time when Cd or LiCl was distilled, the reduced U-Pu alloy was melted completely so as to consolidate into dense ingot. Surfaces of the prepared ingots seemed to be reoxidized slightly, however, all over the cross-section revealed metallic luster. The chemical analysis of the ingots indicated concentration of each oxygen, carbon, nitrogen and chlorine was below 1000 ppm [2]. These impurity levels were allowable in the present study since the U-Pu alloy ingots were diluted with high-purity U and Zr metals in the fuel slug fabrication and the upper limit of sum of carbon, nitrogen, oxygen and silicon in the slug was controlled to be less than 2000 ppm.

Fabrication tests of U-Pu-Zr alloy slugs were carried out using an injection casting furnace installed in high-purity Ar atmosphere glove box. Single quartz mold, approximately 5 mm in inner diameter and 280 mm in cavity length, was used for each casting batch. The ternary alloy composition of 0, 8.5 and 20 wt.% of Pu was selected for the present tests to determine appropriate casting conditions with respect to the composition variety, because liquidus and solidus temperature of the ternary alloy varies with the composion drastically. The casting conditions such as melt alloy temperature at injection and mold immersion time after injection were determined based on the results of engeneering-scale U-Zr injection casting tests [3] and phase diagram of U-Pu-Zr ternary system [4]. The cast U-Pu-Zr alloy slugs met the tentative specifications, which were determined based on those of EBR-II driver fuels [5]. The constituent elements were uniformly distributed in a longitudinal direction, along with americium as a decay product [6].

Based on the casting test experiences, six U-Pu-Zr fuel slugs for the irradation test at JOYO have been fabricated and satisfied the specifications listed in Table 1. Impurity content in the fuel slug were also lower than the values in the specification. Assembling of the U-Pu-Zr fuel pins are now in progress.

Table 1 Specifications of U-Pu-Zr fuel slug for the irradiation tests at JOYO

Item		Specifications	
Fissile content		17.9±1.0	wt. %
Pu content		20.0±1.0	wt. %
Zr content		10.0±1.0	wt. %
U enrichment		0.71±0.10	wt. %
Impurities	Cr	500	ppm
	Fe	1600	ppm
	Ni	500	ppm
	Y	500	ppm
	Pt	500	ppm
	W	500	ppm
	Li	500	ppm
	Cl	1000	ppm
	C+N+O+Si	2000	ppm
Density		15.3～16.1	g/cm^3
Diameter	(Fuel slug I)	5.05±0.05	mm
	(Fuel slug II)	4.95±0.05	mm
Length		200±1	mm

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Fabrication and characterization of U-Zr alloys for SFR fuel by gravity casting

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SFR fuel fabrication by gravity casting system has been designed and installed. The optimization process is being investigated to get the microstructure of a fuel metal to warrant a better reactor performance. In this study, the fuel materials such as U-Zr binary alloys and U-Zr-Ce ternary alloys (10 ø dia. and 6 ø dia.) were fabricated in lower pressure (100~200 torr) Ar environment by gravity casting. The melt temperature was approximately 1,500^{o}C. Density measurement and gamma-radiography for detecting of internal defects such as internal pores and internal cracks were performed. Microstructure analysis was also carried out to observe intermetallic precipitates by using optical microscope and scanning electron microscope. As-cast properties of the fuels were relatively sound, and they will be presented in this paper. And also some thermal properties including specific heat and thermal expansion characteristics were evaluated for U-Zr binary alloys and U-Zr-Ce ternary alloys in the temperature range from 25 to 600^{o}C to characterize the thermal properties of SFR fuel.

The important results are drawn as follows.

First, specific heats of U-10Zr-Ce alloys were higher than those of U-10Zr and U-15Zr alloys above 400^{o}C, which means that Ce element in the fuel can play an important role to increase specific heat of the fuel.

Second, thermal expansion of U-Zr binary alloys and U-Zr-Ce ternary alloys increases linearly with increasing temperature. Alloying effect analysis shows that

addition of Zr element in the fuel decreases thermal expansion of the fuel, whereas addition of Ce element in the fuel increases thermal expansion of the fuel.

Third, There is a transition of thermal behavior in the temperature range of about from 600~700^{o}C, which is believed to be caused by phase transformation of the fuel materials.

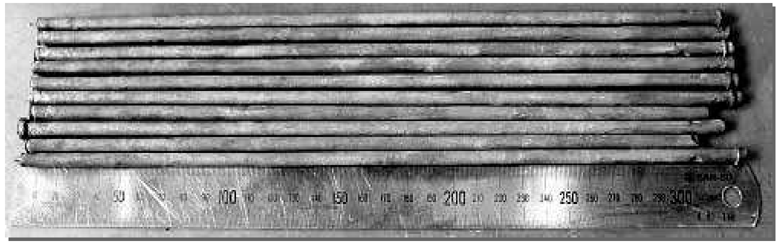

Fig.1 U-10Zr Fuel Slug(6 mm dia. X 300 mm L)

Synthesis of zirconia sphere particles with natural organic material

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In a novel concept of nuclear fuel cycle, zirconia-based oxide fuel, which is so called “inert matrix fuel”, is an attractive one to burn excess plutonium and minor actinides, and fast reactor is more favorable for this purpose. For utilization as a nuclear fuel, zirconia has some advantages such as high melting point, low neutron capture cross-section, stability agaist irradiation. However, its themal conductivity is rather low. As a result, zirconia doped with actinides is anticipated to be used as CERCER (cermic-ceramic) or CERMET (ceramic-matal) fuel in fast reactors.

So far, we have developed the sol-gel method to systhesize zirconia sphere particles [1]. However, a large amount of waste solution, e.g. kerosine and anmonium solution, was generated through this synthesis process. In the present study, we, therefore, aimed to establish the synthesis method with smaller amount of waste solution using a natural organic material, i.e. sodium alginate (Na-Alg). In this method, synthesis processes are very simple, and a major part of waste is $CaCl_2$ solution, which means that this synthesis method is “not” harmful to the environment.

Processes of the synthesis method using Na-Alg were as follows. Firstly, zirconia slurry was prepared as a mixuture of Na-Alg solution and zirconia powder. Secondly, slurry thus obtained was dropped into $CaCl_2$ solution by adding vibration. Thirdly, gelled sphere particles were washed with deionized water and dried. Finally, dried spheres were sintered in air. CERCER pellets were produced as follows. Mixture of sintered zirconia particles and MgO powder was formed into pellet, and consequently, sintered in air. In the present study, as process optimazition in Na-Alg synthesis method, size, density and sphericity of zirconia particles were investigated for experimental parameters such as slurry concentration, Na-Alg concentration, diameter of needle for syringe containing slurry, and so on.

Figure 1 (a) shows dried zirconia particles obtained via Na-Alg synthesis method, and these diameter is less than 0.5 mm. Considering the radiation damage of dispersive fuel with sphere particles, desirable diameter of particles was estimated to be 0.2-0.5 mm [2]. By sintering at high temperature, these particles were anticipated to be shrinked further. Figures 1 (b) and (c) are the photographs of CERCER. Small deformation can be seen during sintering, which results from difference in shrikage rate between zirconia and MgO.

Other resutls and detail disucussion on process optimization will be presented in FR09, e.g. CERMET (zirconia-Mo), relation between particle size and slurry property.

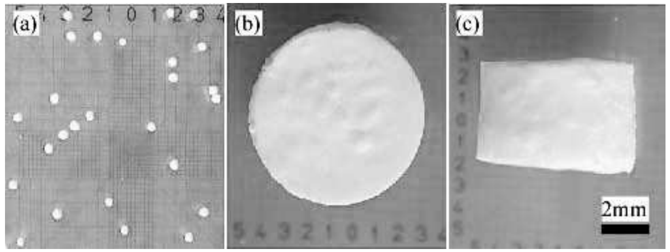

Fig. 1 (a) Dried zirconia particles and CERCER (zirconia-MgO): (b) top and (c) side views.

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Manufacturing Experience for Mixed Uranium-Plutonium Carbide Fuels for Fast Breeder Test Reactor

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The plutonium rich mixed uranium-plutonium carbide pellets of two compositions, namely $(U_{0.3}Pu_{0.70})C$ (MK-I) and $(U_{0.45}Pu_{0.55})C$ (MK-II), are used as the fuel for the Indian Fast Breeder Test Reactor (FBTR) at Kalpakkam. These fuels were developed and are being fabricated and characterized at Bhabha Atomic Research Centre (BARC) and have performed very well with peak burn-up exceeding 155GWd/t. This achievement has been possible through a combination of stringent fuel specifications, quality control during fabrication and inputs obtained from the detailed post irradiation examination of fuel at different stages combined with the modeling of the behaviour of the fuel clad and wrapper materials. The high burn-up and short cooled fuel has also been reprocessed successfully in the reprocessing facility at IGCAR. The fissile material (Pu) recovered from reprocessing has now been used for fabrication of fresh mixed carbide fuel which will be loaded in FBTR in the next reload schedule. Closing the carbide fuel cycle is an important milestone in the fast reactor fuel cycle.

Bhabha Atomic Research Centre, Trombay developed the fabrication flow sheet for MK-I and MK-II carbide fuels for FBTR. Since carbide fuel is pyrophoric and susceptible to hydrolysis, the fabrication has to be carried out in high purity nitrogen cover gas in leak tight glove boxes. Moreover, adequate shielding is provided to minimize the personnel exposure. The carbide fuel are made using powder metallurgy route with UO_2, PuO_2 and graphite as the staring material. The homogeneously mixed oxide and graphite powders are compacted into small tablets at low pressure in order to have handling strength and intimate contact between oxide and graphite particles, and to have sufficient porosities for the easy removal of carbon monoxide. The vacuum and temperature for carbothermic reduction are controlled in order to minimize plutonium losses by vaporization and also to have oxygen, nitrogen, carbon, higher carbide phases within the specified range. The carbide clinkers formed after carbothermic reduction are milled and the resulting fine powders are mixed with suitable binder and lubricant. These powders are finally cold compacted after pre-compaction and granulation. Green pellets are de-waxed and sintered at 1650^{o}C in argon-hydrogen gas mixture. The sintering time-temperature profile is properly maintained to achieve the specified density, dimensions and higher carbide phases. The accepted pellets, after chemical and physical quality control procedures, are stacked and encapsulated in stainless steel type 316M clad tubes along with other components including uranium carbide insulation pellets at both ends of the fissile column. Fuel pins are decontaminated and then subjected to stringent quality checks by helium leak testing, radiography, X-ray gamma autoradiography (XGAR), metrology etc. before wire wrapping. Before the dispatch, the fuel pins are numbered for identification.

During fuel fabrication campaign, the introduction of some novel processing techniques, optimization of process parameters, changes in the end-plug design and use of pulsed TIG

welding technique have resulted in improved yield and enhanced productivity. The process of dry recycling of chemically and physically rejected pellets has resulted in low feed material requirements and hence better economy. The use of attritor instead of ball mill has considerably reduced the processing time and improved the quality of the fuel pellets. Similarly, the repair of end-plug welding saved the time and reduced the amount of active waste. Due to the incorporation of these equipments/techniques, the total processing time from the powder to pellet making has reduced substantially leading to significant decrease in dose to the operators. The complexity involved in the fabrication of plutonium rich carbide has given us confidence to make fuels for the next generation of fast reactors. This paper gives the details of the procedure employed for the fabrication of mixed carbide fuels developed in BARC.

POSTERS OF SESSION 8:

Improvements in fast reactor components and system design

Conceptual Design Study of JSFR (1) – Overview and Core Concept -

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The major concept for FBR in Japan and its core concept will be shown in the full paper.

The abstract of the core concept is shown as follows;

The Japan Sodium-Cooled Fast Reactor (JSFR) at the commercialized stage should have excellent prospects in operation and maintenance as well as the economy. The conceptual core design study on JSFR has been performed from view points of safety, economics, resource utilization, environmental burden reduction, and nuclear proliferation resistance.

The major consideration in core design studies is concentrated on the following requirements.

From the safety consideration of hypothetical accidents, core coolant void reactivity should be low enough to prevent the prompt criticality in the initiating phase of Core Disruptive Accident (CDA), and measures of early discharge of molten fuel should be considered in the core and fuel design to prevent the recriticality in the transition phase of CDA.

The core design is aiming at the core coolant void reactivity of 6$ or less, the core specific power of 40kW/kg -MOX or more and the core height of 100cm or less in order to avoid the excess positive reactivity insertion in the initiating phase of Unprotected Loss of Flow (ULOF) events.

The newly designed FAIDUS(Fuel Assembly with Inner Duct Structure) type subassembly concept is adopted in fuel design of JSFR. The inner duct is installed at corner of subassembly and a part of upper shielding element is removed in FAIDUS. In the transition phase of CDA, the molten fuel enters the inner duct channel and goes out from core region passing through the upper shielding. The FAIDUS type assembly is expected to have superior performance for molten fuel release at CDA.

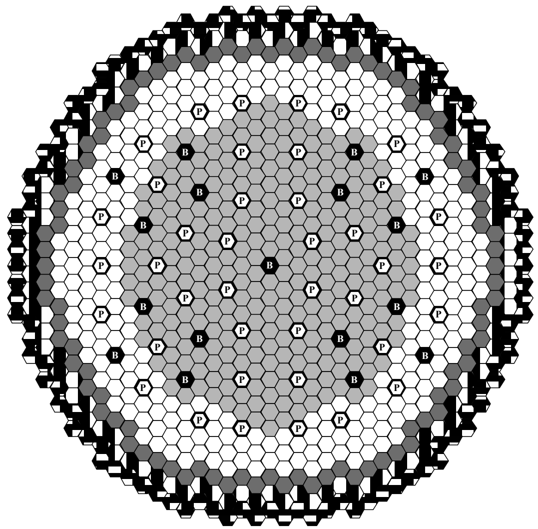
For the reduction of fuel cycle cost due to the economical competitiveness requirement, the target of core average discharge burnup is 150GWd/t, and the total average discharge burnup (including blankets) 60~80 GWd/t. The high burnup contributes to reducing the fuel mass capacity in the fuel cycle facilities like reprocessing plants and fuel fabrication plants.

Cladding and wrapper tube materials of JSFR are ODS (Oxide dispersion strengthened) martensitic steel and PNC-FMS (ferritic/martensitic steel), respectively, which withstand neutron dose of high burnup fuel. Ferritic/martensitic material is selected because of its

dimensional stability up to high neutron dose. ODS martensitic steel is selected because of its excellent high temperature strength as well as its dimensional stability.

From a viewpoint of uranium resource utilization, core should have flexible breeding capability. The target for the maximum breeding ratio of JSFR is 1.1-1.2. The high breeding core with breeding ratio of 1.2 is achieved by changing of fuel specifications under the same fuel assebmly size as the low breeding core and the same core layout. Figure-1 shows the core layout. The core of JSFR is composed of 288 inner core fuel subassemblies, 274 outer core fuel subassemblies, 96 radial blanket fuels and 57 control rods. The major fuel specifications of the low breeding core are as follows. The fuel pin diameter is 10.4mm, number of fuel pin per subassembly is 255, outer flat-to-flat width of wrapper tube is 201.6mm and fuel subassembly pitch is 206.0mm. The fuel pin length is 2690mm.

This paper describes the current study for core and fuel design in JSFR. It shows the core and fuel specifications and core layout of JSFR, and describes the evaluated results of neutronic and thermal hydraulic characteristics and fuel integrity such as CDF.



Inner Core Fuel	288
Outer Core Fuel	274
Radial Blanket	96
Radial Shield (stainless steel)	102
Radial Shield (Zr-H)	108
Primary Control Rod	40
Backup Control Rod	17

Fig.1 Core layout of JSFR

Conceptual Design Study of JSFR (2) - Reactor System -

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Japan Atomic Energy Agency (JAEA) is conducting "Fast Reactor Cycle Technology Development (FaCT)" project in cooperation with Japanese electric power companies. Mitsubishi FBR Systems, Inc. (MFBR) plays an important role for designing and engineering for JSFR.

This paper describes the summary of the conceptual design study on the reactor system.

The reactor system for 1500 MWe plant has following main specifications.
(1) Top entry loop type
(2) Two loops per reactor (two hot leg pipes and four cold leg pipes per reactor)
(3) Inlet temperature: 395 deg. C / Outlet temperature: 550 deg. C
(4) Reactor vessel material: Type 316 FR stainless steel
(5) Reacter vessel diameter: 10.7m / height: 21.2m
(6) Refueling system which consists of a single rotating plug, a shell-less column-type Upper Internal Structure (UIS) with a slit and Fuel Handling Machine (FHM) with a extension arm
(7) Hot vessel without a wall cooling system

Keeping the diameter of the reactor vessel below 11m is required to reduce the construction cost so as to compete with a future commercial light water reactor and. For that purpose, the refueling system that the FHM is entered the arm into the center of the UIS through a slit is adopted.

The flow velocity in the reactor vessel is high, because of the small reactor vessel diameter. Consequently, there is a possibility that gas entrainment, vortex cavitation and flow-induced vibration occur. Consequently, we have been addressed the conceptional design study and the R&Ds for optimizing the coolant flow in the reactor vessel and for inhibiting the vibration of small diameter pipings settled in UIS such as thermowells and a failed fuel detection and location (FFDL) sampling pipings.

Seismic stress is generated at the top of the reactor vessel, because the reactor vessel is hanged at the bottom of the roof deck.The highly increased earthquake loading is required because of the Niigataken Chuetsu-oki Earthquake in 2007. As the result, the seismic design margin of the reactor vessel which has a thin wall is decreased very much. Accordingly, we are developing the advanced seismic isolation system for mitigating the earthquake loading.

Conceptual Design Study of JSFR (3) - Reactor Cooling System -

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Japan Atomic Energy Agency (JAEA) and Mitsubishi FBR Systems, Inc. have promoted the conceptional design of the Japan Sodium-Cooled Fast Reactor (JSFR) in the Fast Reactor Cycle Technology Development (FaCT) project. This paper describes the concept of the reactor cooling system design on the JSFR.

The two-loop configuration is used for the reactor cooling system to enhance economics. The reactor cooling system consists of the primary and the secondary systems to prevent the possibility of contact between radioactive sodium in the primary system and water in the steam generator (SG). There are an intermediate heat exchanger (IHX) and a SG per loop. The decay heat removal system (DHRS) consists of two primary reactor auxiliary cooling systems (PRACS) and a direct reactor auxiliary cooling system (DRACS). The heat exchanger of the PRACS is located in the integrated IHX/PUMP, and that of the DRACS is located in the reactor vessel.

The integrated IHX with pump is incorporating a primary pump and an IHX into a single vessel, and it is possible to eliminate a middle-leg piping connecting a pump and an IHX. The hot-leg piping of primary system is L-shaped piping running from the upper part of the reactor vessel to the IHX. As the result, the reactor vessel and the IHX vessel are positioned closely. The primary and the secondary piping have the guard pipe which ensures preventing fires and maintaining primary sodium level required for decay heat removal if sodium leaks.

The sodium-water reaction hazards in the event of the SG tube failure are considerably reduced by adopting the double-walled tube. The shell and tube type SG with straight tube bundle is adapted to enhance manufacturability of double-walled tubes. The material of SG tube is high Cr-based steel, which is resistant to high temperature and has high thermal conductivity, so it is possible to reduce the heat transfer area. A once-through type steam generator can eliminate a super-heater because high Cr-based steel is free from stress corrosion cracking (SCC).

The DHRS is the cooling system utilizing natural circulation. The DHRS equipment can be operated under fully passive conditions without need for pumps and blowers to maintain the core cooling function after a reactor shutdown. The large temperature difference between core inlet and outlet can ensure core flow rate by buoyancy force. The natural circulation system increases reliability and reduces construction costs.

The target construction cost of the JSFR is lower than that of future light water reactors.

Conceptual Design Study of JSFR (4) - Reactor Building Layout -

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Japan Atomic Energy Agency (JAEA) and Mitsubishi FBR Systems, Inc. have promoted the conceptional design of the Japan Sodium-Cooled Fast Reactor (JSFR) in the Fast Reactor Cycle Technology Development (FaCT) project. This paper describes the concept of the reactor building layout on the JSFR.

Reduction of construction costs is one for the most important matters for commercialization of fast reactor. Miniaturization of reactor building and investigation of construction method for reduce construction term has been carried out in this layout design.

In miniaturization of the reactor building, sharing of facilities (such as Fuel-handling facilities, waste-disposal facilities) and rationalization of components arrangement according to be a rectangular containment vessel has been carried out.

Especially for containment vessel, steel plate concrete structure (SC structure) is planned to adopt. It is expected that conventional works such as steel rod reinforced work and installation / withdrawal of a mold work become unnecessary. For adoption of SCCV (Steel plate Concrete Containment Vessel) , research and development are furthered.

As the result, the size of the reactor building is about 104m(L) x 77m(W) x 70m(H) .

On the other hand, innovated construction method called "Large Unit Construction Method" has been investigated to reduce construction term. The construction method is assembled skeleton building structure unit in factory and transported to the site by ship.

The unit division proposal in Large Unit Construction Method is set up and the weight for every unit is computed. Becoming division of a total of 13 units, maximum weight of the unit is 6,000ton.

Examination which shortens the construction time is performed and the trial calculation of the construction time is made with the second half for 30 months for one plant.

And also, the highly incresed earthquake loading is requierd because of the Niigataken Chuetsu-oki Earthquake in 2007, the seismic isolation system is adopted to the reactor building.

Structural Integrity test on Reflector Cavity of 4S

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The 4S (Super-Safe, Small and Simple) is a small-sized, sodium-cooled fast reactor with a reflector-controlled core. In this paper, the reflector design and the structural integrity test are described.

For 4S, the reactivity and the power are controlled by an annular reflector surrounding the core. The reflector consists of six segments and each segment is separately controlled. Figure 1 shows the structure of the reflector.

The reflector consists of the reflecting region and the cavity region. The cavity region is installed above the reflecting region; its function is to enhance neutron leakage relative to the surrounding sodium coolant. Each segment of the cavity region consists of six cavity cans which are cylindrical shape and filled with argon gas.

The reactivity of 4S reactor is controlled by the reflector. The reflector increases reactivity when it is raised above the reactor bottom to overlap the active region of the core and lowers reactivity when it is lowered back into the reactor bottom.

After the power is reached to the rated power, the reflector is raised at a slight speed to compensate for burnup swing reactivity over core life.

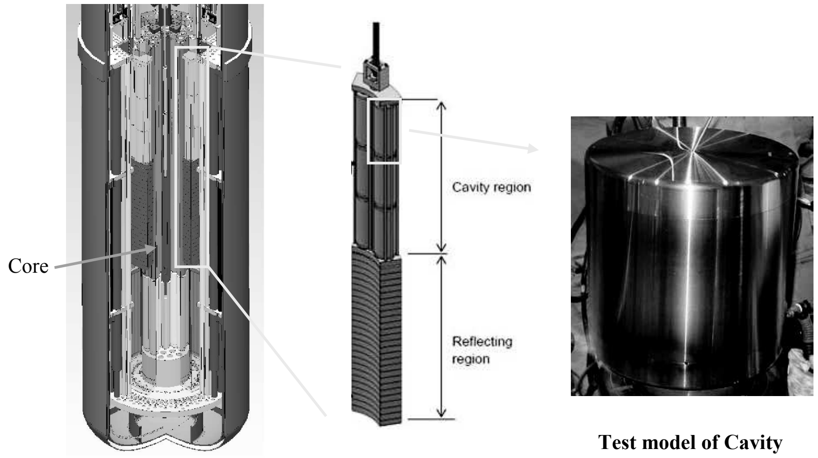
The cavity can is one of most important equipments on safety, which is enhancing the neutron leakage.

The cavity can is such a thin wall structure, which is exposed to the thermal stress and thermal transient in reactor that its integrity is required to prevent from inducing the insertion of the reactivity for core in case of the gas leakage due to rupture.

Figure 1 shows the test model of the cavity can used for confirming the design margin for structural integrity and propriety of structural design method.

A test is in process to examine the integrity of cavity cans by simulating experimentally thermal stress and transient condition in reactor. The heat cycle is imposed on the test model in sodium test tank and the maximum test temperature is up to 550℃ in order to give large thermal load.

From this test, the degree of the creep fatigue damage is evaluated and with the structural analysis, the structural integrity of cavity cans for core life will be confirmed.



4S Reactor **Reflector (1 segment)** **Test model of Cavity**

Figure 1 Reactor of 4S

Structural Design Studies on a Large Pool Type SFR of 1200MWe

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An economic improvement is a hot issue as one of the Gen IV nuclear plant goals[1]. To secure economic competitiveness of a SFR compared to a pressurized water reactor, several structural design concepts are adapted in without loosing the reactor safety level. One is the increase of the plant capacity with the minimum number of component and loop, which leads the reduction of the plant maintenance, repair, and construction costs by a large-size scale effect. Another is the simple system arrangement, compact reactor size for only two loop system for a 1200MWe capacity of a pool type SFR, and the minimization of IHTS piping length through the properly locating the SG and secondary pump. Several researches are also studied to attain the economic improvement target of the NSSS in structural point of view; for example, an integrated concept of a refueling machine and inspection device with a long waveguide sensor for reactor internals.

Fig.1 shows the reactor internals and components arrangement in reactor vessel. The outer diameter of the reactor vessel is 14.5m, which is very compact size compared to other designs, and 0.05m in thickness. It can accommodate the maximum core size of 7.9m. With the internal arrangement, the refueling availability of core assemblies was confirmed, and 36 control rods are supported and guided by upper internal structures[2]. The material of reactor vessel and internal structure is a Type 316 stainless steel.

The primary system consists of 4 sets of primary pump, IHX, and DHX in reactor vessel. The component size is not much larger than the KALIMER-600 design because the numbers of components was increased. The reactor vessel's diameter is relatively so small that the minimum space between the components is 55cm, which may not be enough for equipment maintenance.

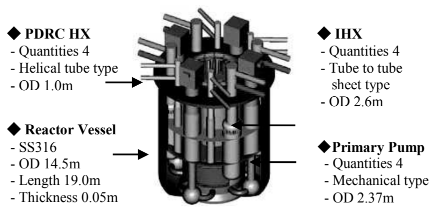

FIG. 1. Component arrangement in RV for two-loop system of a pool type SFR of 1200MWe.

The NSSS has two intermediate heat transport loops. The each of IHTS loops outside of the reactor vessel has 2 mechanical type pumps and 2 SGs[3].

The fabricability of the piping and elbows of the large piping diameter was investigated for realizing this system. The piping diameters for the hot and cold legs of 80 cm and 110cm respectively are within the feasible fabrication range through the both ways of forged pipe and seamless pipe.

The piping length of each loop is relatively long about 180m because of the inverse U shape piping layout adapted to the preventing the pressure propagation to the reactor vessel when a sodium-water reaction accident occurs in SG upper tube. Since this long piping layout increases the maintenance cost, the shortening of the piping total length would be necessary by properly sacrificing over-excessive safety design level.

The pipe material of a Mod.9Cr-1Mo can shorten the piping length about 60m compared to stainless steel, which has also a higher mechanical strength and a low thermal expansion. This has an advantage for obtaining a simple layout of IHTS piping subjected to a high temperature environment. The two-loop system is also advantageous to a compact building size because the numbers of components can be minimized, and the total piping length could be shortened. The height of the SG is about 37.4 m. The SG tube is a double-wall straight type to reduce the possibility of the water injection to the sodium side.

Two new design strategies are adapted for the economic improvement of the NSSS in structural point of view. One is an integrated component of a refueling machine and an in-service inspection (ISI) tool with a long life waveguide sensor for reactor internals. This concept will shorten the overall period of about 2 days through the reduction of in-service inspection time of Rx internals. The other is a LBB technology application for IHTS piping and RV, which will reduce the construction cost because of the unnecessary of a large scale protection facilities against to sodium leak accidents.

The component arrangement and reactor structural sizing for two loop systems for a 1200MWe capacity of a SFR are suggested with several structural design improved concepts to attain an economic improvement of a large size pool type SFR. These concepts will be confirmed on the structural integrity for the operating and design loads, and optimized to a unified conceptual design through some trade-off studies.

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Thermal Analysis on Shipping Cask for JSFR Fresh Fuel

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In the Fast Reactor Cycle Technology Development (FaCT) Project, a commercial sodium cooled reactor named “JSFR” has been developing. In the future fast reactor fuel cycle, trans-uranium elements including minor actinides are closed in the nuclear fuel cycle to reduce environmental burden. Fuel cycle and fuel design analyses have shown that JSFR fresh fuel require certain shielding and cooling because irradiation and decay heat from minor actinide contents. Besides a major requirement of a fresh fuel cask is to keep fresh fuel subassembly conditions as manufactured. In the case of fast reactor fuel, the cladding temperature has to be maintain below creep temperature, because any creep damage is not taken into account in the core design. And the surface condition of the fuel pin cladding is also important for fuel integrity. Since coolant sodium could react with water providing corrosive productions, any residual humidity on subassembly surface has to be prevented. SFR experiences have shown troubles due to water contact before sodium immersion and experiences also show that complete dry out of water immersed components are quite difficult. From this point of view, adoption of conventional water casks isthought to be difficult without long term R&Ds. Therefore, for the first step of SFR fresh cask development, a dry cask has been selected as a reference concept. From the viewpoint of high thermal conductivity, helium gas is selected as filling gas.

In this study, a basic feasibility of the helium gas cask has been evaluated by thermal analyses. There have been conducted two analyses: whole cask and detail inside subassembly analyses. The whole cask analysis has evaluated temperature distribution in the whole cask providing boundary conditions for the detail inside subassembly analysis. The detail inside subassembly analysis has shown that the temperature distribution is mainly governed by thermal conductivity and natural convection of coolant helium hardly contributes heat removal. In the case of a cask with five subassemblies with 2.2kW decay heat per each, the maximum cladding temperature is evaluated to be 361deg-C satisfying cladding temperature limit of 395deg-C. Those results have shown the basic feasibility of the helium gas fresh fuel shipping cask.

ACKOWLEDGMENTS

This paper includs the results of “Development of Fuel Handling System” entrusted to the Japan Atomic Power Company by the Ministry of Education, Culture, Sports, Science and Technology-Japan.

Development of Transfer Pot for JSFR Ex-vessel Fuel Handling

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In the Fast Reactor Cycle Technology Development (FaCT) Project, the Japan Sodium Cooled Fast Reactor (JSFR) attempts to incorporate innovative technologies in order to reduce commodities and to improve plant availability. Among these technologies related to the fuel handling systems, a spent fuel transfer pot for ex-vessel fuel handling, which contains two fuel subassemblies simultaneously and applicable size to compact reactor vessel, has been developing so as to shorten a refueling period leading to an improvement of plant availability. The pot is required to provide sufficient cooling capacity even in case of transportation malfunction in the guide tube cooled indirectly by air flow outside the guide tube. Since preliminary analysis shows that radiation dominates heat transfer, vertical fins on the pot surface and chrome carbide coating on the pot and guide tube have been proposed. Both experimental and analytical studies have been performed to evaluate the cooling capacity of the pot after sodium immersion, which is considered to affect radiation between the pot and guide tube.

The emissivity of the chrome carbide coating after sodium immersion has been measured using plate specimens. The measured results show that the effect of liquid sodium immersion on the emissivity of the coating is negligible. A full scale mock-up of the fuel transfer pot and the guide tube was manufactured and heat transfer test after sodium immersion has been conducted to validate the analytical model, which has been constructed using fluid dynamics code. Temperature distribution of the mock-up test has shown that pot cooling is governed by radiation from the pot to the guide tube and the cooling capacity could be evaluated by the constructed analytical model incorporating the measured coating emissivity data. Those results are considered to demonstrate the feasibility of the pot.

The results of “Development of Fuel handling System” study entrusted to the Japan Atomic Power Company by the Ministry of Education, Culture, Sports, Science and Technology of Japan are included in the present paper.

Development of Spent Fuel Cleaning method for JSFR

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In the design study of Japan sodium cooled Fast Reactor (JSFR), we have been developing a dry spent fuel cleaning method in place of the conventional steam-water cleaning method which are applied in the existing sodium cooled fast reactor (SFR) such as MONJU . In the conventional steam-water cleaning method, spent fuel subassemblies are rinsed to remove residual sodium almost completely. But, because of the rinse process, it provides a certain amount of radioactive liquid waste. In the conventional storage system, spent fuel subassemblies are accomodated in a closed can in the water pool. Then the spent fuel must be cleaned compleately so that residual sdoium could not affect water conditions inside the can. In the case of JSFR, spent fuel subassemblies are acoomodated directly in the water pool. Then certain ammont of residual sodium on the spent fuel subassemblies are acceptable as long as the water pool cooling system can handle it. The amount of the radioactive liquid waste of the dry cleaning method is less than that of the steam-water cleaning method because the dry cleaning method does not have rinse procedure which spends a lot of fresh water. If the dry cleaning method is adopted in JSFR, the radioactive liquid waste could be decrease to tens of percent compared with the steam-water cleaning method.

The schematic diagram of the JSFR dry cleaning concept is shown in Fig.1. When the decay heat of a spent fuel subassembly becomes lower enough in the ex-vessel sodium storage (EVST), the subassembly is transported from EVST to the water pool storage by the ex-vessel transfer machine (EVTM). During the transportation, the dry cleaning is capable to be done right above the EVST using the EVTM gas cooling sytem. Then the EVTM carry the spent fuel to the water pool storage.

In this study, a full-scale mockup of a JSFR fuel pin bundle has been manufactured and dry cleaning performance tests after sodium immersion have been conducted. The subasembly mockup involves fuel pin bunddle, wrapper tube and inner duct. To simulate decay heat of the spent fuel, the fuel pin bundle have heater inside fuel pins. Since the JSFR fuel subassembly has an inner duct, sodium drain from the inner duct have been also evaluated.

The test results showed that mesured sodium amounts on the mockup fuel pin bunndle are less than 200g including residual sodium in the inner duct. And this result is thought to demonstrate basic feasibility of the dry cleaning method.

Now, we are evaluating the results of the test to gain the design data for the spent fuel storage water pool cooling and filtering system.

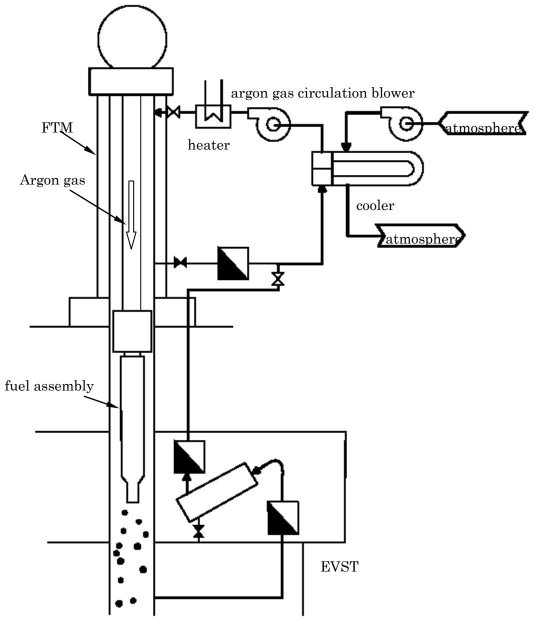

Fig.1 the rough cleaning concept

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Design and layout decisions for refuelling system of advanced fast neutron reactor

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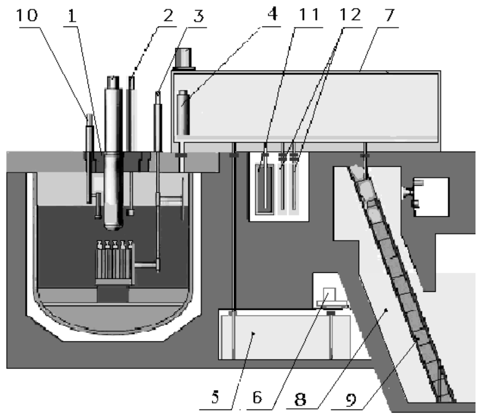
The experience in operation of BOR-60, BN-350 and BN-600 power units, as well as development of refuelling systems for BN-800 power unit, allows developing of refuelling system for BN-1200 advanced reactor of new generation. The refuelling system was developed on the basis of possible technical decisions aimed at improvement of safety and technical-and-economic indices.

Structural layout of BN-1200 reactor refuelling system is given in Fig. 1



1. Rotating plugs
2. "Direct" refuelling mechanism
3. Elevator
4. Refuelling machine
5. Fresh fuel assembly (FA) conveyor
6. FA transfer mechanism
7. Refuelling box
8. Water pool
9. Inclined elevator of discharge cavity
10. "Cantilever" refuelling mechanism
11. Box for washing liquid lead
12. Boxes for washing steam

FIG. 1

Main differences in BN-1200 reactor refuelling system as compared with BN-800 reactor are given in Table 1.

Table 1

Table 1Equipment	Equipment quantity	
	BN-800	BN-1200
1 Rotating plugs	3	2
2 Refuelling mechanism	1	2
3 Elevator	2 (inclined)	1 (vertical)
4. Refuelling machine of refuelling box	1	1 (with enhanced radiation protection)
5. Fresh FA drum	1 with the capacity of 234 FA	Is replaced for fresh FA conveyor with the capacity of ~50 FA
6. FA transfer mechanism	1	no
7. Refuelling box	1	Joint structure with light radiation protection
8. Washing box	1	
9. Refuelling machine of washing box	1	no
10. Spent FA drum (with sodium)	1	no

Design features of refuelling equipment are:

- BN-1200 reactor has a split large rotating plug to allow transporting of its components by railway with subsequent assembling at site;

- the refuelling box is fabricated in the form of sectional parallelepiped to allow transporting of its components by railway with subsequent assembling at site;

- one "direct" refuelling mechanism and one cantilever" refuelling mechanism are used to refuel rarely replaced protection assemblies that allows reducing of overall dimensions of rotating plugs;

- the vertical elevator is arranged on the oval plug installed on the reactor cover. The upper structure with elevator drive rotates together with the elevator plug under rotary drive located on the oval plug. The vertical elevator allows sufficient reduction of refuelling box;

- the refuelling machine runs on straight-line rails.

The vertical elevator, gas gate valve on reactor refuelling channel, non-use of spent FA drum and enhanced radiation protection on the column of refuelling box machine allows reduction of specific materials consumption of BN-1200 reactor refuelling system by more than 10 times as compared with BN-800 reactor.

To verify refuelling equipment operability the following experiments are planned:

- mastering of gripper design for "direct" refuelling mechanism and refuelling machine;

- mastering of "cantilever" for refuelling mechanism;

- mastering of fresh FA conveyor design.

As for the other refuelling equipment, the designs, approved by BN-350 and BN-600 operation experience, are used. This equipment does not require experimental study.

Design validation of the 4S high temperature electromagnetic pump by one pole segment test equipment

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The small fast reactor 4S (Super Safe Small and Simple) adopts high temperature sodium immersed electromagnetic pump (EMP) as a primary pump. The reason is that EMP is able to satisfy the low mantenance, safety and reliability requirements for 4S, because it has simple structure and no moving parts. Technical challenges of 4S EMP are the following three items. The first is to confirm manufacturability of 4S EMP. 4S EMP has the world greatest dimension coil and stator and flattened shape with the aspect ratio (outer diameter / stator length) of 1.8. The 4S prototype EMP as same as practical dimention and the one pole segment test equipment were manufactured in JFY 2008. The second is to construct of the back up power supply system for flow coast down. It will be confirmed by combination test of 4S prototype EMP and the system in JFY 2010. The third is to confirm low maintenance for 30 years. A part of long-term soundness test of coil is tested by heat cycle test with one pole test equipment discussed in the next section. Forthermore, as the future development subject, the seismic assessment and the long-term soundness test of thr EMP body by the sodium fluid test has been planned.

The one pole segment test equipment is used for evaluation of following items. The first is the integrity of the electrical insulation of large diameter coil. Insulation breakdown will be estimated due to thermal expansion in coil during operation, thus, the heat cycle is loaded to the equipment simulating start-and-stop of 4S. Insulation resistance, leak current and tan δ will also be measured in several setps of coil temperature. The second is validation whether stator support system of the EMP would work as designed. In practical 4S EMP, all of Joule heat by energization of the coil during operation is transferred into sodium through core and duct. To keep the heat transfer system, in the outer stator, the core is pressed upon the duct by spring plates. Therefore, it is important to measure thermal displacement in the stator structure continually and compare to the design values. The one pole test equipment is installed into a electrical heating furnace, heat cycle is loaded to 18 times between 200℃ and 500℃, where are in accordance with practical coil operation temperature of 4S EMP. The number of cycles will be enough because start-and-stop cycles of 4S is estimated as 6. Those test is planned to be carried out in JFY 2009 no later than starting of the 4S prototype EMP test.

Fig.1 shows a picture of one pole segment test equipment manufactured and the drawing of stator structure. The diameter of the outer stator is as same as 4S EMP and the height is one-sixth height of 4S EMP. This test equipment contains 6 coils, 7 iron core blocks and minimum configuration of stator support system.

In JFY 2008, the electric and magnetic test was carried out in the test equipment, and the characteristic was confirmed in accordance with designed at room temperature under atmospheric conditions.

This report involves manufacture of the one pole segment test equipment and results of the electric and magnetic test as well as heat cycle test.

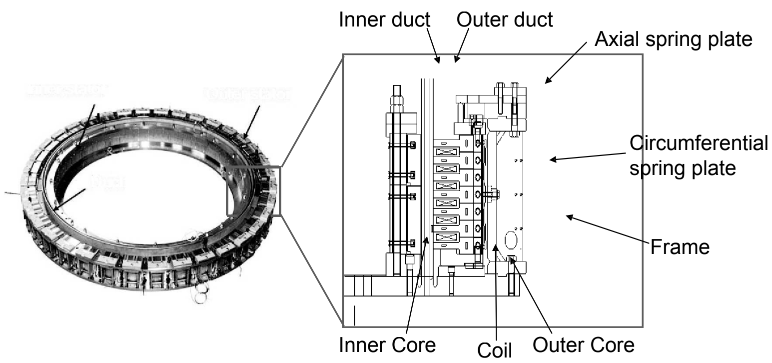

Fig.1 Picture of one pole test equipment and drawing of stator structure

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The part of present study is the result of "Development of high temperature electromagnetic pump with large diameter and a passive flow coast compensation power supply to be adapted into medium and small reactors of GNEP" entrusted " Toshiba Corporation" by the Ministry of Economy, Trade and Industry (METI) of Japan.

Decay heat removal system by natural circulation for JSFR

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Japan Atomic Energy Agency (JAEA) is conducting "Fast Reactor Cycle Technology Development (FaCT)" project in cooperation with electric power companies, Mitsubishi FBR Systems,Inc.(MFBR), major nuclear plant manufacturers, and other organizations in Japan. In the project, a conceptual design study for Japan Sodium-cooled Fast Reactor (JSFR) is being implemented in parallel to development of innovative technologies for JSFR. A decay heat removal system (DHRS) utilizing passive natural circulation was selected as one of the innovative technologies to be possibly applied to JSFR. In order to adopt such a passive DHRS in a large scale SFR, it is necessary to clarify not only core cooling capabilities but also thermal-hydraulic phenomena in the plant system including core, reactor vessel, primary and auxiliary cooling systems.

In this study, a water experiment was performed using a 1/10th scale model which physically simulates the reactor core and vessel, the two sets of the primary loop and the DHRS consisting of one set of Direct Reactor Auxiliary Cooling System (DRACS) and two sets of Primary Reactor Auxiliary Cooling System (PRACS) of the large scale SFR. The model scale was determined from the similarity rule for the natural circulation to match the Richardson number, Peclet number and Euler number between the model and the prototype. One of the typical transient events tested in the experiment is the loss of external electric power supply, and the others are events under unsymmetrical conditions of twin components installed in the two loops. The results of the experiment showed us that stable and sufficient natural circulation flow rate was established in the primary loops for all of the tested events, and it was clarified that rather severe thermal fluctuation may be imposed on some parts of the system during the transient events, which will be remedied in our future study.

A sodium experiment was also performed using Plant Dynamics Test Loop (PLNDTL) test facility in JAEA O-arai Research and Development Center. Sufficient heat transfer coefficient was confirmed for a PRACS heat exchanger with parameter of the primary side flow velocity which covered reactor conditions. The start-up characteristics of natural circulations in air stack of the air cooler, the DHRS loop, and the primary loop were examined by a scram transient experiment. Smooth increases of natural circulation flow rates were obtained in all systems.

Large and complex components adopted in JSFR will enhance multi-dimensional characteristics of phenomena during scram transients, e.g., thermal stratification in the piping and biased flow in the heat exchangers. Thus, a transient analysis method was developed in which three-dimensional computational domains were used for the whole primary cooling system. Verifications are undergoing based on the water and sodium experiments. The core flow rate in a natural circulation system is low and not constant. However, buoyancy force will flatten radial temperature profiles in subassembly scale and also core scale. Then a new evaluation method for core peak temperature was developed to take account of such buoyancy force effects and uncertainty factors under the natural circulation conditions. Basic procedure of the evaluations was defined and sensitivity analyses of major factors were completed.

We summarize current status of this study in full paper. Details of the water and sodium experiments and the code developments will be published near future.

Present study is the result of “Development of evaluation methods for decay heat removal by natural circulation under transient conditions, 2008” entrusted to “MITSUBISHI FBR SYSTEMS,INC.” by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT). The water experiment was performed by CRIEPI and the sodium experiment was performed by JAEA under the contract with MFBR.

Steam Generator with Straight Double-Walled Tube - Development of fabrication technologies of main structures made of high chrome steel-made -

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Steam generator (SG) with straight double-walled tube is adopted for the Japan Sodium-Cooled Fast Reactor (JSFR). This paper describes the reserch and development of the SG with straight double-walled tube.

To improve reliability of the plant against sodium-water reaction caused by SG tube failure, the Double-walled tube is adopted. Double-walled tubes consist of inner and outer tubes, and they are mechanically contacted each other. They have small clearance between inner and outer tubes, and reduce penetration leakage (of pressurized water) due to high flow resistance across the clearance in the event that both inner and outer tube leak at defferent point. Using high chrome steel, which has high strength at elevated temperatures and has a high thermal conductivity, as the material for SG tubes reduces the area of heat-transfer surface. By adopting high chrome steel, which has also tolerance to stress corrosion cracking (SCC), the SG is designed as a once-through type generator, enabling a reduction in the component number. It is important to keep high reliability without incurring a performance penalty. Therefore we put a cap on the amount of clearance and design to keep contact in operating conditions.

Since double-walled tube, welded joint between tube and tube sheet, etc. are special specification and shape, it is necessary to develop these fabrication technologies such as welder for narrow space. Double-walled tubes of 10m length were manufactured for trial, and processability, clearance size and so on were confirmed. About welded joint between tube and tube sheet, tests of welding and tube expansion against tube sheet were performed, and these works and strength data were acquired. Since Double-walled tube is thick for its small diameter, expansion is quite difficult. Therefore we are going to survey best condition for Double-walled tube.

Thermal Hydraulic Design of a Double Wall Tube Steam Generator with an On-line Leak Detection System

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As one way to improve the reliability of a steam generator for a SFR, a double-wall tube steam generator is being developed. The current focuses of this research are an improvement of the heat transfer capability for a double-wall tube and the development of a proper leak detection method for the failure of a double-wall tube during a reactor operation. The ideal goal is an on-line leak detection of a double wall tube to prevent the sodium-water reaction. However, such a method is not developed yet. An alternative method is being used to improve the reliability of a steam generator by performing a non-destructive test of a double wall tube during the refueling period of a reactor [1].

In the conventional pre-stressed double wall tubes, the inner tube and the outer tube are made of the same material [2]. To improve the heat transfer capability of a double tube, it is preferable to form the inner tube with the material having a thermal expansion coefficient about 10 to 15% greater than that of the outer tube. When the inner tube is formed with the $2\frac{1}{4}$Cr-1MO steel and the outer tube is formed with the modified 9Cr-1MO steel, the double-wall tube has the heat transfer efficiency similar to that at room temperature when the temperature difference between the inner tube and outer tube is about 55℃ in the normal operating condition. At an abnormal condition when the tube side is empty while the sodium temperature is 550℃, the stress of about 12MPa is formed according to the ANSYS code analysis, but this value is very lower than the yield stress of those materials.

It is another object of the present study to provide a damage detection system for a heat transfer tube that can detect on-line and real-time whether the heat transfer tube is damaged or not. The method is achieved by that the heat transfer tubes with grooves are radially arranged along the lower tubesheet, and each of the detection holes communicates with each of the heat transfer tube gaps in the lower tubesheet. And each of detection holes is formed in the direction of the radius of the lower tubesheet so as to communicate with the side of that. Therefore, the heat transfer tube gaps and the detection holes meet one to one respectively. If the gaps are filled with about 2MPa Helium gas, One can detect the tube failure by online checking the change of volume of gas in the groove.

A helical coil double wall tube steam generator for SFR with 375MWth capacity is designed thermal-hydraulically by the embodiment of above methods as shown in Fig. 1. The steam generator has the on-line, real-time tube failure detection capability and the heat transfer efficiency is better improved than other DWTSG.

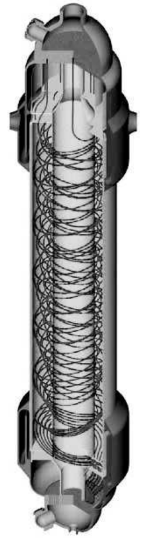

FIG. 1. A 3-D view of the DTWSG with a few selected heat transfer tube

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DESIGN AND MANUFACTURE OF TUBE TO TUBESHEET JOINTS OF STEAM GENERATOR FOR 500 MWe PROTOTYPE FAST BREEDER REACTOR

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Abstract :

Prototype Fast Breeder Reactor (PFBR) is 500 MWe pool type sodium cooled fast reactor. Presently this reactor is at advanced stage of construction at Kalpakkam. The main function of the steam generator (Fig. 1) is to extract the reactor heat through secondary sodium system and convert the feed water into superheated steam in the tubes of steam generators. The steam generator is a vertical shell & tube type heat exchanger with liquid sodium in the shell side and water/steam in the tube side. Operating experience of FBRs have shown that steam generator (SG) holds the key to commercial success of such reactors. Tube leakage is a serious problem and the prevention of sodium water reaction incident in the SG is essential to maintain the plant availability. In case of crack/failure in tube, high pressure water/steam reacts with shell side sodium and results in exothermic reaction with evolution of hydrogen, corrosive reaction products and intense local heat depending on leak size. This high reactive nature of sodium with water/steam requires that sodium to water/steam boundaries of steam generators must possess a high degree of reliability against failure. This is achieved in design and manufacturing by maximising the tube integrity and more importantly by proper selection of tube to tubesheet joint configuration. The principal material of construction of SG is Modified 9Cr-1Mo steel. The tubes are seamless and produced by electric arc melting followed by Electro Slag Refining (ESR) with tight control on inclusion content. Ultrasonic and eddy current testing is done on entire tube length in accordance with ASME SEC III Class I. Long seamless tubes (each 23m) are used in order to reduce the number of tube to tubesheet welds. Each SG has 547 tubes and there are 9 SG in the reactor including one spare module. There is no tube to tube joint as the aim is to minimise the number of welds to increase reliability. Tube to tubesheet joint (Fig. 2) selected for PFBR steam generator is of internal bore welded type with raised spigot. The bore side has been selected as it eliminates crevices between tubes and tubesheets (as found in conventional rolled and welded joints) and moreover it permits 100% radiography inspection. Of the main joint form for the internal bore weld i.e. raised spigot and inset tube type, the former has been selected to a) place the welds in a low stress region, b) easier to weld and offer better repeatability c) provides increased ligament for a given tube pitch as the tubesheet hole will be the tube bore and d) permitting preheating and Post Weld Heat Treatment (PWHT) for individual tube to tubesheet weld. The type of weld joint selected is cup & cone type butt joint between spigot machined from tubesheets and tube ends. Both the ends are machined with very tight tolerances to get specified weld profile. ESR grade material is also used for tubesheet to get clean steel to achieve sound weld. The tube is of 17.2 mm OD, 2.3 mm wall thickness and welded to both the tubesheets at the ends. The welding process selected is autogenous, pulsed GTAW process with different high pulse level of current. To meet the stringent weld profile requirements, process parameters (pre purge time, up slope, speed of welding, down slope, post purge time)

were optimised and welding procedure was established. The welds are subjected to preheat at 473K-523K and PWHT at 1033K±10K. Argon gas purging is provided to avoid oxidation during PWHT. Each joint is subjected to thorough visual examination, dye penetrant testing, radiography using microfocal rod anode x ray, profile check (concavity, convexity and wall thickness), pressure testing and helium leak testing. The acceptance limits of weld joints are kept very stringent. Specially designed dial gauge is used to check the weld profile from inside for all the joints. Replica technique is also used to crosscheck the internal weld profile for 1% of the total joints in addition to dial gauge measurement. External profile measurement is done by replica technique for all the weld joints. Extensive 22 different types of tube to tubesheet welding trials were conducted welding over 500 joints to meet the design requirement of 40 years life. The major defects observed in the tube to tubesheet welds are porosities. In order to overcome these defects, skim cut & innovative cleaning process for the tube & spigot end was developed. Drastic improvement in the quality of weld joints was observed after acid cleaning of spigot end by ultrasonic agitation technique. Till date around 7100 out of 9846 tube to tubesheet joints were welded successfully on the actual job. This paper highlights the issues related to design, materials and experience gained during manufacture of tube to tubesheet joints of steam generators in detail.

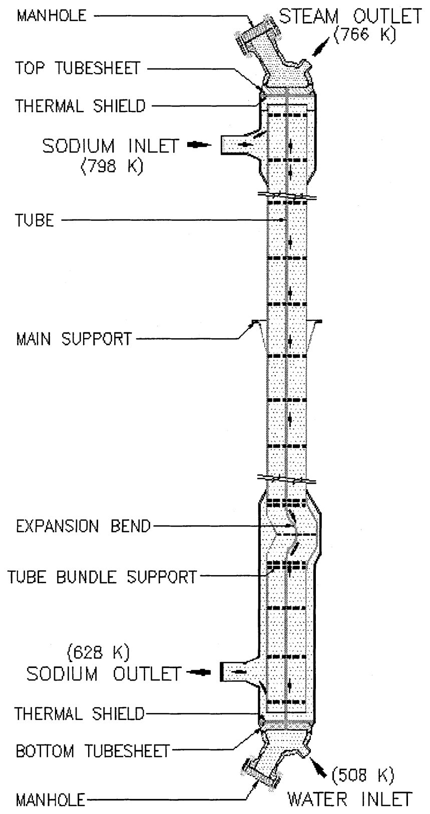

FIG. 1 STEAM GENERATOR

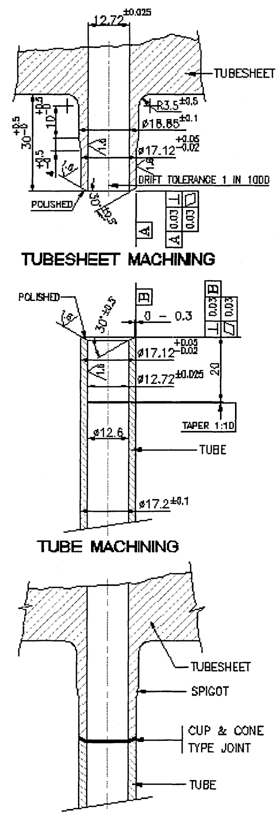

FIG. 2 TUBE TO TUBESHEET JOINT

Componentry, constructive and process solutions of sodium vapour precipitation problem

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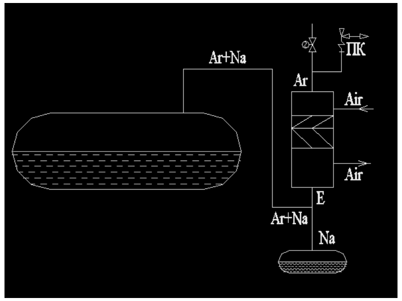
Sodium vapour trap for periodic operation (SVT) is installed to present of sodium vapour emissions after response of safety valve on tanks with sodium and to provide protection from sodium vapour during planned argon blowing from tranks.

It is recommended that SVT be placed directly above tanks with sodium. But the main problem of BN-600 and BN-800 componentry (grouping) is the lack of premises. So, the recommended placement is impossible.

The principal scheme of SVT piping BN-800



Argon purification from sodium vapor is carried out by air refrigerating. Refrigerating degree is regulated by control valve on the air delivery pipe to SVT.

There is montejus tank in the scheme of SVT piping for liquid sodium drainage that is condensed in SVT. Sodium drainage pipe is combined with argon delivery pipe (line E).

There are two main problems with the present construction of SVT based on operation experience of BN-600:

1. The horizontal pipeline "Ar+Na" before the SVT entrance is a dangerous section of this piping. Electric heating of this pipeline is always "on". In spite of this, sodium vapour condenses before SVT. It means that the pipeline becomes progressively clogged up.

2. Two substances (argon with sodium vapour and liquid sodium) are moving towards each other in one pipeline (line E). This is the most probable place of clogging by sodium, especially branch-pipe in the connection point of line E with SVT.

Sodium cruds turn into solid state in the process of argon delivery pipe clogging. In most cases solid scrubs melting temperature exceeds 400-500C depending on chemical composition of crud (for example caustic soda - 1200C). It means that it is almost impossible to heat the pipeline to restore its passability.

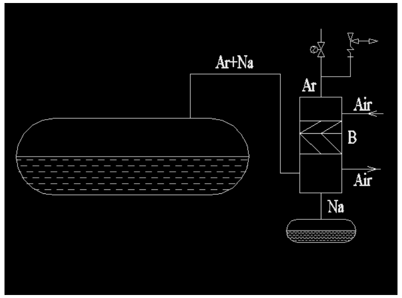
The only measure in this case is the change of pipe section during BN stop.

The first problem can be solved by:

- Increasing the diameter (which was DN40 in the project of BN-600, and is DN80 in BN-800);
- Increasing the temperature of electric heating;
- Minimizing pipeline lengths;

The second problem can only be solved by changing SVT design:

The principal scheme of SVT piping that is optimal in the lack of premises



Argon delivery pipeline is separated from sodium drainage pipeline. Drainage clogging is less probable in such SVT design.

Challenges in PFBR Civil Construction

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Civil works of PFBR has posed numerous design and construction challenges. Eight safety related nuclear buildings of PFBR are founded on a single raft, first of its kind in Indian nuclear Reactors. Construction of eight interconnected buildings on the common raft has been challenge for construction material transportation. The construction of Nuclear Island Connected Buildings (NICB) raft was completed in three and a half months. This massive100mx100m raft for NICB, requiring 35000 cubic meter of concrete was undertaken in twelve pours. This Herculean task required storage of massive quantities of raw materials and ice. The construction of Turbo Generator (TG) building including TG mat and deck and interconnection with NICB posed enormous construction challenges that were well planned and executed. The intake and outfall structures too are exemplary construction handled effectively at PFBR.

The excavation for the nuclear and power islands was large and deep. Rock out crops were found starting at various depths of excavation, requiring extensive planning controlled blasting and mechanization in excavation. This also required innovative dewatering procedures in sandy terrain often mixed with clay. Ground water table at PFBR site being high, water proofing for massive underground structure of PFBR was required to be carefully engineered and handled. This started with innovative grouting of excavated strata below the raft till required permeability was met and bone dry condition was achieved for laying of waterproofing membranes.

Concurrent construction and regulatory reviews and stage wise clearance for civil construction required dynamic construction planning and sequencing.

The scope of civil construction in PFBR expands to critical steel fabricated structures integrated with civil works. This includes huge embedment erected to very close tolerances. Large size panels for upper lateral, lower lateral and bottom shields were fabricated to intricate contours with tight tolerances and were transported to locations from far off distances and erected to tolerances. Massive fixturing had to be resorted to meet tolerances of reactor vault. The reinforcement and concreting placement required protection of the biological shield cooling pipes integrated with the vault liners. Novel construction practices and stringent quality control procedures were implemented to accomplish the critical erection requirement. Integration of cooling coils inside the safety vessel flanges required special contour bending for numerous reinforcement rods and development of special construction methodology. The neutron detector box integration with bottom shield was also a construction challenge well addressed by site.

Many features of Nuclear Island Connected Buildings(NICB) required development of innovative construction techniques. This included construction of underground slabs at Steam Generator Building after erection of the massive sodium storage tanks. Maintaining the stringent requirements such as clean room conditions was a major challenge and was met effectively. Equipment erection in parallel to civil works required innovative solutions for clean condition maintenance while civil works are in progress. Heavy density hatches weighing approximately 150 MT is also a major challenge in the civil construction above argon buffer tanks. The construction of huge cells in reactor buildings require heavy density concrete to be poured in controlled manner to avoid segregation. Three densities of heavy concrete, 3.2, 3.8 and 4.2 gm/cc are used in Reactor Containment Building (RCB).

Many mock ups were carried out for demonstrating the construction procedures and methodology for critical construction activities. The construction engineers ensured that constructability of all critical areas where concrete flow could be doubtful is established on mock ups prior to actual construction.

In conclusion it can be said that the civil construction of PFBR is a marvel of technology handled professionally by competent engineers deploying scientific construction techniques and stringent quality control procedures.

Design of seismic base-isolation for the Super-safe, Small and Simple (4S) reactor building

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This report describes the seismic base isolation (SBI) design of the 4S reactor building. Currently, there are no codes and regulations specifically for the application of seismic base isolation to nuclear power plants in the U.S. Therefore, the present 4S seismic base isolation design is based on a Japanese guideline for the application of seismic base isolation to nuclear power plants.

The current 4S design uses Lead-rubber bearings (LRBs), but the other types could also be used in the 4S design. An LRB device is composed of natural rubber and a lead plug damper. As a whole, an LRB device has very nonlinear dynamic properties. Among the properties, stiffness of the isolators after yielding of the dampers is a key performance indicator of the isolator. Also the yielding force level of the lead dampers is another key.

The design earthquake for the 4S standard design is an Safe Shutdown Earthquake (SSE) as defined in U.S. Regulatory Guide 1.60, scaled with maximum ground acceleration of 0.3g. The Japanese JEAG 4601-2000 guideline[1] states, however, that it is also necessary to pay full attention to the amplitude of the design spectra in the lower-frequency region. Thus, the 4S design earthquake spectra were determined by modifying the Regulatory Guide 1.60 spectra so that they also cover, in the lower-frequency region, another design earthquake spectrum proposed by the Central Research Institute of Electric Power Industry in Japan (CRIEPI) for application of base isolation to nuclear power plants[2].

Result of the dynamic analyses in horizontal direction shows that the maximum acceleration of the isolated building is 231 to 235 cm/sec^2, which is less than the seismic input 0.3g (294 cm/sec^2). The maximum displacement is 212 mm, which will be the design displacement for the isolators. The base shear ratio, the ratio of *total shear force at isolation level to total weight of isolated building*, is 0.236.

The maximum acceleration of the isolated base mat in vertical direction is 337 cm/sec^2, which is larger than the seismic input 0.3g.

Once the design displacement has been obtained, the necessary properties and specifications of the isolators can be determined. This process for 4S is carried out following the Japanese guideline, JEAG 4614-2000. Detailed design of the isolator units are determined.

The effect of changes in soil stiffness on response, viz. the soil-structure interaction, is studied. Two cases of soil properties, the shear wave velocities V_s = 1500 m/sec and V_s = 450 m/sec, are compared. The effect of a change in soil properties on the response of horizontal

direction is very small, while the effect of a change in soil properties on the response of vertical direction is remarkable.

Safety margin of the isolator design is evaluated. According to JEAG 4614-2000, the maximum displacement due to the design earthquake shall be less than 2/3 of the linear-elastic limit of rubber. Beyond the linear elastic limit of rubber, moderate hardening occurs until the rubber ruptures. As the strain, that corresponds to the linear elastic limit, is around 250%, and the strain at rupture corresponds to around 450%, the linear elastic limit of rubber to the rupture strength is around 1/1.8 (2.5/4.5). Therefore, the design displacement is 0.67 (2/3) of the linear elastic limit, and 0.37 (2/3 x 1/1.8) of the rupture strength. A structural allowance for seismic displacement of 500 mm was determined for the 4S standard design. This size is enough even if the seismic input is 0.5G.

Finally, aspects of fire protection, inspection, maintenance are discussed. Concluding remarks include advantage of seismic base isolation and a roadmap to obtain the design approval from Regulatory Committees.

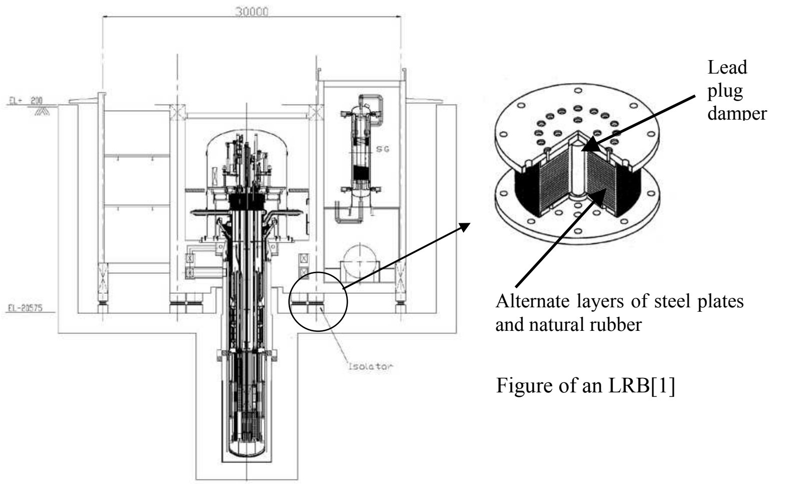

FIG. 1. Section of base-isolated reactor building and an LRB

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Seismic Isolation Design for JSFR

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Japan Atomic Energy Agency (JAEA), the Japan Atomic Power Company (JAPC) and Mitsubishi FBR Systems, Inc. (MFBR) have promoted the conceptional design of the Japan Sodium-Cooled Fast Reactor (JSFR) in the Fast Reactor Cycle Technology Development (FaCT) project.

This paper describes the seismic design of JSFR which consists of seismic condition, seismic isolation system and seismic evaluation of primaly component.

A SFR plant has thin-walled structures, because its thermal stress due to elevated temperature is much higher than that of LWR and its internal pressure is much lower than that of LWR. The thin-walled structure is strongly affected by severe earthquakes. Therefore, JSFR employs the seismic isolation system to mitigate earthquake force.

The design seismic condition became severer than previous conditions because of the *The Niigata-ken Chuetsu-oki Earthquake* in 2007. The earthquake force which affects the primary components had to be mitigated more than that of the previous seismic isolation system. We examined the advanced seismic isolation system by optimizing the performance of the previous seismic isolation system considering the natural frequency of the primary components. The advanced seismic isolation system for SFR adopted laminated rubber bearing which is thicker than rubber of previous one as well as oil damper.

The features of the advanced seismic isolation system is as follows:

- mitigation of the horizontal seismic force by thicker laminated rubber bearing with longer period and improvement of damping performance by adopting oil dumper instead of steel bar dumper, and

- mitigation of the vertical seismic force by thicker laminated rubber bearing with longer period.

We also investigated the seismic evaluation of the nuclear reactor component under the condition of applying the advanced seismic isolation system, and confirmed the performance of that.

POSTERS OF SESSION 9:

Past twenty years with fast reactors and experimental facilities: experience and prospects

Engineering R&D for Sodium Cooled Fast Breeder Reactor in India

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Effective utilization of the moderate uranium resources in India, had lead to the launching of the Fast Breeder Reactor (FBR) programme in the 70's. FBTR (40 MWt, 13.2 MWe) was the first FBR constructed in India in 1985 with French collaboration. While the design was based on the French Fast Reactors, the manufacturing, erection, commissioning and operation was totally indigenous. The R&D for FBTR was minimal in view of the validated design obtained from France. One hundred & fifty tonnes of sodium was purified to nuclear grade and purification techniques were developed with emphasis on safe handling of sodium and sodium fires. Basic heat transfer studies, testing of control rod drive mechanisms, calibration of magnetic flow meters, rupture disc development, calibration of in sodium hydrogen detectors and sodium ionization detector for detecting sodium aerosols were some of the key areas in which substantial R&D was carried out [1,2].

The successful commissioning and operation of FBTR led to the launching of the 500 MWe FBR project (PFBR), whose construction is in an advanced stage. PFBR is a large pool type reactor unlike its predecessor and warranted a design that was safe and techno-economic. The design for the reactor was based on sound concepts, experience of operating reactors and inputs gained from the deeper understanding of the complex phenomena. This necessitated a strong R&D base to cater to the needs of PFBR. Salient R&D activities carried out in support of PFBR after the year 2000 are detailed in this paper. The important R&D activities are briefed below.

Component Development and Testing

Indigenously designed and developed prototype reactor components are tested to qualify the reference designs. Components such as Control Safety Rod Drive Mechanism (CSRDM) and Diverse Safety Rod Drive Mechanism (DSRDM), DC conduction pump and Failed Fuel Localization Module (FFLM) were tested and qualified for reactor operations. Valuable experience was gained in the design, development and testing of other components such as under sodium scanner mechanism, high capacity Annular Linear Induction Pump, vapour condenser, sodium freeze seal valves, cold traps, inflatable seals etc..

A Steam Generator Test Facility was set up and operated for obtaining data to optimize the design of SG for Fast Breeder Reactors. Model of sodium heated once through steam generator and sodium to air exchanger were tested. The facility will also be useful to assess the improved designs of auxiliary equipments used in Fast Breeder Reactors.

Developments in Sodium Technology

Sodium technology forms the vital link in the fast reactor systems and experimental evaluations have been carried out to improvise on the existing state of technological developments. Experiments to study the heat transfer and temperature distribution in the top shield were carried out and a suitable cooling circuit for the top shield was designed. Self wastage and adjacent tubes wastage phenomenon in case of a sodium water reaction in a steam generator was studied in an experimental facility. In house developed hydrogen

sensors were performance tested. Mass transfer studies on primary and secondary circuit structural materials, friction and wear studies, self welding studies, fatigue, creep rupture, creep fatigue interaction studies were undertaken to establish the conservativeness of structural materials selected for PFBR.

Different types of sodium cleaning methods were also evaluated as part of decontamination activities. Programmes for qualification of sodium leak detectors for PFBR were also taken up and completed. A cover gas purification pilot plant was constructed and operated to evaluate its performance.

Thermal Hydraulics

The complex thermal hydraulic behaviour of the pool and the reactor system components have been investigated using experimental thermal hydraulic simulations to understand the different mechanisms that can lead to unsafe thermal loadings or vibration failures. The results have been extrapolated to the prototype either directly or through validation of the computational code.

Studies related to pool thermal hydraulics have been carried out in the SAMRAT model [¼ scale of PFBR] to estimate the velocity distribution near IHX window in the hot pool from flow induced vibration considerations of IHX tube bundle, free level velocities and fluctuations from considerations of gas entrainment, thermal stratification behaviour in hot pool components after reactor trip and flow stability in the main vessel cooling baffles. Flow Induced vibration studies on fuel pins, hydraulic testing of sub assemblies to assess the pressure drop & cavitation performance were conducted. Apart from the above, experimental studies have been able to optimize the design of pressure drop devices for flow zoning. Gas entrainment was analyzed in the surge tank using a large 5/8 scale model in water. A natural circulation sodium loop has also been constructed to evaluate the performance of SGDHR components. Theoretical modeling of transients in the natural circulation sodium loop was also taken up.

Sodium Instrumentation

Developments of advanced instrumentation techniques have been accorded priority for deployment in PFBR. R&D for the development of sensors for sodium flow measurement, void detection, ultrasonic under sodium viewing, sodium temperature measurement, cross wire type leak detector in the sodium water reaction products discharge circuit, leak detectors for main vessel sodium leak detection and temperature sensitive magnetic switch are in an advanced stage.

Eddy Current Position sensor to detect the position of absorber rod in DSRDM (DSR) has been developed and tested. Eddy Current flowmeter (ECFM) for primary sodium flow measurement was tested and qualified for reactor operations. Acoustic measurement technique to determine the free fall time of shut down rods CSR and DSR has been evaluated and was found to be attractive. Long Mutual Inductance type level probe required for PFBR were calibrated in sodium.

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Thirty years operating experience at the experimental fast reactor Joyo

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The experimental fast reactor Joyo at the Oarai Research and Development Center of the Japan Atomic Energy Agency (JAEA) is the first liquid sodium fast reactor in JAPAN. The purpose of constructing Joyo was to obtain technical information about liquid metal fast breeder reactors (LMFBR) through experience with its design, construction and operation, and to use the reactor as a fast neutron irradiation facility for the development of fuels, materials, and other components required for the LMFBR program. Joyo achieved its initial criticality in April, 1977 as a breeder core (MK-I core). After completion of MK-I operation in 1981, core conversion work was carried out in which the MK-I breeder core was replaced by the MK-II irradiation-test-bed core. Joyo achieved initial criticality with the MK-II core in 1982, and attained thermal output of 100 MWt in 1983. MK-II operation was completed by June 2000[1,2]. The reactor has been upgraded to the MK-III core to increase its irradiation capability. The major objects of this project are the increase of neutron flux of the core and the modification of the cooling system related to the power increase. Joyo achieved initial criticality with the MK-III core on July, 2003, and rated power of 140MWt operation was started from 2004[2].

Through design, construction, testing, operation and maintenance experience, Joyo has contributed much to the LMFBR development program. Thirty years of successful operation of Joyo has provided a wealth of experience covering core management, chemical analysis of sodium and cover gas for impurity control, demonstration of inherent safety by natural circulation tests, upgrade of the fuel failure detection system, corrosion product measurements, replacement of major components in the cooling system, etc. The components of Joyo have been performing satisfactorily since the initial criticality in 1977, and many valuable experiences have been obtained regarding LMFBR components and systems. Installation of new components demonstrated the feasibility of modification and improvement of LMFBR plants.

In addition to providing operating experience, many kinds of irradiation tests have been conducted for the development of fuels and materials under the conditions of higher fast neutron flux and temperature than those in LWRs. During the operation of MK-II irradiation test bed core, many kinds of irradiation experience were accumulated, such as monitoring of the driver fuel and control rod performance, and irradiation of test fuels and materials using irradiation test devices to develop the fuels and materials for the prototype reactor MONJU and future fast reactors. Irradiation tests have started again by the MK-III improved irradiation core, and carried out unique irradiation tests such as the demonstration test of holding stability using the reduced-scale experimental equipment of Self actuated shutdown system (SASS), the irradiation behaviour of mixed oxide fuel bearing minor actinides (MA-MOX) and in-pile creep rupture experiment of oxide dispersion strengthened (ODS) ferritic steel with using new irradiation test device named MARICO. The results are utilized in the Fast Reactor Cycle System Technology Development Project in Japan, the development of Monju and

follow-on fast reactors, the international Generation-IV study, and developing initiatives for accommodating worldwide nuclear expansion such as the Global Nuclear Energy Partnership.

Joyo has been operated successfully for thirty years since its criticality was first achieved in 1977 without any major trouble, and this operation has demonstrated the safety and reliability of the sodium cooled fast reactor technology. Given the worldwide trend of fast reactor shutdowns, Joyo is an increasingly valuable world resource for current and future reactor development.

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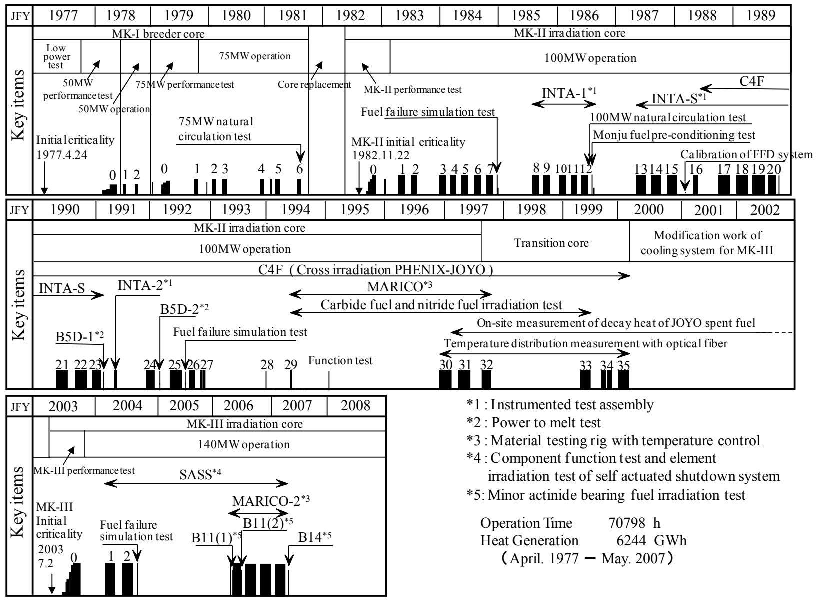

Design validation and BN-800 power unit construction status

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In 2006, the Government of the Russian Federation has approved the Federal target program "Development of nuclear power generation complex in Russia up to 2015". Along with accelerated construction of large VVER-1000 power units, the program envisages the development of nuclear power innovative technologies. They include the activities in fast neutron reactors, which are involved in the perspective to change nuclear power over to closed fuel cycle, which provides the most efficient use of uranium resources and solution of highly actual political and ecological issues of SNF and RW handling. The central stage of the innovation component of target program deals with construction of fast sodium reactor BN-800 that shall take the important stage in development of fast breeder reactor technology and in generation of closed fuel cycle of nuclear power industry. BN-800 design has been developed on the basis of BN-600 reactor, which operates successfully at Beloyarsk NPP starting from 1980. During this period, the reactor plant shows high and stable indices of reliability, safety and efficiency. Average annual plant capacity factor of power unit from the moment of reaching of design rate (from 1982) is 75.7%. In addition, unscheduled loss of plant capacity factor value due to process equipment failures is small (~ 2%). The final technical decisions on BN-800 were made in 1990th after Chernobyl NPP accident and following strengthening of requirements for nuclear-power safety. For qualitative safety enhancement, BN-800 design implements some predominantly passive systems (additional emergency protection rods hydraulically suspended in sodium coolant, emergency cooling air HX, tray for confining of core melt under hypothetic accident). During design upgrading, the improvements aimed at enhancement of technical-economic indices were also introduced. The main fact that allows sufficient decrease (by 34%) of RP specific metal consumption is BN-800 power increase by ~45% at RP vessel overall dimensions close to BN-600. In addition, technical-economic indices are improved due to implementation of one turbogenerator instead of three ones as in BN-600 and due to some other new design decisions.

BN-800 RP and power unit design as a whole is based on large scope R&D work, experimental check results and BN-600 operation experience. The nowadays R&D work is aimed at the development of missing documentation and special-purpose RP equipment among other things:
- development and mastering of MCP speed regulation system;
- development and mastering of SG automatic protection system;
- development of process fluid parameter measuring equipment;

- testing of sodium valves;
- development of system of ultrasonic observation in sodium;
- development of core design for the initial reactor loading;
- validation of power unit service life up to 45 years.

Large amount of R&D work is related to the core.

BN-800 is directed to mixed uranium-plutonium fuel while BN-600 that was developed to master nuclear reactor technology, uses enriched uranium dioxide. As for BN-800, mixed uranium-plutonium fuel will be fabricated on the basis of both depot stocks of power-producing plutonium and weapon-grade plutonium that will be utilized as excessive for defence. Development of BN-800 spent mixed uranium-plutonium fuel reprocessing production will realize industrial demonstration of closed fuel cycle using fast reactors.

As for the first period of fast reactor operation, MOX fuel is acceptable considering its commercial production; however, in future it will be necessary to change over to more dense fuel that provides increased breeding. For BN-800, the nitride-fuel core option has been developed that ensures breeding ratio close to 1 directly in the core. From a perspective of technical-economic indices improvement, the important issue is fuel burnup increase that is connected with improvement of radiation resistance of fuel cladding materials.

As for the first period of reactor operation, fuel claddings of austenitic steel ChS-68 c.d. (06Cr16Ni15Mo2Mn2TiVB) mastered in BN-600 with maximum possible burnup of 10% h.a. (damage dose is ~90 dpa) will be used. Further, it is planned to use improved austenitic steel EK-164 c.d. with maximum possible burnup increase up to 13% h.a., and then, ferrite-martensitic steel will be used. ferrite-martensitic steel EP-450 will be used for FA wrapper from the very beginning, same as in BN-600 reactor.

To validate new steel applications for fuel element claddings, corresponding experimental studies are performed in BN-600 reactor.

Preparation for construction of the 4th power unit started at Beloyarsk NPP site in 1984; however, new NPP construction was frozen after Chernobyl accident. Now, three concrete-mixing plants and nine fixed tower cranes operate at 150 ac site, "Demag" crane with the lifting capacity of 600 t has been assembled to install reactor vessel and heavy equipment tied together. More than 2000 people participate in construction and assembling operations; it is planned to increase personnel quantity. In 2008, a separate building for reactor assembling has been put into operation where reactor vessel components are preliminary assembled. It enables reactor installation acceleration and at the same time, ensures the required quality of assembling and assembly-welding work. BN-800 RP equipment is entirely fabricated at the Russian enterprises. Fifteen large plants are involved as a whole. OKBM bears general responsibility on RP equipment set delivery. This company is RP designer. The effective part of this equipment will be fabricated in OKBM. To reduce the period of power unit construction, the schedule for equipment fabrication and supply is combined with the schedules of building construction and large equipment installation. It is planned complete power unit construction in 2012.

From MYRRHA/XT-ADS to MYRRHA/FASTEF: the FP7 Central Design Team project

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SCK•CEN in association with 18 Europeans partners from industry, research centra and academia, responded to the second FP7 call from the European Commission to establish a Central Design Team for the design of a **FA**st **S**pectrum **T**ransmutation **E**xperimental **F**acility (FASTEF) able to demonstrate efficient transmutation and associated technology through a system working in subcritical and/or critical mode. The proposal prepared by the consortium coordinated by SCK•CEN has been accepted for funding and the project has started on April 01st, 2009 for a period of three years.

Since 1998, SCK•CEN has been designing a multipurpose Accelerator Driven System (ADS) for R&D applications, called MYRRHA, which consists of a proton accelerator delivering its beam to a liquid Pb-Bi spallation target that in turn couples to a Pb-Bi cooled, sub-critical fast core.

MYRRHA in its 2005 version (also called "Draft-2") has been offered as a starting basis for the XT-ADS design within the EUROTRANS project in the FP6 in the context of Partitioning and Transmutation. EUROTRANS aims to deliver a design of a short-term, small-scale ADS, called XT-ADS, and also looks at the long-term, larger-scale European Facility for Industrial Transmutation, called EFIT. This project has started in April 2005 and is scheduled to be completed in March 2010.

As a further upgrade of XT-ADS, MYRRHA/FASTEF is proposed to be designed to an advanced engineering level for decision to embark for its construction at the horizon of 2012-2013 with the following objectives:

- be operated as a flexible and high-flux fast spectrum irradiation facility;
- be an experimental device to serve as a test-bed for transmutation by demonstrating the ADS technology and the efficient transmutation of high level waste;
- contribute to the demonstration of the Lead Fast Reactor technology without jeopardizing the two above objectives.

The work of the Central Design Team (CDT) has been organised in 4 technical work packages.

Work package 1 deals with the definition of specifications and detailed work programme of MYRRHA/FASTEF. The purpose of this work package is to analyse and review the XT-ADS design choices made in view of the different objectives set forward for MYRRHA/FASTEF as outlined above. This work package is divided in three separate tasks:

- input and review of design choices and methodology for XT-ADS;
- design modifications due to the introduction of a critical operation mode;
- analysis to which extent MYRRHA/FASTEF can cover the objectives of a demonstration reactor for the Lead Fast Reactor technology.

At the end of WP1, a definition of specifications and design choices together with a detailed work programme for MYRRHA/FASTEF will be presented. As WP1 is scheduled to occur in the first six months of the project (April – September 2009), the full paper will present more in detail its conclusions.

Work package 2 deals with the design of MYRRHA/FASTEF in sub-critical & critical mode. The starting point will be the XT-ADS design provided at the end of IP-EUROTRANS. In IP-EUROTRANS, the conceptual design of MYRRHA "Draft-2" has been taken as starting point; it has been critically reviewed, which led to some significant conceptual design modifications. Dedicated parts within the primary system, the core and the windowless target have then been further detailed to obtain an advanced design. Other components will only reach a conceptual design level at the end of IP-EUROTRANS. The purpose of this work package is to obtain an advanced level of design for all these components at the end of the project.

Work package 3 deals with plant requirements. These studies will comprise all the MYRRHA/FASTEF facility infrastructures, except the primary and secondary systems, spallation loop, accelerator and the beam line. This study will result in a comprehensive functional description complemented with the characteristics, and main technical requirements, of the auxiliaries to fulfil all plant functions and requirements for both the sub-critical and the critical options as well as in an overall plant layout. This work package will also include a specific task on Instrumentation & Control for sub-critical & critical mode operation. In this task, the necessary instrumentation for the nuclear island and accelerator will be identified. The necessary reactor control and scram logic and interlocks will also be defined.

Finally, work package 4 deals with the key issues towards realisation. Several key issues can indeed hinder the future realisation of the facility and are addressed in this work package:

- fuel design, procurement, demonstration & qualification;
- global cost of the facility and financing scheme;
- licensing procedures in both modes of operation;
- operation mode analysis, R&D needs and activities.

The design of MYRRHA/FASTEF with the above mentioned objectives is completely in line with the vision on the European Research Area on Experimental Reactors (ERAER) prepared in the framework of the Strategic Research Agenda (SRA) of the Sustainable Nuclear Energy Technology Platform. The ERAER is based on three pillars:

- the Jules Horowitz Material high performance Testing Reactor (JHR), mainly devoted to the Generation II and III reactors needs and in a back-up role for radio-isotope production;
- MYRRHA/FASTEF for the Generation IV and fusion reactor needs and in a back-up role for radio-isotope production;
- PALLAS for providing the radio-nuclides for medical applications with some possibilities to address GEN II & III reactor needs.

The ENEA TAPIRO Fast-Source Reactor for Neutronic Research

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TAPIRO is a fast neutron source reactor operating at CASACCIA Research Center since 1971. The project, entirely developed by ENEA's staff, is based on the general concept of AFSR (Argonne Fast Source Reactor - Idaho Falls).

The reactor is basically composed of a cylindrical uranium-metal critical core (radius ≈ 6.3 cm, height ≈ 10.9 cm) enriched to 93.5% in uranium-235, surrounded by a natural copper reflector (external radius = 41 cm) and a large concrete biological shield; it is cooled by helium; reactivity control is achieved by adjustment of the core neutron leakage from the reflector.

The cylindrical uranium-metal core is made up by two overlapped cylindrical blocks: the upper one is fixed to the reactor structure whereas the lower one is movable and can drop (gravity driven) in order to rapidly shut-down the system.

A 60° sector of the external copper reflector is removable allowing insertion of fissile spectral conversion zones feeding the thermal column.

Four experimental channels take place within the system: three different channels at the reactor mid-plane and one tangential (to the top edge of the core) channel. One mid-plane channel crosses over the core allowing measures of small samples (internal diameter of the channel in correspondence of the core ≈ 1 cm) in an almost pure U-235 fission spectrum.

A large experimental cavity, labelled thermal column (parallelepiped 110 × 110 × 160 cm), is provided within the shield zone.

The TAPIRO possibilities for reactor experiments with energies up to 1.35 MeV will be illustrated.

Sodium Test Plan and Facility for JSFR developments

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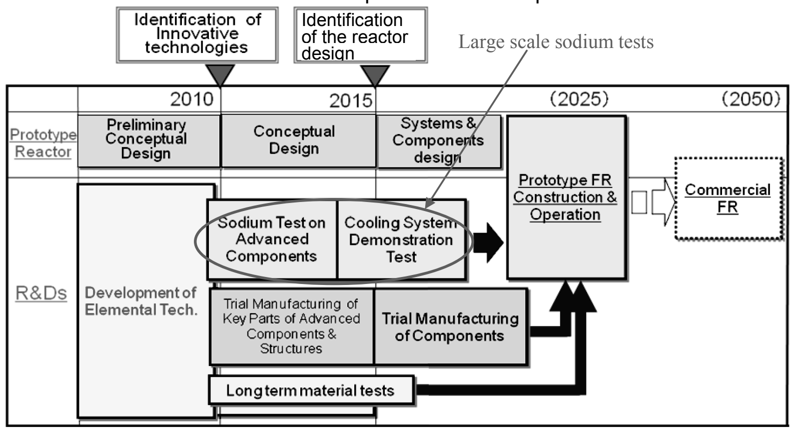
The Japan Atomic Energy Agency (JAEA) and electric utilities in Japan have proposed the sodium cooled fast reactor concept (Japan Sodium Cooled FR; JSFR) as one of the candidates of future commercialized power reactors. JSFR conceptual design studies and experimental research have been carried out from JFY2006, and this developing project is called as "FaCT" (Fast reactor Cycle Technology development) project [1].

FaCT project has been carried out with the road map shown in table one [2]. The project includes cooling components tests and system demonstration, and they need sodium test facilities that can install demonstration scale sodium components and systems.

Table 1 Roadmap of JSFR Development



In order to perform demonstration scale sodium experiments, new sodium test facility has been built from JFY2009. Experiments in the sodium test facility are planed from JFY2012 for cooling components development of JSFR.The facility has following features.

- Flexible configuration and schedule for experimental demand
  - "Mother loop" has integrated utility functions, such as sodium storage, charge, drain, and purification, etc.
  - "Daughter" test loops are independent each other and sodium is supplied from the mother loop.

- ✓ Several experiments at the same time
- ✓ Construction and maintenance schedule of test loop is flexible
- ✓ Less construction and operation cost

- Total solutions for FBR development in Oarai R&D center
  - ✓ Cooperative with other tests on safety, thermo-hydraulics research test facility or irradiation experiments in one site.
  - ✓ Latest equipments for measurement and analysis
- Experiments with Reactor grade sodium and SFR condition
  - ✓ Sodium inventory is over 300 ton
  - ✓ Temperature range is ～600℃
  - ✓ Maximum thermal output is ～60MW

The sodium test facility has large sodium capacity, thermal output, and building. It has flexibility in the configuration and schedule. This is also very useful for the next generation SFR development programs.

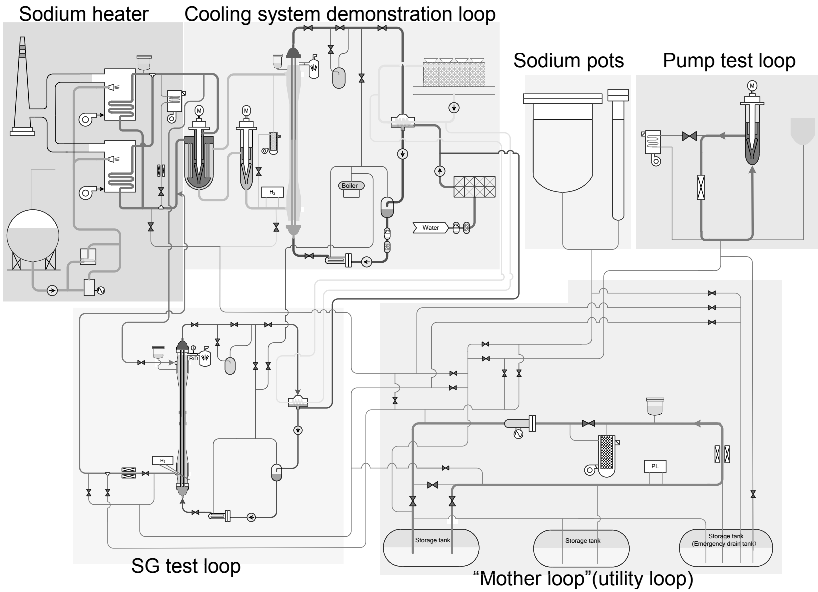


Fig.1 Schematic diagram of the sodium test facility

References:
[1] K. Aoto, et.al. "Current Status of Japanese Sodium Cooled Loop Type Fast Reactor (JSFR) Development", ENC2007
[2] "An Argument of R&D and demonstration process on FBR", MEXT, METI, FEPC, JEMA, and JAEA, 2006

Overview of sodium test facilities in Takasago R&D Center of Mitsubishi Heavy Industories, Ltd.

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Sodium test facilities were constructed and operated in MHI since 1975. MHI had several sodium test facilities, which were multi-purpose sodium testing facility (MTL-3, MTL-4), thermal transition test facility in Sodium (SASS), materials test facility in sodium, and thermal hydraulics test facility (SGTF). MHI have MTL-3, SASS and SGTF at present.

MTL-3 was originally constructed in 1975. It was transferred to Takasago R&D center and fabricated in 1987. MTL-3 can simulate the concentration of oxygen and the temperature of the primary sodium loop. It is suitable for material tests or compatibility tests with sodium. The control temperature of its cold trap is 120 deg. C. The design temperature is 550 deg. C. Test pot in 0.7m dia. and 1.9m height has flexibility for various tests.

SGTF was constructed in 1985 for heat transfer and thermal hydraulic test of steam generators. SGTF have two sodium systems, which are elevated temperature sodium system (800l/min) and low temperaure sodium system (400 l/min). It is able to set a test condition of high temperature and low temperature. Maximum sodium inventry is about 8 tons. Sodium heater capacity is 800 KW. SGTF specification is suitable for simulating sodium hydraulic, cover gas hydraulic in reactor vessel, demonstration of components, heat transfer performance test of SG tube or erosion test of piping.

Various sodium tests for fast breeder reactors (FBRs) have been carried out using these test facilities. They were used for fatigue test, creep fatigue test, immersion test of materials and piping bellows, characteristic test of the motor using under sodium, heat resistant test of self actuated shutdown system (SASS), sliding tests of slide joint, the hydrogen behavior test in the cover gas of sodium, test for heat extension control rod (ETEM), heat transfer flow characteristics of reactor vessel wall (reactor wall cooling test), steam generator tube of 9Cr-1Mo steel, and an ultrasonic sensor under sodium (USAM, etc), for example.

MHI has accumulated techniques on sodium by operating sodium test facilities.
・Purity control, and rinsing method
・Heat transfer character, and thermal hydraulics of sodium
・Leakage and burning behavior of sodium
・Material properties in sodium

SGTF and MTL-3 are now being renewed to improve its reliability and flexiblility of operation because it became superannuated. The renewal will be completed in 2009. The control console will be replaced to digital control console. Electro magnetic pump will be replaced. Heaters, cold-traps, and plugging meter will be replaced. Automatic control will be adopted widely, such as heaters, argon gas system, and plugging meter.

MHI is going to carry out a lot of sodium tests as the core company of FBR development. Fatigue test, slide or wearing test, immersion test of materials, FSW (Friction Stir Welding) test in sodium will be carried out.

Out-of-pile Experimental Base to Justify Fast Reactors and Prospects of Its Further Development

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The paper makes an overview of the current status of the experimental and test base that is used to justify fast reactors (FR) being in operation and under development in Russia. The paper gives the description of the largest test facilities and out-of-pile experimental rigs that are in operation at the leading research institutes and enterprises (SSC RF-IPPE, OKBM, SSC RIAR, RDIPE, OKB "GIDROPRESS", etc.).

Consideration is also given to the prospects of further development of the out-of-pile experimental and test base aimed at justification of advanced FR designs of the fourth generation that are under development at the Russian institutes and enterprises.

The task to develop the out-of-pile experimental base is one of the key trends in implementation of the Federal Target Program (FTP) "Nuclear power technologies of a new generation" for 2010-2020, that is under approval in the Government of the Russian Federation. The FTP envisages development of three innovative FR technologies, i.e. a sodium-cooled fast reactor (SFR), a lead-cooled fast reactor (BREST) and a lead-bismuth fast reactor (SVBR), and the respective fuel cycles. In compliance with these goals the FTP presupposes development, upgrading and reconstruction of the required experimental base of Rosatom in terms of the following aspects:

- core physics;
- reactor system technology;
- materials.

In the field of core physics studies it is envisaged to technically reequip the complex of big test facilities (BFS) at the SSC RF-IPPE.

In the area of reactor system technology it is planned to concentrate the research work in the following areas:

- hydraulics and heat exchange in the flow-through part of the equipment;
- coolant technologies (control and purification systems);
- operability of mechanisms and systems;
- technological safety of equipment and systems;
- endurance testing.

In the field of material science the research work is envisaged in the following aspects:

- justification of material efficiency in the core and circulating loops in contact with the coolant;
- material radiation resistance studies.

Current State of RIAR Experimental Base for Fast Reactors Development

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SSC RIAR

At present, Research Institute of Atomic Reactors (Dimitrovgrad) is a large research center that has unique capabilities for experimental research into the problems of fast reactors.

RIAR site hosts Europe's largest complex for materials testing and studies of nuclear reactor core elements, samples of irradiated materials and nuclear fuel, as well as a complex for doing research in the nuclear fuel cycle area.

The RIAR material science complex performs a full set of reactor materials research, fabricates samples, fuel element mock-ups and experimental devices for irradiation in research reactors at a given temperature, flux density and neutron spectrum, as well as performs PIEs of full-size BN-reactor FAs. The equipment and procedures available at the complex provide safe conduction of reactor core elements research under accident conditions, including research of melted spent fuel.

The complex for doing research in the nuclear fuel cycle area is the only one in the world that implements the safest and ecologically-clean "dry" pyrochemical method of reprocessing nuclear fuel in molten salts and where an automated remotely controlled process line to produce fuel elements and FAs of fast neutron reactors operates.

The Radiochemical Division develops and justifies methods of reprocessing, transmutation, disposal and storage of MAs and RW from pyrochemical production.

The RIAR research complexes and hot labs are a basic instrument in developing closed fuel cycle technologies of fast neutron reactors on which the right decision making depends when designing fuel cycle facilities, including the commercial ones.

The report covers the RIAR experience in operation of experimental complexes and facilities. It tries to highlight near- and long-term tasks to be solved during the future development of experimental base of fast reactors.

RUSSIAN FAST RESEARCH REACTOR BOR-60 REACTOR: EXPERIMENTAL INVESTIGATIONS

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SSC RIAR

Experimental fast reactor BOR-60 is one of the leading experimental facilities in Russia used to test a large of number fuel pins, fuel assemblies, and control rods of different designs, fuel compositions and structural materials. It is also widely used for trying out the elements of closed fuel cycle, transmutation of actinides and plutonium utilization. BOR-60 reactor and high-capacity experimental base available at RIAR allow various experimental investigations to be performed. Since the BOR-60 startup (in 1969), a large scope of experiments have been done at RIAR practically in all directions that are of interest for the nuclear power engineering and related areas of science and engineering. In addition, a wide experience has been gained in calculation support of experimental investigations.

During its 40-year operation, the BOR-60 core underwent multiple changes. There were more than 120 micro-runs, each micro-run being a reactor state different from others. The experimental investigations performed in different periods of time may be of interest for a researcher.

Results obtained at this reactor contributed greatly to the development of the nuclear power engineering and made a basis for a successful startup and operation of reactors BN-350 and BN-600 as well as for long and safe operation of BOR-60 itself. At present, both the reactor and experiment gained at it are widely used for justification of promising fast reactors.

CALCULATION SUPPORT OF RUSSIAN FAST RESEARCH REACTOR BOR-60 OPERATION AND EXPERIMENTS

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SSC RIAR

Different codes covering practically all calculations are used at RIAR to support experiments performed at BOR-60. All the codes were verified in a large amount of experiments and compared to other codes. A part of codes has been certified or is being certified now.

The codes are based on the following programs used to calculate the reactor neutron and physical characteristics: TRIGEX is used for on-line engineering calculations; MCU-RFFI/A and JARFR (certified) are used for detailed precise calculations.

Other programs – thermo-hydraulic codes DINBOR (certified), GREAT and Vikond2 – allow us to calculate thermal capacity, pressure, coolant flow rate and temperatures in the primary and secondary circuit components (fuel rods, FAs, pipelines, heat exchanges, etc.) under stationary and transient operating conditions as well as to calculate stresses-strained state and simulate the distribution of porosity and fuel components in a fuel rod.

A shell uniting all the above codes and some other programs is software KAR for automated calculations. On the one hand, this software is a flexible instrument for the calculation support of the reactor operation and experiments; from the other hand, it is used to archive various reactor states beginning from its startup.

The above software is widely used at BOR-60 for quick and reliable solution of any operational and experimental task of the experimental fast reactor.

PURPOSE AND PRIORITY TASKS OF A RUSSIAN MULTI-FUNCTION FAST RESEARCH REACTOR

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A rise of commissioning of nuclear reactors is expected in the world as well as a boost of innovative projects activities. Due to the renaissance observed in the nuclear power engineering, research reactors begin to play an important role in selecting promising directions for the development of nuclear reactors, enhancement of their safety and prolongation of lifetime of the existing nuclear reactors. Only a research reactor is suitable to simulate to the maximal extent the effect of damaging factors to justify the reliability of components important for the NPP safety.

Nowadays, the Russia’s needs in performing experimental research are satisfied by four research reactors commissioned more than 40 years ago: MIR, BOR-60, IVV-2M and SM.

According to assessments of the Russian specialists, a fast reactor with sodium coolant is the most universal and efficient nuclear reactor that can provide the highest neutron flux density and hardest neutron spectrum, good thermo-physical characteristics and wide range of operating temperatures.

One of these is BOR-60 that is used for the majority of experiments in justification of innovative nuclear reactors and closed fuel cycle. At present, BOR-60 is the only fast research reactor under operation. However, its lifetime expires in 2016 and an urgent task of the near future is to develop a new research reactor.

A conceptual design of a multifunction fast research reactor (MBIR) has been developed in Russia to show its basic tasks and experimental capacities.

In particular, MBIR should be a powerful source of energy (thermal capacity – 150 MW) to provide a high neutron flux (more than 5×10^{15} $cm^{-2}s^{-1}$); it should have a hard neutron spectrum to provide 50 dpa per year. There also should be several loops cooled autonomously with different operating media (sodium, lead, lead-bismuth and gas), cells for in-pile material tests, instrumented cells and vertical and horizontal experimental channels out of reactor vessel.

MBIR is to be a multi-function reactor to allow investigations in material science, reactor physics and safety, testing of new nuclear reactor components and monitoring and analysis means, accumulation of radioisotopes, etc. This research reactor will provide investigations in both fast and thermal nuclear reactors as well as in other promising directions of the nuclear power engineering for the next decades.

POSTERS OF SESSION 10:

Fast reactor knowledge management, education and training

Model of Fast Reactor Knowledge Preservation System

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Despite lack of the energy market today, fast reactors (FR) in the closed nuclear fuel cycle are the basis of a full-scale development of nuclear power in future. However, there are serious problems conserning the future R&D of these reactor techologies related to the following obstacles.

All research on FR was stopped in Germany, Italy, United Kingdom and the United States and the work performed only dealt with the decommissioning of FR. Many experts who participated in R&D programs to create FR have retired or are approaching retirement age. In France, Japan and Russia work on the development of FR still continues, but there is a lack of young scientists and engineers.

Due to all this factors IAEA launched the initiative to combine efforts of the leading nuclear countries to develop a project for the preservation of knowledge in the field of scientific and technological problems of FR development. Efforts of IAEA and national experts resulted in a model of FR information search and classification (so called «taxonomy of the Fast Reactor Knowledge Preservation System»). This work has initiated a systematic process of creation and filling of information data bank on various aspects of FR design and operation.

As the next step it would be logical to develop self-consistent mathematical models of FR-based NPP and closed NFC with their subsequent introduction into the system of knowledge preservation. So, it will serve as an important step towards preservation of knowledge in the field of FR design through joint development and to ensure open access to software. Such a project may lay the groundwork for the future development of distance learning courses and training on the optimal FR design, with the participation of leading specialists in this field.

The report provides a mathematical and logical model for the preservation of knowledge concerning FR science and technology: taxonomy, an engineering model of FR-based NPP, a FR NFC model.

Network Representation of Design Knowledge of Prototype Fast Breeder Reactor

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A method of design knowledge representation was studied for the Japanese fast breeder reactor Monju, aiming at enhanced understanding of engineering considerations with mutual relations. Taking over design knowledge of Monju to next generation designers/engineers to be in charge of design of future FRs is by no means easy, in contrast with operation and maintenance knowledge which can be acquired in the real plant operation and maintenance. Specifications of the as-is Monju contains only a small part of the entire design knowledge, mainly by two reasons. Firstly, reasons for selecting the as-is specifications can not be understood until reaching proper knowledge source. Secondly, there are many passed-over options on the design specifications. Reasons for passing-over these options are not always technical inferiority. A large part of the current specifications are selected because the worst possible technical value can be foreseeable or guaranteed to be acceptable within limited R&D period and resource, not because the expected value is estimated to be the lower. In other words, in the future where new materials with improved properties, faster and more accurate analysis/prediction methods, rationalized technical standards or regulatory requirements, and/or some other environment for thorough comparison among specification options are available, these passed-over options are likely to be worth reconsidering.

There are a huge number of technical documents on diversified engineering studies, such as calculation of maximum possible temperature gradient of important structures, necessary sodium flow rate in particular sub-assemblies, etc. for validation of each decision making in design. A large part of these documents are scanned and stored in a data base with each catalogue data for electronic browse.

The authors propose a network representation of these items of design decision making, where the items are mutually connected by directed arcs, where nodes stand for decision making(including validations) and arcs stand for handed-over information between decision makings. This representation is expected to help integrated understanding of the plant design, by browsing the technical documents along with the decision making sequence.

After surveying the existing database, three dimensions of categorization are introduced as listed below:

1) scale or granularity of related hardware range: "Plant level", "System level" and "Component level", etc.

2) related plant function: "fuel", "reactor core", "heat transport system", "instrumenation/control", "building" and "miscellaneous", etc.

3) purpose of information: "plant standard", "plant design", "characteristics evaluation", "safety evaluation", "integrity evaluation", etc.

A prototype software to visualize this knowledge representation has been built on a commercial personal computer to validate this concept, as shown in Fig.1. The software draws the network by automated layout of the nodes and arcs, representing user-specified set of local dependencies of the knowledge items.

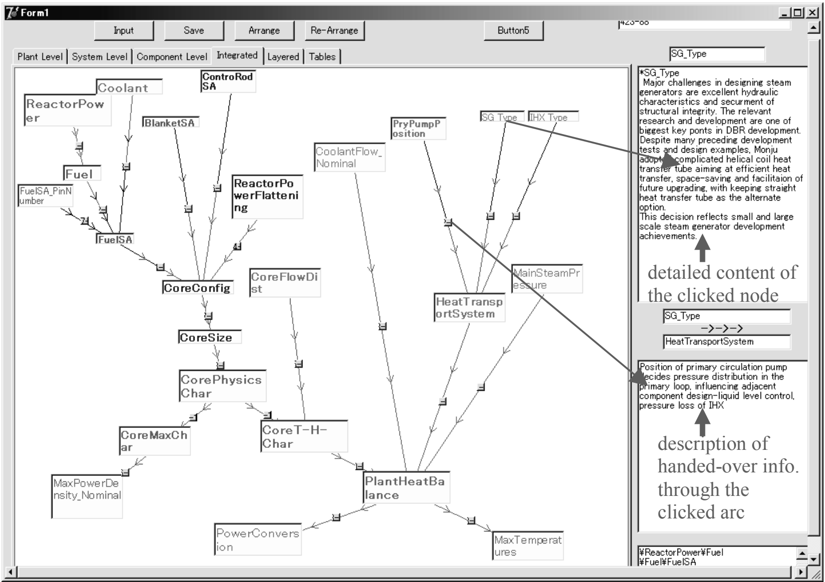

Fig.1: Sample execution result of the prototype software for network representation of design knowledge

A set of user-selected knowledge items in all the levels are shown in single plane in this sample, while related plant functions are distinguished by node colors. The upper box on the right hand side shows the detailed content of the clicked node and the lower box shows description of handed-over information through the arc with a small in the middle.

The prototype software verified that the proposed concept of knowledge representation is feasible on market-available computer resources, while user-friendliness, objectivity of specifying inter-dependency of knowledge items still have a significant room for improvement.

An important possibility to pursue on this concept is support function of hypothetical case study of passed-over options, because inter-relation of specification selection plays a key role. For example, locating the primary circulation pump on the hot leg allows larger pressure loss in hot leg coolant piping and primary side of intermediate heat exchanger, leading to possibility of shorter hot leg pipings and smaller intermediate heat exchanger. This makes some room for more flexible component layout inside the reactor containment vessel. Of course this option requires R&Ds to secure pump and the peripheral device integrity against potential thermal shocks and this is the main reason for Monju to select the cold leg pump option. However, this kind of eased hypothetical case studies can enhance knowledge take over to coming generations of FR designers.

Student internship program using the experimental fast reactor Joyo and related facilities

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The student training courses using the experimental fast reactor Joyo of the Japan Atomic Energy Agency (JAEA) and related facilities have been initiated based on the JAEA's mission to contribute to the human resources development program of the Japanese Ministry of Education, Culture, Sports, Science and Technology (MEXT) and Ministry of Economy, Trade and Industry (METI). The development of the student training courses was also strongly supported by the faculty of nuclear engineering of domestic universities in two reasons: one is that the nuclear related curriculum has recently been reduced due to the trend of decreasing interest by the younger generation in nuclear research and industry, and the other reason is that the aging research reactors and nuclear facilities owned by the universities are very difficult to keep operating. Considering this situation, JAEA decided to cooperate with the universities in developing the student training course.

The experimental fast reactor Joyo[1] of JAEA is a sodium cooled fast reactor with plutonium-uranium mixed oxide (MOX) fuel, which has two primary sodium loops, two secondary loops, and an auxiliary system. An intermediate heat exchanger (IHX) separates radioactive sodium in the primary system from non-radioactive sodium in the secondary system. The secondary sodium loop transports the reactor heat from the IHX to the air-cooled dump heat exchanger (DHX). Joyo has a full-scope type core and plant simulator, which duplicated all the main control panels located in the Joyo central control room. The simulator enables to offer a real time simulation of the plant behaviors under normal and abnormal conditions by applying the plant dynamic analysis code Minir-N2[2] and the same interlock system as the Joyo reactor system.

The sodium analysis facility is located apart from Joyo complex to primarily conduct impurity measurement of Joyo cooling system. These data were measured by chemical analysis, gas chromatography, beta-ray scintillation and gamma-ray spectrometry. In addition to this role, an innovative instrumentation device by means of laser resonance ionization mass spectrometry[3] (RIMS) and the helium accumulation fluence monitor (HAFM) dosimetry system[4] were installed and research study has been initiated with the domestic universities and research institutes.

The Joyo and related facilities prepared five student training courses consisting of five curriculums as follows:

(1) Reactor core physics analysis and comparison with the measured data obtained from the actual Joyo operational cycle
(2) Reactor core physics and plant dynamic experiments using the Joyo full-scope training simulator (Fig. 1.)
(3) Neutron dosimetry by means of multiple foil activation and HAFM methods
(4) Trace amount of tag gas measurements using RIMS
(5) Radiochemical analysis of liquid solution including the tritium measurement in the Joyo secondary coolant sodium (Fig. 2.)

These courses will be provided for the graduate and/or undergraduate students not only for the nuclear engineering course but also for other college students studying science and technology. The training term is one week for item (2) and four weeks for the others. The text book and user's manuals of the reactor core physics analysis code system using engineering workstation or PC were prepared along with the teaching notes. All students can attend the on-site reactor plant lectures at the Joyo facility in order to learn and utilize the actual nuclear reactor and plant systems.

The preparation started in 2006 and the trial was carried out in 2007 for 13 students from four universities. The contents were modified based on the comments and advice of the teachers of those students, and the formal courses were carried out in 2008 for 20 students from six universities. It is emphasized that the reactor core physics analysis of Joyo was verified by comparing with the experiments using the training simulator, and the feedback reactivity experiments are very useful to understand the physical phenomena which occurred in the high power reactor. The established course has a wide scope not only for the nuclear engineering but also the science and technology department of the domestic university, and that meets the JAEA's mission to provide our own facilities for the human resources development.

FIG. 1. Lecture on the reactivity feedback experiment due to the secondary coolant temperature change

FIG. 2. Sodium impurity measurement (Tritium in the secondary coolant of Joyo)

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