International Conference on Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development (FR17)

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Book of Abstracts

Contents

Poster Session 2 / 0

Nuclear Reactor Modelling and Simulation Toolkit (NuReMoST) –Numerical Reactor Model Configuration System with Interface to Simulation Codes

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Nuclear Reactor Modeling and Simulation Toolkit (NuReMoST) is a new generation system for numerical simulation of the nuclear reactor. The system integrates typical simulation tasks, such as preparation of the design and geometry data, setup of the initial and boundary conditions, meshing tools, visualization functions, as well as several other related utilities and services. The core point of the system is to provide a standard interface for coupling of numerous simulation codes. The reactor data storage system collects simulation-related data in a consistent database with all possible and necessary details. NuReMoST provide an access to the stored reactor data and a standard interface for coupling with reactor simulation codes.
The key feature of the system is to provide easy and effective tools for preparation of the reactor simulation model. Easy design with instant visualization provide quick and effective tool for the model preparation and quick update thanks to the instant visual user interface.
NuReMoST is designed to provide easy coupling capabilities. The coupling tools with several popular reactor simulation codes are already included in NuReMoST, many others will be included in the near future. In addition, several simplified simulation tasks including steady-state thermal hydraulic and power distribution calculations are natively integrated into the system. Thus, even standalone REMCOS version can be used for a quick evaluation and for educational purposes.
At the moment, a conceptual prototype of the NuReMoST system is available. The samples of coupling interface with simple thermal hydraulics and neutronics models are presented and discussed. The coupling with reactor neutronics codes, such as ERANOS and SERPENT are under development.

Country/Int. Organization:

International Atomic Energy Agency (IAEA)

Poster Session 1 / 1

Computational investigation of nuclear waste incineration efficiency in a subcritical molten salt driven by 50-100 MeV protons

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Molten salt reactors were designed and operated at 1960s. The subcritical accelerator driven MSRs are being considered recently. In the present work, accelerator driven homogeneous subcritical core configuration was Modelled using MCNPX code. The composition of NaF-BeF2-ThF4-TRUF4 and NaF-233UF4-ThF4-TRUF4 was selected as the fuel loaded inside a 58×60 cm cylindrical core respectively. NaBF was selected as coolant salt of the fuel salt circuit. Accelerated proton particles were used to induce fission in the transuranic nuclei. The projectiles energy was changed from 50

MeV up 100 MeV in five steps. TRU fission rate, deposited heat distribution and neutron flux distribution were determined inside the subcritical core. Neutron and proton flux distribution inside the subcritical molten salt core was compared with each other. Energy gain, source multiplication factor and proton importance parameters were calculated for any different projectile energy. Optimized proton energy was suggested to be applied for nuclear waste incineration using such system. Burn-up calculations were carried out for the cores with different fuel loading.

Country/Int. Organization:

Islamic Republic of Iran

Poster Session 2 / **3**

EBR-II SHRT-17 and SHRT-45R Benchmark Analyses

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In 2012, the International Atomic Energy Agency (IAEA) established a coordinated research project (CRP) on EBR-II Shutdown Heat Removal Tests (SHRT). The objectives of the CRP, which concluded in 2016, were to improve design and simulation capabilities in fast reactor neutronics, thermal hydraulics, plant dynamics, and safety analyses through benchmark analysis of two landmark tests performed during the EBR-II SHRT program. The selected tests were SHRT-17 and SHRT-45R, the most severe protected and unprotected loss of flow tests performed during the SHRT program, respectively. Nineteen organizations representing eleven countries participated in the CRP.

The benchmark was performed in two phases. During the first phase, participants had no access to the recorded data from either test. Once all phase 1 calculations were completed in February 2014, phase 2 was initiated with participants receiving experimental data.

In addition to its role as the lead technical organization for the CRP, Argonne also performed analyses as a participant in the CRP. Argonne simulated the SHRT-17 and SHRT-45R tests using the sodium fast reactor safety analysis code SAS4A/SASSYS-1. Although Argonne's SHRT-17 simulation during the blind phase of the CRP predicted similar trends as the measured test data, overpredicted flow rates after the beginning of the test led to underpredicted temperatures. Predictions of the SHRT-45R flow rates through the core inlet piping agreed much better with the measured data for SHRT-45R than for SHRT-17.

Argonne's modeling efforts during Phase 2 focused primarily on improving the predicted flow rates for SHRT-17. During phase 1, the initial shape of the flow curve was well matched but the magnitude of the flow rate was too high. This discrepancy was corrected during phase 2 by properly accounting for the resistance of the locked pumps and the circumstances under which the pumps were assumed to lock.

For SHRT-45R, additional analyses were performed with the reference power level used as a boundary condition to assess the performance of the systems model, allowing for more accurate analysis of the primary system model without power discrepancies affecting predicted flow rates and temperatures. This poster will present the analyses performed by Argonne during the two phases over the four-year duration of the CRP. Argonne's contributions to the neutronics component of the SHRT-45R benchmark will also be presented.

Note: EBR-II Benchmarks CRP Invited Poster Session

Country/Int. Organization:

USA

6.11 IAEA Benchmark on EBR-II Shutdown Heat Removal Tests / 4

EBR-II Passive Safety Demonstration Tests Benchmark Analyses

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In 2012, the International Atomic Energy Agency (IAEA) established a coordinated research project (CRP) on EBR-II Shutdown Heat Removal Tests (SHRT). The objectives of the CRP, which concluded in 2016, were to improve design and simulation capabilities in fast reactor neutronics, thermal hydraulics, plant dynamics, and safety analyses through benchmark analysis of two landmark tests from the EBR-II SHRT program: SHRT-17 and SHRT-45R, the most severe protected and unprotected loss-of-flow transients, respectively. Nineteen organizations representing eleven countries participated in the CRP.

The benchmark was performed in two phases. During phase 1, participants had no access to recorded data from either test. Once all phase 1 calculations were completed in February 2014, phase 2 was initiated with participants receiving the experimental data.

This paper will summarize the SHRT-17 and SHRT-45R tests and the benchmark specifications for each test that were developed for the CRP. An overview of the major simulation results from all CRP participants will be presented, and lessons learned from the CRP will be discussed.

Two companion papers in this conference will cover in detail the comprehensive analysis results for SHRT-17 and SHRT-45R, respectively. Two additional conference papers will address 1) the optional

neutronics benchmark that was conducted for the SHRT-45R test and 2) the computational fluid dynamics and subchannel models that were used by some of the CRP participants.

Note: EBR-II Benchmarks CRP Invited Session

Country/Int. Organization:

USA

5.3 Advanced Fast Reactor Cladding Development I / 5

IAEA activities in the area of Nuclear Power Reactor Fuel Engineering

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The main IAEA program implementation tools in the area of fuel engineering are Coordinated Research Projects (CRP), Technical Meetings (TM), Expert Reviews, and NEA-IAEA International Fuel Performance Experiments (IFPE) Database. This report provides information about organization and implementation practices of these activities, and summarizes their major outputs including ongoing CRPs and TMs in the area of Fuel Engineering. The first announcement and preliminary information on the new CRP "Fuel Materials for Fast Rectors"to be launched in 2018 will be presented, including main topics for International collaboration within the CRP.

Country/Int. Organization:

IAEA

Poster Session 1 / 6

Model validation of the ASTERIA-FBR code related to core expansion phase based on THINA experimental results

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Mechanical consequences which might be caused by core disruptive accidents (CDAs) are one of the major concerns in FBR safety. Once a severe re-criticality occurs, core materials are vaporized creating a CDA bubble which consists of fuel vapor, steel vapor and fission gases. Rapid expansion of the CDA bubble drives a sodium slug of the upper plenum and threatens integrity of the shielding plug. Energy conversion from thermal energy to mechanical energy plays an important role for the boundary integrity during core expansion phase.
This paper describes model validation study of ASTERIA-FBR*related to the thermal-to-mechanical energy conversion process, focusing on calculation models such as interfacial area model through the THINA** test simulation. As a result, it was found that the energy conversion process and its ratio were in good agreements with the experimental results. Mechanism of CDA bubble expansion and uncertainty brought by calculation models were also discussed.

[References]

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**F. Huber, et.al., "Experiments on the Behaviour of Thermite Melt Injected into Sodium –Final Report on the THINA Tests Results,"Proceedings of IAEA/IWGFR Technical Committee Meeting on Material-Coolant Interactions and Material Movement and Relocation in Liquid Metal Fast Reactors, 167-198, Oarai, Japan, 1994.

Country/Int. Organization:

Japan

7.1 Sustainability of Fast Reactors / 7

Performance and sustainability assessment of nuclear energy deployment scenarios with fast reactors: advanced tools and application

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A performance and sustainability assessment of nuclear energy deployment scenarios with fast reactors is a multi-criteria problem which is determined by a wide spectrum of criteria characterizing resource consumption, economic performance, risks of unauthorized proliferation, safety and waste management etc. In determining the most promising scenario, it is necessary to consider the conflicting nature of the criteria because an improvement in one of the criteria causes, as a rule, a deterioration of values in the others. To increase the validity of judgments formulated on the basis of calculations, an uncertainty analysis is also required. There is a need to use multiple criteria decision making methods.
Multiple criteria decision making methods are a support tool aimed to help experts and decision makers who are faced with numerous conflicting assessments to highlight conflicts and perform proper trading off during the decision making process. A multi-criteria decision analysis and multi-objective decision making constitute the main components of multiple criteria decision making. The major distinction between these groups of methods is based on whether the solutions are defined explicitly or implicitly. Although tasks that can be solved using these methods are different, their combined use seems to be appropriate to identify the most suitable and balanced nuclear energy deployment scenarios with fast reactors despite various contradiction criteria.
The both groups of methods may be applied to assess the performance and sustainability of nuclear energy deployment scenarios with fast reactors by searching for compromises between the conflicting system factors that determine the nuclear energy deployment scenarios and selecting the trade-off option, to carry out multi-objective optimization of nuclear energy structures, taking into account the nuclear energy system evolution, the structure and the organization of nuclear fuel cycle and the most important system constraints and restrictions.
The paper presents the toolkit developed in the National Research Nuclear University MEPhI for a performance and sustainability assessment of nuclear energy deployment scenarios with fast reactors providing a solution to the problem of optimizing and comparing nuclear energy deployment scenarios with fast reactors in multiple criteria formulation. Some results of implementation of this toolkit for the performance and sustainability assessment of nuclear energy deployment scenarios with fast reactors are presented which demonstrate that technologically diversified nuclear energy structures in which several different fast reactor technologies are present may produce a synergetic effect in terms of nuclear energy system sustainability and performance improvement.

Country/Int. Organization:

Russian Federation

Poster Session 2 / 8

Detection and analysis of fuel cladding damages using gamma ray spectroscopy

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The cleanliness of the primary circuit is a safety requirement for Generation IV nuclear power plants. During operation, fission product concentration into the sodium coolant has to be monitored by dedicated radiation monitoring systems. These systems are based on neutron counting and gamma spectroscopy. Neutron detection allows detecting clad failure when its opening is large enough to enable sodium/oxide interaction and then to release volatile fission products as Br, I and Cs [1]. Gamma spectroscopy is an earlier detection system which measure gaseous isotopes of Xe and Kr released during the first step of the failure. Moreover, gamma spectroscopy is a useful technique to examine the status and the evolution stage of the failure. Indeed, using the isotopic information provides by the system, a figure of merit named release to born ratio R/B as a function of the radioactive constant provides an estimation of the failure stage [2].
The study deals with an X/γ rays spectroscopy system named ADONIS and its associated spectrum analysis software SINBAD [3, 4]. This system has been installed at the ISABELLE-1 test loop of the OSIRIS reactor (CEA) [5]. A complete fuel failure has been monitored by the system. The recorded data have been processed and results concerning the fuel analyses will be presented and discussed.
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[3] Barat, A., et al. "A bimodal Kalman smoother for nuclear spectrometry", Nuclear Inst. and Methods in Physics Research A 567, 350–352, 2006.
[4] Rohée, E., et al., "Benchmark of the non-parametric Bayesian deconvolution method implemented in the SINBAD code for X / γ rays spectra processing."Nuclear Inst. and Methods in Physics Research, A. In Press, 2016.
[5] "Research Facilities for the Future of Nuclear Energy", Proc. ENS Class 1 Topical Meeting, 261-270, 1996.

Country/Int. Organization:

France.

8.1 Professional Development and Knowledge Management - I / 9

GEN IV Education and Training Initiative via Public Webinars

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An increasing number of countries are opting for new nuclear energy as an important step towards economic development and environmental protection. According to the IAEA, electricity from nuclear energy may triple by 2050 as evidenced in the report IAEA-RDS-1/33; therefore, the projected use of this carbon-free technology will require many new nuclear engineers and scientists. In addition, countries such as France, the United States, which are the world's largest producers of nuclear energy, are experiencing a decline in the nuclear energy workforce both in their national laboratories and in the private sector. The future vigor and prosperity of nuclear energy and associated nuclear science, clearly depend on continued use of available nuclear reactors as well as the development of advanced nuclear reactor technologies. To maintain the know-how in this field, to increase the knowledge of new advanced concepts, and to avoid the loss of the knowledge and competences that could seriously and adversely affect the future of nuclear energy, the Generation IV International Forum (GIF) established the GIF Education and Training Task Force. The task force serves as a platform to enhance open education and training as well as communication and networking in support to GIF. Indeed, its first initiative is the organization of a webinar series on the next generation of nuclear energy systems (Sodium Fast Reactor, Supercritical Water Reactor, Molten Salt Reactor, Lead Fast Reactor, Very High Temperature Reactor, and Gas-cooled Fast Reactor) and other cross-cutting subjects such as the basics on nuclear reactor systems, thorium fuel cycle, and nuclear fuel and materials.

By exploiting modern internet technologies, the GIF Education and Training Task Force is reaching out to a broad audience and is raising the interest and strengthening the knowledge of participants in topics related to advanced reactor systems and advanced nuclear fuel cycles.

This achievement is the direct result of partnering with internationally recognized subject matter experts and leading scientists in the nuclear energy arena who conduct live webinars on a monthly basis (for more details on the webinar series please see https://www.gen-4.org/).

Besides opening the classroom to everyone in the world, the webinars offer earlier opportunities for interdisciplinary networking and educational and research collaboration. The details and examples of the GIF webinar modules will be presented in our paper.

Country/Int. Organization:

USA/Department of Energy

6.3 Neutronics - 1 / 10

INTEGRAL EXPERIMENTS WITH MINOR ACTINIDES AT THE BFS CRITICAL FACILITIES: STATE-OF-THE-ART SURVEY, REEVALUATION AND APPLICATION

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The paper presents the results of a computational and experimental analysis of a systematized and revised series of experiments carried out between 1990-2013 on measurements of absolute fission rate of minor actinides (from 237Np to 245Cm) in different neutron spectra at the BFS-1,2 facilities. A total of 25 critical configurations, i.e., reactor core models with different fuels and coolants were examined. The earlier experimental data have been revised according to more accurate data processing methods with account of permanent chamber deformations and introduced corrections to the efficiency of detecting fragments (fission events in the chambers). The computational models of assemblies presented in the international handbooks were supplemented with evaluated fission rate ratio models using non-analog algorithms. The resulting consistent set of experimental data and computational models can be used in solving various applied and fundamental problems. A generalization and re-evaluation of a series of integral experiments at the BFS facilities can serve as an information base for the verification and refinement of evaluated minor actinides neutron data. The analysis of all the available set of data on minor actinides fission rate ratio measurements can be used for supporting rationale and planning of research programs for critical assemblies.

Country/Int. Organization:

Russian Federation

Poster Session 2 / 11

Study about the transient characteristics of the unprotected loss of flow accident in a metal fuel sodium cooled fast reactor based on the SAS4A code

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Unprotected loss of flow accident (ULOF) is the most typical severe accident in sodium cooled fast reactor, which is focused by scholars civil and abroad. Metal fuel has different safety characteristics with the oxide fuel as the important development direction of future sodium fast reactor, accident analysis of which is also a research focus at home and abroad. This paper bases on one Cooperation Research Project proposed by ANL and organized by IAEA, analyses the Shut-down Removal Test-45R of the metal fuel sodium cooled fast reactor EBR-II in the US with SAS4A code, to research the transient characteristics of it in ULOF accident. Studies have shown that, metal fuel sodium cooled fast reactor has very good inherent safety performance, which can reduce the reactor power in ULOF accident through the negative feedback itself.

Country/Int. Organization:

China/China Institute of Atomic Energy.

Poster Session 2 / 12

The U.S. Knowledge Preservation Program for Fast Flux Test Facility Data

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An important goal of the U.S. Department of Energy's Office of Nuclear Energy is to preserve the knowledge that has been gained in the United States on Liquid Metal Reactors by collecting, organizing and preserving technical information that could support the development of an environmentally and economically sound nuclear fuel cycle. The FFTF is the most recent LMR to operate in the United States and its 10 years of operation provide a very useful framework for testing the advances in LMR safety technology based on passive safety features. Such information may be of increased importance to new designs after the events at Fukushima. This report describes the knowledge preservation activities related to FFTF legacy information including data from the design, construction, startup, and operation of the reactor and summarizes the current status and accomplishments of the FFTF knowledge preservation activities and lessons learned.

Country/Int. Organization:

USA

Poster Session 2 / 13

Lessons Learned from Fast Flux Test Facility Experience

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The Fast Flux Test Facility (FFTF) is the most recent liquid-metal reactor (LMR) to operate in the United States, having operated from 1982 to 1992 and played a key role in LMFBR development and testing activities. In addition to irradiation testing capabilities, FFTF provided long-term testing and evaluation of plant components and systems for LMFBRs. The FFTF was highly successful and demonstrated outstanding performance during its nearly 10 years of operation. The technology employed in designing and constructing this reactor, as well as information obtained from tests conducted during its operation, can significantly influence the development of new advanced reactor designs in the areas of plant system and component design, component fabrication, fuel design and performance, prototype testing, site construction, and reactor operations. Efforts have been made to preserve important lessons learned during the nearly 10 years of reactor operation, and a brief summary of these Lessons Learned will be discussed.

Country/Int. Organization:

USA

Poster Session 2 / 14

Passive Safety Testing at the Fast Flux Test Facility Relevant to New LMR Designs

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Significant cost and safety improvements can be realized in advanced liquid metal reactor designs by emphasizing inherent or passive safety through crediting the beneficial reactivity feedbacks associated with core and structural movement. This passive safety approach was adopted for the Fast Flux Test Facility (FFTF), and an experimental program was conducted to characterize the structural reactivity feedback. The FFTF passive safety testing program was developed to examine how specific design elements influenced dynamic reactivity feedback in response to a reactivity input and to demonstrate the scalability of reactivity feedback results to reactors of current interest. Benchmarks based on empirical data gathered during operation of the FFTF as well as design documents and post-irradiation examination will aid in the validation of software packages and the models and calculations they produce. Evaluation of these actual test data could provide insight to improve analytical methods which may be used to support future licensing applications for LMRs.

Country/Int. Organization:

USA

Panel 1: Development and Standardization of Safety Design Criteria (SDC) and Guidelines (SDG) for Sodium Cooled Fast Reactors / 15

The Safety Design Guideline Development for Generation-IV SFR Systems (Japan/GIF)

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The Generation-IV International Forum (GIF) Safety Design Criteria Task Force (SDC TF) has been developing a set of safety design guidelines (SDG) to support practical application of SDC since completion of the "SDC Phase I Report"that clarifies safety design requirements for Generation IV SFR systems. The main objective of the SDG development is to assist SFR developers and vendors to utilize the SDC in their design process for improving the safety in specific topical areas including the use of inherent/passive safety features and the design measures for prevention and mitigation of severe accidents. The first report on "Safety Approach SDGs"aims to provide guidance on safety approaches covering specific safety issues on fast reactor core reactivity and on loss of heat removal. The second report on "SDGs on key Structures, Systems and Components (SSCs)"focuses on the functional requirements for SSCs important to safety; reactor core system, reactor coolant system, and containment system.

Country/Int. Organization:

Japan

3.5 General Safety Approach / 16

Evolution of the collective radiation dose from the nuclear reactors through the 2nd to the 4th generation.

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During operation of a nuclear reactor, the individual external doses received by staff are measured and
recorded in accordance with regulations. The annual collective dose expressed in man.Sv/year can be
assessed by summation. This information is a practical tool in order to compare different types of reactors. This article collects the trends of collective radiation dose for several reactor systems, relying
mainly on the publications of the NEA and the IAEA for GEN II reactors and on a specific bibliography for sodium-cooled fast reactors and GEN IIL
Doses from various sources received by the staff show a decrease, year after year, that seems mainly the result of two factors: the dissemination of good practice (optimization of operating conditions and
organization, sharing of experiences, etc.) and a continuous improvement in the design of these reactors.
In the case of sodium-cooled fast reactors, a compilation and synthesis of various information, is provided to allow a comparison with the second and third generation reactors. Considering the results,
it appears that the doses received in SFR operation are significantly lower.

Country/Int. Organization:

France / C E A

Poster Session 1 / 17

Mechanical Design Evaluation of Fuel Assembly for PGSFR

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The PGSFR core is a heterogeneous, uranium-10% zirconium (U-10Zr) metal alloy fuel design with 112 assemblies: 52 inner core fuel assemblies, 60 outer core fuel assemblies (FA), 6 primary control assemblies (CA), 3 secondary control assemblies, 90 reflector assemblies (RA), and 102 B4C shield assemblies (SA). The core is designed to produce 150 MWe with an average temperature rise of 155 ℃. The inlet temperature is 390 ℃ and the bulk outlet temperature is 545 ℃. The core height is 900 mm and the gas plenum length is 1,250 mm. The fuel assembly is composed of the several structural parts, which are the handling socket, upper/lower reflector, nose piece, hexagonal duct and fuel rods. The face to face dimension and the length are 132.36 mm and 4,550 mm, respectively.
In this paper, there are two kinds of analyses for mechanical design and evaluation of FA. One is the dynamic behavior analysis of FA and the other is the structural analysis of FA components as design level. All of these analyses results will be verified by out-pile test of actual size test FA.

Country/Int. Organization:

Rep. of Korea

Poster Session 2 / 18

Modeling of Phenix End-of-Life control rod withdrawal tests with the Serpent-DYN3D code system

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The nodal diffusion code DYN3D is under extension for Sodium cooled Fast Reactor (SFR) applications. As a part of the extension a new model for axial thermal expansion of fuel rods was developed. The model provides a flexible way of handling the axial fuel rod expansion that is each sub-assembly and node can be treated independently. In the current paper the new model will be described in details. The performance of the model will be assessed with the help of the benchmark on the control rod withdrawal tests performed during the PHÉNIX end-of-life experiments. The DYN3D results will be tested against the experimental data as well as against the numerical results provided by other participants to the benchmark.

Country/Int. Organization:

Germany

1.6 CORE AND DESIGN FEATURES - 2 / 20

Fundamental Approaches to High-power Fast Reactor Core Development

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The article examines approaches to developing a core for a high-power lead-cooled fast reactor in compliance with the formulated safety requirements and technical and economic specifications. A particular focus is made on achieving inherent safety properties, such as: a negative void effect, negative reactor power and coolant temperature coefficients, uniform coolant heating in the reactor core, as well as minimal operating reactivity margin and overshoot of reactivity during the core lifetime, which eliminates the potential risk of prompt neutron power excursion.
Constraints of reactor core characteristics in terms of acceptable power level, heating temperatures and lead coolant flow rate are considered in relation to the development level of available structural materials and lead coolant technology.
A comparative analysis of the impact of various factors (fuel rod geometry, spacing of triangular fuel rod lattice, fuel density) on characteristics of BR 1200 lead-cooled reactor core with a thermal power of 2,800 MW (void effect, reactivity margin during core lifetime, critical core loading) is conducted and trends of these characteristics are plotted depending on the fuel weight fraction.
Solutions aimed at reducing the irregularity of coolant heating and radial power peaking factor in a

high-power reactor are evaluated.
Solutions aimed at reducing the irregularity of coolant heating and the radial power irregularity in a high-power reactor are evaluated.

Country/Int. Organization:

Russian Federation

Poster Session 2 / 21

Burnup Analysis for BN-600 Reactor Core fueled with MOX fuel and Minor Actinides

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The BN-600 reactor is a sodium-cooled fast breeder reactor, built at Russia. Designed to generate electrical power of 600 MW in total. IAEA has considered the reactor for many phases of benchmark problems.The coordinated research project activities were started in 1999 and included studies fora so-called hybrid BN-600-reactor-type core model, partially fuelled with highly enriched uranium and partially, about 20% of "fuel subassemblies (FSAs),"with MOX (Phases 1 to 3), a full-MOX core model with weapons-grade plutonium (Phase 4), a model of the BFS-62-3A experimental critical configuration, a mockup of the hybrid core (Phase 5) and, finally, a full-MOX core model with plutonium and Minor Actinides coming from spent nuclear fuel (Phase 6).

Computer Code Model.

MCNPX code , based on Monte Carlo method , is used to design a three dimensional and typical computer model to the reactor , all core zones radially and axially are represented in the model . Figure 1. illustrate typical horizontal and vertical layout of the model. BN-600 core MOX contains three radial fuel zones , Low enrichment (LEZ) , Medium enrichment(MEZ) and Higher enrichment (HEZ) zones respectively. Radially beyond the HEZ outer zone are two steel shielding zone(SSA1 and SSA2) [1]. The reactor core is controlled by two types of control rods SHR and SCR. Minor actinides 5 % [2, 3] was added to each fuel zone to analyse and study burnup and transmutation of Minor actinides inside the fast reactor core. A core cycle of 140 days is considered for the analysis. The results calculate the reactor multiplication factor , flux and power distributions. Kinetic and safety parameters of the reactor core, also Incineration and burn up of minor actinides at EOC are evaluated.

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Country/Int. Organization:

Egypt

Poster Session 1 / 22

Advanced Energy Conversion for Sodium-Cooled Fast Reactors

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Advanced energy conversion using the supercritical carbon dioxide (sCO2) Brayton cycle has been under development at Argonne National Laboratory (ANL) for over twelve years. It has been shown to enable the SFR capital cost per unit output electrical power ($/kWe) or Levelized Cost of Electricity (LCOE) to be significantly reduced improving SFR economics (a U.S. DOE SFR goal) and eliminating sodium-water reactions, although there still remains a need to understand potential sodium-CO2 interactions that is being addressed through ongoing sodium-CO2 interaction tests. It has been shown that the cycle enables the use of dry air cooling whereby heat is rejected directly to the air atmosphere through the use of finned tube air coolers. A Plant Dynamics Code for system level dynamic analysis of sCO2 cycles has been developed, coupled to the SAS4A/SASSYS-1 SFR transient analysis code, and is being validated through comparison with data from sCO2 integrated cycle test loops.

Country/Int. Organization:

USA, Argonne National Laboratory

Poster Session 2 / 23

Uncertainty Quantification of EBR-II Loss of Heat Sink Simulations with SAS4A/SASSYS-1 and DAKOTA

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Argonne has developed SAS4A/SASSYS-1 models for the benchmark analyses of the Experimental Breeder Reactor II (EBR-II) Balance-of-Plant (BOP) tests that represented protected and unprotected loss of heat sink conditions. The analyses were performed to support the validation of simulation tools and models used for SFR development. Previous benchmark results for the two BOP tests were in good agreement with selected measured data. Some assumptions had to be made for the models because of uncertainties related to the cooling system. In addition, the reactivity feedback coefficients also have uncertainties due to the nuclear data. These uncertainties may contribute to discrepancies observed between the simulation results and the measured data.

The objective of this study is to apply the recently developed coupling between Dakota and SAS4A/SASSYS-1 to investigate the impact of uncertainties on the simulation results. The Dakota software is an uncertainty quantification and optimization toolkit. It was coupled with SAS4A/SASSYS-1 via a Python interface to meet an increased need to perform sensitivity analyses and uncertainty quantification in the advanced reactor domain. Dakota was used to sample user-specified parameters, drive SAS4A/SASSYS-1 transient simulations, and quantify statistical metrics as part of post processing. The studies described in this paper include the uncertainty quantification of the EBR-II simulations and the calibration between the simulation results and the experimental data. By applying Dakota

for uncertainty propagation, it is found that the radial expansion, the control rod drive expansion, and the stagnant sodium mixing models have significant impacts on the benchmark results. Following the uncertainty quantification, parameters in the EBR-II model that were identified to have significant impacts were optimized by Dakota in order to improve the agreement between the simulation results and the measurements.

Country/Int. Organization:

USA

5.5 Large Component Technology I / 24

Experimental qualification of rotatable plug seals for Sodium Fast Reactor on a large scale test stand

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In the framework of the ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) project, the CEA Sealing Laboratory with its partner TECHNETICS (TGF) was involved to propose a new concept of rotating plug seals to replace the commonly used liquid-metal seals. An innovative combination of static, dynamic and inflatable seals in silicone rubber ensuring double tightness-barriers for the cover gas was developed. Following the design phase and materials studies, a dedicated test stand was built to qualify the technical performances of these seals. The large size of the test stand composed of a 2.5 m diameter rotating plates was chosen to provide a small profile height on seal diameter ratio, and a volume of enclosed gas large enough to allow representative qualification of tightness test methods. After a description of the test stand, the paper presents the mains outcomes of the technical qualifications (mechanical behavior, sealing performance, endurance test) led on several seals design.

Country/Int. Organization:

France

Poster Session 1 / 25

A Mechanistic Source Term Calculation for a Metal Fuel Sodium Fast Reactor

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A mechanistic source term (MST) calculation attempts to realistically assess the transport and release of radionuclides from a reactor system to the environment during a specific accident sequence. The U.S. Nuclear Regulatory Commission (NRC) has repeated stated its expectation that advanced reactor vendors will utilize an MST during the U.S. reactor licensing process. As part of a project to examine possible impediments to sodium fast reactor (SFR) licensing in the U.S., an analysis was conducted regarding the current capabilities to perform an MST for a metal fuel SFR. The purpose of the project was to identify and prioritize any gaps in current computational tools, and the associated database,

for the accurate assessment of an MST. The results of the study demonstrate that an SFR MST is possible with current tools and data, but several gaps exist that may lead to possibly unacceptable levels of uncertainty.

Country/Int. Organization:

USA/Argonne National Laboratory

Poster Session 1 / 26

Advanced Reactor PSA Methodologies for System Reliability Analysis and Source Term Assessment

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Beginning in 2015, a project was initiated to update and modernize the probabilistic safety assessment (PSA) of the GE-Hitachi PRISM sodium fast reactor. This project is a collaboration between GE-Hitachi and Argonne National Laboratory (Argonne), and funded in part by the U.S. Department of Energy. Specifically, the role of Argonne is to assess the reliability of passive safety systems, complete a mechanistic source term calculation, and provide component reliability estimates. The assessment of passive system reliability focused on the performance of the Reactor Vessel Auxiliary Cooling System (RVACS) and the inherent reactivity feedbacks mechanisms of the metal fuel core. The mechanistic source term assessment attempted to provide a sequence-specific source term evaluation to quantify offsite consequences. Lastly, the reliability assessment focused on components specific to the sodium fast reactor, including electromagnetic pumps, intermediate heat exchangers, the steam generator, and sodium valves and piping.

Country/Int. Organization:

USA/Argonne National Laboratory

Poster Session 1 / 27

TRANSMUTATION TRAJECTORY ANALYSIS IN THE MODELLING OF LFR FUEL CYCLE

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The closed nuclear fuel cycle belongs to the most promising options
for the efficient usage of the nuclear energy resources with the full recycling of the long-lived transuranic elements. However, it can be implemented only in fast breeder reactors of generation IV. The paper shows our methodology applied to the analysis of lead-cooled fast reactor equilibrium fuel cycle with the application of the continuous energy Monte Carlo Burnup code –MCB. The implementation of novel modules for nuclear transmutation trajectories folding, allows us to trace the life cycle of crucial minor actinides from the reactor beginning of life to the state of adiabatic

equilibrium. The numerical calculations were performed for the reactor core designed within the European Lead-cooled SYstem (ELSY) project and redefined in the follow-up Lead-cooled European Advanced DEmonstration Reactor (LEADER) project.

Country/Int. Organization:

Poland

3.1 Safety Program / 28

The Status of Safety Research in the Field of Sodium-cooled Fast Reactors in Japan

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This paper describes the status of safety research activity in the field of sodium-cooled fast reactors (SFRs) in Japan, mainly on severe accident related issues. Core damage sequences are analyzed by applying probabilistic risk assessment methodology and categorized into typical accident phases, i.e., initiating phase, transitions phase, and material relocation and cooling phase. In order to utilize superior characteristics of sodium as coolant, achievement of in-vessel retention is one of important objective of safety design and evaluation for SFRs. Focus is on the later phases of accidents for which experimental data acquisition and code development are going on. A series of out-of-pile and in-pile experiments in EAGLE-3 and related tests in the MELT facility are being conducted for molten fuel discharge and cooling. Study on debris bed formation and self-leveling effect is also conducted. A fast-reactor safety analysis code, SIMMER, is developed to enhance its capability to be applicable such phenomena in the later phase of accidents.

Country/Int. Organization:

Japan

Poster Session 1 / 29

”Peculiarities of behavior of Coated Particle fuel in the core of Fast Gas Reactor BGR-1000”

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Fast Gas Reactor BGR-1000 with thermal power of 2 GW is cooled with high-pressure helium (16 MPa), heated in the core from 350 to 750oC. In a steam generator of the power conversion system the thermal power is transferred to the SCW-coolant of secondary circuit, which goes to the turbine

with pressure of 30 MPa and temperature of 650oC. Reactor core contains Fuel Assemblies (FA) having perforated shrouds. FA's inside cavity among shroud, control rod guide tubes and central perforated collector is filled with pebble-bed of micro-fuel coated particles (CP). Helium coolant goes into FA through the perforated shroud, passes over CPs removing heat from them and goes then to the FA outlet collector through its perforated wall. The mix-carbide fuel UPuC with mean plutonium content of 16.5% is dispersed in the core in the form of CPs kernels. While loading of heavy atoms is 3640 kg, reactor average burnup amounts 9.7% h.a. Having a breeding ratio of 1.025 reactor can operates in the regime of self-provision of the secondary fuel in the closed fuel cycle. Calculational optimization of CP design has given the following performance of the CP kernel and coatings: CP outer diameter of 2000 um, kernel diameter of 1640 um, nondense pyrocarbon buffer coating of 125 um, dense pyrocarbon inner layer (IPyC) of 10 um and outer protective SiC layer of 50 um. In the paper the basic positions of the model of the thermo-mechanics of BGR-1000 coated particles are presented and calculational results revealing the effect of CP design on their behavior during irradiation are demonstrated. It is shown, that in the result of the viscous deforming the summarized volume of the kernel and buffer, limited by the elastic SiC, keeps practically invariable. In an equilibrium state volume changes of the fuel (due to its swelling) and of the pyrocarbon layers (due to radiation-induced size changes) are compensated by changing of the volume fraction of porosity in the fuel and buffer owing to their viscous deformations.

Country/Int. Organization:

Russian Federation

8.2 Professional Development and Knowledge Management - II / 31

IAEA's Fast Reactors Knowledge Portals and Catalogues

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Committed to the sustainability and addressing the technology progression, the IAEA provides support to the Member States covering all technical aspects of current, evolutionary and innovative fast reactors and subcritical hybrid systems R&D, design, deployment, and operation. The IAEA has been carrying out a dedicated fast reactor knowledge preservation initiative since 2003.

The main objectives of the initiative are to, (a) halt the information related to Fast Reactors (FR), and (b) collect, retrieve, preserve and make accessible existing data and information on FR in the dedicated Fast reactor Knowledge Portal (FRKP). The interest for this initiative has been reconfirmed at several technical meetings held in this area along with a constant support from the IAEA technical working group on fast reactors.

The portal incorporates the existing set of knowledge and information and also provides a systemic platform for further preservation of new developments. The FRKP is capable to control and manage both publicly available as well as controlled information. In the portal a taxonomy based search tool is implemented, which helps using new semantic search capabilities for improved conceptual retrieve of documents.

The FRKP includes fast reactor document repositories (approximately 30.000 documents), project workspaces for the IAEA's coordinated research projects, technical meetings, forums for discussion and databases.

One of these databases is "A catalogue of facilities in the IAEA member states in support of development and deployment of liquid metal cooled fast neutron systems (LMFNS catalogue)". This newly developed living database in its current form presents an overview and detailed information on 150 experimental facilities in 14 IAEA Member States, currently operating, being designed or under construction, in support of the development and deployment of the innovative LMFNS, both critical and subcritical.

The LMFNS Catalogue contains data on 79 facilities in support of development of sodium cooled fast reactors (SFR) and 71 facilities in support of lead/LBE cooled fast reactors (LFR). Besides, the facilities in support of SFR and LFR there are a few dual-application facilities in support of both designs. An overview and detailed facility profiles are reported and classified according to the reactor type, their most relevant research fields. Multiple choice filtering and sorting are available.
By providing the end-users with detailed information on existing and future experimental facilities able to support innovative LMFNS, the open LMFNS database will facilitate cooperation between organizations with an active programme on these nuclear energy systems.

Country/Int. Organization:

International Atomic Energy Agency (IAEA), Vienna, Austria

5.3 Advanced Fast Reactor Cladding Development I / 32

Results of monitoring, using high-resolution neutron diffraction, of radiation-induced damages in claddings of fuel pins after their performance in the reactor BN-600 as a ground for prolongation of their life expectancy

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Fast neutron irradiation gives rise to rather complex processes developing in the fuel element claddings that lower their technical characteristics and restrict time of safe exploitation. Neutron diffraction studies open up a possibility to monitor them even at a stage of incubation period and thereby to promote development of reliable methods for life expectancy prolongation of the reactor components. An important advantage of these methods is minimal manipulations needed for work with high radioactive samples.
We carried out neutron-diffraction studies of the samples prepared from the fuel elements claddings made of austenitic steel EK-164 both in the initial state and after real exploitation in different parts of the reactor BN-600 core at temperatures (360 –630) C up to the dose of 75 dpa. It was found that the samples cut from the fuel elements of different lots had small distinctions in their structure state (inner stresses, texture, dislocation density). It was confirmed that up to the highest fluences in the study, the FCC lattice is retained, with the microstate depending of three parameters: fluence, neutron flux density and temperature. Our neutron diffraction data saying that maximal concentration of defects takes place at high fluence within the core part with the temperature of (400-550) C are in agreement with data on the fuel element claddings swelling. At the same time low neutron flux densities, temperature about 375 C and dose up to 10 dpa result in the annealing of initial defects and decrease of microstresses (dislocation densities). It is interesting that within the core part at a temperature of 628 C and 75 dpa defect concentration is shown to decrease again, down to the level being lower initial.
Now we continue our studies of the claddings materials up to 105 dpa.
The research was carried out at IMP Neutron Material Science Complex within the state assignment of FASO of Russia (theme "Flux"No. 01201463334), supported in part by Ural Branch of Russian Academy of Sciences (Project № 15-17-2-3).

Country/Int. Organization:

Russian Federation

5.8 Structural Materials / 33

Fabrication and Evaluation of Advanced Cladding Tube for PGSFR

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Ferritc/martensitic steels are being considered as cladding materials for PGSFR. So KAERI has developed cladding material (FC92) which has superior thermal creep property to HT9. In order to verify the performance of cladding tube, KAERI has manufactured FC92 cladding tube in connection of the steelmaking industry. Out-of pile tests like mechanical tests (uniaxial tensile, biaxial burst, pressurized creep) and simulated transient test are now being conducted. Thermo-physical properties (density, Young's modulus, Poisson's ratio, thermal expansion, specific heat capacity and thermal conductivity) of FC92 material is being performed to use in fuel design. Quality assurance program has been introduced in all out-of pile tests to acquire reliability of the test data. Evaluating in-pile property of cladding tube is essential for not only usage in fuel design but also demonstrating integral performance of developed cladding under irradiation condition. To verify its performance, KAERI has launched irradiation test program of fuel cladding tube using an experimental fast reactor. In 2014, KAERI has made the contract between RIAR in Russia for the irradiation tests in BOR-60, where irradiation creep and swelling will be mainly performed. Obtained dataset will be used for developing creep model of FC92 cladding tube, together with out-of pile creep data.

Country/Int. Organization:

Republic of Korea/Korea Atomic Energy Research Institute

Poster Session 1 / 34

ECOLOGICAL ASPECTS OF THE USE OF FAST REACTORS IN A CLOSED NUCLEAR FUEL CYCLE UNDER THE "PRORYV"PROJECT

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The development of nuclear power engineering with closing of a fuel cycle and the use of fast reactors must ensure a higher level of ecological safety of the population and the environment. The highest ecological effect is achieved by recycling of spent fuel and isolation of long-lived radionuclides (90Sr, 137Cs and 99Tc) and transmutation of 99% of americium.
In normal operation of CNFC facilities exposure doses to the population are formed via different critical pathways: for a rector plant –due to inhalation intake of 3H, for fabrication and refabrication

module –due to inhalation of Pu aerosols, for SNF recycling module –due to external radiation from the soil and ingestion in case of surface contamination of plants.
The use of nitride fuel generates large amounts of 14C (270 g per 1 ton fuel). Insolubilizing most of 14C ensures compliance with the project standards for the population exposure to the gas phase of release.

Country/Int. Organization:

RUSSIA/Institution «ITC «PRORYV» Project»

Poster Session 1 / 35

Thermodynamics and separation factor of lanthanides and actinides in system "liquid metal-molten salt"

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Pyrochemical separation processes in molten salt media have been proposed as a promising option in the future nuclear fuel cycle. The major steps of these processes include the electrorefining or reductive extraction of the recovering actinides in molten chloride/liquid metal systems and for recovery of minor actinides from spent fuel or high level radioactive liquid wastes. The goal of these investigations is to find the extraction systems with high values of separation factor An/Ln using bimetallic liquid metals.
Thermodynamic properties of lanthanum, praseodymium, neodymium and uranium were studied in systems (Me)Ga-In/3LiCl-2KCl and (Me)Ga-Al/3LiCl-2KCl vs. of the composition of liquid alloys and the temperature. The influence of the nature of elements on activity coefficients and separation factor was determined. The calculated values of separation factor of U/La, U/Pr and U/Nd lies in the region 4.0·10^4 –5.0·10^5.

Country/Int. Organization:

Russian Federation

Poster Session 2 / 36

ANALYSIS OF VARIOUS APPROXIMATIONS IN NEUTRONIC CALCULATIONS OF TRANSIENT IN FAST REACTORS

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A module of transient neutron transport problem solution is an essential part of complex codes designed for the analysis of nuclear safety in different modes of operation of a nuclear reactor. The neutron transport problem can be solved with a variety of approximate schemes, such as point kinetics, adiabatic, quasi-static and improved quasi-static approximations or direct numerical solution of the original equations. Approximate schemes of solution because of their inherent assumptions have limited application area. Therefore, it is necessary to study a possibility of application of the approximate scheme of solution for each specific task.
The paper considers the kinetic calculations of several tests associated with movement of control rods in fast reactors and characterized by the maximum local perturbation of a reactor environment. Each test has been calculated by a direct numerical solution of the transient neutron transport equation based on a diffusion theory and by different approximate schemes of solution of the original equation.
Error of numerical solution obtained using each approximate scheme has been obtained by comparison of calculated reactor parameters (e.g. power or reactivity) with direct numerical solution. The calculation results demonstrate that some approximations that are successfully used for the calculation of similar problems in the thermal reactor are not able to provide acceptable solution accuracy for the fast reactor. Results of the analyses of the different solution schemes are presented. It is found that solutions obtained using combined schemes based on an improved quasi-static approximation are preferred because of high solution accuracy and admissible time expenditure.

Country/Int. Organization:

Russian Federation

1.4 CORE AND DESIGN FEATURES - 1 / 37

ALLEGRO Core Neutron Physics Studies

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Status of neutron-physical analysis of ALLEGRO - demonstrator of the gas cooled reactor is characterised at this article.
Benchmarking of existing neutronic codes utilised for PWR analyses mainly, is first task, solved at running projects. As there are available no neutronic experiments with He coolant at fast spectrum, code to code comparison was selected as first stage of validation process.
First ALLEGRO oriented neutronic benchmark was split into two phases. Definition, solution and partial conclusions of first phase concentrated on pin calculation - Methodological benchmark with simplified geometry for the group constant generating tools - are described at the article. Definition of second phase oriented on assembly calculations and its first evaluations are treated as well.
Evolution of ALLEGRO core evolution is driven by two factors - problems with DHR proportional to power density and by better availability of UOX fuel for first cycles (in comparison with MOX). First round of calculations oriented on fuel and power density optimisation including resulting direction of core modifications is characterised in the paper.

Country/Int. Organization:

Slovak Republic
VUJE Inc., SK91864 Trnava

1.8 INNOVATIVE REACTOR DESIGNS / 38

A Concept of VVER-SCP reactor with fast neutron spectrum and self-provision by secondary fuel

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In recent few years RF Concern "Rosenergoatom has promoted R&D in support of designing of innovation VVER with supercritical parameters of water coolant (VVER-SCP). Main goals of VVER-SCP have been the followings: possibility of operation of reactor in a regime of self-provision by fuel in the closed cycle; energy efficiency of NPP should be not less than 40-42%. One of VVER-SCP concepts has been a variant of two-circuit NPP with fast reactor, cooled by light-water steam of supercritical pressure –SCPS-600 with electrical power of 600 MWe. Reactor SCPS-600 has a vessel with diameter like VVER-1000, but thicker wall of 350 mm. Combination of quite tight fuel lattice and SCP steam coolant (with the inlet/outlet reactor temperatures of 388oC and 500oC respectively and pressure of 24.5 MPa) allows realizing quite fast neutron spectrum in the core. The core is formed with three groups of Fuel Assemblies with different content of PuO2: 16, 18.5 and 24% weight respectively. Butt and side blankets comprise dioxide of depleted uranium with content of 235U of 0.2%. Central zone of the core consists of the ThO2 fertile fuel that provides appropriate void reactivity coefficient. With a load of 32.3 m.t. of heavy atoms the averaged burnup in reactor amounts 54.2 MW-days/kgha. The ratio of unloaded-to-loaded fissile atoms in reactor amounts 1.01 –1.03 that make it possible to get the regime of self-provision of reactor by its own secondary fuel in the equilibrium closed fuel cycle. In the secondary circuit of the installation it is planned to use a quite compact supercritical steam turbine with intermediate steam separation without overheating. The steam going from Steam Generator to the turbine has pressure of 23.5 MPa and temperature of 480oC. Net efficiency of the turbine installation amounts 42.5%. The paper considers effect of the design and technology solutions upon the main neutron-physics and thermal characteristics of the reactor. Special care is taken to ways of lowering a positive void reactivity effect by decreasing of parasitic neutron absorption in the core due to thinner cladding, use of new structure materials, like ferritic-martensitic steels and SiC, use of different variants of the spatial distribution of ThO2 and solid moderators (like ZrH2 and BeO) in the core.

Country/Int. Organization:

Russian Federation

Poster Session 2 / 39

Design and Fabrication of Closed Loop Systems (CLS) for the Fast Flux Test Facility (FFTF)

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The FFTF was designed to accommodate up to four CLS. Each CLS was an irradiation testing system capable of operating at 2.3 MWt with its own independently controlled coolant system. An irradiation test in a CLS was inserted into the core within a Closed Loop In-Reactor Assembly (CLIRA). An entire CLS consisted of a CLIRA, a primary and a secondary cooling loop, and a Dump Heat Exchanger (DHX) as the ultimate heat sink. Four CLS were designed and two were fabricated. All connections needed to install a CLS were likewise fabricated including the branch arm piping within the reactor vessel. The CLS were not installed prior to startup of the FFTF due to resources being shifted to achieving full power as soon as possible. This paper will describe the design and fabrication of the FFTF CLS as well as the lessons learned during design and fabrication.

Country/Int. Organization:

USA

5.4 Advanced Fast Reactor Cladding Development II / 40

Creep resistance and fracture toughness of recently-developed optimized Grade 92 and its weldments for advanced fast reactors

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Optimized Grade 92 has been developed at Oak Ridge National Laboratory in support of the USA Sodium-cooled Fast Reactor program. Composition modification and processing optimization successfully achieved the development of optimized Grade 92 with desired microstructures for superior properties. A variety of properties have been assessed for optimized Grade 92, which include tensile, creep, fatigue, creep-fatigue, impact and fracture toughness, weldability, thermal aging resistance, and sodium compatibility. This paper focuses on presenting the results of creep and fracture toughness tests of optimized Grade 92 and its weldments. Compared to the literature data of Grade 92 and similar 9Cr ferritic-martensitic steels, optimized Grade 92 exhibited significantly enhanced creep resistance, together with superior or comparable fracture toughness. Creep rupture ductility of the ruptured samples is discussed by comparing to the reference steels. Samples extracted from tungsten-inert-gas fabricated weldments showed slight reductions in creep life and creep strength compared to the base metal of optimized Grade 92. The reductions, however, are noticeably smaller than that of the reference steels. Satisfactory fracture toughness was observed for the weldments of optimized Grade 92. Hardness measurements and microstructural characterization following the tests shed light on the superior properties of optimized Grade 92 and its weldments. The enhanced properties are expected to favor the application of optimized Grade 92 for advanced fast reactors.

Country/Int. Organization:

Oak Ridge National Laboratory, USA; Argonne National Laboratory, USA

6.10 Other issues of code development and application / 41

U.S. Sodium Fast Reactor Codes and Methods: Current Capabilities and Path Forward

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The United States has extensive experience with the design, construction, and operation of a variety of sodium cooled fast reactors (SFRs) over the last six decades. Despite the closure of various facilities, the U.S. continues to dedicate research and development (R&D) efforts to the design of novel experimental, prototype, and commercial facilities. Accordingly, in support of the rich operating history and ongoing design efforts, the U.S. has been developing and maintaining a series of tools with capabilities that envelope all facets of SFR design and safety analyses. This paper will provide an overview the current U.S. SFR analysis toolset, including codes such as SAS4A/SASSYS-1, MC2-3, SE2-ANL, PERSENT, NUBOW-3D, and LIFE-METAL, as well as the higher-fidelity tools (e.g. PROTEUS) being integrated into the toolset. Current capabilities of the codes will be described, and key ongoing development efforts will be highlighted.

Country/Int. Organization:

USA/Argonne National Laboratory

3.3 Probabilistic Safety Assessment / 42

Dynamic probabilistic risk assessment at a design stage for a sodium fast reactor.

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The design of a new reactor is an iterative process. At a design stage, each reactor system may exist in several variants : combined together, these variants give different reactor designs.
Probabilistic risk assessment is a tool that gives a measure (risk, and risk space) that may help to compare different reactor designs. Conservative or macroscopic way to perform probabilistic risk assessment at a design stage may be insufficient to provide enough information to distinguish between different design variants.

The classical way to construct a PRA model with boolean Event Trees/Fault trees (ET/FT) is well applicable for a PWR type reactors at a design stage. Nevertheless ET/FT formalism finds its limits for a Sodium Fast Reactor if one considers long mission times and possible system recoveries. Due to thermal inertia and simplicity of thermal-hydraulic behavior, the decay heat removal function of a Sodium Fast Reactors (SFR) is a good candidate for dynamic probabilistic risk assessment.

In this paper, we present a new approach to construct the dynamic probabilistic model of the Decay Heat Removal (DHR) system of a Sodium Fast Reactor using Stochastic Hybrid Automata with PyCATSHOO modeling tool.The proposed approach allows to construct a dynamic probabilistic model of a SFR DHR system that incorporates the time evolution of physical parameters and dynamic changes of DHR system state (failure or recovery of DHR system components) and presents enough flexibility to be applied at the design stage.

Below, we list the important properties of a dynamic models in more details:
- We define core damage as function of primary sodium temperature during an accidental sequence.

The evolution of primary sodium temperature follows the simplified equations that can be validated by a qualified thermohydraulic code. Evolution of primary sodium temperature is automatically calculated for every accidental sequence to predict the end state of the sequence (OK/Core Damage).

- We explicitly treat the dependency of DHR system trains on different support systems: electrical, I&C, ventilation etc.
- We explicitly model Common Cause Failures between different DHR components.
- We explicitly model component recoveries.

Our modeling approach allows to:

- have a detailed model e.g. it can be as detailed system components,
- easily change the system architecture to test and compare in a realistic way different design variants,
- perform an uncertainty analysis of the risk as a function of uncertainty in reliability and physical parameters

Country/Int. Organization:

France/EDF R&D/CEA

Poster Session 2 / 43

Basic Visualization Experiments on Eutectic Reaction of Boron Carbide and Stainless Steel under Sodium-Cooled Fast Reactor Conditions

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This paper describes basic visualization experiments on eutectic reaction and relocation of boron carbide (B4C) and stainless steel (SS) under a high temperature condition exceeding 1500℃ as well as the importance of such behaviors in molten core during a core disruptive accident in a Generation-IV sodium-cooled fast reactor (750MWe class) designed in Japan. At first, a reactivity history was calculated using an exact perturbation calculation tool taking into account expected behaviors. This calculation indicated the importance of a relocation behavior of the B4C-SS eutectic because its behavior has a large uncertainty in the reactivity history. To clarify this behavior, basic experiments were carried out by visualizing the reaction of a B4C pellet contacted with molten SS in a high temperature-heating furnace. The experiments have shown the eutectic reaction visualization as well as freezing and relocation of the B4C-SS eutectic in upper part of the solidified test piece due to the density separation.

Country/Int. Organization:

Japan Atomic Energy Agency

Poster Session 1 / 45

Sodium compatibility of Recently-Developed Optimized Grade 92 and its Weldments for Advanced Fast Reactors

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This paper presents the results of sodium compatibility of the base metal and weldment of optimized Grade 92 steel recently developed in the United States for applications in advanced sodium-cooled fast reactors. Optimized Grade 92 (Fe-9Cr-0.5Mo-2W-V,Nb) is a variant of commercially available Grade 92 ferritic-martensitic steels with tighter control of chemistry and thermo-mechanical treatment to achieve improved high temperature performance. Several heats of optimized Grade 92 were investigated in liquid sodium environments to evaluate their long-term performance. The data are used to assess optimized Grade 92 for the American Society of Mechanical Engineers (ASME) code qualification and Nuclear Regulatory Commission (NRC) licensing.
Sodium exposure experiments were conducted at 550, 600 and 650C in forced convection sodium loops at the Argonne National Laboratory (ANL). The oxygen content of sodium was controlled by the cold-trapping method to achieve ~1 wppm oxygen level. Specimens were removed from the sodium loop after a pre-determined exposure time for post sodium-exposure examination, including weight and thickness measurements, cross-sectional examinations, microhardness measurements, microstructural characterization, and tensile tests at the sodium exposure temperature. Data of sodium-exposed specimens were compared with thermal aging data of the same heat of optimized Grade 92 to separate the effects of sodium exposures and thermal aging. Optimized Grade 92 showed an insignificant weight loss after exposures to sodium at 550-650C. Sodium exposures at 650℃ have a much stronger effect on the tensile strength than thermal exposures at 650℃, but not at 600 and 550C with available data. Microstructural characterization of optimized Grade 92 after sodium exposure at 650C showed drastic microstructural changes manifested by reduction in dislocation density, subgrain coarsening, M23C6 particle coarsening, MX dissolution, precipitation and coarsening of Laves phase. Weldment of optimized Grade 62 was also characterized after sodium exposures at 600C.

Country/Int. Organization:

USA

Panel 1: Development and Standardization of Safety Design Criteria (SDC) and Guidelines (SDG) for Sodium Cooled Fast Reactors / 46

The Safety Design Criteria Development and Summary of Its Update for the Generation-IV SFR Systems (USA/Japan/GIF)

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The Generation-IV International Forum (GIF) Task Force completed development of Safety Design Criteria (SDC) for the Generation-IV SFR systems in May, 2013. SDC reflects high level GIF safety

and reliability goals (excellence in operational safety and reliability, and reduced likelihood and degree of core damage) and follows GIF basic safety approach (application of defense-in-depth and emphasis on inherent and passive safety features so that safety is built-in to the design, not added-on). The SDC report aimed to establish reference criteria for safety design of structures, systems and components and achieve harmonization of safety approaches among GIF member states. Following its public release, SDC report was distributed to international organizations and national regulatory bodies for review and feedback. Based on comments received during the following two year period, SDC report underwent a revision reflecting important feedback received from IAEA, NRC (USA), IRSN (France), and NNSA (China). This paper will provide an overview of SDC development effort, and summarize the comments/suggestions received from its international review along with their resolutions by the GIF Task Force.

Country/Int. Organization:

U.S.A./Argonne National Laboratory

Poster Session 2 / 47

Research on modeling and simulation of the primary coolant system for China Experimental Fast Reactor

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Based on the structure and physical characteristics of primary coolant system (PCS) of China Experimental Fast Reactor (CEFR), a series of reasonable mathematical and physical models were set up. A set of stable and highly effective numerical methods were used to solve the models. Then the real-time thermal-hydraulic analysis codes for PCS of CEFR have been developed with modular method by using FORTRAN programming language. The codes have been merged into SimExecTM real-time simulation platform and could be linked to the modules of other systems.
The models include the basic thermal-hydraulic model of coolant, the heat transfer model of fuel pellet, the heat thermal model of intermediate heat exchangers (IHX), the model of primary pump, the flow friction and heat transfer correlations, the thermo-physical properties, etc. "Control body" method was put forward in the set of models and Gear method was applied to solve the thermal-hydraulic model of coolant and heat transfer model of fuel pellet. Quasi-newton iteration method was used to solve the flow rate distribution equations. The model and numerical method in this research contribute to the accurate and effective of calculation to meet the requirement of real-time simulation.
The design parameters of CEFR were used to validate six different steady-state conditions from 26.5%FP to 100%FP by this code, and the steady-state calculation of the reactor main vessel cooled system was also finished and the results were compared with the datum obtained by the references. Thus the validity and applicability of this code was proved. The normal operation conditions were calculated and validated by the manual of PCS of CEFR. The reactivity insertion accident, the loss of coolant accident and the loss of heat sink accident were also simulated. The results showed that the trend of simulation curves for the steady and transient conditions are reasonable, which are in accordance with the actual physical process. The real-time characteristics of the code were analyzed and could meet the simulation requirements.

Country/Int. Organization:

P.R.China

Poster Session 1 / 48

Thermal Annealing Effect on Recovery of Corrosion Properties of EP-450 Steel Irradiated IN BN-600 Reactor to High Damage Doses

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High radiation-resistant ferritic-martensitic steels are prospect materials for fast reactor cores. Chromium ferritic-martensitic steel 12Cr-13Mo-2Nb-B-V (EP-450) was successfully used as a shroud tube material for BN-350 reactor fuel assemblies and is still used for BN-600 and BN-800 reactors. It is expected to use ferritic-martensitic steel EP-450 for BN-1200 reactor shroud tubes and ferritic-martensitic oxide dispersion strengthened steel for claddings. Wet storage of BN-1200 reactor spent fuel assemblies in a cooling pool is supposed.
However during the storage of BN-600 spent fuel assemblies in the cooling pool high rates of corrosion and corrosion product (CP) release into the cooling pool water for spent fuel assembly structural elements made of 12%-chromium ferritic-martensitic steels were observed. Along with higher corrosion and corrosion product release rates radionuclide concentration increases, thereby increasing radiation hazard level. Localized (pitting) corrosion may occur and lead to cladding depressurization with high active fission product (cesium, strontium, iodine) release into the environment. Cladding depressurization risk diminishes environmental safety of spent fuel assembly storage in the cooling pool.
The research results of water corrosion resistance of 12Cr-13Mo-2Nb-B-V steel after irradiation in BN-600 reactor in the range of temperatures between 360 and 520 oC to different damage doses and isothermal annealing in the range of temperatures between
650 and 750 oC during 1-10 hours are given. It is shown that annealing at a temperature above 700 oC and during more than 3 hours reduces corrosion and corrosion product release rate of 12%-chromium ferritic-martensitic steels in the cooling pool water by more than an order of magnitude. Also it inhibits localized pitting corrosion occurrence. A new technique of the storage of fast reactor spent fuel assemblies with structural elements made of 12%-chromium ferritic-martensitic steels in the cooling pool has been developed. A patent for the technique No. 2555856 (RU) dated 10.06.2015, IPC G21C19/06, was granted.

Country/Int. Organization:

Russian Federation

Poster Session 1 / 49

Investigation of Radiation-Induced Swelling of EK-164 Steel, an Advanced Material for BN-600 and BN-800 Claddings

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One of the main issues of fast reactors is to improve their economic efficiency. Nowadays austenitic stainless steel ChS-68 is used as a state material for BN-600 reactor claddings ensuring damage doses

up to 87 dpa. Post irradiation examination results show the residual life of fuel elements with possible maximum damage dose up to 92-95 dpa corresponding to fuel burn-up to 12-13 % FIMA. At JSC "VNIINM" a prospective austenitic stainless steel EK-164 (16Cr-19Ni-2Mo-Ti-Si-V-P-B), more resistant to radiation-induced swelling than ChS-68 steel, has been developed to extend service life of fast reactor fuel assemblies.
A trial operation of ChS-68 and EK-164 combined fuel assemblies has been carried out to ensure the operating capacity and assess the capability to improve operational characteristics of fuel elements with EK-164 claddings. At the initial stage maximum damage dose during operation of two test fuel assemblies is 74 and 77 dpa, respectively. Post irradiation examination confirms the advantage of EK-164 steel in terms of radiation-induced swelling resistance.
Manufacturing technology of claddings used as a material for fuel elements operated to maximum damage dose in the range between 84 and 96 dpa has been improved considering structure and EK-164 cladding properties investigation results. Post irradiation examination shows that fuel elements retain their operating capacity and have sufficient residual life. Therefore it is possible to predict the operation resource at damage doses above 110 dpa.
The paper aims to investigate radiation-induced swelling of EK-164 claddings at different temperatures and neutron irradiation doses, and to distinguish radiation-induced porosity characteristics at different irradiation temperatures.

Country/Int. Organization:

Russian Federation

Poster Session 2 / 50

Low-void-effect sodium-cooled core: Uncertainty of local sodium void reactivity as a result of nuclear data uncertainties

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Safety robustness by all means is still an open issue for Generation IV Sodium-cooled Fast Reactors (SFR) that needs to be demonstrated in particular with respect to severe accidents. For reliable safety analyzes it is important among other things not only determining 3D maps of reactivity coefficients which can then be used in corresponding transient analyzes within a point kinetics code; in view of a first step towards design optimizations, it is equally paramount assessing uncertainties resulting from nuclear data uncertainties. These uncertainties can be propagated to the overall transient behavior together with uncertainties from other sources.
It is in this framework, since sodium boiling a priori cannot be excluded by means of safety measures alone, that for one of the promising French SFR low-void-effect cores, the uncertainty of regional reactivity effects due to coolant density reductions including the void effect in those regions has been studied in different situations. In the calculations, ERANOS (Edition 2.2-N) has been used in conjunction with JEFF-3.1 cross-sections and COMMARA-2.0 variance/covariance data in 33 neutron energy groups which is an ENDF/B-VII.0 based library. The sensitivity coefficients required for uncertainty assessments have been obtained by means of Equivalent Generalized Perturbation Theory (EGPT) on the basis of nodal diffusion-theory applied to a 3D model of the full heterogeneous core.
It turns out that the uncertainty of the void coefficient due to nuclear data uncertainties may result particularly large especially in relative terms, sometimes making the sign of the analytical coefficient even questionable in cases where the uncertainty exceeds 100%. This effect occurs primarily in supposed scenarios in which only lower parts of the upper axial plenum near the upper core region are voided; more precisely when the loss of coolant involves just portions of the core/plenum interface. Whereas the uncertainty of the overall negative reactivity effect resulting from hypothetical voiding of the whole core is of the order of 20%. Another interesting result allowing extrapolations to non-explicitly computed voided configurations is that such uncertainties are largely adding with

respect to space and Na mass removed.
On the basis of unadjusted COMMARA-2.0 data, the main contributors to the uncertainty of the void coefficient due to nuclear data uncertainties are in general 238U, especially the inelastic scattering cross-section, and 23Na, particularly the elastic scattering cross-section.

Country/Int. Organization:

Switzerland /Paul Scherrer Institute

1.1 SFR DESIGN & DEVELOPMENT - 1 / 51

Feasibility of Burning Wave Fast Reactor Concept with Rotational Fuel Shuffling

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Burning wave fast reactor is very attractive concept. It is possible to achieve very high burnup using natural uranium or depleted uranium as fuel. It does not need fuel reprocessing facility. This kind of reactors can be categorized into two groups. One is the reactor whose burning wave is moving for radial direction, for example, Traveling Wave Reactor. The other is that whose burning wave is moving for axial direction, for example CANDLE burning reactor. The advantage of the reactor concept with radial direction wave movement is that fuel shuffling is easy, but high burnup fuels will exist in high neutron importance region at the center of core. It can be a disadvantage from the view point of neutron economy. The advantage and disadvantage of reactor concept with axial direction movement wave is vice versa. One of the ideas to solve the problems is the concept of shuffling of fuel pins or fuel elements rotationally so that high burnup fuel can be located in low neutron importance region. In the concept, fuel shuffling, load and reload are easy. The purpose of study is to show the possibility to apply for the concept of rotational fuel shuffling in burning wave fast reactor concept. Preliminary analysis was performed using continuous energy Monte Carlo code MVP2.0, JENDL-4.0 nuclear data library, and newly developed additional programs to simulate movement of fuel elements by the shuffling. The results of preliminary analysis showed the stable neutron flux profile can be existed in the core if the shuffling procedure is proper. The detail of shuffling procedure and the burnup characteristics will be presented in the conference.

Country/Int. Organization:

Japan

4.1 Fuel Cycle Overview / 52

Comparison of fast reactors performance in the closed U-Pu and Th-U cycle

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Breeding as well as burning capabilities of a reactor operated in the U-Pu or Th-U closed fuel cycle can be estimated from its equilibrium cycle parameters. In this study the equilibrium parameters were simulated for 8 selected fast reactors and both U-Pu and Th-U closed fuel cycles. For simplicity, the fission products were neglected and the reactors were represented only by infinite lattice.
It was found that the mass flow is stabilized in equilibrium closed cycle. The fuel composition does not differ between two consecutive cycles and determines the excess reactivity. This reactivity can serve as a measure for breeding or burning capabilities of each reactor. For a breeder reactor it should be high enough to accommodate the expected fission products and the presumed neutron leakage. The study provided insight for the differences between the 8 fast reactors and also between the U-Pu and Th-U closed fuel cycles.

Country/Int. Organization:

Switzerland / Paul Scherrer Institut

Poster Session 2 / 53

Development of Safety, Irradiation, and Reliability Databases based on Past U.S. SFR Testing and Operational Experiences

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Inherent and passive safety is a key aspect to achieve licensing assurance and reduce plant costs, but it requires demonstration and validation of key features. To address this need, DOE-NE's Advanced Reactor Technologies (ART) program supported development of EBR-II, FFTF, and TREAT safety testing databases, metal fuel irradiation database, EBR-II physics analysis database and materials information system, and SFR component reliability database. These activities complement broader knowledge preservation and management efforts to facilitate the science-based R&D goals of the DOE-NE by providing data needed for validation of the state-of-the-art codes and advanced methods [such as those pursued under the Nuclear Energy Advanced Modeling & Simulation (NEAMS) program] for design and analysis of advanced fast reactors. This paper summarizes the progress made during the fiscal year 2016 to achieve these goals.

Country/Int. Organization:

U.S.A./Argonne National Laboratory

6.10 Other issues of code development and application / 54

USDOE NEAMS Program and SHARP Multi-Physics ToolKit for High-Fidelity SFR Core Design and Analysis

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Under the Reactors Product Line of U.S. DOE's Nuclear Energy Advanced Modeling and Simulation (NEAMS) program, 3-D, high-fidelity multi-physics simulation capabilities are being developed to address the needs of designers and analysts in studying advanced, non-water reactor systems in general, and SFRs in particular. Simulation-based High-fidelity Advanced Reactor Prototyping (SHARP) toolkit is a suite developed under the Reactors Product Line of NEAMS, and it consists of pin-by-pin neutronics, thermal hydraulic, and structural mechanics modules, as well as the capabilities to integrate these modules for multi-physics analyses. Physics modules currently include the PROTEUS neutronics code, the Nek5000 computational fluid dynamics (CFD) code for thermal-hydraulics, and the DIABLO implicit finite element analysis code for structural mechanics. Each module can be utilized as a standalone code component or as part of an integrated analysis. The development philosophy for the modules is to incorporate as much fundamental physics as possible in order to extend functionality to general reactor types, rather than developing tools for a limited set of specific reactor analysis applications. This paper summarizes the initial efforts focusing on SFR design and analysis in demonstration of the inherent and passive safety characteristics resulting from multi-physics thermal-structural-neutronics phenomena.

Country/Int. Organization:

U.S.A./Argonne National Laboratory

Poster Session 1 / 55

An Assessment of Fission Product Scrubbing in Sodium Pools Following a Core Damage Event in a Sodium Cooled Fast Reactor

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The U.S. Nuclear Regulatory Commission has stated that mechanistic source term (MST) calculations are expected to be required as part of the advanced reactor licensing process. A recent study by Argonne National Laboratory has concluded that fission product scrubbing in sodium pools is an important aspect of an MST calculation for a sodium cooled fast reactor (SFR). To model the phenomena associated with sodium pool scrubbing, a computational tool has been developed. This tool was developed by applying classical theories of aerosol scrubbing, developed for the case of isolated bubbles rising through water, to the decontamination of gases produced as a result of a postulated core damage event in an SFR. The model currently considers aerosol capture by Brownian diffusion, inertial deposition, and gravitational sedimentation. The effects of sodium vapor condensation on aerosol scrubbing are also treated. This paper provides details of a parametric study performed to determine key modeling uncertainties and sensitivities.

Country/Int. Organization:

USA/Argonne National Laboratory

3.2 Core Disruptive Accident / 56

Advances in the Development of the SAS4A Code Metallic Fuel

Models for the Analysis of PGSFR Postulated Severe Accidents

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The SAS4A safety analysis code, originally developed for the analysis of postulated Severe Accidents in Oxide Fuel Sodium Fast Reactors (SFR), has been significantly extended to allow the mechanistic analysis of severe accidents in Metallic Fuel SFRs. The new SAS4A models track the evolution and relocation of multiple fuel and cladding components during the pre-transient irradiation and during the postulated accident, allowing a significantly more accurate description of the local fuel and cladding composition. The local fuel composition determines the fuel thermo-physical properties, such as freezing and melting temperatures, which in turn affect the fuel relocation behavior and ultimately the core reactivity and power history during the postulated accident. The models describing the fission gas behavior, fuel-cladding interaction, clad wastage formation and cladding failure models have been also significantly enhanced. The paper provides on overview of the SAS4A key metal fuel models emphasizing their new capabilities, and presents results of SAS4A whole core analyses for selected PGSFR postulated severe accidents.

Country/Int. Organization:

U.S.A. / Argonne National Laboratory

6.10 Other issues of code development and application / 57

Validation of Advanced Metallic Fuel Models of SAS4A using TREAT M-Series Overpower Test Simulations

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The SAS4A safety analysis code has been extended to include mechanistic and physics-based models of U-Pu-Zr and U-Zr metallic fuel pins. The simulation of various phenomena such as metal fuel component migration, fission gas behavior, clad wastage formation, gas swelling induced axial fuel expansion, in-pin and ex-pin molten fuel relocation, and clad failure models has been significantly enhanced. The integrated code is validated through analyses of eight metal fuel TREAT M-Series overpower experiments. In this study, the SAS4A calculated fuel reactivity and clad failure data are compared with the corresponding experimental data. The results show that the code satisfactorily predicts solid fuel axial expansion, molten fuel in-pin relocation, cladding loss due to rapid eutectic penetration, cladding creep fracture and molten fuel ejection to the coolant channel. The study shows that the uncertainties in transient response tend to be higher for the lower burnup fuel.

Country/Int. Organization:

U.S.A. / Argonne National Laboratory

6.6 Coupled Calculations / 59

Simulating circulating-fuel fast reactors with the coupled TRACE-PARCS code

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Fast reactors that circulate liquid fuel exhibit a strong coupling between neutronics and thermal-hydraulics that necessitates the use of coupled multi-physics codes to study dynamic behaviour. Presently, most such tools employ computational fluid dynamics (CFD) to resolve thermal-hydraulics. This paper concerns an alternative approach in which the system code TRACE is used to compute two-dimensional flow patterns and temperature distributions of liquid-fuel fast reactors using coarse-meshes and a simplified set of equations. As such, computational requirements are greatly reduced compared to CFD-based solvers. In the coupled tool, the thermal-hydraulic variables are sent to the spatial neutronics solver PARCS that calculates power using cross-sections from the Serpent Monte Carlo code. We report the application of TRACE-PARCS to the primary and secondary circuits of the Molten Salt Fast Reactor, and compare the results with alternative multi-physics tools. Reasonable agreement is found, which paves the way for whole-plant simulations including tertiary turbine circuits.

Country/Int. Organization:

Switzerland

Poster Session 1 / 60

CHALLENGES IN THE FABRICATION AND RECYCLING OF MIXED CARBIDE FUEL

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Abstract

Mixed (U0.3, Pu0.7)C fuel is the driver fuel for Fast Breeder Test Reactor (FBTR) at Kalpakkam, India. This fuel is being fabricated at Radio metallurgy Division, Bhabha Atomic Research Centre (BARC). The reactor was made critical with Mark-I fuel having composition (U0.3, Pu0.7)C in year 1985. The fuel has seen a maximum burn up of 165Gwd/t. The carbide fuel is phyrophoric in nature and very much susceptible to hydrolysis. Hence the handling of fuel is done inside alpha leak tight glove-boxes having N2 as cover gas. The fuel is fabricated by classical powder pellet route. In the recent past a new fuel fabrication facility has been commissioned and improvement over the existing equipments and process steps have been carried out to make the fuel fabrication process more efficient resulting in higher productivity and lesser contact between personnel and radioactive powder. The use of liquid binder and lubricant has eliminated dewaxing step from the process flow sheet for UC pellet

fabrication. Dry recycling of the fuel is carried out on regular basis by oxidizing the mixed carbide powder. Chemically accepted pellets having physical defects are directly recycled by crushing and milling the pellets to powder form and subsequently following other regular process steps to produce sintered pellets.

Country/Int. Organization:

India

5.1 Advanced Fast Reactor Fuel Development I / 62

Development of innovative fast reactor nitride fuel in Russian Federation: state-of-art.

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The nitride fuel is selected as an advanced fuel for fast reactors in Russia. Within the framework of "PRORYV" project a comprehensive program of calculation-experimental study of mixed uranium-plutonium nitride performance for BN-1200 and BREST-OD-300 reactors has been developed. The program provides for works to improve the fabrication technique, composition and structure of nitride fuel, to measure out-of-pile properties, to carry out reactor tests in the MIR, BOR-60 research reactors and in the BN-600 commercial reactor, as well as post-irradiation examination (PIE) of all experimental fuel assemblies (FA). Reactor tests are accompanied by pretest calculations by DRAKON and KORAT fuel codes.
For the nitride fuel fabrication the carbothermal synthesis technology of nitride oxide powders, which are the product of the current radiochemical industry, is used. The laboratory technique of carbothermal synthesis of starting powders developed at JSC "VNIINM" is implemented on a larger scale at JSC "SCC" in Seversk, where the possibility of full-scale production of experimental FAs of BN-600 reactor is created. Nitride fuel pellets have been fabricated for more than 500 fuel pins for all BN-600 experimental FAs. Today 9 FAs are under irradiation in the BN-600 reactor. PIEs of one FA have been completed.
7 dismountable FAs with 7 nitride pins in each are under irradiation in the BOR-60 reactor. Fuel and fuel pins have been fabricated at JSC "VNIINM".
All BOR-60 and BN-600 experimental nitride fuel pins are intact.

Country/Int. Organization:

Russian Federation

5.10 Fuel Modeling and Simulation / 63

PROBLEMS OF CALCULATION MODELLING OF NITRIDE FUEL PERFORMANCE: DRAKON CODE

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The main life limiting factor of nitride fuel pins at high burn-up is fuel cladding mechanical interaction (FCMI) leading to strong deformation or even cladding destruction. The consequences of FCMI depend on fuel and cladding swelling rates, cladding creep rate, cladding long-term stress rupture etc. The calculation modelling problem arise from not enough data on nitride out-of-pile properties and in-pile behavior in dependence on plutonium content, fuel density, irradiation temperature, as well as lack of reliable data on irradiation steel cladding properties. Within the framework of the PRORYV project a comprehensive program for calculation and experimental studies of mixed nitride fuel for BN-1200 and BREST-OD-300 reactors has been designed to provide the required data.

The DRAKON code is designed for numerical simulation of temperature and stress-strain state of fast

Currently DRAKON code is used to study performance of the experimental nitride fuel pins of BN-600 r

Country/Int. Organization:

Russian Federation

3.6 Safety Analysis / 64

Thermal-hydraulics and Decay Heat Removal in GFR ALLEGRO

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One of the key issues in the design of the Gen IV GFR ALLEGRO, a helium-cooled experimental fast reactor, is the core cooling in accident conditions, mainly due to the low thermal inertia of the coolant.
After a brief description of the reactor, this paper presents the currently adopted approach to decay heat removal, and the analysis of some of the most penalizing pressurized and depressurized scenarios.
The results of the benchmarking activities on system codes (CATHARE, RELAP, MELCOR), carried out to optimize the modeling capabilities, are presented.
Starting from the reference design studied up to 2009, the project now explores new possibilities of further development, with a new target nominal power (in the range of 30 –75 MW thermal) and power density (in the range 50 –100 MW/m3), which will be compatible with the safety limits and the design requirements linked mostly to the steel cladded oxide start-up core fuel.
The decay heat removal systems (DHR loops), and their main components must be studied under such conditions to check and improve their efficiency in both forced and natural circulation operation. In addition, the performance of gas injection from the accumulators for the depressurized

conditions are studied.
In addition, the paper summarizes planned experimental programs on validation of the analytical tools and testing of thermal-hydraulics in different flow regimes, using the existing and scheduled experimental helium facilities.

Country/Int. Organization:

France

1.5 LFR DESIGN & DEVELOPMENT / 65

Status of Generation-IV Lead Fast Reactor Activities

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Since 2012 the Lead Fast Reactor provisional System Steering committee (LFR-pSSC) of Generation IV International Forum (GIF) has developed a number of top level activities with the aim to assist and support member countries developments of Lead Fast Reactor technology.
The current full members (MoU signatories) of the GIF-LFR-pSSC are: EURATOM, JAPAN, the RUSSIAN FEDERATION and the REPUBLIC OF KOREA. The pSSC benefits also from the active participation of its observers: the UNITED STATES and the PEOPLE'S REPUBLIC OF CHINA.
The paper highlights some of the main achievements of LFR-pSSC starting from the development of LFR System Research Plan, LFR white paper on safety and LFR Safety Assessment as well as Safety Design Criteria development.
After the presentation of LFR-pSSC top level activities the status of the development of LFR in GIF countries is presented. The collaboration among partners of GIF-LFR-pSSC has proved its effectiveness to help the development of LFR through an open and friendly environment, developing important synergies and exchange of both technical and strategic information.

Country/Int. Organization:

Italy

3.1 Safety Program / 67

The SAIGA experimental program to support the ASTRID Core Assessment in Severe Accident Conditions

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The CEA, together with the NNC, has carried out a feasibility study with regard to conducting an in-pile test program - the future SAIGA program (Severe Accident In-pile experiments for Gen-IV reactors and the Astrid prototype) - on the degradation of an ASTRID-like fuel in the IGR reactor (Impulse Graphite Reactor operated by NNC). The purpose of the SAIGA program is to qualify the SIMMER computer code on the SEASON platform based on tests conducted with axially heterogeneous CFV type ASTRID inner core pins or pin bundles in hypothetical severe accident situations. These tests should be representative, as much as possible, for the phenomena encountered during severe accident sequences considered for ASTRID. The feasibility study aimed to study the generic accident families of loss of coolant and power excursion situations. It is important to point out that the fuel used for these tests can only be a non-irradiated fuel.
The feasibility study focused on tests based on the degradation of one or more fuel pins during Total Instantaneous Blockage (TIB) sequences in a sub-assembly and power excursion (Transient Over-Power: TOP) sequences as in SCARABEE and CABRI with homogeneous pins.
For both scenarios, the feasibility study defined the main characteristics of the experimental devices and the operating conditions for the tests to be conducted in the IGR reactor. The purpose of the studies was to assess the capacity of the IGR reactor to provide the necessary neutron flux during all the transients, to demonstrate the capacity to carry out on-line or post-test measurements of the variables of interest, to study the feasibility of the sodium loop feeding the test device and to assess the cost and timetable for a program of 3 tests.
Preliminary calculations carried out using the SAS-SFR and SIMMER codes were used to simulate the degradation of the fuel during TOP and TIB type tests, respectively.
Based on the information obtained during the feasibility study, specification requirements were given to perform three useful and potentially feasible tests inside the SAIGA program i.e.:

1) Ejection and relocation of fuel in a narrow hydraulic channel (CFV type) with a heterogeneous fuel during a power excursion (TOP type scenario)

2) Loss of flow test on a CFV-type fuel sub-assembly

3) Propagation of a corium pool outside the sub-assembly in the presence of a corium discharge area filled with sodium

Country/Int. Organization:

France

Poster Session 1 / 69

Investigation of steel corrosion products mass transfer in sodium

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The report observes the behavior of the system sodium - oxygen - stainless steel with regard to the sodium cooled circulation loop. Computational and theoretical analysis of mass transport of corrosion products in the channels of non-isothermal circuit in view of chemical interaction of the components of steel with oxygen in sodium, including the reaction of sodium oxide with chromium

in sodium is prepared.
In the proposed model, we consider the reaction of sodium oxide with chromium in sodium in chromium- nickel steel circuit, taking into account the transfer of the reaction products in sodium and dynamics of sodium flow. The processes of impurities interaction with channel walls, formation and transport of suspended particles in the flow of coolant are considered. Closing relations include the equations describing the mass transfer of impurities between the coolant flow and the channel walls, the deposition of particles on the channel surface, the heat exchange between the coolant flow and channel walls.
On the basis of the calculation and the theoretical analysis is refined information about physical and chemical constants, characterizing the mass transport of corrosion products in sodium at presence of increased content of impurities such as oxygen and hydrogen.
Experimental study of mass transfer components of steel in sodium at low and high oxygen content in sodium is carried out. At low oxygen content the composition of the deposition is similar to that of steel dissolved. For the case of high oxygen concentration are performed two experiments: the oxygen content in sodium of 80 ppm and 140 ppm. The comparison of the calculation results with the experimental data on distribution of the chromium deposits in the cooling channels is completed, on which basis are defined updated values of the constants that characterize the mass transfer of chromium by dissolving stainless steel in sodium.
It was found that an increase in the level of dissolved oxygen in sodium increases the solubility of chromium also.

Country/Int. Organization:

Russian Federation / ROSATOM

6.11 IAEA Benchmark on EBR-II Shutdown Heat Removal Tests / 70

IAEA NEUTRONICS BENCHMARK FOR EBR-II SHRT-45R

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A Coordinated Research Project (CRP) initiated by the International Atomic Energy Agency (IAEA) aimed to benchmark Shutdown Heat Removal Tests (SHRT) conducted at Experimental Breeder Reactor II (EBR-II). Two SHRT tests (SHRT-17 and SHRT-45R) representative, respectively of Protected Loss of Flow (PLOF) and Unprotected Loss of Flow (ULOF) transients were considered. The SHRT-45R benchmark included both safety analyses and an optional neutronics benchmark for SHRT-45R. Only the activities carried out for the neutronics benchmark are described in this paper.
The objective of the neutronics benchmark was to provide reactivity feedback coefficients for the thermal hydraulic analysis of SHRT-45R. Several institutes participated in this benchmark, including: Karlsruhe Institute of Technology (KIT), University of Fukui, Paul Scherrer Institute (PSI), and Argonne National Laboratory. The parameters compared code-to-code were keff, βeff, reactivity feedback coefficients (axial, radial and control rod expansion, sodium density, and Doppler) and the power distribution in each subassembly, including fission and gamma heat. The fission and decay heat power for 15 minutes after a postulated scram at the beginning of SHRT-45R were also calculated.
Several stochastic and deterministic codes were used: MC2-3/TWODANT, DIF3D, VARI3D, and PERSENT by Argonne, SERPENT by PSI, and the ECCO/ERANOS codes by the University of Fukui and by KIT. KIT also used the PARTISN code.
Results obtained for keff and βeff were in good agreement (1.2% maximum difference) among the

participants. The reactivity feedback coefficients initially showed a large spread that was reduced by establishing consistency among the definitions used by the participants. However, some spread remains, partially due to the different linear thermal expansion coefficients used in converting the change in reactivity (pcm) to change in reactivity per change in temperature (pcm/K), and will be discussed in the full paper. Differences due to core modeling options (detailed fuel pin modeling vs. homogenized subassembly modeling) and neutron cross-section preparation were also analyzed. Differences among the calculated power distributions were large (up to 80%) in the non-fueled subassemblies, where photon heating dominates, while differences were less than 5% in the fueled subassemblies. No recorded data are available for the detailed power distribution.

Country/Int. Organization:

Germany

Poster Session 2 / 72

SIMMER ANALYSES OF THE EBR-II SHUTDOWN HEAT REMOVAL TESTS

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The Karlsruhe Institute of Technology (KIT) and the Kyushu University (KU) have participated to the Coordinated Research Project (CRP) of the International Atomic Energy Agency (IAEA) on the Experimental Breeder Reactor II (EBR-II) Shutdown Heat Removal Test (SHRT). Two SHRT tests (SHRT-17 and SHRT-45R) representative, respectively, of Protected Loss of Flow (PLOF) and Unprotected Loss of Flow (ULOF) transients have been considered.
For the study, the SIMMER-III ver. 3E code has been employed. The SIMMER-III code is a 2D, multi-velocity-field, multiphase, multicomponent, Eulerian fluid-dynamics code tightly coupled with a space- and time-dependent neutron kinetics model and a fuel pin model, jointly developed by JAEA, KIT and CEA.
For SHRT-17, oriented mainly to investigate the effectiveness of natural circulation, only the fluid-dynamics modules of SIMMER were employed resulting in a reduced calculation time that has allowed testing several modeling options (e.g. radial and axial thermal conduction, effect of IHX position, gap conduction, fuel porosity, etc.).
For SHRT-45R, the standard SIMMER-III version already modified by KU for taking into account a set of specific Equations of State (EOS) and the Thermo-Physical Properties (TPP) for the EBR-II metal fuel has been further extended at KIT by introducing a new core thermal expansion reactivity feedback model and a new PARTISN-based spatial kinetics model.
In order to take into account the peculiarities of the core layout of the two tests (several types of sub-assemblies composing the core), two different 2D (RZ) SIMMER core models were established. In both cases, all the reactor components have been taken into account in the assessment of the models. Unavoidable approximations (necessary for 2D geometry) have been introduced for modeling the reactor outlet Z-pipe and the sodium inlet pipes.
The SIMMER results obtained for the two transients are in good agreement with the available experimental data. The study has allowed, through the simulation of the PLOF case, to get a better understanding of the modeling options and via the simulation of the ULOF case to a further validation of the SIMMER-III code neutronics extensions performed at KIT.
An overview of the main results obtained for the two tests is presented in the paper.

Country/Int. Organization:

Germany

Poster Session 1 / 73

Controlling FCCI with Pd in metallic fuel

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A major factor limiting the lifetime of U-Zr based fuel is fuel-cladding chemical interactions (FCCI). As the fuel is burned, fission product lanthanides (Ln) interact with the Fe-based cladding, causing thinning of the cladding wall and eventual breach of the cladding. In order to extend the lifetime of the fuel in reactor, FCCI must be controlled. Palladium has been shown to be a promising metallic fuel additive to control FCCI due to the stable Pd-Ln intermetallics formed. The current investigation is focused on the characterization of U-Zr-Pd fuel, with and without added lanthanides. Characterization includes as-cast fuel as well as annealed fuel, and comparison to recent postirradiation examination results from U-10Zr fuel. Preliminary diffusion couple results between the fuel (with and without Ln) and iron will also be presented.

Country/Int. Organization:

United States/Department of Energy

Poster Session 1 / 74

Code Qualification Plan for an Advanced Austenitic Stainless Steel, Alloy 709, for Sodium Fast Reactor Structural Applications

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Sodium Fast Reactor (SFR) is one of the leading advanced reactor concepts that would provide a low-carbon energy option to a diverse U.S. power sources. Nuclear energy releases zero carbon emissions during electricity production, and thus is essential in reducing CO2 emissions from the U.S. power sector. SFR also supports other possible missions, including recycling of used fuel for closing the fuel cycle.

Improved structural material performance is one way to improve the economics of SFRs; by increasing thermal efficiency, power output, and design lifetimes of the reactor system. Improved performance and reliability of structural materials could also enable greater safety margins and more stable performance over longer times, and reduce down time of the reactor plant. Advanced materials could also spur improvements in high temperature design methodologies and thereby allowing design simplifications and more flexibility in plant operations. Thus, they could have a significant, positive impact on levelized electricity production cost even if the commodity costs for the advanced materials are higher. Capital cost reduction and improvement in economic return are important incentives for commercial deployments of SFRs.

Alloy 709 is an advanced austenitic stainless steel with enhanced creep strength relative to Code-approved reference construction materials (Type 304 and 316 stainless steels) and that makes it an

attractive candidate material for SFR structural applications.
In this paper, some preliminary data for Alloy 709 will be presented and a qualification plan for developing an ASME nuclear code case will be reviewed.

Country/Int. Organization:

USA/Argonne National Laboratory

3.5 General Safety Approach / 75

Development of Safety Design Criteria for the Lead-cooled Fast Reactor

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The Generation-IV International Forum Lead-cooled Fast Reactor (LFR) provisional System Steering Committee has developed a set of Safety Design Criteria (SDC) dedicated to LFRs. The objective of the LFR SDC is to present a set of reference criteria for the design of systems, structures, and components of LFR systems with the aim of achieving the safety goals of the Generation-IV reactor system. The work has been based on the SDC for the Sodium-cooled Fast Reactor (SFR), since the GIF LFR and SFR systems share a number of design solutions and some safety-related phenomenology. For the development of LFR SDC it was also found useful to use the same structure and methodology of the already existing SFR SDC. A set of eighty two (82) reference safety design criteria for LFRs are systematically and comprehensively laid out in the SDC. The paper summarises results of the steps taken to draft the present set of LFR SDC and provides outlook for their further review and development, in particular towards the individual sets of detailed Safety Design Guidelines.

Country/Int. Organization:

European Commission, Joint Research Centre, Netherlands

4.1 Fuel Cycle Overview / 76

Reprocessing of fast reactors mixed U-Pu used nuclear fuel: studies and industrial test

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Mixed U-Pu used fuel of fast reactors has a high Pu content, high burn-up (50 GW/days*ton and more), and short cooling time (no more than 3 years) as compared with the thermal reactors used fuel. Combined (pyro + hydro) and hydrometalurgy reprocessing technologies are developed in Russian Federation for the close nuclear fuel cycle. These technologies provide reprocessing of used fuel with 1-3 years cooling time, 10-15 % of Pu content and burn-up up to 100 GW/days*ton.
The aim of reprocessing is production of purified mixture of actinides oxides. The purification of actinides from fission products should be around 10.000.000.
The dry reprocessing technology based on pyroelectrochemical refining is under development as well. This technology should be suitable for reprocessing of fast reactors used fuel with burn-up more than 100 GW/days*ton and 1 year cooling time*
The main results of the studies are being discussed in a current paper.
*Within studying the analysis of products and operations of reprocessing 8 MOX irradiated assemblers from BN-600 reactor at RT1 plant with burn-up from 73 up to 89 GW/days*ton were performed. The technology steps were made under standard conditions. The increasing of Pu loses during MOX used fuel of BN-600 as compared with thermal reactor used fuel reprocessing was not found.
The design of reprocessing facility for mixed U-Pu nitride fuel started at 2015. This facility is planed to be built at Siberian Chemical Combine as a part of experimental and demonstrating energocomplex with reactor BREST–ED-300.

Country/Int. Organization:

Russian Federation

5.3 Advanced Fast Reactor Cladding Development I / 77

Development of core and structural materials for fast reactors

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This paper summarizes ongoing efforts in Japan Atomic Energy Agency on the development of core and structural materials for sodium-cooled fast reactors. For core materials, oxide dispersion strengthened (ODS) steels and 11Cr ferritic steel (PNC-FMS) will be applied to fuel pin cladding and wrapper tube, respectively. As for ODS steel, 9Cr,11Cr-ODS steels have been extensively developed. Their laboratory scale manufacturing technology has been developed including reliability improvement in tube microstructure and strength homogeneity. Large scale manufacturing technology development and mechanical testing for codification of material strength standard are on-going. As for PNC-FMS wrapper tube, development of dissimilar joining technique with type 316 steel and properties evaluation of dissimilar welds have been carried out. For structural materials, 316FR stainless steel and Modified 9Cr-1Mo steel are being code qualified. Long-term data have been accumulated and the properties are analysed to establish a technical basis for 60-year design. Also described is the current status of codification of structural materials standards in the design code of fast breeder reactors published form the Japan Society of Mechanical Engineers.

Country/Int. Organization:

Japan

Poster Session 2 / 78

Assessment of Creep Damage Evaluation Methods for Grade 91 Steel in the ASME and JSME Nuclear Codes

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Grade 91 steel is a Code-approved construction material in the ASME and JSME nuclear codes. Applications of Grade 91 steel include intermediate heat exchanger, piping, steam generator tubing and shell, etc. for sodium fast reactor systems. Current creep-fatigue damage evaluation method in the ASME and JSME nuclear code differs in the method to calculate creep damage. In the simplified inelastic approach of the ASME Code, the creep damage is calculated using the isochronous stress-strain curves. In the JSME Code, the creep damage is evaluated using a creep strain equation combined with the strain hardening formulation. In this paper, these two approaches will be reviewed and creep damage predictions from illustrative examples using these two approaches will be presented. Approaches to possible harmonization will be discussed.

Country/Int. Organization:

USA/Argonne National Laboratory

Poster Session 2 / 79

DEPENDENCE OF INTERMEDIATE HEAT EXCHANGER LIFE ON PRIMARY SODIUM HEATING RATE DURING POWER RAISING

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During start-up of the reactor, the rate at which the power raising is to be done is a critical issue. As a safe practice, the power raising can be done slowly by increasing the sodium temperature at the rate of 5K/hr, 10K/hr or 20K/hr. But this takes a long time for start-up, i.e., 80, 40 and 20 hours per cycle respectively to raise the sodium temperature from the initial temperature of 453 K to the full power temperature of 820 K. On the other hand, to minimize this time, the power raising can be done at a higher rate, i.e, 40 K/hr, 60K/hr, 80K/hr or maybe 100K/hr or even higher. But higher heating rate causes creep and fatigue damages to the reactor components like Control Plug, Intermediate Heat eXchanger (IHX) and Inner Vessel, which are in contact with the hot primary sodium. Hence, a thermo-mechanical analysis has been carried out to optimize the heating rate during power raising. In this study, the damage possible in the IHX as a function of the heating rate at a critical region has been determined by developing a numerical model, and the dependence of IHX life on the heating rate is estimated. The heating rates considered in this investigation are 20K/hr, 40K/hr and 60 K/hr. The fatigue damages caused due to power raising at the free level of sodium for heating rates 20 K/hr, 40 K/hr and 60 K/hr are 7.19E-17, 1.64E-08 and 4.95E-05 respectively and the corresponding creep damages are 0.013, 0.099 and 0.145 for 861 cycles. The number of allowable cycles determined by considering the creep and fatigue damage values at full power, along with the values at power

raising are 2798, 2187 and 1958 cycles at heating rates 20 K/hr, 40 K/hr and 60 K/hr respectively. Hence, a sodium heating rate of 60 K/hr is acceptable during power raising in view of IHX life. This reduces the power raising duration from 20 hours to 6.6 hours.
The full paper will present the methodology adopted for identifying the critical location in IHX, procedure for estimation of the damage and the final conclusion about power raising duration in a typical medium size pool-type fast breeder reactor.

Country/Int. Organization:

India

Poster Session 1 / 80

Examination of ChS-68 Steel Used as a BN-600 Reactor Cladding Material

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Austenitic stainless steel has been used as a standard material for BN-600 fast reactor claddings for many years. High-temperature strength is one of the advantages of austenitic steels over ferritic-martensitic ones. Nevertheless, corrosion damages, radiation-induced swelling, creep, embrittlement, and strength reduction are topical problems for claddings made of austenitic steels. In this respect at high burn-up levels swelling is the main problem limiting operation of the material.
Originally thoroughly studied EI-847 steel was used for developing new austenitic steels. After boron modification EP-172 steel was obtained. ChS-68 steel doped with elements reducing radiation-induced swelling, such as boron, silicon, and titanium, was based on EI-847 steel.
At the initial stage significant radiation-induced deformation of fuel assembly elements was one of the factors limiting fuel burnup to 7.2 % FIMA and damage dose to 44 dpa. Transition to new steels and reactor core modifications made it possible by 2000 to attain burnup level of 9.2 % FIMA and damage dose level of 73 dpa per cladding.
ChS-68 properties optimization was carried out by High-technology Research Institute of Inorganic Materials (VNIINM) using the results of post irradiation examination made by Beloyarsk NPP and ROSATOM materials testing enterprises, including Institute of Nuclear Materials (INM).
Irradiated in BN-600 claddings of standard and test fuel assemblies, as well as materials test assembly samples (ChS-68 and other austenitic steels, similar in composition) were examined at INM hot cells. Characteristics and properties of claddings made with variation of steel chemical composition within specifications, at different cold work levels, according to different metal manufacturing and tube production technologies, were determined.
Results of post irradiation examination of physical, mechanical and corrosion properties, and structural characteristics were used to analyze processes leading to structure and properties changes and to predict residual life. On the basis of the examination INM researchers have published a number of articles and presented a number of papers at the conferences of different levels.
Using the obtained results VNIINM in collaboration with Machine-Building Plant (MSZ) has improved cladding manufacturing technology. It resulted in a stepwise (during 15 years) extension of standard fuel assembly service life up to 73, 78, 82, 87 dpa with burnup increase from 9.2 to 13.2 % FIMA. It is expected to increase damage dose at least to 92 dpa and fuel burnup to 14-15% FIMA.

Country/Int. Organization:

Russian Federation/Joint-Stock Company "Institute of Nuclear Materials"

5.2 Advanced Fast Reactor Fuel Development II / 81

Analysis of experimental data on fission gas release and swelling in mononitride fuel irradiated in BR-10 reactor

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Uranium mononitride fuel was used in the fourth and fifth BR-10 reactor core loadings. The total number of irradiated fuel pins was 1250 (660 and 590 fuel pins in fuel loadings IV and V respectively). Most fuel pins were irradiated up to design burnup (8%) without cladding failure.

In addition to standard FAs, some experimental FAs were irradiated in BR-10 reactor. The post irradiation examination (PIE) of 8 standard and 3 experimental fuel assemblies (FA) was done in the IPPE hot lab.

The paper presents the results of study of fission gas release and nitride fuel swelling in standard and experimental fuel pins irradiated in BR-10 reactor. These two phenomena have a significant impact on cladding stress level and therefore on the fuel life time.

Substantial fission gas release from BR-10 nitride fuel starts at a burnup of more than 3at%. Irradiation temperature increase and fuel density decrease both lead to increase of gas release rate. N14(n,α)B11 nuclear reaction in nitride fuel causes formation of quite big amounts of helium. This fact should be taken into account in computer codes used for nitride irradiation behavior modeling. The paper presents the measured nitride swelling rate values in the temperature range from 760 to 1115 C. Increase of fuel temperature leads to increase of fuel swelling rate.

Country/Int. Organization:

Russian Federation

7.2 Economics of Fast Reactors / 82

Fast Reactors and Nuclear Cogeneration: A Market and Economic Analysis

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Fast reactors are typically considered for their potential to make optimal use of natural resources or for their potential to minimize the amount and level of nuclear waste. The additional opportunity of fast reactors designed for cogeneration applications (i.e., production of electricity and process heat), which can bring an enormous reduction in CO2-emissions, is made possible by the elevated temperatures characterizing the primary circuit of such reactors, compared to traditional light water reactors. This article will provide a state-of-the-art overview on the cogeneration market with emphasis on opportunities for lead, gas, and sodium fast reactors, summarize recommendations for these fast reactor systems and their interfaces with a cogeneration application, and discuss the results of a top down cost estimate for a lead fast reactor system with a typical cogeneration application. The economic analysis clearly shows that coupling a small or medium sized fast reactor to a cogeneration application seems attractive.

Country/Int. Organization:

Netherlands

6.11 IAEA Benchmark on EBR-II Shutdown Heat Removal Tests / 84

Final Results and Lessons Learned from EBR-II SHRT-17 Benchmark Simulations

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In 2012 the IAEA initiated a 4-year coordinated research project (CRP) "Benchmark Analyses of EBR-II Shutdown Heat Removal Tests", with Argonne National Laboratory serving as the lead technical institution. Nineteen participants from eleven countries were involved in the project. The overall purpose of the CRP was to improve validation of state-of-the-art sodium-cooled fast reactor (SFR) computer codes through comparisons of the analytical predictions against whole-plant recorded test data. A secondary purpose was training of the next generation of SFR analysts and designers through participation in international benchmark exercises. Numerical simulations were performed for the two most severe experiments conducted in the 1980s during Argonne's EBR-II Shutdown Heat Removal Tests program. The first test was SHRT-17, where a PLOF (Protected Loss Of Flow) accident scenario was performed, and the second –SHRT-45R, where a ULOF (Unprotected Loss Of Flow) scenario was performed. This paper describes the results (blind and final) of the SHRT-17 experiment simulation, findings of the CRP benchmark exercise associated with the EBR-II SHRT-17 test, improvements proposed by the participants, and the lessons learned within the project.

Note: EBR-II Benchmarks Invited Session

Country/Int. Organization:

Russia

Poster Session 1 / 85

Study on the limits of confinement leakage rates of pool-type sodium-cooled fast reactor

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Confinement including containment and primary vessel are barriers of radioactive gas containing.To select the appropriate confinement leakage rate can ensure the pool-type sodium-cooled fast reactor operating safety.In the paper,large pool-type sodium-cooled fast reactor was the object of research.The impact of different primary vessel and containment leakage rate to radioactive material release behavior were researched by using the ORIGEN2 to calculate the burden of fast reactor and researching the radioactive gas release routes of normal operating and accident condition.Hence,the method to analyze the containment and primary vessel leakages reate design value ranges were found. It was concluded that normal operating condition's environmental of radioactive gas was the main limiting factor;containment could make mitigative effect to accident obviously on the accident condition,especially to environmental impact at site boundary on the early state of accident

Country/Int. Organization:

P. R. China

Poster Session 1 / 87

X-RAY DIFFRACTION STRUCTURAL ANALYSIS OF STRUCTURAL AND FUEL MATERIALS FOR BN-600 REACTOR

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Operation of BN fast reactor structural and fuel materials under hard irradiation and at high temperatures leads to significant irreversible changes in material structure.
X-ray diffraction analysis is a common physical method for materials testing and condition monitoring. Different techniques help to determine crystal phase composition, form, internal stress, crystal preferred orientation (grain orientation) and other parameters.
At the Institute of Nuclear Materials (INM) a shielded cabinet of the D8 ADVANCE remote machine (BRUKER, Germany) is used for the examination of samples with activity up to 5.6⊠1011 Bq for 60Co.
A curved monochromator and a scintillation counter are mounted on the stationary arm covered with a lead protective shroud to reduce ionizing irradiation impact of the examined highly radioactive materials.
At INM there is a lot of information collected on the characteristics of fine structure of austenitic steels used for claddings, irradiated fuel compositions including uranium dioxide and MOX fuel.
Dependencies of the effect of the change (within specifications) in main alloying element concentration and low irradiation doses on the lattice parameter of ChS-68 steel solid solution were obtained.
Determination of the contribution of the cladding initial state structural factors to the swelling is in process.
The study of possible determination of plutonium concentration in MOX fuel and division of plutonium, oxygen enhancement ratio and fission product contribution to the lattice parameter changes has been carried out.

Country/Int. Organization:

Russian Federation /Joint Stock Company «Institute of Nuclear Materials»

6.7 Experimental Thermal Hydraulics / 88

NACIE-UP: a HLM loop facility for natural circulation experiments

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The NACIE-UP facility at ENEA-Brasimone R.C. is a large scale loop operating with Lead Bismuth Eutectic (LBE) in the range of 180°C-450°C in free and mixed convection. The difference in height between heat source and heat sink is about 4.5 m and allows the establishment of natural circulation regime inside the loop. Moreover, a gas-lift system provides the pressure head to enhance the circulation. The facility comprises also a secondary system in pressurized water with air-cooler to cool the primary LBE. The primary side is instrumented with a prototypical thermal flow meter, a pressure transducer to measure pressure drops across the test section and several thermocouples.
A wire-spaced 19-pins fuel bundle is actually installed inside the NACIE-UP loop. The pin bundle has a maximum wall heat flux of 1Mw/m2 and is equipped with 67 thermocouples to monitor temperatures and analyze the heat transfer coefficient in different sub-channels and axial positions. Another

test section has been designed in order to study the thermal-hydraulic behavior of a pin fuel bundle cooled by HLM in a flow blockage accident scenario. The bundle is composed of 19-pins with two spacer grids and is equipped with 100 thermocouples in order to monitor pin wall temperatures both with and without blockage, the presence of hot spots and to evaluate the thermal mixing above the pin bundle. The experimental campaigns related to these two sections aim to study outstanding thermal-hydraulic phenomena such as the heat transfer during transient from forced to natural circulation flow and the flow blockage accident in a fuel assembly. These activities are in support of the front-end engineering design (FEED) of GEN. IV/ADS prototypes and demonstrators. Some experimental data on heat transfer coefficient obtained in mixed and natural circulation flow regime are also presented in the paper.

Country/Int. Organization:

Italy

6.8 Experimental Facilities / 89

CIRCE-ICE EXPERIMENTAL ACTIVITY IN SUPPORT OF LMFR DESIGN

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The CIRCE-ICE experimental facility, designed and hosted at ENEA Brasimone R.C., is a large pool multi-purpose integral test facility devoted to support the Thermal-Hydraulic design of Gen IV Liquid Metal Fast Reactor.
Thermal stratification phenomena in HLM pool may induce significant thermal loads on the structure in addiction to existing mechanical loads. Moreover, vertical temperature gradient in accidental scenario conditions could opposes to the establishment of natural circulation which is a fundamental aspect for the achievements of safety goals required in the GEN-IV roadmap.
In this work the LBE temperature field of the pool is investigated during a series of experiment aimed at simulating a protected loss of heat sink with loss of flow. Obtained results are here presented and discussed.
The other important aspect here considered is the chemistry of the coolant. This topic is deeply connected with thermal-hydraulic behaviour of the pool in fact, the oxygen distribution in the coolant is strongly affected by the thermal stratification, posing relevant issues related to the coolant chemistry control and corrosion of structural materials in the pool.
Concurrently with the thermal hydraulic experimental activity, the calibration of various potentiometric oxygen sensor was performed in the chemical laboratory of ENEA Brasimone R.C. in oxygen-saturated liquid LBE and Lead. In particular, different oxygen sensor with various reference systems (Pt-air (gas), Bi/Bi_2O_3 (liquid) and Cu/Cu_2O (solid)) were manufactured and their performances investigated in a wide range of temperature.
Finally, different zirconia electrolytes: Yttria Partially Stabilized Zirconia (YPSZ, with ≈ 5 mol. % of Yttria) and Yttria Totally Stabilized Zirconia (YTSZ, with ≈ 8 mol. % of Yttria) were tested. In this work experimental measurements in the temperature range 160-550 °C are reported and collected data compared with the theoretical expected values.

Country/Int. Organization:

Italy

7.4 Fuel Cycle Analysis / 90

SVBR Project: status and possible development

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SVBR Project deals with small modular fast reactor with heavy liquid metal coolant (SVBR-100) - the eutectic alloy lead-bismuth (LBC). LBC technology was mastered and applied in an industrial scale for submarine applications at the beginning of 1950th. SVBR Project is a pilot project in terms of implementation of large-scale high-tech projects in the nuclear industry jointly with a commercial partner.
Detailed designs of both reactor unit and nuclear power plant (NPP) have been finished in such extent that allowed JSC "AKME-engineering" to obtain the license for the placement a nuclear power plant facility - pilot unit with 100 MW lead-bismuth coolant fast reactors SVBR-100 in the Ulyanovsk region.
Engineering designs of the nuclear power plant and the reactor unit have been examined by industry experts and then considered by scientific and engineering council of SC "Rosatom". The scientific and engineering council advised to carry out cost optimization of the pilot unit and to analyse SVBR NOAK NPP economics based on its conceptual design.
Pilot unit analysis made evident the necessity to reduce specific costs, size of the NPP site, nuclear island building volume and some others in order to reduce pilot unit expenditures and to rich desired competitiveness of the NOAK NPP.
The Basic requirements for reducing of capital and operational expenditures are described in the article. Implementation of these requirements into the NOAK NPPs allow us to get the target values of their levelized cost of energy (LCOE) and other project value economic indicators.
Feasibility study carried out for possible means facilitated the Implementation of the Basic requirements results in the following list of such means for NOAK NPP:
 Increasing the reactor unit capacity by means of increasing average core output coolant temperature, decreasing nonuniformity factor of the core power distribution, increasing fuel rod cladding temperature;
 Decreasing own need expenses (electricity);
 Decreasing equipment costs due to economy of scale and learning factors;
 Decreasing specific costs due to modularization factor;
 Optimization number of NPP personnel;
 Simplifying core reloading procedure;
 Decreasing construction costs due to modularization factor, phased commissioning of the interim spent fuel storage and utilization of simplified core reloading procedure.
NOAK NPP economic indicators (LCOE, NPV, IRR, DPP) assessment is presented subject to implementation of mentioned basic requirements.

Country/Int. Organization:

Russian Federation, JSC AKME-Engineering

Poster Session 2 / 91

Development and Validation of EBRDYN code by Benchmark Analysis of EBR-II SHRT-17 Test

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Experimental Breeder Reactor (EBR-II) was a U-Pu-Zr metal-alloy fueled liquid-metal-cooled fast reactor, extensively used for conducting safety experiments. Out of several tests conducted, the SHRT-17 loss of flow test conducted in 1984 demonstrated the decay heat removal capability by natural circulation in sodium cooled fast reactor with no core damage. In order to utilize the data recorded during these tests for improving the computer codes by extensive code validation, IAEA has initiated a coordinated Research Project (CRP) in which IGCAR, INDIA is one of the participants. IGCAR has developed a plant dynamics code EBRDYN, on the same principles as Indian safety codes FBRDYN, DYANA-P and DHDYN used for the safety analysis of Indian Fast Reactors FBTR and PFBR. The EBRDYN consists of thermal hydraulic models of various components of EBR-II primary heat transport system viz., core, hot upper plenum, Z-pipe, intermediate heat exchanger (IHX), cold pool, primary sodium pumps and associated piping. All the subassemblies (SA) of the core are grouped into a convenient number of radial zones receiving sodium from bottom plenum and discharging into the top hot plenum. The instrumented SA models can predict the thimble flow and its heat transfer with the SA sodium. The mixing of sodium in the hot plenum, dynamics of Z-pipe, heat transfer in IHX from primary to the secondary sodium have been modeled. The primary sodium pumps have been modeled using homologous characteristics and with previous EBR-II experience. The pump models are capable of handling negative flows. The primary sodium circuit has been modeled with the capability to handle two primary pumps operating in parallel. The initial conditions of the reactor and transient boundary conditions viz., core decay power, primary pumps speed, IHX secondary sodium flow rate and inlet temperature as provided by Argonne National Laboratory (ANL) have been used. The steady state results are comparing well with the measured data. Most of the transient parameters viz., primary pumps flow rate, core outlet temperature evolution, Z-pipe inlet temperature, cold pool temperature are comparing well with the measured data. The full paper gives the details of the thermal hydraulic modeling of the primary heat transport system, the transient results and their comparison with the measured data and the effect of uncertainties in various parameters on the transient results.

Note: EBR-II Benchmarks CRP Poster Session

Country/Int. Organization:

INDIA

5.8 Structural Materials / 92

PREDICTION OF CREEP-RUPTURE PROPERTIES FOR AUSTENITIC STAINLESS STEELS UNDERGONE NEUTRON IRRADIATION AT DIFFERENT TEMPERATURES

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Early the physical-and-mechanical model of integranular fracture has been developed that allows the prediction of creep-rupture properties of austenitic stainless steels at different neutron fluxes and temperatures. The model is based on the equations of void nucleation and growth on grain boundaries caused by inelastic deformation (creep and plastic strain) and diffusion of vacancies. The model has been verified when using available published data for austenitic stainless steels of 18Cr-9Ni and 18Cr-10Ni-Ti grades.
The aim of the present work is further verification of the model for austenitic stainless steels under

neutron irradiation. For this in-reactor tests are carried out for gas-filled tubes at different temperatures (550 oC and 600 oC) in RBT-6 reactor with neutron flux equal to 5e+13 n/cm2. Specimens are made from austenitic stainless steels of 18Cr-9Ni and 16Cr-11Ni-3Mo grades. The results calculated by the model are compared with the obtained experimental data, and their good agreement is shown.

Country/Int. Organization:

Russia/CRISM "Prometey",
Russia/JSC "SSC RIAR

3.4 Sodium leak/fire and other safety issues / 93

Identification of important phenomena under sodium fire accidents based on PIRT process

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Since sodium has high chemical reactivity with oxygen and moisture, sodium fire accident is one of key issues in sodium-cooled fast reactor (SFR) plants when the sodium leaks out of a coolant circuit. In order to evaluate the consequence of the sodium fire event numerically, JAEA has developed sodium fire analysis codes such as SPHINCS and AQUA-SF. This paper describes a PIRT (Phenomena Identification and Ranking Table) process for a sodium fire event. The present PIRT is aimed to utilize for validation and improvement of the sodium fire analysis codes. Because a sodium fire accident in an SFR plant involves complex phenomena, various figures of merit (FOMs) for importance ranking could exist in the PIRT process. Therefore, the FOMs are specified through factor analysis. Associated phenomena in a sodium fire event are identified through the element- and sequence-based phenomena analyses. Then importance of each associated phenomenon is evaluated by considering the sequence-based analysis of associated phenomena related to the FOMs. Finally, we have established the ranking table through the factor and phenomenon analyses.

Country/Int. Organization:

JAPAN

5.3 Advanced Fast Reactor Cladding Development I / 94

Examination of Fast Reactor Materials and Structural Elements at JSC "INM"Premises

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At the Institute of Nuclear Materials post irradiation examination has been carried out since BN-600 reactor commissioning to justify safety and reliability of different core elements during routine operation and to search for new ways of extending their service life. The examination is carried out in close cooperation with the design bureau (Afrikantov OKBM), nuclear operator (Beloyarsk NPP), materials testing enterprise (VNIINM), fuel assembly manufacturer (MSZ) and other enterprises. It helps to use post irradiation examination results promptly to advance reactor structural elements and improve economic efficiency. Main aspects of the examination are as follows:
- examination of fuel elements and shroud tubes of standard, trial and test fuel assemblies;
- examination of reactor control and safety units (control rods including absorber elements and shroud tubes);
- examination of materials science assembly samples irradiated in BN-600 reactor;
- investigation of possible service life extension from 30 to 45, and then to 60 years.

The examination carried out at INM is unique because it is not limited with statement of fact of changes in structural elements and their material properties. The aim of the examination is to predict their further behaviour and find out the cause of the changes. It is not sufficient to carry out separate post irradiation examinations, there should be a systematic result set based upon theoretical concepts on the process mechanisms, descriptive modeling of structural evolution processes and corresponding changes in physical and mechanical properties. It is also necessary to improve existing examination techniques and develop new ones to obtain characteristics used to predict residual and limited life for core elements and the reactor as a whole.
The paper aims to show the main results of BN-600 structural element examination at INM, demonstrate their practical application, and make a review on the developed theoretical concepts and the development of the techniques correlating with the examination.

Country/Int. Organization:

Russian Federation

5.4 Advanced Fast Reactor Cladding Development II / 95

Modeling of Processes in Austenitic Steel Produced Under Irradiation in Fast Reactors and Possibilities of Model Practical Application

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Nowadays austenitic stainless steels are used as a cladding material for BN-600 and BN-800 reactors. Examinations carried out at the end of fuel element service life give the information about cladding state. On the basis of the examination it is necessary: 1 –to determine residual life and 2 –to find the way for its extension.
Extrapolation methods are generally used for the first aspect. The results obtained for fuel elements of different fuel assemblies at attaining different damage doses and fuel burn-ups are used. As a rule the results are limited by linear extrapolation. The prediction accuracy is quite low for several reasons. The initial state of claddings from different lots (and casts) is not the same, therefore there is some error even when processes of properties changes linearly depend on irradiation parameters (for example, damage dose). Moreover, the dependence of some processes, radiation-induced swelling in particular, on dose and temperature is quite nonlinear. Therefore linear extrapolation is unacceptable.
Extrapolation used for the second aspect almost gives no results as the characteristics to be correlated are not defined.
At JSC "INM" a technical description of the processes occurring in metal materials under irradiation has been developed for a long time. Description of point defect formation (vacancies and interstitials) is the key concept. All further microstructural changes are determined by the formation intensity, migration and interaction with other microstructural elements (impurity atoms, dislocations, grain

boundaries and other sinks). A machine for quantitative description of point defect migration and concentration has been developed and is used for austenitic stainless steels.
Based on the developed theoretical concepts different stages of structural changes, radiation-induced swelling in particular, as well as the effect of structural changes on physical and mechanical properties have been modeled. These models were used to predict changes in material structure and properties of the claddings operated in BN-600 reactor core.
The paper aims to show the developed at JSC "INM" models of changes in austenitic steel structure and properties under irradiation in fast reactors and to demonstrate their application for BN-600 reactor claddings.

Country/Int. Organization:

Russian Federation

Poster Session 1 / 97

Numerical Investigation of Sodium Spray Combustion Test with SPHINCS code

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As a collaboration on the field of advanced reactor modeling and simulation in Civil Nuclear Energy Research and Development Working Group (CNWG) between Japan Atomic Energy Agency (JAEA) and Sandia National Laboratory (SNL), information exchange of sodium combustion modeling and experimental data has been carried out.
In the collaborative work, a benchmark analysis of Surtsey spray combustion experiments done by SNL has been conducted using SPHINCS code at JAEA and CONTAIN-LMR code at SNL. In this paper, the numerical result of SPHINCS code and the comparison between SPHINCS and CONTAIN-LMR codes are discussed. Furthermore, sensitivity analyses have also been carried out to investigate the influential factor on the experiments. As a result, it is demonstrated that the average droplet diameter of sodium spray has a strong influence on the gas temperature and pressure peaks appearing at an early stage on the experiment. On the other hand, parameters concerning with the pool combustion model affect the experimental result at the later stage.

Country/Int. Organization:

Japan/Japan Atomic Energy Agency

7.2 Economics of Fast Reactors / 98

Providing the competitiveness of nuclear energy in the implementation of PRORYV project

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In the report said about reducing the share of nuclear power generation in the world. In conditions inter-fuel competition highest growth rates show renewables. The key to advancing the development of nuclear energy is to ensure the competitiveness of NPPs whith solving systemic problems. Modern NPPs with LWR in an open NFC, practically exhausted the potential of improving the competitiveness.
In the report considered the comparative competitiveness of NPPs and power plants whith fossil fuel, renewable energy sources. All calculations are made for different countries by monitoring data of technical and economic. Singly considered the comparison conditions for the Russian Federation. Considered criteria of competitiveness for NPPs, allowing to ensure the effective development of nuclear power, taking into account of improving the technical and economic performance of alternative generation. Fixed the requirements for the technical and economic parameters of NPP FRs and closed NFC. The data on the assessment of the achievement the required indicators for NPPs with FRs and closed NFC on the basis of actual work within the PRORYV project. Considering the model of PRORYV project management, identify areas to improve efficiency of the risk-management during creating innovative facilities.

Country/Int. Organization:

Russian Federation

6.1 CFD and 3D Modeling / 99

Modelling and Simulation of Heat Transport System and Steam Power Transition System of CEFR

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In this paper, the graphics interface and parameter modeling real-time simulation software, Jtopmeret, was used to model and simulate the intermediate heat transport system and the steam power conversion system of China Experimental Fast Reactor (CEFR). The two-phase, multi-component models were taken into consideration to simulate the flow and heat transfer of working medium sodium and steam-water Ranking circle. The matrix solving method was used in this paper to solve the mass, momentum, and energy conservation equations accurately, quickly and steady. Operating characteristics under steady, transient and malfunction operations of CEFR were researched. The simulation results showed that the errors of main parameters under different steady operations were less than 1%, the trend curves under transient operations and malfunction operations were reasonable, and the response of the secondary and third loop could show the operating and safety characteristics of CEFR. The models had been applied to full scale simulator of CEFR.

Keywords: China Experimental Fast Reactor (CEFR), modeling and simulation, two-phase, multi-component models

Country/Int. Organization:

P.R.China

Poster Session 2 / 100

CALCULATION OF NEUTRONIC PARAMETERS IN SUPPORT OF A BOR-60 EXPERIMENTAL FA WITH MODERATING ELEMENTS

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At present, different nuclear fuels (NF) to be used in advanced fast neutron reactors (AFR) are tested in the BOR-60 reactor. In such in-pile testing the top priority is to ensure the maximum possible compliance of the target NF irradiation parameters with the design operating parameters. The key monitored parameters in testing experimental fuel elements are the fuel burnup rate and linear heat rate that depend on the nuclear fission rate in the fuel elements.
Rather low enrichment of tested fuel compositions (as compared to BOR-60 standard fuel) and low neutron flux density (as compared to big fast neutron reactors) in the BOR-60 core make it difficult or even impossible to provide the target heat and fuel burnup rates of NF.
To increase the nuclear fission reaction rate in the experimental fuel elements it is suggested to install neutron moderating elements in an experimental fuel assembly (EFA).
We considered three EFA design options:

- Option 1: EFA contains 19 fuel elements and no moderating elements;
- Option 2: EFA contains 13 fuel elements and 6 moderating elements;
- Option 3: EFA contains 6 fuel elements and 13 moderating elements.

The outcome is a calculated data analysis confirming the effectiveness and safety of the suggested design solution. Patent #2560919 was obtained for this invention in 2015.
This EFA design option enables wider BOR-60 capabilities in testing advanced nuclear fuels due to high fuel burnup and heat rates in the experimental fuel elements.

Country/Int. Organization:

Russian Federation

2.3 Decommissioning of Fast Reactors and Radioactive Waste Management / 101

Dependability of the fission chambers for the neutron flux monitoring system of the French GEN-IV SFR

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Sodium-cooled fast reactors (SFR) have been selected by the Generation IV International Forum, thanks to their capability of reducing nuclear waste and saving nuclear energy resources by burning actinides. With reactors such as RAPSODIE, PHENIX and SUPER PHENIX, France gained a 50 year

experience in designing, building and operating SFR: since 2006 the CEA leads the development of an innovative GEN-IV nuclear-fission power demonstrator. As a part of it, the neutron flux monitoring system must, in any situation, permit both reactivity control and power level monitoring from startup to full power. It also has to monitor possible changes in neutron flux distribution within the core region in order to prevent any local melting accident. This implies to install the neutron detectors inside the vessel, putting severe constraints on the detector design to ensure its dependability, that is, both its reliability and maintainability.

In this paper, we present the Photonis high-temperature fission chambers (HTFC) featuring wide-range flux monitoring capability and justify their specifications with the use of simulation and experimental results.

We show that the HTFC dependability is enhanced thanks to a robust physical design and that the mineral insulation is insensitive to any increase in temperature. Indeed, the HTFC insulation is subject to partial discharges at high temperature when the electric field between their electrodes exceeds 200 V/mm or so. These discharges give rise to signals similar to the neutron pulses generated by a fission chamber itself, which may bias the HTFC count rate at startup only. However, we have experimentally verified that one can discriminate neutron pulses from partial discharges using online estimation of pulse width.

In order to satisfy the requirement of wide-range capability, we propose to estimate the count rate of a HTFC using the third-order cumulant of its signal. The use of this cumulant can be seen as an extension of the so-called Campbeling mode, based on the variance, hence the name high order Campbelling method (HOC). The HOC ensures the HTFC response linearity over the entire neutron flux range using a signal processing technique that is simple enough to satisfy design constraints on electric devices important for nuclear safety. The simulations are supported by experimental results that demonstrate that the HOC agrees with the simple pulse counting estimation at low count rates and provides a linear estimation of the count rates at higher power levels.

Country/Int. Organization:

France

3.4 Sodium leak/fire and other safety issues / 102

Numerical –experimental research in justification of fire (sodium) safety of sodium cooled fast reactors

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On the basis of the normative documents requirements on fire safety of nuclear power facilities the build concept and the composition of fire protective system in fast reactors premises with sodium equipment are presented. The main purpose of sodium fire safety system is to protect the technological areas of nuclear power plants with sodium cooled fast reactors from hazards of sodium fire. The hazards of sodium fire are: increasing the pressure and temperature of gas environment in emergency rooms, raise of building structures temperatures upon burning of sodium. Another hazard of sodium fire is spreading of sodium aerosols in the premises of the plant which are harmful to human health.
The numerical justification of the sodium fire extinguishing system effectiveness in case of possible accidents with sodium burning in certain areas of fast reactor is performed.
During the formation of the fire safety conception on sodium cooled fast reactors the special attention is focused on the nature of the outflow and sodium burning and on the sodium leakage limitation.

The numerical and experimental researches aimed at the performance possibility and efficiency of system for early detection of leaks and sodium burning based on automatic smoke fire detectors VESDA are performed.
The issues of jet outflow and spray burning of the sodium coolant and related problem with increasing of the gas pressure and temperature in the room are considered. In the framework of these issues the results of experimental works for sodium spray burning made by French experts are considered. The results of the analysis and processing of experimental data are presented. The method is developed for the gas pressure raising calculation in the room based on processing of experimental data. The main experimental results with sodium flow through the defects in the pipeline under the insulation are presented. A possibility is shown for safe localization of the jet outflow and sodium spray burning in the presence of pipelines and equipment insulation and cladding based on this experimental data.

Country/Int. Organization:

Russian Federation

Poster Session 2 / 103

Potential Capabilities in Transmutation of Minor Actinides of the BOR-60 Reactor and MBIR Reactor under Construction

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The fast research reactor BOR-60 is one of the world's leading research reactors in large-scale testing of fuel elements, FAs and control rods of different design options, advanced fuel compositions and structural materials, as well as in tryout of the closed fuel cycle technologies and transmutation of minor actinides. BOR-60 is a unique experimental reactor with a neutron spectrum ranging from the hard one in the core up to the intermediate one in the blanket; the neutron flux in single cells can differ by three times. Since BOR-60 commissioning there has been large-scale irradiation testing of different fuel compositions including minor actinides.
At present, RIAR is making efforts to extend BOR-60 lifetime over the licensed period. In parallel, the MBIR reactor is being constructed. The MBIR design envisages wider experimental capabilities compared to the ones of BOR-60. It should be noted that the main parameters of MBIR related to transmutation of minor actinides are as good as BOR-60 parameters.

Country/Int. Organization:

Russian Federation
JSC "State Scientific Center - Research Institute of Atomic Reactors"

7.4 Fuel Cycle Analysis / 104

Fast Neutron Reactors, Fuel Cycles and Problem of Nuclear Non-Proliferation

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Long-term development of nuclear power assumes using energy resources of uranium-238 and thorium-232. The valuable property of fast reactors (FR) is their possibility to involve fertile isotopes uranium-238 and thorium-232 into nuclear fuel cycle (NFC). The use of regenerated products in FRs, received during reprocessing of spent fuel (SF), will provide closure of a NFC.
Currently limited cases of partial closure of NFC for thermal reactors (TR) was organized in Russia and France by reusing regenerated uranium in RBMK reactors and plutonium in PWRs in Western Europe. The full-scale closure of a NFC includes two main technologies –reprocessing of SF and fabrication of fresh fuel using regenerated nuclear materials. During these technological operations large amounts of fissile isotopes will be circulated in NFC and this point should be taken into consideration in order to provide nonproliferation regime and physical protection of nuclear materials. It is especially important for FRs and their NFC where concentration of fissile materials is several times higher as compared to TRs and their NFC.
During expansion development of NP in the world it is predicted that many countries will have intension to use FRs. The nuclear states, according to the NPT, are obliged to provide to non-nuclear states a complex of services for peaceful use of nuclear energy including the use of FRs. And the non-nuclear states are obliged to fulfill requirements of non-proliferation regime and physical protection of nuclear materials. Therefore it is expedient and very important to follow the IAEA safeguards at an early stage of development of new nuclear reactor designs and appropriate technologies of nuclear fuel cycle –so-called Safeguards by Design. At a stage of reaching commercial attractiveness of FRs in Russia the important issue will be raised in the country concerning the possibility of export FRs to other countries.
The issues of proliferation resistance of FRs and related nuclear fuel cycles due to diverse of nuclear knowledge, technologies, and nuclear materials of civilian NP for other purposes are discussed in the paper. Features of a closed NFC to maintain global non-proliferation regime in comparison with open NFC of thermal reactors are presented.
Optional versions of start FRs with the use both of plutonium and enriched uranium are considered. Features of on-site deployment of NFC infrastructure in comparison with centralized one for FRs are given from nonproliferation point of view.

Country/Int. Organization:

Russian Federation

Poster Session 2 / **105**

Calculation and Experimental Data Analysis of Neutron Spatial/Energy Distribution in the BOR-60 Blanket

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At present, a wide range of tests is performed in the BOR-60 reactor in support of reactors under operation, construction and design in Russia and worldwide. Most of the tests are performed in the reactor core regions of the peak dose accumulation rates. However, there is a high demand for irradiation testing to be performed in the BOR-60 blanket.

An important feature of any nuclear facility is neutron spatial/energy distribution in the reactor. An experimental data analysis of neutron spectra takes much effort and time. The effective volume of an irradiation rig is rather limited, which makes it difficult to install dozens of neutron activation

detectors at the expense of tested samples. Therefore, irradiation parameters are confirmed experimentally using several detectors; and spatial/energy distribution of the neutron field is obtained in calculation.

RIAR's experience in thorough calculations and experiments in support of BOR-60 operation shows good agreement between the calculated and experimental core parameters. The deviation of the calculated values from the experimental ones in the blanket is higher, and there are much less experimental data. Therefore, verification of the applied calculation codes, models and methods is a crucial relevant issue.

Country/Int. Organization:

Russian Federation

5.1 Advanced Fast Reactor Fuel Development I / 106

Fuel Cladding Chemical Interaction Tests of Irradiated Metallic Fuel

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To investigate the fuel cladding chemical interaction for the irradiated metallic fuel, high temperature heating tests were performed. The fuel rod consisting of U-10Zr-5Ce fuel with T92 cladding were irradiated in HANARO reactor. After the irradiation, the fuels was cut into cylindrical specimens, and then the top and the bottom plates of the specimens were put into contact with FC92 and HT9 plates, respectively. The specimens were exposed at high temperature in the range of 650–800 oC for one hour. Microstructural examinations were conducted by utilizing optical microscope, scanning electron microscope, and electron prove micro-analysis. Migration phenomena of U, Zr, Fe, and Cr as well as Nd lanthanide fission product were observed at a melting region. Elements distribution at the melting region demonstrates that eutectic melting occurs during high temperature experiment. The penetration depth of the eutectic melting in FC92 and HT9 were compared with that for T92 cladding.

Country/Int. Organization:

The Republic of Korea/Korea Atomic Energy Research Institute (KAERI)

5.4 Advanced Fast Reactor Cladding Development II / 107

Preliminary Inspection of Spent Fast Reactor Fuel Claddings

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Electrical potential testing which is a primary nondestructive testing method is used at JSC "INM" hot cells as an incoming inspection of spent BN-600 and BN-800 reactor fuel elements.
Electrical resistivity curves demonstrate the level of the fuel element defectiveness and help to work out a cladding dismantling plan for further post irradiation materials examination of the cladding problem areas.
Electrical potential testing through the electrical resistivity distribution profile enables immediate evaluation of cladding structural changes resulting from material swelling under irradiation.
Depending on the damage dose it is possible to evaluate and compare values of cladding swelling by resistograph images.
Theoretical dependence and experimental results showing the correlation between material radiation-induced swelling as well as cladding corrosion thinning and the change of electrical resistivity are shown.
The effect of radiation and technological defects on electrical resistivity change of the claddings made of ChS-68, EK-164, and EP-450 steels are discussed.

Country/Int. Organization:

Russian Federation

Poster Session 2 / 108

Analysis of Irradiation Ability of China Experimental Fast Reactor

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China Experimental Fast Reactor (CEFR) has completed physics start-up tests in 2010 and connected the grid on FP in 2014. Characteristic of neutron field for irradiation in CEFR has been researched by calculation and experiments. In future, CEFR will been operated as an irradiation test facility for fuel, material and other application, and some irradiation projects, such as irradiation of cladding material, MOX fuel and (U, Np)O2 pellet have been planned. Now some irradiation rigs have been developed and some irradiation experiments have been carried out in CEFR. In the future, some irradiation channels in rotation plug will be updated for strengthening the irradiation ability of CEFR.

Country/Int. Organization:

China, China Institute of Atomic Energy

Poster Session 2 / 109

Analyses of unprotected transients in GFR (ALLEGRO) and SFR reactors supporting the group constant generation methodology

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In this study, the fuel, the coolant, the cladding and the wrapper temperature reactivity coefficients were calculated with Serpent Monte Carlo code for the ALLEGRO demonstrational GFR core and for an SFR core with 3600 MWth power. The results were compared with each other and with thermal reactor reactivity coefficients, and it was found that the thermal expansion of the core structural elements has significant effect on the reactivity for fast spectrum reactors. Detailed explanation was given for the reactivity coefficients.
Additionally, the importance of the reactivity coefficients during unprotected transients were determined with thermal-hydraulics simulations using ATHLET 3.1A code. The calculations were based on the determination of evolving maximal fuel, cladding and coolant temperatures. The uncertainties were also calculated for these parameters considering the uncertainties of the reactivity coefficients. Our results can be used for further group constant parametrization.

Country/Int. Organization:

Hungary

Poster Session 1 / 110

A Preliminary Study of P&T Scenario on a Sustainable Energy System in China

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1.radioactive harmfulness of the spent fuel
2.capability of MA burning in 600 MWe FBR
3.a study of nuclear energy system scenario
4.summary

Country/Int. Organization:

P.R.China

4.3 Partitioning and Sustainability / 111

Fuel cycle studies of Generation IV fast reactors with the SITON v2.0 code and the FITXS burn-up scheme

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Due to the high computational cost of detailed burn-up calculations, most scenario codes use burn-up tables or parametrized few group cross-sections to calculate fuel depletion in reactors. As a special parametrization approach, a fast and flexible burn-up scheme called FITXS was developed at the BME Institute of Nuclear Techniques, which is based on the fitting of one-group cross-sections as polynomial functions of the detailed fuel composition. The scheme was used to develop burn-up models for the Generation IV Gas-cooled Fast Reactor (GFR), Lead-cooled Fast Reactor (LFR) and Sodium-cooled Fast Reactor (SFR), which are able to calculate spent fuel compositions with high accuracy for a wide range of initial compositions in very short computational time. The models were also integrated into the nuclear fuel cycle simulation code SITON v2.0, developed at the Centre for Energy Research, and several fuel cycle scenarios were investigated and compared with the different fast reactor models concerning the reduction of transuranium inventories and the stabilization of the plutonium inventory.

Country/Int. Organization:

Hungary

4.2 Reprocessing and Partitioning / 114

The actinide oxides preparation by thermal denitration

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Pyrochemical, hydrometallurgical and combined (pyro + hydro) technologies for reprocessing of mixed nitride uranium-plutonium fuel are under development in RUSSIA now. The main product of hydrometallurgical and combined processing is a mixed U, Pu and Np oxides.
The following technologies of actinides oxides preparation have been investigated for the choice of most promising: precipitation of oxalate, plasma and flame denitration, microwave denitration under as well as precipitation by ammonium or hydrazine. The samples of oxide types U-Th and U-Pu have been got and the following oxides properties have been identified: bulk density, specific surface, fractional composition, the chemical content uniformity and oxygen ratio. Microwave denigration was chosen for the future development on a base of previous studies results.
The preparation of actinides oxides by microwave radiation was performed in two stages. During the first stage the nitric-acid solution of actinide was evaporated to the mixture of uranium trioxide, hydrated plutonium oxide and hydrated neptunium oxide. During the second stage, the calcination of actinides was made in Ar-H2¬ atmosphere. The obtained dioxide is grinded into powder mechanically.
The full-scale pilot setup was made.

Country/Int. Organization:

Russian Federation

Poster Session 2 / 115

Scoping Analysis of STELLA-2 using MARS-LMR

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To support the development of Prototype Gen IV Sodium-cooled Fast Reactor (PGSFR), the Sodium Integral Effect Test Loop for Safety Simulation and Assessment (STELLA) program has been launched and the basic design of STELLA-2 facility was completed in 2015. The STELLA-2 is a scaled facility including all the major systems and components in PGSFR and is able to simulate the transient behavior. For the scoping analysis of STELLA-2, the representative design basis event (DBE) analysis was conducted and evaluated by using MARS-LMR code with the same assumption and approach of PGSFR. The Loss of Flow (LOF) accidents with the Loss of Offsite Power (LOOP) was the target event and the result of PGSFR and STELLA-2 were compared. In general, the flow trend well-followed the PGSFR behavior whereas the temperature trend was inconsistent with the PGSFR result. Several design issues and analysis issues were found and the solution for each problem was also suggested. After the improvement/modification of the STELLA-2 input, it was verified that the both flow and temperature trend well-follows the PGSFR transient behavior. These issues are expected to be handled in the installation and manufacture stage of STELLA-2. For further study, various sensitivity tests on key factors are planned. Furthermore, more DBEs are under consideration to be analyzed and evaluated.

Country/Int. Organization:

Korea/KAERI

6.11 IAEA Benchmark on EBR-II Shutdown Heat Removal Tests / 118

Thermal Hydraulic Investigation of EBR-II Instrumented Subassemblies during SHRT-17 and SHRT-45R Tests

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Experimental Breeder Reactor (EBR-II) was a U-Pu-Zr metal-alloy fuelled liquid-metal-cooled fast reactor, extensively used for conducting safety experiments. EBR-II was heavily instrumented to measure sodium flows and temperatures at various locations in the primary circuit including the temperature distribution inside the subassemblies (SA). Several transient tests were conducted on the reactor to improve the understanding of thermal hydraulics and neutronics of fast reactors. The shutdown heat removal tests (SHRT-17 & SHRT-45R) conducted in 1984 and 1986 demonstrated mechanisms by which fast reactors can survive severe accident initiators with no core damage. In order to utilize the data recorded during these tests and facilitate computer code validation, IAEA has initiated a coordinated Research Project (CRP) wherein 19 organizations representing eleven countries participated. Several participants simulated parts of the primary heat transport system using CFD codes. Amongst theses studies, the sub-channel/CFD analysis of the instrumented SA (XX09 & XX10) are very important. XX09 was a 61-pin (59 fueled) SA with helically wound spacer wire

over each pin and XX10 was a 19-pin non-fuelled SA without spacer wire. The instrumented SA are additionally cooled by a small amount of thimble flow around the SA. These SA were instrumented with wire wrap thermocouples, flow meters (below the core) and thermocouples at the SA inlet and outlet. Participants used sub-channel analysis codes and CFD codes for predicting the thermocouple temperatures at various locations. It is seen that the CFD studies are computationally intensive and transient studies could not be continued for long duration. The core top and SA top temperatures predicted by the sub-channel analysis codes and CFD codes are in reasonably good agreement with the measured values. The temperature distributions at the middle of the core predicted by CFD codes are in closer agreement with the measured values as compared to the predictions by sub-channel studies. The studies brought out the importance of thimble flow, inter-subassembly heat transfer, the effect of spacer wire and the power distribution inside the SA. The full length paper gives modeling details of the SA with various codes and the comparison of the results obtained.

"Track 3: Fast Reactor Safety": EBR-II Benchmarks Invited Session

Country/Int. Organization:

INDIA

Poster Session 2 / 119

Thermal design of double helium gas gap conduction test facility

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Abstract:In order to obtain the heat transfer characteristics of the helium gap in the conditions of different thickness and power line in the high temperature range, on the basis of the previous research, the original test device was improved, through the theoretical design of double helium clearance, the test device can perform experiments under high temperature conditions. Compared with the experimental results,the theoretical design values are in good agreement with the experimental results. According to the design results of the test device, the helium gap test can be carried out in a high temperature range, and the test results can provide reference for the design of the material irradiation assembly.

Country/Int. Organization:

China/China Institute of Atomic Energy

Poster Session 1 / 120

INSERTION RELIABILITY STUDIES FOR THE RBC-TYPE CONTROL RODS IN ASTRID

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This paper reports on preliminary studies performed regarding the insertion reliability of the RBC-type Control Rods designed for the ASTRID Sodium-cooled Fast Reactor. At this stage, the primary aim of the analysis is to evaluate the mechanical behaviour of RBC Control Rods under Emergency Shutdown conditions, for which reactor core structures are subjected to significant misalignments (including earthquake-imposed displacements). Using a Finite Element Model based on the Cast3M solver and developed specifically for these studies, computations are performed that allow assessing contact reactions (and the associated friction forces and contact pressures), deformations and stresses (mostly due to bending-induced deformations) which are considered for design. Based on these preliminary results, some optimisation of the Control Rod design is proposed that ensures some stable behaviour all along the rod drop, with substantial design margins.

Country/Int. Organization:

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Poster Session 2 / 122

EXPERIENCE OF COMMISSIONING OF BN-800 CORE DIAGNOSTIC SYSTEM (SDRU)

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Diagnostic system of the core of the reactor facility BN-800 (SDRU) of Unit 4 of Beloyarsk NPP is an automated system as part of automatic process control system (APCS), designed for complex control of processes taking place in the reactor in normal operating conditions and violations of normal operation, detection at an early stage of violations of normal operation and damage to the reactor core (the deformation of the core elements, sodium boiling in the fuel assembly, the melting of fuel rods in the fuel assembly, a violation of the core cooling and other anomalies).
SDRU includes an abnormal reactivity detection system (SOAR), neutron noise diagnostic system (SNSHD), core temperature control system (STKAZ), complex analysis system (SCA).
When entering SDRU into operation at different power levels, measurement channels (MC) as a part of SDRU performance is confirmed (temperature MC, neutron chambers current and current fluctuations MC, reactivity MC), the definition of neutron-physical characteristics of the reactor is carried out, and also the basic system algorithms efficiency is investigated:
 abnormal reactivity control based on continuous determination of reactivity effects due to changes in the parameters of the reactor, the fuel burn-up, moving the control rods, etc .;
 sodium current temperature values control over the heads of the fuel assembly, the inlet and outlet of the reactor core, temperature fluctuations and power distribution loops for heat exchange;
 current fluctuations neutron ionization chambers control;
 a comprehensive analysis of the diagnostic information and the formation of the early signs of normal operation violations and damage to the reactor core on the basis of the testimony of SDRU systems and operational information from higher-level process control systems (ARMS, TLS-U).
Currently SDRU system is in trial operation and in-service functions on the Unit 4 of Beloyarsk NPP, the possibility of expanding the functions of the system for the calculation of neutron-physical characteristics of the core of the reactor BN-800 is based on the software and hardware SDRU.

Country/Int. Organization:

JSC «SSC RF-IPPE», FSUE «Alexandrov Research Institute of Technology», Russian Federation

3.6 Safety Analysis / 123

Sensitivity studies of SFR unprotected transients with global neutronic feedback coefficients

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Improvements on SFR design are expected to meet the safety goals of GEN IV reactors. One main objective is to enhance the core behavior during unprotected transients to increase the level of prevention of severe accidents. However, performing a detailed safety analysis for all initiators requires many multiphysical analyses and is a rather lengthy process when designers need to assess safety trends quickly. To compare some core design options from a safety point of view, simplified modeling using the global neutronic feedback coefficients is able to estimate the core behavior during unprotected transients. The paper explains how to use these coefficients to provide some trends for these transients like loss of flow (ULOF) or loss of heat sink (ULOHS). Main parameters to optimize the inherent safety of SFR cores are discussed to show for example that the primary pumps halving time is not always the key for improving the ULOF behavior. The paper shows that the ULOF inherent behavior of a core can be driven by one single coefficient. The paper gives also some validation insights of this methodology and an analytical comparison of some French SFR cores is made.

Country/Int. Organization:

France
CEA Cadarache

Poster Session 1 / 124

Synergetic mechanism of high temperature radiation embrittlement of austenitic steels under long term neutron irradiation at high temperatures

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The present report represents the study results of mechanisms of fracture and embrittlement of austenitic steels of 18Cr-9Ni and 18Cr-10Ni-Ti grades after long term neutron irradiation at high temperatures. The effect of irradiation temperature, irradiation time and neutron dose is considered on the fracture strain and fracture mechanisms.
It has been found in the present research that long term neutron irradiation (near 120,000 h, neutron dose is 1dpa) at high temperature (near 500 degrees Celsius) results in significant decrease of the fracture strain for uniaxial tensile specimens tested at temperature above degrees Celsius. This decrease is accompanied by the transition from transcrystalline ductile fracture to intercrystalline quasi-brittle fracture.
Possible reasons of high temperature radiation embrittlement have been analyzed. It has been shown that the known mechanism of high temperature radiation embrittlement connected with accumulation and growth of helium bubbles on grain boundaries is not the only reason of this embrittlement

for the investigated steels and irradiation condition.
It has been revealed in the present study that high temperature radiation embrittlement is caused by synergetic action of two factors - helium and thermal aging. Thermal aging results in formation of various phases on grain boundaries and, hence, in decrease of grain boundary strength. Helium diffusion at high test temperatures stimulates accumulation and growth of helium bubbles on weakened grain boundaries. Thus, thermal aging promotes the helium brittleness development. However, separately neither thermal aging nor helium bubbles results in high temperature radiation embrittlement for the investigated steels and irradiation condition.

Country/Int. Organization:

Russia, Saint-Petersburg, Central Research Institute of Structural Materials "Prometey"

Poster Session 1 / 126

EVALUATION OF COBALT FREE COATINGS AS HARDFACING MATERIAL CANDIDATES IN SODIUM FAST REACTOR

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The need for materials having good tribological properties in Sodium Fast Reactors has been identified (SFR) from the first reactors operation. Where galling or adhesive wear cannot be tolerated, hardfacing alloys or galling-resistant coatings are usually applied on rubbing surfaces. The most used coating is the cobalt base alloy named Stellite because of its outstanding friction and wear behavior. Nevertheless, cobalt is an element which activates in the reactor leading to complex management of safety during reactor maintenance and decommissioning. As a consequence, a collaborative work between CEA, EDF, AREVA and French academic laboratories has been launched for selecting promising cobalt free hardfacing alloys for SFR applications.
Several nickel base alloys and aluminides have been selected from literature review then manufactured on two candidate steel grades: 9Cr ferritic-martensitic steel EM10 and 18Cr austenitic steel AISI 316L(N). Nickel base alloy coatings were deposited through Plasma Transferred Arc or Laser Cladding, and the aluminides coatings, through pack cementation or slurry. Among the numerous properties required for qualifying their use as hardfacing alloys in SFR, good corrosion behaviour and good friction and wear behaviour in sodium are essential. The results obtained on these properties are shown in this presentation.
First, the corrosion behaviour of all coatings was evaluated through exposure tests in purified sodium for 5000 h at 200 °C and 550 °C. The degradation of the surface was carefully measured thanks to several complementary analysis techniques (GD-OES, FESEM, XRD, ⋯).
Finally, the friction and wear properties of all candidates were evaluated in sodium in a newly designed facility. The influences of temperature and of oxygen content in sodium on these properties are detailed.

Country/Int. Organization:

FRANCE/CEA

5.1 Advanced Fast Reactor Fuel Development I / 128

Conceptual design of fuel and radial shielding sub-assemblies for ASTRID

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The French 600 MWe Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) project reached in 2015 the end of its Conceptual Design phase. The core design studies are being conducted by the CEA with support from AREVA and EDF. Innovative design choices for the core have been made to comply with the GEN IV reactor objectives, marking a break with the former Phénix and SuperPhénix Sodium Fast Reactors.
The CFV core of ASTRID demonstrates an intrinsically safe behavior with a negative sodium void worth achieved thanks to a new fuel sub-assembly design. This one comprises (U,Pu)O2 and UO2 axially heterogeneous fuel pins, a large cladding versus small spacer wire bundle, a sodium plenum above the fuel pins, and an upper neutron shielding with both enriched and natural boron carbide. The upper shielding also maintains a low secondary sodium activity level and is made removable on-line through the sub-assembly head for washing compatibility. Calculations have been performed to increase the stiffness of the stamped spacer pads in order to analyse its effect on the core mechanical behaviour during hypothetical radial core compaction events.
Concerning the radial shielding sub-assemblies surrounding the fuel core, heavy iterative studies were performed in order to fulfill ASTRID requirements of minimising the secondary sodium activity level and maximising the in-core life-time. Evaluated options were reflectors sub-assemblies made of steel or MgO rods, and radial neutron shielding sub-assemblies made of B4C or borated steel, with different configurations in the design and in the core layout.
This paper describes the design of the fuel and radial shielding sub-assemblies for the ASTRID CFV v4 core at the end of the Conceptual Design phase. Focus is placed on innovations and specificities in the design compared with former French SFRs.

Country/Int. Organization:

CEA
France

5.8 Structural Materials / 130

Basic principles for lifetime and structural integrity assessment of BN-600 and BN-800 fast reactors components with regard for material degradation

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The present paper overviews the basic principles of Russian Standard elaborated for justification of lifetime prolongation of BN-600 fast reactor (FR) and for justification of design lifetime of BN-800

FR. These principles are based on the analysis of the main mechanisms of material damage under service and formulation of the limit conditions for different components of FR of BN type.
Various mechanisms of material damage under service are considered. In particular, intergranular fracture is considered for a case when material is undergone mechanical loading and neutron irradiation simultaneously. Fatigue under neutron irradiation and creep is also considered. Embrittlement of a material is taken into account caused by thermal aging, neutron irradiation and swelling. Different critical events are formulated as corresponding to different damage mechanisms, and the methods for their analysis are developed. The trend curves are presented for prediction of the physical and mechanical properties of the materials used for the BN reactor components.
Limit condition of a reactor component is formulated as a loss of structural integrity or serviceability of this component that is caused by some set of the critical events for this component. The limit conditions are formulated and represented for the main components of FR of BN type.

Country/Int. Organization:

Russia, Central Research Institute of Structural Materials "Prometey", JSC "OKBM Afrikantov"

3.7 Core Disruptive Accident Prevention / 131

Minimisation of Reactivity Margin for Equilibrium Core of Liquid Metal Cooled Fast Reactors

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Maximum reactivity margin describes the scale of potential nuclear danger of reactor unit at its release and the scale of measures that must be taken to compensate by control and safety systems, to minimize the probability of such accidents and to limit the consequences. "PRORYV"project not only sets the goal of minimizing consequences but also eliminating the root cause of reactivity accident.
This paper presents the results of systematic analysis of consequences of reactivity accidents caused by an uncontrolled escape of the total accumulated reactivity in fast reactors with lead and sodium coolants. The following criteria describe minimization of reactivity margin:
• conditions for preserving the wholeness of fuel rods or reactor body;
• elimination of accidents requiring people evacuation and relocation.
In the first case it's shown that if the induced reactivity is at □eff (0.6–1 □eff) level the power of feedback is enough to avoid exceeding the safety operation limits of rod shell temperature. Therefore, this conservative criterion rules out that the initial event of reactivity implication will grow into a nuclear accident.
The second criterion is less conservative. DINAR and COREMELT computational software that have proper mathematical models has been used for analysis of accidents caused by higher reactivity implication. It's shown that there can be set less strict reactivity limits. Particularly for sodium reactor reactivity margin can be increased up to 0.75%□k/k.
Maximum reactivity margin has been analyzed for influence on requirements to CPS devices. It was shown that reactivity margin should be lowered in terms of nuclear safety especially if reactor unit power increases to 1200 MW (and higher).
Minimization of reactivity margin is possible in terms of the so-called concept of "equilibrium reactor core"(decrease of reactivity margin for burn out) and with the use of high heat transfer fuel with liquid metal layer (temperature to power phenomena). The paper specifies "equilibrium reactor core"term, analyzes implementation possibilities and "transition"problems solutions: dependency from initial isotope composition of Pu, uncertainties of fuel parameters and forecasted properties of reactor core.

Country/Int. Organization:

Russia/Private institution «Innovation and technology center for the «PRORYV» project»

8.2 Professional Development and Knowledge Management - II / 132

IAEA NAPRO Coordinated Research Project: Physical Properties of Sodium Overview of the Reference Database and Preliminary Analysis Results

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The International Atomic Energy Agency (IAEA) recently launched a coordinated research project (CRP) on "Sodium properties and safe operation of experimental facilities in support of the development and deployment of Sodium Cooled Fast Reactors - NAPRO", to be carried out in the period 2013 –2017. The first phase of the CRP is focused on the collection and assessment of sodium properties, and it will lead to a consistent property data set which will be published in the form of a handbook. This work is carried out by several participating organizations from 10 Member States through the review and evaluation of the existing available data, the identification of the data gaps and the development of recommendations for the experimental programmes to support closing these data gaps.

A specific work package (WP 1.1), under the leadership of Argonne National Laboratory, is focused on the analysis of physical properties of sodium: thermodynamic properties (including gaseous state) and transport properties. The expected outcome includes the improved understanding of the availability, accuracy and range of applications of sodium properties focused on fast reactors and other technological applications.

In this work a detailed overview of the collected references is presented, identifying the most reliable and useful sources of information and showing the depth and breadth of the collected reference database. In addition, the limited availability of recent experimental data and the lack of information on data uncertainty are discussed as well as the approach the CRP nevertheless adopted to characterize data quality. Significant findings related to WP 1.1 are also presented in this work, focusing in particular on inconsistent data sets of properties that are commonly considered well determined and known and that the CRP has instead found to be characterized by unexpected variability between different data sources.

Country/Int. Organization:

ANL, CNEA, KIT, JAEA, IGCAR, IPPE, CIAE, NRG, CEA, IAEA

3.1 Safety Program / 133

RECENT ACTIVITIES OF THE SAFETY AND OPERATION PROJECT OF THE SODIUM-COOLED FAST REACTOR IN THE GENERATION IV INTERNATIONAL FORUM

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The Generation IV (GEN-IV) international forum is a framework for international cooperation in research and development (R&D) for the next generation of nuclear energy systems. Concerning the sodium-cooled fast reactor (SFR) system, there are five cooperation projects for R&D. The SFR Safety and Operation (SO) project addresses the area of the safety technology and the reactor operation technology developments. The aim of the SO project includes (1) analyses and experiments that support establishing safety approaches and validating performance of specific safety features, (2) development and verification of computational tools and validation of models employed in safety assessment and facility licensing, and (3) acquisition of reactor operation technology, as determined largely from experience and testing in operating SFR plants. The tasks in the SO topics are categorized into the following three work packages (WP): WP-SO-1 "Methods, Models and codes"is devoted to the development of tools for the evaluation of safety, WP-SO-2 "Experimental Programs and Operational Experiences"includes the operation, maintenance and testing experiences in experimental facilities and SFRs (e.g., Monju, Phenix, BN-600 and CEFR), and WP-SO-3 "Studies of Innovative Design and Safety Systems"relates to safety technologies for GEN-IV reactors such as active and passive safety systems and other specific design features.
In this paper, recent activities in the SO project are described.

Country/Int. Organization:

France,US,Korea, Russia,China, Japan, Euratom

Poster Session 2 / 134

Results of old and program of new experiments on the small-sized fast multiplying systems with HEU / LEU fuel for receiving the benchmark data on criticality

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Benchmark criticality experiments on small-sized fast multiplying systems with HEU fuel were performed using "Giacint"critical facility of the Joint Institute for Power and Nuclear Research –Sosny of the National Academy of Sciences of Belarus. The critical assemblies'cores comprised fuel assemblies, each of which consisted from 19 fuel rods of two types and had no the clad. The first one is metallic U (90% U-235); the second one is UO2 (36% U-235). The active area length is 500 mm. The clad material is stainless steel. Three types of fuel assemblies with different content fuel rods were used. Side radial reflector: an inner layer –Be, an outer layer –stainless steel. The top and bottom axial reflectors –stainless steel. The analysis of the experimental results obtained from these benchmark experiments by developing detailed calculation models and performing simulations for the different experiments is presented. On "Giacint"critical facility are being prepared benchmark criticality experiments on multiplying systems modeling physical features of cores with LEU fuel for use in works on fast reactors with gaseous and liquid-metal coolants. Critical assemblies represent uniform hexagonal lattices of fuel assemblies, each of which consist from 7 fuel rods and do not have the clad. The fuel is UZrCN (19.75% U-235). The active fuel length is 500 mm. The clad material is stainless steel or Nb. Three types of fuel assemblies with different matrix material (air, aluminium and lead) were investigated. Side radial, top and bottom reflectors –Be (internal layer) and stainless steel (external layer). The description of construction and composition of critical assemblies, the results of calculation and the program of works on critical assemblies are presented.

Country/Int. Organization:

Republic of Belarus / Joint Institute for Power and Nuclear Research-Sosny of NAS of Belarus

5.8 Structural Materials / 135

TEM CHARACTERIZATION OF A SWELLING-RESISTANT AUSTENITIC STEEL IRRADIATED AT HIGH TEMPERATURE (>600°C) IN THE PHENIX FAST REACTOR

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In the framework of the sodium fast-reactor (SFR) project ASTRID, an optimized austenitic steel 15%Cr-15%Ni stabilized with titanium was chosen for the fuel-pin cladding. Besides irradiation swelling, one of the main issue of irradiated austenitic steels in fast reactor is the embrittlement at high temperature (>600°C).

To investigate the long-term behaviour of cladding material irradiated at temperatures greater than 600°C, fuel pins cladded with optimized 15-15Ti, were irradiated in experimental assemblies of the Phénix SFR to reach a cumulative irradiation time of 941 days (EFPD Equivalent Full Power Day).

Thin foil specimen suitable for Transmission Electron Microscope (TEM) were taken from the upper part of the fuel pin where the irradiation temperature ranges from 600 to 630°C. The final dose of the irradiated thin foil is about 40 dpa NRT.

TEM observation was carried out to characterize irradiation defects, dislocation network and precipitation to identify possible embrittlement mechanisms. Faulted Frank loops were observed and characterized. A population of nanometric bubbles was also detected in grains and grain boundaries. In addition, fine nanometric secondary precipitation of titanium carbides and its interaction with dislocation network was investigated.

A significant intergranular precipitation was observed in the specimen. Combination of EDX (Energy Dispersive X-ray Spectroscopy), EFTEM (Energy Filtered Transmission Electron Microscopy) and micro-diffraction techniques were used to identify the various types of precipitates at grain

boundaries. The role of this intergranular precipitation in the embrittlement mechanisms is discussed in the paper.

Country/Int. Organization:

France/ CEA Saclay

Poster Session 2 / 136

Uncertainty Analysis of Kinetic Parameters for Design, Operation and Safety Analysis of SFRs

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Uncertainty Analysis of Kinetic Parameters for Design, Operation and Safety Analysis of SFRs

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An OECD/NEA sub-group on Uncertainty Analysis in Modelling (UAM) for Design, Operation and Safety Analysis of Sodium-cooled Fast Reactors (SFR-UAM) has been initiated in 2015 under the NSC/WPRS/EGUAM with the objective to study the uncertainties in different stages of Sodium Fast Reactors.

Best-estimate codes and data together with an evaluation of the uncertainties are required for that purpose, which challenges existing calculation methods. Neutronic feedback coefficients as well as the kinetic parameters are being calculated for transient analyses. Experimental evidence in support of the studies is also being developed.

The use of the Iterated Fission Probability method in the Monte Carlo codes such as Tripoli4® Serpent-2 and MCNP-5 gives reference values for calculating βeff. Deterministic codes like ERANOS and PARTISN/SUSD3D are also used for nuclear data sensitivity analysis and uncertainty propagation. The derived values are validated against experiments and their uncertainties. A vast series of experiments has been selected and analysed leading to recommendations on the tools, procedures and data to be used for beta-eff calculating of the benchmarks including uncertainties.

Country/Int. Organization:

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Poster Session 1 / 138

Passive Complementary Safety Devices for ASTRID severe accident prevention

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Sodium-cooled Fast Reactors (SFR) is one of the Generation IV reactor concepts. It has been selected to secure the nuclear fuel resources and to manage radioactive waste. In this context, the CEA (French Commission for Atomic Energy and Alternative Energy) with its partners is involved in a substantial effort on the ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) Project.
ASTRID core design is mainly guided by safety objectives. The first one is prevention of the core meltdown accident, at first through natural favourable behaviour of the core and of the reactor, and with the addition of passive complementary systems if natural behaviour is not sufficient for some transient cases. The second one is the mitigation of the severe accident to guarantee that core melting accidents do not lead to significant mechanical energy release.
The robust safety demonstration is supported by complementing ASTRID core with two types of Complementary Safety Devices dedicated to core damage prevention that would passively shut down the reactor. The first type is based on the Curie point use of electromagnetic devices that hold some specific ASTRID shutdown systems to address unprotected loss of heat sink transients (ULOHS). The second type is a hydraulically suspended absorber rod subassembly, called RBH, dedicated to unprotected loss of flow (ULOF) transients; under normal operation, the absorber rod subassembly is hydraulically suspended above the core by the upward flow of the sodium coolant. Should an ULOF event and the associated drop in flow rate occur, this upward force would become insufficient, thus allowing the absorber insertion into the active core region by gravity.
This paper presents a state of the art on simulations of accidental transients using CATHARE2 system code. It has been demonstrated that these Complementary Safety Devices, for ASTRID severe accident prevention, achieve absence of sodium boiling (ULOF) and limitation of the reactor temperature (ULOHS).

Country/Int. Organization:

Commissariat à l'Énergie Atomique et aux Énergies Alternatives
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5.6 Liquid Metal Technologies / 139

Development and Demonstration of Ultrasonic Under-Sodium Viewing System for SFRs

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In a sodium-cooled fast reactor (SFR), an under-sodium viewing (USV) system will be essential for real-time monitoring of core operation and/or in-situ inspection and repair of components. The USV system must be capable of operating in the high-temperature, high radiation, and highly corrosive environment of SFRs. Argonne National Laboratory (ANL) has successfully developed ultrasonic waveguide transducers (UWTs), brush-type ultrasonic waveguide transducers (BUWTs), and submergible transducers that can be used for defect detection, component identification, loose part location, and operation monitoring in the harsh sodium environment. A USV facility was constructed for the development and in-sodium testing of instruments and nondestructive evaluation techniques that potentially could be used for SFRs. An integrated USV imaging system, including data acquisition, signal and imaging processing, and different automated scanning mechanisms, was developed for real-time and faster defect detection and visualization.

Special UWTs were designed and used as a waveguide to isolate a conventional ultrasonic transducer from the harsh core environment. The Argonne UWTs have shown a detection resolution of 0.5 mm in width and depth in water and hot oil, and have also demonstrated defect detection and component recognition capabilities in sodium at elevated temperatures up to 343℃. Prototypes of high-temperature submergible transducer were also developed and tested successfully in water, hot oil, and sodium. Different piezoelectric elements and backing materials were evaluated. The prototypes have demonstrated a detection resolution of 1 mm in width and 0.5 mm in depth in sodium at elevated temperatures up to 343℃. To reduce imaging time, we have also developed BUWTs. Multiplexing technique was tested first. The results generated from a water mockup have shown great defect detection capability. Argonne is currently integrating the BUWT and phased array (BUWT-PA) techniques for better defection resolutions and faster inspection. Feasibility study of BUWT-PA was conducted in water and preliminary results have shown that the inspection speed is 10 times faster. However, the resolutions of BUWT-PA are not as good as that of the UWTs and submergible transducers. The future USV R&D plan is presented.

Country/Int. Organization:

USA/Department of Energy - Argonne National Laboratory

1.5 LFR DESIGN & DEVELOPMENT / 140

Simplification, the atout of LFR-AS-200

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LFR-AS-200 is under development by Hydromine in cooperation with ENEA. LFR stands for Lead-cooled Fast Reactor, AS stands for Amphora-Shaped, referring to the shape of the inner vessel and 200 is the electrical power in MW. The project has been carried out by a team of engineers who had participated to the construction of SPX1.
The strengths of the LFR-AS-200 are safety, simplicity, cost-competitiveness and operational simplicity.
Safety relies on lead properties and is enhanced by innovative solutions including passively actuated

and operated decay heat removal systems and a Steam Generator (GV) featuring a spiral-tube bundle, partially raised above the cold collector free level.
The SG features a lower inlet window and an upper outlet window in correspondence of the lead free level, in order to drastically reduce the mass of displaced lead in case of SGTR.
LFR-AS-200 dispenses with several components, hitherto typical of fast reactors, up to achieving a volume of the primary system per unit power of less than 1 m3/MWe, i.e. about 4 times lower than that of the SPX1 Sodium-cooled Fast Reactor (SFR), and also several times less than other international LFR projects, a key-factor for cost competition.
The smaller size has been achieved through design simplification, that did mainly consist in the elimination, besides of the intermediate circuits (feature common to any other LFR project), of several components typical of SFRs and also of previous LFRs, namely (i) the in-vessel refueling machine, (ii) the above-core structure, (iii) the diagrid, (iv) the strongback, (v) the shielding elements, (vi) in-lead bearings of the pumps, (vii) the "LIPOSO"or equivalent tubular hydraulic connection between the pumps and the core and (viii) the "Deversoir"or equivalent system aimed at keeping the reactor vessel at the temperature of the cold collector.
Several operational benefits pertaining to the proposed LFR-AS-200 technology are the result of insightful choices, typically the adoption of Fuel Assemblies (FAs) with a stem that extends above the lead free surface, and hung by a support system which is integral part of the FA's head, i.e. located in the gas plenum and therefore visible by the operators. This keeps the support system free from neutron damage and thermal loads and strongly reduces the burden of in-service inspection of the primary system.

Country/Int. Organization:

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1.2 SFR DESIGN & DEVELOPMENT - 2 / 141

FASTER Test Reactor Preconceptual Design

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The FASTER test reactor was designed as part of the U.S. Advanced Demonstration and Test Reactor Options (ADTR) Study in 2015/2016. The ADTR study provided an assessment of advanced reactor technology options and is intended to provide a sound comparative technical context for future decisions concerning these technologies. Point designs for a select number of concepts were commissioned.

One of the two test reactor point designs was a sodium-cooled fast test reactor called FASTER. FASTER is a sodium-cooled, metal alloy fueled fast reactor with a core thermal power rating of 300MW. The FASTER plant was designed with extended testing capabilities in mind while trying to keep the reactor plant as simple as possible. The main function of the FASTER plant is to provide high neutron flux irradiation capability for both fast neutron spectrum and thermal neutron spectrum applications.

The FASTER reactor plant incorporates an innovative core arrangement that also provides for irradiation testing in closed loops with different working fluids. This paper will describe the design characteristics of the FASTER plant and provide background information on the ADTR study and its objectives.

Country/Int. Organization:

United States of America

6.3 Neutronics - 1 / 142

Benchmark Evaluation of Dounreay Prototype Fast Reactor Minor Actinide Depletion Measurements

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Historic measurements of actinide samples in the Dounreay Prototype Fast Reactor (PFR) are of interest for modern nuclear data and simulation validation. Samples of uranium, neptunium, plutonium, americium, curium, and californium isotopes were irradiated for 492 effective full-power days and radiochemically assayed at Oak Ridge National Laboratory (ORNL) and Japan Atomic Energy Agency (JAERI). Limited data were available regarding the PFR irradiation; a six-group neutron spectra was available with some power history data to support a burnup depletion analysis validation study.

Under the guidance of the Organisation for Economic Co-Operation and Development Nuclear Energy Agency (OECD NEA), the International Reactor Physics Experiment Evaluation Project (IRPhEP) and Spent Fuel Isotopic Composition (SFCOMPO) Project are collaborating to recover all measurement data pertaining to these measurements, including collaboration with the United Kingdom to obtain pertinent reactor physics design and operational history data.

These activities will produce internationally peer-reviewed benchmark data to support validation of minor actinide cross section data and modern neutronic simulation of fast reactors with accompanying fuel cycle activities such as transportation, recycling, storage, and criticality safety.

Country/Int. Organization:

USA/INL, USA/ORNL, France/OECD-NEA, Japan/JAEA

Poster Session 2 / 143

Overview of Experiments for Physics of Fast Reactors from the International Handbooks of Evaluated Criticality Safety Benchmark Experiments and Evaluated Reactor Physics Benchmark Experiments

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Specialists involved in the process of validation and verification of codes and cross sections for the physics of fast reactors traditionally used the benchmarks presented in the "Cross Section Evaluation Working Group Benchmark Specifications"BNL-19302 (ENDF-202) handbook first issued in 1974 and last updated in 1991. This handbook presents simplified homogeneous models of experiments with appropriate corrections of the experimental data. This approach was relevant to the codes and computational possibilities existed during the design of the first generations of fast reactors.

The Nuclear Energy Agency (NEA) of the Organisation for Economic Cooperation and Development (OECD) coordinates the activities of two international projects on the collection, evaluation and documentation of experimental data - the International Criticality Safety Benchmark Evaluation Project (ICSBEP) (since 1995) and the International Reactor Physics Experiment Evaluation Project (IRPhEP) (since 2003). The result of the activities of these projects are, every year updated, the International Handbooks of critical (ICSBEP Handbook) and reactor physics (IRPhEP Handbook) benchmark experiments. The handbooks present detailed models of experiments with minimal corrections and comprehensive evaluation of their uncertainties. Such models are of particular interest in terms of implementation of possibilities of the modern calculational codes and systems of automated prediction of the uncertainties of the design parameters and margins.

The Handbooks contain a large number of experiments which are suitable for the study of physics of fast reactors. Many of these experiments were performed at specialized critical facilities, such as BFS (Russia), ZPR and ZPPR (USA), ZEBRA (UK) or the experimental reactors - JOYO (Japan), FFTF (USA). Other experiments, such as compact metal assemblies, are also of interest in terms of the physics of fast reactors, were performed at the multipurpose critical facilities in Russia (VNIITF and VNIIEF) and the US (LANL, LLNL, and others.). Also worth mentioning is the critical experiments with fast reactor fuel rods in water, interesting in terms of verification of criticality safety during transportation and storage of fresh and spent fuel.

This report provides a detailed overview of the mentioned experiments, designates the areas of their applications and includes the results of calculations with modern cross sections in comparison with the evaluated benchmark data.

Country/Int. Organization:

USA/INL, France/OECD-NEA, Russia/IPPE

144

Neutron Thermalization in the FASt TEst Reactor

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Over the past couple years Argonne National Laboratory has been developing a FASt TEst Reactor as part of the DOE-NE Advanced Reactor Campaign in order to address the growing needs for fast neutron irradiation capabilities. A unique feature of FASTER is that it not only offers very high fast fluxes and large irradiation volumes, but it can also offer very high thermal fluxes by making use of the large neutron leakage probability.

The thermalization of leaking neutrons to achieve a high thermal flux is discussed in this paper. In order to meet this objective, a number of materials could be used. The performance characteristics, relative to the thermal flux level, are compared for the various thermalizing materials envisioned. Approaches used to mitigate the risk for a large peaking power to occur when thermal neutrons are re-entering the fuel region are also discussed in this paper.

Country/Int. Organization:

USA / Argonne National Laboratory

145

A Versatile Coupled Test Reactor Concept

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A Versatile Coupled Test Reactor (VCTR) is a highly reconfigurable, sodium-cooled, coupled fast and thermal spectrum test reactor, which provides (1) a fast flux irradiation environment prototypical of potential fast reactor designs, (2) thermal and epithermal flux irradiation environments complementary to those of ATR and HFIR, and (3) other possibilities, including beam tubes for scientific experiments, irradiation vehicles for isotope production, etc. The ongoing design of a new versatile coupled thermal-fast test reactor at INL that can accommodate both fast and thermal irradiation, technology and safety features tests is described in this paper.

A coupled reactor is defined as a reactor with two distinct spectral zones (Fast and Thermal), which are neutronically coupled to each other: some neutrons born in fast zone cause fission in thermal zone and vice-versa. Only fast neutrons are allowed to go from one zone to the other, which is assured by a thermal neutron filter. A distinction of coupling can be made depending on the main purpose of using fuel assemblies with some level of neutron moderation (thermal zone) together with fuel assemblies with as little moderation as possible (fast zone). It can be used mainly to minimize the core power and fissile inventory, or it can be used mainly to provide additional control of the irradiation conditions in the thermal zone.

The steady-state neutronics calculations were performed using the three-dimensional continuous-energy Monte Carlo reactor physics burnup calculation code Serpent2. Although, a few changes were necessary to support the Serpent code in calculating the coupling coefficients. Steady state and transient thermal analyses were performed with RELAP5-3D.

One of the core configurations using only low enriched uranium fuel could provide both a high fast neutron flux and a high thermal neutron flux—respectively 3.5E15 n/s.cm2 and 1E15 n/s.cm2 —in large volumes while maintaining core power below 300 MW. The use of plutonium in the fast zone provides more flexibility in fuel assembly design than when low enriched uranium is used, the designer has more degrees of freedom to work with; in particular it could provide additional control of the irradiation conditions in the thermal zone.

The results showed that the current design fulfills most R&D needs requirements providing high thermal and fast fluxes and wide possibilities for diversified future reactor features test in appropriate loops, both in steady state and during transients.

Country/Int. Organization:

USA/Idaho National Laboratory

Poster Session 1 / 147

Modeling technologies of fuel cycles

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There exist different variants of organizing the closure of nuclear fuel cycle (CNFC) depending on fast reactor type, fuel types, station or centralized allocation of closed nuclear fuel cycle stages. Many processes and engineering solutions used for implementation of chosen technologies for reprocessing spent fuel are little-studied. One of the ways to verify and estimate engineering solution is mathematical modeling of radiochemical technology which in the end will allow to optimize composite technological process in order to increase effectiveness and reduce cost.
The mathematical models for key processes of spent fuel reprocessing, fuel refabrication and radioactive waste managing are being developed in the frames of "Proryv"project for these purposes. Also codes VIZART and KOD TP to validate realizability and optimize parameters of CNFC processing lines are being developed. The codes use integrated library of technology models and allow to calculate material balance, create cyclograms, determine the most loaded parts of processing lines, estimate accumulation of fissile materials in devices and intermediate vessels, estimate the influence of control actions on technology process.

Country/Int. Organization:

Russian Federation, 1) RFNC-VNIITF,
2) ITCP «PRORYV»

Poster Session 2 / 149

Evaluation of the OECD/NEA/SFR-UAM Neutronics Reactivity Feedback and Uncertainty Benchmarks

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One of the tasks of the OECD/NEA sub-group on Uncertainty Analysis in Modelling (UAM) of Sodium-cooled Fast Reactors (SFR-UAM) under the NSC/WPRS/EGUAM is to perform a code-to-code comparison on neutronic feedback coefficients and associated uncertainties calculated for transient analyses. This benchmark exercise benefits from the results of a previous Sodium Fast Reactor core Feed-back and Transient response (SFR-FT) Task Force work under the NSC/WPRS/EGRPANS. Two SFR cores have been selected for the SFR-UAM benchmark, the 3600MWth oxide and the 1000MWth metallic SFR cores.

Results from four and seven participating international institutes were received for respectively, the metallic and oxide SFR cores, using a wide range of calculation methodologies. The preliminary results display good agreement in the reactivity coefficients estimated, with remaining discrepancies explained by different nuclear data libraries, modeling approximations for deterministic solutions, and statistical convergence for stochastic evaluations on small perturbations. Nuclear data uncertainty evaluations on the reactivity coefficients are compared and display consistent results.

Country/Int. Organization:

USA/ Argonne National Laboratory

Poster Session 1 / 150

Assessment of the reactivity effects of Gas cooled Fast Reactor

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Presented paper assess standard reactivity effects, as coolant void effect and Doppler effect, of the power scale Gas cooled Fast Reactor (GFR 2400) in a comprehensive manner by application of the perturbation theory. To achieve high validity of the results the conventional SCALE 6 system and adapted computational scheme (ACS) are utilized. The ACS is based on standard computational package incorporating codes like TRANSX, PARTISN, DIF3D, PORK and STUUP with cross section data library optimized for fast reactor applications. The reactivity effects of the GFR 2400 core were calculated in a range of the pre-defined temperatures and coolant pressures. Coolant void effect for nominal and lowest operational pressure and Doppler effects for highest temperature increase and decrease were identified as most important reactivity effects. Spatial distribution and reactivity components decomposition of the selected reactivity effects is analyzed and presented in this paper for further evaluation. In the next step, sensitivity and uncertainty analysis is performed for these reactivity effects where the sensitivity coefficients are validated via direct perturbation calculation and energy profile comparison. In case of possible optimization of selected reactivity effects the most sensitive isotopes and contributors to the overall uncertainty are identified. The final part of the paper is dedicated to the first optimization studies and preliminary results are presented. Two possible options of optimizations are proposed; homogeneous and heterogeneous. In the heterogeneous case the rod follower volume is used for application of materials which can possibly influence the reactivity effects. In the second case the core design modifications are homogeneously distributed over the entire core volume. Finally in the conclusion recommendations and some drawbacks are collected for further analyses.

Country/Int. Organization:

Slovakia/Slovak University of Technology in Bratislava, Institute of Nuclear and Physical Engineering

Poster Session 1 / 152

Tradeoff Study of Advanced Transmutation Fuels in Sodium-cooled Fast Reactors

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Advanced transmutation fuels are being developed for Sodium-cooled Fast Reactors (SFRs) with reduced chemical interaction between the fuel and cladding and higher burnup achievable. Different diluent materials such as Zirconium, MTZ (5Mo-4.3Ti-0.7Zr), with addition of Palladium are considered, together with Cr-coating inside the cladding. The use of advanced transmutation fuels was assessed in this study based on the ABR-1000 concept.

This study confirms the significant impact of using advanced transmutation fuels on the reactor physics parameters due to the reduction in heavy nuclei density and to the addition of more absorbing elements. The addition of Palladium leads to increase in the fissile content while decreasing the conversion ratio. The reduced neutron flux in the low-energy range affects the neutronic feedback coefficients by reducing the Doppler effect and increasing the sodium void worth. Using MTZ diluent is also found to affect the reactor physics parameters by requiring higher fissile content, and decreasing the conversion ratio. In this case, however, no significant changes in the feedback coefficients are found despite the large shift in spectrum observed, and caused by the elastic scattering cross-section of Ti-48. A 20 µm coating of Chromium had a minor effect on the reactor physics performance of the SFR.

Country/Int. Organization:

USA/Argonne National Laboratory

5.3 Advanced Fast Reactor Cladding Development I / 153

FRACTURE STRAIN AND FRACTURE TOUGHNESS PREDICTION FOR IRRADIATED AUSTENITIC STEELS OVER WIDE RANGE OF TEMPERATURES TAKING INTO ACCOUNT THE EFFECT OF SWELLING AND THERMAL AGEING

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Fracture toughness and fracture strain of austenitic steels are the important performance properties, which control serviceability of the components of fast reactors including fuel assemblies. It is known that the above properties decrease under irradiation and thermal ageing. Especially strong decrease of fracture toughness occurs under irradiation accompanied by swelling. It is necessary to note that ductile transgranular fracture mechanism dominates at temperature less than 500 degrees Celsius even for highly embrittled material with high swelling. Such type of fracture can be predicted by the ductile fracture model proposed by authors early.
At the temperature higher than 500 degrees Celsius fracture of irradiated austenitic steels occurs by intergranular mode. Such embrittlement is known as high temperature radiation embrittlement (HTRE). There are only few experimental data on HTRE, and models are practically absent for prediction of fracture toughness and ductility for different levels of stress triaxiality.
The present work considers the features of prediction of fracture strain and fracture toughness for irradiated austenitic steels over wide range of temperatures with regard for swelling and thermal ageing. Model is developed for prediction of both quasi-brittle intergranular and ductile transgranular fracture and the fracture mechanisms transition. The model allows one to predict fracture strain and fracture toughness of material for different stress triaxiality taking into account the influence of neutron irradiation, swelling and grain boundary damage by He. The data are represented for fracture modeling of the material of decommissioned fuel assemblies.

Country/Int. Organization:

Russia/ Central Research Institute of Structural Materials "Prometey",
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Poster Session 1 / 155

High temperature design and evaluation of forced draft sodium-to-air heat exchanger in PGSFR

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In PGSFR (prototype-gen IV sodium-cooled fast reactor), two kinds of DHRSs (decay heat removal systems) are employed for emergency decay heat removal during a loss of normal heat sink accident, which are ADHRS (active decay heat removal system) and PDHRS (the passive decay heat removal system). The ADHRS is a safety-grade active system, which is comprised of two independent loops with a single sodium-to-sodium decay heat exchanger (DHX) immersed in cold pool region and a single forced-draft sodium-to-air heat exchanger (FHX) located in upper region of the reactor building. The total heat removal capacity of the DHRS is 10 MWt which amounts to about 2.5% of the rated core thermal power. The DHRS is capable of cooling the plant from an initial temperature corresponding to any power operation condition to the safe shutdown condition within 72 hours after reactor shutdown with a single failure. The FHX employed in the ADHRS is a shell-and-tube type counter-current flow heat exchanger with M-shaped finned-tube arrangement. Liquid sodium flows inside the heat transfer tubes and atmospheric air flows over the finned tubes. During normal plant operation, small amount of heat loss through the FHX is permitted to prevent potential flow reversal or stagnation in each decay heat removal sodium loop. After the reactor shutdown, heat removal rate increases by opening dampers located in air flow paths of FHXs, and then the heat transferred to the decay heat removal system is finally dissipated into the atmosphere. In this study, high temperature design and creep-fatigue damage evaluation for a FHX were conducted. A creep-fatigue damage evaluation was performed according to the elevated temperature design codes of ASME B&PV Section III Division 5 based on a full 3D finite element analysis. The integrity of the heat exchangers under creep-fatigue loading was confirmed.

Country/Int. Organization:

Republic of Korea/Korea Atimic Energy Research Institute

1.1 SFR DESIGN & DEVELOPMENT - 1 / 156

Current status of GIF collaborations on sodium-cooled fast reactor system

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The SFR system arrangement Phase II became effective on 16 February 2016 by signatures of CEA, JAEA, KAERI, USDOE, and Rosatom), and was extended for additional 10 years. China signed the SFR SA Phase II on 3th August 2016 and Euratom is expected to sign near future. Collaboration of GIF SFR is growing adding new reactor concepts and related R&Ds.
In 2015, a project arrangement on SFR System Integration and Arrangement (SI\&A) has been signed by 7 members : China, EU, France, Japan, Korea, Russia and US. In the SI\&A project, R\&D needs from the SFR design will be shown to the R&D project, and R&D results from each R&D project will be integrated into the designs. Presently there are four SFR design concepts as shown in ATFR-2015, 1) JAEA Sodium Fast Reactor (JSFR, loop) Design Track, 2) KALIMER-600 (KAERI, pool) Design Track, 3) European Sodium Fast Reactor (ESFR, pool) Design Track, 4) AFR-100 (DOE, modular) Design Track, are proposed from each signatory. China is going to propose CFR-1200, and Russia is going to propose BN-1200 as new design tracks.
After SI\&A project started, the GIF-SFR has completed the function of R\&Ds and integration of R&D result to the SFR design. These strong collaboration network is expected to provide the promising generation IV SFR concepts. This paper describes SFR design concepts in SFR project and interactions with R\&D projects under GIF framework.

Country/Int. Organization:

GIF-SFR-SSC

Poster Session 2 / 157

Computational Analysis Code Development for core and primary system thermal hydraulic design of SFR

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This study developed a thermal hydraulic design code based on powerful capabilities of computer hardware and advanced numerical simulation technology for the optimization design of reactor core assemblies, big and small grid plates and the primary circuit fluid network, for the use of core assemblies thermal hydraulic design, special design of flow distribution in the big and small grid plates, as well as the design of complex throttle network in the primary circuit of fast reactors, providing a necessary tool for engineering design while supporting safety analysis of the reactor core and the primary circuit.

Country/Int. Organization:

China/China Institute of Atomic Energy

1.1 SFR DESIGN & DEVELOPMENT - 1 / 158

Advanced sodium-cooled fast reactor development regarding GIF safety design criteria

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Design studies on a next generation sodium-cooled fast reactor (SFR) considering the safety design criteria (SDC) developed in the generation IV international forum (GIF) was summarized. To meet SDC including the lessons learned from the TEPCO's Fukushima Dai-ichi nuclear power plants accident, the heat removal function was enhanced to avoid loss of the function even if any internal events exceeding design basis or severe external event happen. Several design options have been investigated and auxiliary core cooling system using air as ultimate heat sink has been selected as an additional cooling system regarding system reliability and diversification. Even though the next generation SFR already adopts seismic isolation system, main component designs have been improved considering revised earthquake conditions. For other external events, design measures for various external events are taken into account. Reactor building design has been improved and important safety components are diversified and located separately improving independency. Those design studies and evaluations on the next generation sodium-cooled reactor have contributed to the development of safety design guidelines (GIF) which is under discussion in the GIF framework.

Country/Int. Organization:

Japan

Poster Session 2 / 159

Analysis of the EBR-II SHRT-45R neutronics benchmark with ERANOS-2.0

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A Coordinated Research Project (CRP), initiated by the International Atomic Energy Agency (IAEA), consisted of benchmark calculations with the goal to analyze two of the Shutdown Heat Removal Tests performed in EBR-II, namely SHRT-17 and SHRT-45R. Test SHRT-45R concerned an Unprotected Loss Of Flow (UOLF) scenario. In this case, only the inherent feedback mechanisms due to the change of the temperatures are available to reduce the power of the reactor during the transient. The SHRT-45R benchmark thus included a neutronics benchmark, the goal of which was to establish the neutronic feedback coefficients to be used in the transient calculations. We used ERANOS-2.0 to calculate the feedback coefficients for SHRT-45R.

Since the standard cross section library of ERANOS-2.0 is outdated, we used our in-house cross section processing software to generate cross section libraries for SHRT-45R, based on JENDL-4.0, in 1968 and 33 energy groups. Our processing software consists of a script, coupling the various calculation steps to generate an ECCO library. Since the SHRT-45R also specified some special materials (e.g. "fissium" as well as lumped fission products), the necessary cross sections were determined with ERANOS.

The SHR-45R benchmark specifies 3 unique mixtures for each fuel and blanket assembly, and since there are about 400 assemblies in total, there more than 1200 unique mixtures to take into account. Since each mixture is "user defined" in this case, some custom routines were added to the ERANOS software to create and manage the necessary data sets (EDLs).

Core calculations were done with diffusion theory and transport theory and generally gave good results for the major parameters, i.e., the multiplication factor, Doppler feedback effect, as well as the effects of radial and axial expansion.

The power distribution showed large deviations in non-fuel assemblies compared to the benchmark values. It is generally believed that this is due to the treatment of the gamma ray transport.

Some weak points of ERANOS-2.0 were highlighted: the management of a large number of user-defined mixtures, the determination of uncertainty due to delayed neutron parameters, the inconsistent treatment of gamma transport, and issues related to the treatment of thermal expansion.

Overall, the benchmark analysis showed a generally acceptable performance of our cross section processing and core analysis methods.

Country/Int. Organization:

Japan

Poster Session 1 / 161

DECAY HEAT REMOVAL SYSTEM IN THE SECONDARY CIRCUIT OF THE SODIUM-COOLED FAST REACTOR AND EVALUATION OF ITS CAPACITY

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Decay heat removal system (DHRS) option for the secondary circuit of the sodium-cooled fast reactor (SFR) by means of air cooling the outer surface of piping and equipment of heat removal loops of the SFR secondary circuit is proposed.
The DHRS option under consideration implies case mounted around main piping and equipment of heat removal loops of the SFR secondary circuit and divided into a number of sections connected in parallel to each other with an exhaust chimney. This case performs certain containment function under normal operation condition, and it is arranged a natural circulation of air through the gap between piping and equipment of the secondary circuit and this case under emergency cooling modes by opening the air dampers.
Effectiveness of this decay heat removal system is evaluated by using specially developed computational code that allows modeling transient emergency cooling modes and optimization of the DHRS characteristics to reduce the maximum value of coolant temperature in these transients. Results of effectiveness evaluation for the proposed decay heat removal system applied to fast reactor with sodium coolant are presented.

Country/Int. Organization:

Russian Federation

Poster Session 1 / 162

Structural Design and Evaluation of a Steam Generator in PGSFR

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A once-through steam generator(SG) in SFR(Sodium-cooled Fast Reactor) coverts the sub-cooled feedwater to superheated steam by transferring heat from the IHTS(Intermediate Heat Transport System) sodium to water/steam and provides superheated steam during normal power condition. It is a heat exchanger as well as structural barrier between liquid sodium and water/steam and thus is regarded as one of the most critical components in SFR deciding the plant reliability and availability. The PGSFR(Prototype Gen-IV SFR) employs a vertical once-through shell-and-tube heat exchanger which has a sodium-to-water counterflow with single-walled straight heat transfer tubes. A PGSFR SG is designed to have 196 MWth power capacity and generates steam at 16.7MPa and 503℃. Its construction material is 9Cr-1Mo-V for high heat transfer performance and allowable strength at high temperature region. It uses the straight tube so that it is possible to apply single piece tube without any weld for tube-to-tube joint. Since straight tubes do not provide enough flexibility to accommodate the longitudinal expansion difference between tubes, flow distributors are applied for uniform flow at inlet shell. And, expansion bellows joint was joined on the main shell to provide a large flexibility to compensate for differential thermal expansion between shell and tube bundle.
In this study, structural design for a PGSFR SG was described and structural integrities against representative operating duty cycle event were evaluated by ASME B&PV Section III Division 5. From the evaluation results, the structural design issues were found and structural applicability with design improvement was ensured.

Country/Int. Organization:

South korea/

5.6 Liquid Metal Technologies / 163

Mass Transfer Simulation Model for Justification Sodium Purification System Characteristics

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Provision of the safe and reliable operation of Nuclear Power Plant with sodium fast reactor requires maintaining of the quality of the sodium coolant in regards of oxygen and hydrogen contents at the specific level, since even a few parts per million of sodium oxide can have a deleterious effect on the performance and reliability of a fast reactor cooled by sodium and its heat transfer system.

The main component of sodium coolant purification system is a cold trap. In a cold trap the sodium is cooled and impurities present in amounts exceeding those which will dissolve at the lower temperature precipitate. Cold traps are designed to promote controlled precipitation and to retain the precipitate in the trap.

Justification of cold trap operational parameters at the existing sodium fast reactors was performed 30-40 years ago based on experiments and semi-empirical calculations. Today's simulation technologies can be used to perform detailed computer analysis of cold trap performance taking in account accumulated experimental experience. But due to complexity of the heat and mass transfer processes effecting on each other and occurring in the cold trap simultaneously, there is a need to upgrade and validate standard calculation models incorporated in the existing codes.
The method of consecutive heat and masstransfer simulation using custom OpenFOAM solver to determine the retain coefficient of the cold trap mock-up is discussed in this work. The new solver has

been developed, which calculates fields of dissolved oxygen and suspended sodium oxide particles and the rate of their accumulation in the cold trap as well.

Calculations were performed in two steps: first, using standard thermal-hydraulic OpenFOAM solver and modified scalartTransportFoam solver at the second step, setting results of the first step of simulation as initial conditions.

Formation of particulate in the cold trap volume was accounted by introducing the oversaturation mechanism in to the solver. Saturation concentration usually defined as temperature function, which derived from experiment data approximation. Temperature dependence of the diffusion coefficient was incorporated in the solver source code in order to get more accurate results.

In the result of performed simulations the distributions of velocity, pressure, temperature and concentration of both dissolved and particulate phases inside the cold trap were obtained as well as the rate of impurity accumulation. It has been demonstrated that suggested methodology allows one to analyze operational parameters of a purification systems of sodium facilities.

Country/Int. Organization:

Russian Federation

3.1 Safety Program / 164

Study on Safety Design Concept for future Sodium-cooled Fast Reactors in Japan

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This paper describes safety design concept for future sodium-cooled fast reactors (SFRs) in Japan, which is based on the safety design criteria and safety design guidelines under development in the international forum of generation IV nuclear energy systems. The future safety design of SFRs should be advanced taking the feedback of experiences, achievement of existing technology, and innovative technology into account. Inherent and/or passive design features are utilized based on SFRs characteristics such as low pressure, high thermal inertia of the system. Lesson learned from the Fukushima Dai-ichi accident is one of important issue to be incorporated into the safety design concept. In order to realize commercial SFRs in the future, robust and rational safety design should be pursued by integration of various factors in the design, limiting additional specific systems, structures and components. Existing engineering bases for design and manufacturing of SFRs components, and innovative technologies introduced in the FaCT project are keys to realize the safety concept.

Country/Int. Organization:

JAPAN

Poster Session 2 / 165

Using of computer code GEFEST800 at the initial stage of NPP operation with BN-800

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The GEFEST800 code has been developed to carry out neutronic calculations of nu-clear power plant operation for sodium cooled fast breeder reactor BN-800 (stationary and transient from minimum controllable power level to full reactor power with drive control rods and fuel burning). The code allows to calculate the following parameters: keff; maximum reac-tivity reserve; effective reactivity of control rods and control rod groups; full, specific and lin-ear power of energy release in fuel assemblies; unevenness coefficients of energy release in fuel assemblies and reactor core; damaging irradiation dose for fuel assemblies; burning; reac-tivity coefficients; effective fraction of delayed neutrons; transient processes characteristics; decay energy release and many other parameters.
The code has diffusion, transport and Monte-Carlo modules. The CONSYST code with ABBN-93 library is used for constants' preparation. The code has thermomechanical module to take into account changes in the size of the cells. The presence of such module al-lows to consider these changes in calculations of reactivity coefficients at different power lev-els. Services such as calculated parameters control, graphics, data preparation, analysis of cal-culated results are provided in interactive mode using specially developed graphical shell.
Some results of using the code at the initial stage of NPP operation with BN-800 are presented in the paper.

Country/Int. Organization:

RUSSIAN FEDERATION

Poster Session 1 / 166

APPLICATION OF PHYSICAL MODELING WHEN CALIBRATING HIGH RANGE ELECTROMAGNETIC FLOWMETERS

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At high flowrates the «drift» of magnetic field characterized by the criterion Rem = μ0σvD can be revealed in the readings of the electromagnetic flowmeters if it's magnetic field is insufficiently extensive. The modeling of high range flowmeters presented involves using a small on scale transducer sample as compared to full scale flowmeter maintaining similarity of measuring section and magnetic field distribution. The hydrodynamic and MHD criteria (Re,Rem), corresponding to full scale flow conditions can be provided at much lower flowrate values. The experiments were carried out at sodium calibration test facility IRS-M, using its main loop and two parallel auxiliary loops supplied by calibrated electromagnetic flowmeters. On the model of a measuring section with the pipe DN150 at a total flowrate G=360 m3/h the value Rem=7,5 has been achieved, that corresponds to parameter of the flowmeter installed in accident heat removal system of BN-800 reactor (Gmax = 720 m3/h, DN300). Calibration characteristics have been determined for different electrode pairs the longitudinal extension of magnetic field being Lm=0,7DN. An estimate of the nonlinearity introduced by the quadratic term in the dimensionless representation is obtained, E=k0(1 - αRem)Rem. The coefficient α can be used further to adjust the characteristic of the full scale flowmeter.

Country/Int. Organization:

Russian Federation, JSC "SSC RF –IPPE"

2.2 Commissioning and Operating Experience of Fast Reactors II / 167

Proposal of Basic Principles of Maintenance Management for Prototype Reactors

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Basic principles of maintenance management for prototype reactors are proposed in this paper. One of main missions of prototype reactors is R&D for commercializing advanced reactors, which shall be appropriately considered in maintenance management for prototype reactors. Development of maintenance programs suitable to a reactor type is one of key features of the proposed basic principles of maintenance management for prototype reactors. It is important to identify risks specific to the reactor type, and maintenance grade of structures, systems and components should be determined considering the risks by applying the graded approach. Degradation mechanisms specific to the reactor type shall be also taken into account in the maintenance program. Progressive development of maintenance programs by accumulation of operation experiences is another key feature of the proposal. Maintenance programs have to be modified and improved frequently by reviewing results and knowledge obtained during operations. The graded approach will be useful to control risks corresponding to revisions of maintenance programs. Standardization is one of effective ways to utilize operation experiences for maintenance of prototype reactors and also development of commercialized reactors.

Country/Int. Organization:

Japan

Poster Session 2 / 168

More precise definitions of the perturbation theory formulas for reactivity effects calculations

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More precise definitions of the perturbation theory formulas
for reactivity effects calculations

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Abstract

The paper is devoted to the issues of development and modification of methods of e

There have not any references to this effect in the publications available. Howeve

Country/Int. Organization:

Russia, Obninsk, IPPE

Poster Session 1 / 169

Extending the grid plate life - Incorporation of lower axial shield for FBTR

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Operational life of Fast Breeder Test Reactor (FBTR) is limited by the grid plate life. An irradiation experiment was carried out in FBTR to determine changes in the mechanical properties of specimens of grid plate material at the desired low fluence irradiation conditions. Based on the analyses of these experiments and flux measurements at the grid plate location, the residual life of FBTR was estimated to be 6.52 EFPY at the end of 18th campaign. Possibility of reducing the neutron damage by including lower axial shields has been considered. Neutronics studies on the effectiveness of materials such as tungsten, tungsten carbide, boron carbide and ferro-boron have been conducted. A suitable arrangement of enriched boron carbide and stainless steel has been analyzed too. Based on these studies, tungsten carbide emerges as the best option. Chemical and metallurgical studies indicate that the material is compatible with sodium, has good thermo-physical properties and, hence suitable for introducing in the FBTR core as the lower axial shield. On implementation, it is expected that the life of FBTR would be increased by 35% of its remnant life.

Country/Int. Organization:

Reactor Facilities Group, Indira Gandhi Centre for Atomic Research, Kalpakkam, India

Poster Session 1 / 171

The behavior features of fuel elements with nitride fuel - theory and experiment

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The atomistic models of the swelling, creep, gas release, and nitride fuel decomposition are presented.
The models were verified on the results of nitride fuel elements irradiation in the Russian fast reactors BR-10, BOR-60 and BN-600.
At temperatures <1150 ℃ nitride fuel operate under athermal radiation processes that provide the best performance of the fuel element.
At high temperatures, a thermally activated processes leads to accelerated swelling, which is not compensated by the increased fission gas release.
Features of uranium mononitride define sensitivity of irradiation behavior to technology impurities - oxygen and carbon, including the transition temperature from athermal to thermally activated processes.
The calculations were done by the code NMUP-F (Nitride Mixed Uranium Plutonium Fuel), which combines the above mentioned atomistic models of radiation damage in the nitride fuel. Calculation of the cladding irradiation behavior was carried out by codes EDPA (Effective Displacement Per Atom) and VACS (Vacancy Activated Condensed System).
As a critical stage of fuel operation, the mechanical interaction with the cladding was considered. Under the operating conditions in developed fast reactors initial porosity of nitride fuel does not fully compensates swelling even at high temperatures.
The forecast of the marginal operating parameters of nitride fuel is done, depending on the design features of the fuel elements, the irradiation conditions, the cladding material, the structural parameters and content of impurities in fuel.
Based on calculations for fast reactor BN-1200 and BREST proposed the following design and materials for the "ideal" nitride fuel element of container type.
Fuel: density of pellet >95% of theoretical density, the oxygen content <1000 ppm and the carbon above stoichiometry (C + N) / (U + Pu) = 1 also less than 1000 ppm.
Cladding: ferritic-martensitic steel having tempered martensite structure with interlamellar distance of 0.1-0.5 microns.
Fuel element: diameter of 9-10 mm, the ratio of cladding thickness to diameter about of 1:20, the radial helium gap between the fuel and cladding approximately 0.1 mm. The ratio of free volume in the fuel element to the fuel volume is approximately 1: 1.
For the average fuel rod linear power of 40 kW/m (maximum 46 kW/m) this design provides burnup of nitride fuel up to 10 at.% (the equivalent burnup for oxide fuel about of 14 at. %) at a stress in the cladding less than 20 MPa.

Country/Int. Organization:

Russian Federation
National Research Nuclear University "MEPhI"

3.2 Core Disruptive Accident / 172

An assessment of transient over-power accident in the PGSFR

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KAERI (Korea Atomic Energy Research Institute) has been developing a preliminary specific design of the PGSFR (Prototype Gen-IV Sodium-cooled Fast Reactor), which is a pool-type sodium cooled fast reactor with a thermal power of 392.2 MW. The PGSFR has an inherent safety characteristic owing to the design to have a negative power reactivity coefficient during all operation modes and it has a passive safety characteristic due to the design of a passive decay heat removal circuit.
For an evaluation of the safety features of the PGSFR, a sensitivity analysis has been performed for TOP(Transient Over-Power) which is one of most important DBEs in the PGSFR using MARS-LMR code. MARS-LMR contains the sodium property table including dynamic properties, heat transfer correlations for the liquid metal, and the models describing the flow resistance by wire-wrap spacer in the core, which 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 a sensitivity analysis, some design variables are applied to be conservative. An effect of uncertainties is evaluated on a Doppler reactivity and a sodium density. Conservative assumptions are applied to the analysis of the plant responses during the postulated DBAs, which are 102 % of power condition with ANS-79 decay power model, 5.0 seconds delay in opening of AHX and FHX dampers, and loss of off-site power (LOOP) is taken into account. Additionally, one PDHRS and one ADHRS are available in accordance with a single failure criterion and maintenance.
As a result, the preliminary specific design PGSFR meets safety acceptance criteria with a sufficient margin during the TOP event and keep accidents from deteriorating into more severe accidents.

Country/Int. Organization:

Republic of Korea/Korea Atomic Energy Research Institute

5.1 Advanced Fast Reactor Fuel Development I / 174

Development of Electromagnetic Devices for Sodium Cooled Fast Reactor Application

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Liquid sodium is used as coolant due to its suitable neutronic and thermal properties in fast reactors. Good electrical conductivity of sodium is used for development of electromagnetic devices such as electromagnetic pumps & flowmeters and level probes for use in sodium cooled fast reactors, where conventional devices used in chemical plant cannot be used due to high chemical activity of sodium and high temperature. Design, development and testing of a Sodium Submersible Annular Linear Induction Pump (ALIP) was carried out recently. The developed pump can be used for sodium draining from main vessel of pool type of Sodium Cooled Fast Reactor (SFR) and any other application where pump has to be submerged in sodium. The developed pump does not require any external cooling when submerged in radioactive sodium of 200oC. The winding of submersible ALIP can withstand 550oC. The submersible ALIP was tested in sodium loop for obtaining pump characteristics. AC Conduction pump for low flow application in sodium loop has been developed. Design, analysis and manufacturing aspects are brought out in the paper. Development of three different types of compact electromagnetic flowmeters based on Samarium Cobalt permanent magnet, electromagnet formed from soft iron in combination with mineral insulated cable and small probe type permanent magnet flowmeters were successfully demonstrated. Samarium Cobalt magnet helps in reducing the size and weight of flowmeter due to its high energy product. Flowmeter having electromagnet coil made from mineral insulated cable has high temperature withstand capability of around 500oC. The electromagnet coil in combination with soft iron replaces permanent magnet, hence it provides diversity in flow measurement in critical applications. The probe type flowmeter uses small permanent magnet encapsulated in a slender probe which can be inserted inside the pipe where sodium flow measurement is required. Eddy current based ex-vessel level probe was developed for measurement of sodium level in the vessel without insertion of probe inside the vessel. It works on the principle of eddy current and using this probe, sodium level inside the stainless steel vessel can be obtained by keeping the probe outside the vessel. This technique of discrete sodium level measurement is first of its kind. This paper enumerates development of sodium submersible ALIP, newly developed flowmeters and development of ex-vessel sodium level probe. Test results obtained from sodium testing are also brought out in the paper and FEM analysis carried out for different devices are also depicted.

Country/Int. Organization:

Indira Gandhi Centre for Atomic Research, Kalpakkam (T.N.) - India

Poster Session 1 / 176

Fabrication process of NpO2 pellets

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In order to increase dissolution ratio of the irradiated NpO2 targets, it's necessary to add a little diluent into NpO2 pellet. In this paper, pressureless sintering processes and microstructures of NpO2-10% CaO, NpO2-10%SrO, NpO2-10%MgO and NpO2-5%MgO pellets were studied, sintered at 1730℃ for 2 hours in Ar-5%H2 gases. Only NpO2 solid solution phase structure was found in all the pellets. NpO2-10%CaO pellet melts at the sintering process. NpO2-10%SrO pellet has a sintered density of 60.0% TD with cracking and porous microstructures. NpO2-10%MgO pellet has a sintered density of 83.1%TD with irregular grains. NpO2-5%MgO pellet can be sintered to 90.0%TD with cobble grains. Density of NpO2-5%MgO pellet will increase to 92.5%TD using UO2 powder embedded sintering process.

Country/Int. Organization:

China Instituate of Atomic Energy

Poster Session 1 / 177

Benchmark Between EDF And IPPE On The Behavior Of Low Sodium Void Reactivity Effect Sodium Fast Reactor During An Unprotected Loss Of Flow Accident

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The validation of severe accident analysis codes for Sodium Fast Reactors (SFR) is a difficult task as it is not possible to carry out full scale integral experiments. Therefore, in addition to the validation of specific models with dedicated experiments, it is of the utmost importance to increase the confidence we have in these codes by performing benchmarking exercises with independent codes and by independent teams. As EDF R&D and IPPE are both interested in the analysis of the behavior of low Sodium Void Reactivity Effect (SVRE) cores during severe accidents, whether to support R&D on the ASTRID project (conducted by CEA) or to support R&D on the BN family reactors, a benchmarking exercise has been launched in this purpose.

As a first step, a low SVRE core design has been developed especially for this benchmark. Its main neutronics properties related to severe accident behavior - sodium density and void effect and fuel Doppler effect - have been evaluated with the CEA code ERANOS for EDF and with TRIGEX for IPPE and are compared in this article. Finally, the primary phase of an Unprotected Loss Of Flow (ULOF) accident has been simulated by each partner. On EDF side, the SIMMER code has been used whereas IPPE performed its calculations with its code COREMELT. Main results concerning power evolution and sodium boiling are compared.

Country/Int. Organization:

France

Russian Federation

Poster Session 1 / 178

POSTREACTOR STATE OF THE STANDARD AND EXPERIMENTAL BN-600 FUEL KINDS

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A large number of post irradiation examinations of the state of spent fuel element composition have been carried out for more than ten years of successful operation of BN-600 core of the third 01M2 modification. The paper aims to substantiate operating capacity of standard and certain experimental oxide fuel kinds with service life characteristics increase, in particular, burn-up depth increase.
Examinations include analysis of gamma-emitting fission product distribution, swelling and fuel porosity measurements, metallography of kernel structural changes, kernel physicochemical and thermomechanical interaction with cladding material, X-ray diffraction analysis and oxygen enhancement ratio assessment.
It is shown that with burn-up range 8.9–12.4 % FIMA uranium dioxide pellets are compatible with ChS-68 and EK-164 cladding steels. It leads to structural changes and fuel creep under restricted swelling inducing high-temperature corrosion not exceeding 65 μm. There is a tendency to residual gap broadening with burn-up increase due to high-porous unstable rim of the pellets.
Fuel film generation on the cladding internal surface is typical for MOX fuel pellets with burn-up to 11.6 % FIMA. Kernel microstructure contains low-porous globular which interpreted as depleted uranium dioxide. Significant internal corrosion increase regarding standard fuel is not detected.
Vibropac uranium gettered MOX fuel with burn-up to 10.1 % FIMA shows no high-temperature interaction with ChS-68 steel cladding. However an abnormal thermomechanic deformation of the cladding with a swelling kernel at the core-reflector boundary is observed due to getter nonuniform distribution and oxidation, and high local concentration of splitted cesium.

Country/Int. Organization:

Russian Federation

Poster Session 2 / 179

Numerical Analysis of EBR-II Shutdown Heat Removal Test-17 using 1D Plant Dynamic Analysis Code coupled with 3D CFD Code

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After reactor shutdown in sodium-cooled fast reactors, natural circulation in the heat transport systems can be expected to remove the core decay heat in the case of station blackout. For reactor safety, the core hot spot temperature during decay heat removal by natural circulation should be evaluated. In order to evaluate the core hot spot temperature, Japan Atomic Energy Agency is developing a plant dynamics analysis code Super-COPD coupled with a CFD code AQUA to simulate the thermal-hydraulics in the whole plant under natural circulation conditions. As a code validation, the coupled analysis code was applied to an analysis of EBR-II shutdown heat removal test in the cooperation with Argonne National Laboratory. The experiment simulated a protected loss of flow accident by simultaneous trip of the two primary pumps and control rod scram. The numerical results showed good agreement with the measured data.

Country/Int. Organization:

Japan/Japan Atomic Energy Agency

6.4 Neutronics –2 / 180

Stability Analysis of a Liquid Metal Cooled Fast Reactor

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Under specific transients, fast reactor cores often show significant deviation in their power distribution which leads to spatial instability. As a quantitative indication of these decoupling characteristics, the λ-mode eigenvalue separation has been frequently employed. The physical interpretation of eigenvalue separation provides a measure of the spatial neutronic coupling among various parts of a reactor and, hence is indicative of the space-time dynamic behaviour. In this paper the core-wide and regional stability of a Korean Prototype GEN-IV Sodium-cooled Fast Reactor (PGSFR) design is investigated using deterministic approaches. To calculate higher mode eigenvalues and associated eigenvectors the methodology of flux higher eigen-modes calculation was implemented into DIF3D 10.0 code and is thoroughly described in the paper. This specific DIF3D modification is denoted as DIFHH where the decontamination (or in some literature known as deflation) method was adopted as the simplest solution. In order to validate and demonstrate the performance of DIFHH code modification, the simple benchmark problem based on paper prepared by Mr. Obaidurrahman was chosen and investigated. The comparison of achieved trends and absolute values confirmed a favourable consistency between the reference and calculated results. The D/H ratio of the reactor

core was identified as an indicator of the extent of core stability, therefore the present analyses include the investigation of eigenvalue separation and flux distribution of various core D/H ratios. The findings and the results are deeply discussed in the paper.

Country/Int. Organization:

Slovakia, B&J NUCLEAR ltd.

Poster Session 2 / 181

Impact of nuclear data uncertainties on the reactivity coefficients of ALFRED

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The advancement of the design of ALFRED –the Advanced Lead-cooled Fast Reactor European Demonstrator –beyond the conceptual phase, passes through the analysis of the impact of uncertainties, notably to what concerns safety-related conditions.
Focusing on the design of the core, nuclear data are the main source of uncertainties, so that their evaluation is of utmost importance in order to assess the favourable behaviour of the system under beyond-Design Basis transients, as resulting from previous best estimate analyses standing on nominal values of the system parameters.
This work presents the results of the sensitivity/uncertainty (S/U) analysis of the ALFRED core on the reactivity (k-eff) and safety coefficients. The sensitivity analysis allowed pointing out firstly the most relevant cross sections for every response function and the key regions where safety parameters needed to be evaluated. Uncertainty analysis allowed then establishing a possible range of confidence for the reactivity coefficients. The adjoint-based technique involved in TSUNAMI-3D module from SCALE6 system was used.
The confidence intervals identified for each reactivity coefficient will permit transient calculations to propagate uncertainties into transient behaviour, after pointing out the most unfavourable –yet physical –set of reactivity coefficients for the selected transient scenarios. This will in turn provide an exhaustive picture of the influence of nuclear data uncertainties on core performance, identifying key parameters and possibly indicating specific actions required to achieve the aimed safety performances.

Country/Int. Organization:

Italy

3.7 Core Disruptive Accident Prevention / 182

Impact of an accidental control rod withdrawal on the ALFRED core: tridimensional neutronic and thermal-hydraulic analyses

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In the last years an international effort has been pursued for the development of the Advanced Lead Fast Reactor European Demonstrator (ALFRED), whose design was initially conceived in the EURATOM FP7 LEADER project and now is being carried on by the FALCON International Consortium signed by Italian, Romanian and Czech organizations. In line with the vision of the Gen-IV initiative, the LFR concept following ALFRED is expected to excel in safety and economics while allowing the closure of the nuclear fuel cycle.
The article - dealing with the ALFRED core design - analyses the local and global effects due to the accidental extraction of one Control Rod (CR) during nominal plant conditions. The impact on the core neutronics was evaluated with the ERANOS deterministic code in terms of:
• reactivity balance and power distribution among the Fuel Assemblies (FAs);
• distortion of the local power distribution inside the hottest FA.
By means of specific ERANOS procedures, a detailed 3D power map was evaluated at the level of single fuel pins. The steady state neutronic results were then used as input for accurate transient and thermal-hydraulic studies made by:

• the RELAP5 system code, to investigate the evolution of the worst conceivable scenario, associated with the un-adverted withdrawal of the CR and failure of the scram actuation by the reactor protection system, thereby resulting in an Unprotected Transient of Over-Power (UTOP);

• the ANTEO+ sub-channel code, developed and validated by ENEA, to evaluate the temperature distributions in all the pins and surrounding sub-channels at key instants of the transient.

The main objective of the study was the assessment of the new thermal conditions of the hottest FA in order to verify the compliance with the safety limits. By adopting a credible maximum withdrawal velocity of the CR, a margin of about 250 °C resulted from the melting point of the MOX fuel even in a completely unprotected scenario. Such margin should be high enough to accommodate the modelling, material and fabrication uncertainties. Similarly, by looking at the stainless steel cladding temperature behaviour and its ability to withstand creep, the grace time for operator intervention resulted far higher than the minimum target of 30 minutes to allow for operators' intervention.

Country/Int. Organization:

Italy

5.5 Large Component Technology I / 183

Heat Transfer Performance Test for a Sodium-to-Air Heat Exchanger with an Inclined Finned-Tube Banks

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A separate effect test facility called SELFA (Sodium thermal-hydraulic Experiment Loop for Finned-tube sodium-to-Air heat exchanger) using liquid sodium and air as operating fluids has been developed. SELFA is one of the requisite sodium thermal-hydraulic test facilities within the framework of STELLA (Sodium Test Loop for Safety Simulation and Assessment) program, which is indispensible for the support of PGSFR (Prototype Gen IV Sodium-cooled Fast Reactor) development. The model heat exchanger (M-FHX) of SELFA was designed for performance demonstration of FHX (Forced-draft sodium-to-air Heat eXchanger) in PGSFR, which has three-row inclined finned-tube banks with staggered arrangement. Using this dedicated sodium heat exchanger test facility, several sets of heat transfer performance test have been conducted for validation of computational codes such as the heat exchanger thermal-sizing code (FHXSA) and the safety analysis code (MARS-LMR). In this study, we carried out performance tests for the M-FHX at the design point (i.e., thermal duty of 320 kWt). The test results obtained from this test have been used for its heat transfer performance evaluation through comparisons with the computational analyses results obtained from both a commersial CFD analyses as well as in-house thermal design and analysis computational codes developed by KAERI. Finally, it was confirmed that we have got reasonable experimenal datasets through this work.

Country/Int. Organization:

Rep. of Korea

6.6 Coupled Calculations / 184

Coupled calculations for the fast reactors safety justification with the EUCLID/V1 integrated computer code

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The EUCLID/V1 integrated computer code is designed for the safety analysis and justification of the new generation NPPs with liquid metal cooled fast reactors under normal operating conditions, design basis accidents and beyond design basis accidents. The EUCLID/V1 code includes the system thermohydraulics module, spatial time-dependent neutronics module, quasi two-dimensional fuel rod module and the module of burnup and decay heat calculations. In the neutronics module the improved quasistatic method is employed to solve the transport equation in the multigroup diffusion or discrete ordinates (Sn) forms.
The extensive V&V of the single modules have been carried out on analytical and numerical tests, experimental results and benchmarks. To validate a coupled modeling of physical processes in a reactor core and its loops, the experimental data on the BN-600 and BOR-60 transient regimes have been used.
Some of the BN-1200 and BREST-OD-300 reactor facilities design basis accidents and beyond design basis accidents have been simulated by means of the EUCLID/V1 code. Particularly, the loss of offsite power accident in the BN-1200 reactor has been modeled. In this design basis accident the reactor shutdown cooling with the emergency heat removal system is considered. It has been shown that after the control rods drop and reactor pumps shutdown the efficiency of two of four channels of the emergency heat removal system is sufficient to prevent the maximum fuel, cladding and sodium temperatures from exceeding the design limits. In frame of the test calculations the accident caused by the introduction of the total positive reactivity margin via withdrawal of all control rods from the reactor core at the maximum design speed during full power operation without scram operation (UTOP+ULOF) has been simulated for the BREST-OD-300 reactor facility. The obtained results indicate that after the reactor coolant pumps shutdown the total power decreases due to the thermohydraulic reactivity feedbacks and operation of the passive feedback system. By 100 s of the scenario natural circulation of lead progresses in the primary loop, it is equal to 9.6% of the nominal total flow. The core disruption does not happen.

Country/Int. Organization:

Russian Federation

Poster Session 1 / 185

The core of the LFR-AS-200: robustness for safety

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The LFR-AS-200 is a 200 MW(e) Lead Fast Reactor (LFR) standing on simplicity to target the objective of representing a commercially viable option for an innovative Small Modular Reactor (SMR). To fulfil its envisaged role, which is particularly meaningful for multi-units sites, the design has to enhance the safety performances; this is achieved by exploiting the relevant favourable intrinsic properties of lead, and by implementing engineered features, passively operating to permit a robust response of the system even in challenging beyond-design accidental conditions resulting as a consequence of multiple failures of the reference lines of defence.

The design of the core is here presented with a particular emphasis on the encompassed safety provisions, both intrinsic and engineered. Notably, the largely negative reactivity coefficients of the core will be presented along with a passive provision enhancing the flowering of the core, thereby the anti-reactivity insertion upon transients resulting in an increase of the core outlet temperature. The performances of the system in two main unprotected transients –a transient of over-power and a combined loss of flow-loss of heat sink –are finally presented. The results prove the effectiveness of the design to withstand such challenging conditions and to ensure extremely large grace times for actuating countermeasures without incurring in the failure of any of the first two engineered barriers for the confinement of radioactivity –the fuel cladding and the primary circuit boundary –thereby protecting not only the environment and the population, but also the investment itself.

Country/Int. Organization:

Italy

2.2 Commissioning and Operating Experience of Fast Reactors II / 186

EXPERIENCE OF COMMISSIONING OF THE SECTORAL MONITORING TIGHTNESS SYSTEM OF FUEL ELEMENTS CLADDINGS (SSKGO) OF RF BN-600, RF BN-800

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Sectoral monitoring tightness system of fuel elements claddings (SSKGO) is important for the reactor facility (RF) safety and designed for fuel cladding state and appearance and development of the «fuel-coolant contact» defects operational control. Developers of the system are: JSC «IPPE» and JSC «Afrikantov OKBM».
Modernized SSKGO system (3N safety class) of RF BN-600 and SSKGO system (3NU safety class) of RF BN-800 were put into trial operation at Beloyarsk NPP by IPPE specialists in conjunction with Beloyarsk NPP specialists from 2014 to 2015. SSKGO modernization is based on a new design of the detection blocks (DB), which include the supports with the ionization fission chambers and the use of modern measuring and computing facilities. As a result of modernization it was possible to increase the sensitivity of the measurement channels to the neutron flux density by more than 60%, and significantly reduce background from the core neutrons RF BN-800.
The certification of the neutron flux density models of SSKGO detection blocks that will significantly reduce the amount of work and measurement time on the reactor during neutron flux density certification from neutron sources, based in the in-house DB was conducted in IPPE in conjunction with specialists VNIIFTRI in 2013-2014. According to the results of certification the neutron spectrum shape impact on the measuring channels sensitivity has been studied. Programs and methods of periodic verification of regular neutron source have been developed.
Stability studies of technical and metrological characteristics of measuring channels SSKGO based on ionization fission chambers were carried out on the BN-600 and BN-800. When working at various power levels, the indications background according to reactor power was determined. SSKGO tests showed high reliability and immunity systems during commissioning at BN-600 and BN-800.
Currently SSKGO systems BN-600 and BN-800 are nominally operated on blocks 3 and 4 of Beloyarsk NPP.

Country/Int. Organization:

JSC «SSC RF-IPPE», JSC «Afrikantov OKBM», Russian Feredation

1.2 SFR DESIGN & DEVELOPMENT - 2 / 188

Analysis of the Characteristics of the Fast Breeder Reactor with Metallic Fuel

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A lot of approaches are considered to increase a marketability of fast breeder reactors producing two products –electricity and exceeding nuclear fuel. To increase a production of exceeding nuclear fuel it is proposed to switch from widely used oxide fuel to carbide, nitride and the densest metallic uranium fuel. In a fabrication chain of the exceeding nuclear fuel a cost of spent nuclear fuel refabrication is also important. From all kinds of nuclear fuel, considered worldwide at the fast breeder reactors'area, the metallic fuel provide the highest values of the exceeding nuclear fuel production i.e. the highest value of the breeding rate (BR) and the lowest refabrication cost for spent nuclear fuel due to melting technology.
But the reactors with metallic fuel have issues which lead to the absence of completed projects and their realizations. The main problem of the safety assurance of such reactors is related to a weak reactivity feedback by fuel temperature. To solve this problem an approach with heterogeneous placement of the fuel at the axial direction is suggested. Layout of the depleted metallic fuel is proposed at the bottom blanket region and at the top blanket region above the sodium cavity to receive high breeding rate. In addition, placement of the oxide fuel with central thin layer made of metallic fuel in the core is proposed to provide sufficient level of temperature feedback. An improvement of this approach with replacement of the oxide fuel from the bottom part of the core by a metallic

plutonium fuel is considered at the paper.
It is shown by calculations that the suggested approach together with the replacement of the oxide fuel by the metallic depleted uranium fuel at the assemblies of a radial blanket region ensures the high reactor BR with sufficient level of the temperature feedback. The high BR value is provided by using of the metallic fuel in the majority of reactor's volume. Substantial feedbacks are provided by the utilizing of the oxide fuel at the area of high coolant, fuel and cladding temperatures. At the same time the metallic plutonium fuel is placed at the area of high power density and low temperature of the core components.

Country/Int. Organization:

Russian Federation/JSC ("VNIIAES JSC")

3.3 Probabilistic Safety Assessment / 189

Development of Smart Component Based Framework for Dynamic Reliability Analysis of Nuclear Safety Systems

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Dynamic reliability methodologies account for the safety system's time dependent characteristics while estimating the reliability. Time dependence can arise due to interaction of process variables with the hardware and hardware failure on process conditions. Though static reliability models often capture the average behavior and try to make conservative estimates, it is inadequate from a number of perspectives. First, this requires that the analyst needs to establish that the model is conservative. Second, such modeling requires more expertise and experience in the appropriate domain of the problem, rather than in the reliability methods. Third, approximate methods may be inadequate to establish reliability enhancements or degradations due to subtle alterations in the system design. In spite of the significant effort in the reliability community to establish dynamic reliability analysis methods, there are no general purpose tools similar to that available for fault tree event tree modeling. A methodology based on 'Smart Components' is being developed for dynamic reliability evaluation of safety systems involving digital IC systems interacting with process and hardware. Smart Component based dynamic method uses elements of object oriented and relational data base architecture and is suitable for being developed into a general purpose tool. The paper demonstrates the capability of the method to evaluate reliability of systems having various types of time dependence, interaction between hardware failure and process evolution and complexity by means of few case studies. The method is found to be promising for accurate modeling of dynamic as well as static scenarios.

Country/Int. Organization:

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190

Optimization problem for characteristics of fast reactors operating in a closed fuel cycle

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Results of the analysis of the optimization problem for characteristics of fast reactors operating in a closed fuel cycle are presented.
The optimization problem is proposed to solve in two stages. At the first stage, multiple calculations of characteristics of fast reactors are carried out using a variety of characteristics of the fuel returned to reactor from a closed nuclear fuel cycle, such as the content of fission fragments, the uncertainty in determination of the concentration of fissile nuclides in the returned fuel, etc. At the second stage, the optimization problem is solved directly by selecting the most appropriate characteristics of the reactor from multiple sets of obtained characteristics.
There are different approaches to solve the optimization problem for an objective function, which does not have predetermined analytical dependence on its parameters. One way or another, multiple calculations of function values at different sets of parameters are assumed. The following approach seems to be the least complicated in the case of a large number of optimization variables. Analytical approximation of the dependence is based on a previously calculated set of examples of the parameter values corresponding to the function values. Finally, when the approximation is constructed, the classic gradient methods are used to solve the optimization problem.
As an example of solving the optimization problem the reactor EBR-II with a metal fuel returned to the reactor with the contents of the fission products after recycling is considered. The influence of the uncertainty in determination of the nuclides concentration on the reactor characteristics is shown.

Country/Int. Organization:

Russia

Poster Session 1 / 192

Decay-heat removal in accidents in fast reactors with liquid metal coolant

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The problem of decay-heat removal from a shutdown reactor is still pending and Fucushima accident proved it. The complexity of this problem grows with the increase reactor power.
High power reactors with sodium and lead coolants were analyzed and compared in terms of decay -heat removal using 3D thermo-hydraulic calculations of reactor cooldown.
Two DHRS designs are compared that differ by the location of emergency heat-exchanger. In the first design emergency heat-exchanger is located in the upper reactor chamber and heat is removed from reactor core due to following circulation path: "emergency heat-exchanger –upper plenum –inter-wrapper space of reactor core –upper plenum". In the second design emergency heat-exchanger is located in the downtake slit of reactor and design includes backflow valve that in cooldown mode allows "hot"coolant from the upper plenum to enter emergency heat-exchanger and blocking this flow while the reactor operates in power mode.
DHRS of a sodium reactor results to be more effective for both DHRS designs. As for the lead cooled reactor the second DHRS design also allows to remove after-heat without exceeding the allowed temperature limits. With the 1st DHRS design fuel rods overheat for a short period of time.

Country/Int. Organization:

Russia/«Innovation and technology center for the «PRORYV» project»

Poster Session 1 / 193

SIBYLLA CODE: ASSESSMENT OF WATER BODIES CONTAMINATION AND DOSES RECEIVED BY POPULATION DUE TO RADIOACTIVITY DISCHARGES INTO THE HYDROSPHERE

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SIBYLLA code is intended for calculation of the radionuclides content in water and bottom sediments of water bodies undergoing radioactive contamination during normal operating conditions or accidents at the nuclear facilities. Also SIBYLLA enables to calculate the doses resulted from use of the water-bodies including water-supply. The code contains set of models for water bodies of different types (lakes, rivers, water-reservoirs, etc). SIBYLLA can be used for radiation safety assessment of nuclear facility on the all stages of its lifecycle: design, operation, decommissioning.
Wide range of sources of radioactive contamination and pathways to the water-bodies can be taken into account –fallouts from the atmosphere, discharges, leakages, wash-out from contaminated catchments, waters of contaminated tributaries.
Internal and external exposure doses received by population are estimated. SIBYLLA takes into account consumption of drinking water, fish, agricultural products that can be contaminated due to watering of cattle or use of contaminated irrigated lands or flood-lands. It also takes into account inhalation of tritium, swimming, fishing, being at irrigated lands, flood-lands or in the vicinity of water-bodies.
SIBYLLA models main processes that determine the migration of radioactivity in water-bodies: radioactive decay; advection and dispersion; sorption and desorption on suspended particles and bed sediments; deposition and resuspension of sediments; diffusion on the water –bottom interface, etc.
The results of code validation against experimental data are presented in the paper. The validation is performed against data on radioactive contamination of eight water-bodies of three different types: the Kiev water-reservoir, a river and three lakes contaminated in the result of Chernobyl accident; the Tygish lake situated on the axis of East Ural Radioactive Trace; the Techa river; the Tom river influenced by discharges of Siberian Chemical Combine. It is shown that in 52 % of cases calculated and observed data differed less than in 1.5 times; in 95 % of cases less than in 3 times; in 100% of cases less than in 10 times. The main reason of discrepancies is uncertainty of input data.
The quality of the SIBYLLA code is confirmed by expert council for software accreditation of ROSTECHNADZOR (Federal Environmental, Industrial and Nuclear Supervision Service of Russia), where SIBYLLA is certified in 2016.

Country/Int. Organization:

Russian Federation

7.1 Sustainability of Fast Reactors / 194

Assessment of a nuclear energy system based on the integral indicator of sustainable development

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Ability of nuclear power to meet the requirements of sustainable development is a critical point for public acceptance of the energy technology and its further development. The method for assessment of the nuclear energy system (NES) compatibility with the requirements of the UN concept of sustainable development was developed in the IAEA within the activities of the INPRO international project. The approach discussed in the paper is aimed at decreasing the uncertainty of the method in assessing NES sustainability in order to duly arrange the NESs from those at the low limit of sustainability up to advanced systems with much better sustainability features.
It is shown that the integral index determined with the use of common approaches of the multi-attribute theory may provide a measure of the NES sustainability to be used for quantitative comparison of sustainability of NESs based on different technical and institutional solutions. The value of the index depends on the reaching by an assessed NES of certain key developments in the seven subject areas: economy, nuclear safety, resource supply, waste management, non-proliferation, public acceptance and international cooperation.
It is shown that assessment of the NES based on OTFC with thermal reactors and UOX fuel on a large time horizon will be characterized by a low value of the sustainability index in case of lack of natural uranium resource and continued accumulation of spent nuclear fuel and plutonium therein. The index radically increases due to synergistic use of thermal and fast reactors with closed nuclear fuel cycle. This effect can be gained within the country that has mastered both technologies or in the system in which user country receives services on closed NFC from supplying country. It is noted that the model needs further development in order to take into account more factors related to the sustainability notion.

Country/Int. Organization:

Russia

Poster Session 2 / 195

NEW NEUTRONIC CALCULATION CODES BASED ON DISCRETE ORDINATES METHOD USING METHODS OF FINITE DIFFERENCES AND FINITE ELEMENTS

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CORNER and ODETTA codes for neutrons and photons transport based on discrete ordinates using finite differences and finite elements methods have been developed as a part of the new generation codes for the construction and validation of the perspective FBR safety.
Modern CONSYST software is used for the preparation of the macroscopic cross sections. Both eigenvalue (keff) and fixed source problems can be solved, including joint calculations of neutrons and gamma rays. The principal application is solving transport problems with deep penetration. OpenMP technology is applied for parallel computing.
The CORNER code allows calculations in three-dimensional hexagonal and combined geometry (to account for the heterogeneous features of the computational model). Weighted Diamond Difference

and nodal schemes are used to approximate the spatial dependence. The calculations have been performed for models of BN-800 and BN-1200 reactors, and for BFS critical assemblies.
ODETTA code uses discontinuous linear finite element method on unstructured tetrahedral meshes, based on the selected CAD model with Salome and Gmsh programs. Space rebalance method and δ-process are used to speed up the inner and outer iterations respectively. Results of code validation against safety experiments ASPIS and EURACOS from SINBAD database and cross-verification on a test model of the reactor BN-1200 are presented.

Country/Int. Organization:

Nuclear Safety Institute of the Russian Academy of Sciences, Moscow

6.9 Research Reactors / **196**

Detailed engineering neutron codes for calculations of fast breeder reactors

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The necessity in detailed engineering codes to perform neutron-physical calculations of the core of nuclear facility is caused by the problems arising during fast reactors design and operation. Diffusion program complexes of neutron-physical calculations with different degrees of assembly representation has been developed and used in IBRAE RAN.
The FUBUKI program is intended for three-dimensional neutron-physical calculation in the fuel element approximation. There are two versions of FUBUKI program. In the FUBUKI-1 fuel assembly is represented by a set of regular hexagonal prisms modelling the fuel cluster surrounded by a set of irregular hexagonal prisms which in their turn simulate the fuel assembly blanket and the surrounding space. At that the boundary microcells belong to two (three for angle) assemblies simultaneously. In the FUBUKI-2 irregular pentagonal prisms belonging only to a single assembly surround the set of prisms that simulates fuel elements. The first version of the program allows performing the calculation of the reactor core in which all fuel assemblies have the same number of fuel elements. The second version of the program allows to perform the calculations of the reactor core in which there are fuel assemblies with different number of fuel elements. Fuel element approximation provides the characteristics of each fuel element.
Program G-7 uses the hex view of the hex assembly in the form of a hexagonal prism as a central cell surrounded by six trapezoidal prisms as external cells. It allows to determinate the fuel assembly blankets and the control and protection system sleeves to the external cells but fuel elements and absorber elements to the central cell. Each cell in the models can have their material content and size.
The paper presents results of existing and planning reactors calculations obtained by the specified programs.

Country/Int. Organization:

Russian Federation

5.1 Advanced Fast Reactor Fuel Development I / **198**

Fabrication Characteristics of Injection-cast Metallic Fuels

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The fabrication process of metallic fuels for sodium-cooled fast reactor (SFR) was developed using the injection casting. U-Zr-RE(Nd-Ce-Pr-La) fuel slugs were fabricated and characterized to optimize the injection casting process. The microstructure examined by SEM showed that precipitates were uniformly distributed over the fuel slug. The fuel weight loss after the injection casting was measured to be about 1.5%. The reaction between the melt and the crucible was found to be significant in the fabrication of RE-containing fuel slugs compared to U-Zr fuel slugs. The pressurized injection casting method was also developed to fabricate the fuel slugs containing volatile elements. U-Zr-Mn fuel slugs were fabricated as a surrogate for Am-bearing metallic fuels under three different melting pressure conditions. From the chemical composition analysis by the ICP-AES method, no evaporation of Mn was detected in the fuel slugs fabricated under Ar atmosphere higher than 400 torr.

Country/Int. Organization:

Republic of Korea

Poster Session 1 / 199

Modeling of hydrodynamic processes at a large leak of water into sodium in the fast reactor coolant circuit

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A description is given of a physico-mathematical model of the processes that occur in the sodium circuit with a variable flow cross-section in case of a water leak into sodium. The application area for this technique includes a possibility to analyze consequences of this leak as applied to sodium-water steam generators in fast neutron reactors.
Hydrodynamic processes that occur in the sodium circuit in the event of a water leak are described within the framework of a 1-D thermally-nonequilibrium three-component gas-liquid flow model (sodium-hydrogen-sodium hydroxide). In this case hydrogen is assumed to be an ideal gas and its solubility in sodium is taken into account. Consideration is also given to dependence of sodium and sodium hydroxide on pressure and temperature.
In the proposed improved approach the sodium circuit is presented in the form of combination of two models:
• a 1-D model with distributed parameters, that describes dynamics of the parameters in all the circuit elements (sodium-water reaction region included), with the exception of expansion tank volume;
• an expansion tank model built as part of the model with lumped parameters.
These two models are cross-linked in the expansion tank inlet and outlet points.
The proposed model and calculation technique have been realized in the form of a computer code.
A computer code was tested on experimental data obtained from the injection of water vapor into sodium at the Russian sodium loop.

Results gained from a comparison of calculations with experimental data, lead us to conclude that the proposed technique adequately reflects the transient behavior of the relevant parameters during the hydrodynamic processes that occur in sodium-water interaction in a sodium circuit.

Country/Int. Organization:

Institute for Physics and Power Engineering named after A.I. Leypunsky"

Poster Session 2 / 200

The relative yields and half-lives of precursors of delayed neutrons in the fission 241Am by fast neutrons.

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At the present time, the most perspective processes, which could form the basis of the technology of transmutation of radionuclides, are the processes associated with using of nuclear reactors, as well as sub-critical systems with high neutron flux generated using charged particle accelerators. The delayed neutrons have an important role in the safe management and kinetics of nuclear power plants. Therefore, the development of any of the above concepts of transmutation of nuclear waste requires the information on nuclear-physical characteristics of delayed neutrons for minor actinides in the reactor energy range of primary neutrons. In this paper the energy dependence of the relative delayed neutron yields and half-lives of their nuclear precursors in the fission 241Am by neutrons in the energy range of 1-5 MeV was measured. The assembly of 3He-counters in neutron moderator block was used as a detector. The measurements of decay curves of delayed neutron activity were carried out in a cyclic mode. The obtained decay curves of delayed neutron activity have been processed in order to obtain the values of the relative yields and half-lives of delayed neutron precursors. The energy dependence of the detection efficiency of neutron detector was obtained as a result of a series of measurements of spectra of monoenergetic neutrons.

Country/Int. Organization:

Russian Federation / JSC "SSC RF –IPPE"

Poster Session 2 / 201

Features of the time dependence of the intensity of delayed neutrons in the range of 0.02 s in the fission 235U by thermal and fast neutrons.

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In the present work the set-up created on the basis of the accelerator Tandetron (IPPE) for the experimental studies of the time dependence of delayed neutron activity from neutron induced fission of 235U is described. Measurements were carried out with neutron beam generated by the 7Li(p,n) reaction. The lower limit of the investigated time range was governed by the proton beam switching system that was 20 ms. The neutron detector is an assembly of three SNM-18 counters (working gas: a mixture of 97% He-3 + 3% Ar, pressure of 405 kPa.) mounted in the polyethylene box. It was shown that the temporary characteristics of delayed neutrons from the fission of 235U by epithermal neutrons is consistent with the time dependence which at present is recommended as a standard. In case of the fast neutron induced fission of 235U the measured decay curve of delayed neutrons shows excess of counting rate in the time interval 0.01-0.2 s as compared with the decay curve corresponding to the recommended data. The microscopic approach using the data on the probability of emission of delayed neutrons and cumulative yields of fission products for 368 nuclei precursors also indicates the existence of short-lived component (T1/2 <0.2 s) in the decay curve of activity of delayed neutrons emitted in the fission of 235U.

Country/Int. Organization:

Russian Federation / JSC "SSC RF –IPPE"

Poster Session 2 / 203

Validation of the evaluated fission product yields data from the fast neutron induced fission of 235U, 238U, 239Pu

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The evaluated fission product yields data are an important characteristic. The validation method of the evaluated fission product yields data is based on comparing the characteristics of delayed neutrons produced by the summation method with appropriate recommended data. The total delayed neutron yields and the mean half-life of delayed neutron precursors were used as the delayed neutron characteristics. It was shown in the present work that the use of fission product yields presented in the JEFF library allows to obtain values of the total neutron yields and the mean half-life precursors closest to accordingly recommended data for fission 235U, 238U, 239Pu by thermal neutrons and for fission by fast neutrons except 239Pu. The using of the fission product yields of libraries ENDF/B and JENDL gives the values of total delayed neutron yields and half-lives of delayed neutrons materially different from the recommended data. The macroscopic characteristics of delayed neutrons were calculated by the summation method. The modeling of time-dependent activity decay of delayed neutrons was carried out. The sensitivity of the macroscopic parameters was studied for used microscopic data. The most reliable sets of microscopic data were chosen.

Country/Int. Organization:

Russian Federation / JSC "SSC RF –IPPE"

Poster Session 1 / 204

Assessment of accuracy from the use of point kinetics when analyzing transition processes in high power fast reactor

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"Point kinetics"approximation is widely used in reactor justification for calculation of transient and emergency modes in the first place. The point kinetic model is used as the base model for Russian DINROS, GRIF, SOKRAT-BN software programs used for safety justification of fast reactors. Its popularity is explained by its relative simplicity and physical transparency (possibility to interpret results on the language of reactivity effects and easily demonstrative verification).
Computational study of errors caused by the use of point kinetic model is performed with the use of UNICO multi-physical software (3D neutronics in diffusional approximation + 3D thermohydraulics) for three non-static test example problems for BN-1200 reactor:
• The problem of sudden change of coolant temperature at the inlet of pressure header of reactor core (in one of 4 first circuit loops).
• Emergency protection rods drop at nominal power (example of fast running process).
• Self-act of one of the control rods.
It is shown that fuel rod temperature estimation error during self-act of one of the control rods can reach 100°C.

Country/Int. Organization:

ITC "PRORYV"Project, Moscow, Russian Federation

Poster Session 1 / 205

Passive Shutdown Systems for Liquid Metal-Cooled Fast Reactors

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A major focus of the design of modern fast reactor systems is on inherent and passive safety. Inherent safety means that the reactor design is such that the plant remains in a safe condition solely on the basis of the laws of nature; these laws ensure that all performance characteristics remain

within safe bounds under all conceivable circumstances. The definition of passive safety is broader, and implies that no human intervention, no triggering signals and no supply of external energy are required for the reactor to remain in a safe condition. Inherent and passive safety features are especially important when active systems such as the SCRAM-systems for reactor shutdown are not functioning properly. Passive shutdown systems can operate either continuously (analogous to reactivity feedback mechanism or function as a backup actuation method for the conventional reactor SCRAM system.

Numerous passive shutdown systems designs have been developed over the years in fast reactor research programs across the world. To summarize the state-of-the-art in this field, the members of the technical working group on fast reactors at the IAEA (TWG-FR) has been collecting the various approaches, design principles, engineering solutions and their impact on reactor safety and operation, in a joint collaborative project over the last year. A review of a total of 20 different systems, divided in to five categories depending on the way they are actuated, will eventually be presented in a Nuclear Energy Series report written jointly by the TWG-FR group. These systems and the general findings of the TWG-FR group will be presented in a dedicated poster session at the FR17 meeting.

Country/Int. Organization:

Technical Working Group on Fast Reactors, IAEA

Poster Session 1 / **207**

Overview of the U.S. DOE fast reactor fuel development program

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Research and development activities on fast reactor fuels in the US are focused on their potential use for actinide transmutation in future sodium fast reactors. As part of this application, there is a desire to demonstrate a multifold increase in burnup potential.
A number of fuel design innovations are under investigation with a view toward significantly increasing the burnup potential of fuels, since higher discharge burnups equate to lower potential actinide losses during recycle. Promising innovations under investigation include: 1) lowering the fuel smeared density in order to accommodate the additional swelling expected as burnups increase, 2) utilizing an annular fuel geometry for better geometrical stability at low smeared densities, as well as the potential to eliminate the need for a sodium bond, and 3) minor alloy additions to immobilize lanthanide fission products inside the metallic fuel matrix and prevent their transport to the cladding resulting in fuel-cladding chemical interaction.
This paper presents results from these efforts to advance fuel and cladding technology in support of high burnup and actinide transmutation objectives. Highlights include examples of fabrication of low smeared density annular metallic fuels, experiments to identify alloy additions effective in immobilizing lanthanide fission products, and early postirradiation examinations of annular metallic fuels having low smeared densities and palladium additions for fission product immobilization. Advanced cladding development activities include the investigation of high dose tolerant steel alloys. An overview of the U.S. Department of Energy fast reactor fuel research and development activities will be provided.

Country/Int. Organization:

Idaho National Laboratory - USDOE

Poster Session 2 / 209

Actual Status of the Development of Multigroup XS Libraries for the Gas-cooled Fast Reactor in Slovakia

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Slovakia is involved in the development of the ALLEGRO reactor, the demonstrator of the unique GFR technology. Since the Gas-cooled Fast Reactor lacks any applicable experimental data, the design and optimization of its core must rely on data from similar reactor concepts and on calculations using Monte Carlo and deterministic methods. Although these two methods differ in their nature, both require appropriate nuclear data libraries. The present paper describes the actual status of the development of multigroup XS libraries, optimized for fast, but precise deterministic calculations of the GFR 2400 reactor. The optimization of the XS library starts with a similarity assessment, to identify benchmark experiments, which could provide experimental data relevant to GFR 2400. The selected benchmarks are evaluated in the discrete ordinates PARTISN transport code, based on integral parameters as well as sensitivity profiles. In order to gather the required sensitivity data, the benchmark calculations are performed in very fine energy structure. For these calculations the previously developed SBJ v2015 XS library is used. In order to obtain the overall sensitivity profile of the GFR 2400 reactor the same approach is applied, using the SBJ v2015 XS library. The overall sensitivity profile is the sum of absolute sensitivity profiles of all relevant nuclides and reactions. This sensitivity profile is used to analyze the GFR 2400 neutron spectrum and to develop an optimized intermediate-group energy structure for the SBJ v2016 XS library. This intermediate-group energy structure is used to reevaluate the selected benchmark experiments and to compare the performance of the SBJ v2016 XS library with MCNP5, with the SBJ v2015 XS library as well as with other multigroup XS libraries available for fast reactor calculations. In the final step the new optimized SBJ v2016 XS library is applied on the full core GFR 2400 reactor model in DIF3D and compared with MCNP5 results based on core-wide and group-wise distributions of neutron physical parameters.

Country/Int. Organization:

Slovak University of Technology in Bratislava, Faculty of Electrical Engineering and Information Technology, INPE

5.7 Chemistry Related Technology / 211

The Conditioning and Chemistry Programme for MYRRHA

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In 2010 the Chemistry and Conditioning Programme (CCP) was established to provide R&D support for the engineering and licensing of the MYRRHA nuclear system under development at the Belgian Nuclear Research Centre (SCK-CEN). MYRRHA is an accelerator-driven subcritical nuclear reactor with fast neutron spectrum, using liquid lead-bismuth eutectic (LBE) as spallation target material and coolant. The CCP team studies various chemistry-related aspects of the LBE which are important for safety, operation and decommissioning of MYRRHA. In this paper we present an overview of our activities and achievements.

One of the programme's main goals is to develop methods for accurate measurement and control of dissolved oxygen in the LBE coolant. Oxygen control is needed to reduce corrosion of reactor components exposed to LBE and to avoid precipitation of lead oxide in the primary circuit. Our achievements in this domain include the development of new oxygen sensors that measure reliably down to 200 ℃ in both loop and pool configuration, several advanced designs of lead-oxide based solid mass exchangers for oxygen supply to LBE and a new electrochemical oxygen pumping system to precisely regulate dissolved oxygen. Accurate oxygen control on pilot scale has been recently demonstrated in the LBE chemistry loop MEXICO. Experimental studies on oxygen control are supported by various theoretical calculations. Thermochemical calculations have been successful in predicting the influence of temperature and impurities such as corrosion products on oxygen control. Detailed CFD calculations coupled with chemical reactions are used to assess oxygen distribution and transport in the complex geometries of the primary system of MYRRHA.

A second priority is the study of evaporation of several safety-critical radionuclides from LBE. These radionuclides are formed in the LBE by activation of the coolant (polonium), by spallation (mercury, …) or may be released into the LBE through leaking fuel pins (fission products such as iodine). For the experimental study of polonium release, considered to be one of the most important safety issues of LBE-cooled reactors, a dedicated polonium lab has been set up. The evaporation of other elements is typically studied using stable isotopes. In close collaboration with colleagues from especially the Swiss Paul Scherrer institute, we have discovered several physicochemical mechanisms by which polonium can be released from LBE and we have performed studies on methods to capture especially volatile gaseous Po molecules.

Country/Int. Organization:

Belgian Nuclear Research Centre

Poster Session 1 / 212

Evaluation of Anticipated Transient without Scram for SM-SFR using SAS4A/SASSYS-1

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Small Modular Sodium-cooled Fast Reactor (SM-SFR) was developed in UNIST as a breeder reactor with the target of ultra-long cycle operation. The depletion analysis and quasi-static reactivity balance analysis were performed to see its inherent safety in the neutronics point of view. In this study, the inherent safety evaluation is performed in the thermal-hydraulic point of view by using transient analysis for LMR code SAS4A/SASSYS-1 which was developed in Argonne National Laboratory. Three major events of Anticipated Transient without Scram (ATWS) were tested for this research; Unprotected Loss of Flow (ULOF), Unprotected Loss of Heat Sink (ULOHS), Unprotected Transient Over Power (UTOP). Every perturbation for each transient event occurs at 10 second and each simulation time is 100 minute. The power to flow change, the reactivity profiles, and the temperature changes were investigated to trace each transient trend. It has been confirmed that SM-SFR has inherent safety from the fact that any of the events doesn't have a clad failure or a coolant boiling.

Country/Int. Organization:

Republic of Korea

Poster Session 2 / 213

Expriment and Analysis of Flow distribution of MOX Assembly

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China experimental fast reactor (CEFR) uses uranium dioxide as its first fuel, then it switchs to the MOX fuel. The MOX fuel have been great changes in thermal hydraulic characteristics, therefore it needs to study. In this paper, an resistance characteristic expriment and the CFD simulation of the CEFR-MOX assembly was carried on. Through comparison and analysis, an empirical formula was fitted out, and the size of assembly was determined. The experience formula after curing can be directly used for assembly selection, while reducing the testing time and saving money.

Country/Int. Organization:

China, CIAE.

Poster Session 1 / 214

Mathematical modeling of the mononitride nuclear fuel production processes

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Implementation program of fast reactors in nuclear power engineering provides for the use of new types of nuclear fuel, in particular nitride. RFNC-VNIITF in cooperation with VNIINM develops models and codes for the simulation of physical and chemical processes for the purpose of software development and implementation of nitride mixed uranium-plutonium fuel fabrication technology. The main objective is to create software that allows you to select and optimize the process conditions.

To date, developed and implemented the mathematical models of the basic processes of fuel fabrication:
- grinding of powders;
- carbothermal synthesis;
- granulation;
- pressing;
- sintering tablets.

Models are designed to calculate the basic characteristics of the products according to the characteristics of the starting materials and the process conditions.
Also designed auxiliary thermodynamic program module that allows calculating the thermodynamic equilibrium of multicomponent and multiphase systems with different initial conditions, the thermodynamic functions of the individual reactions, allows you to work with the database of the thermodynamic properties of substances. In this work module is used for modeling systems, typical for carbothermal synthesis and sintering processes.

Country/Int. Organization:

Russian Federation/Russian Federal Nuclear Center - Zababakhin All-Russian Scientific Research Institute of Technical Physics

6.3 Neutronics - 1 / 216

The APOLLO3 scientific tool for SFR neutronic characterization: current achievements and perspectives

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ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) is a Sodium Fast Reactor design that will be France's Flagship 4th Generation Reactor.
Its innovative core contains many axial and radial heterogeneities (in order to obtain a negative void coefficient) and interfaces that are challenging for current deterministic codes to simulate correctly. Hence there is the need for new improvements in modeling (3D simulations, parallel processing) like those being elaborated within the APOLLO3 platform.
The APOLLO3-SFR package built with APOLLO3 solvers defines reference calculation schemes associated with a nuclear data library to calculate all neutronic parameters (critical masses, sodium void, Doppler coefficient, βeff, etc…) together with certified biases and uncertainties derived from the VV&UQ process. This VV&UQ process incorporates numerical validation, a-priori uncertainties based on nuclear data covariances as well as experimental validation mainly from MASURCA, a fast mock-up reactor, located at CEA Cadarache. A future programme called GENESIS will be performed in support to the prototype ASTRID to validate the CFV core specificities. In addition, a part of the GENESIS experimental program contains integral experiment underway at the BFS facility.
The paper presents the various VV&QU activities which are currently conducted to derive all neutronic characteristics with a certified uncertainty.

Country/Int. Organization:

CEA Cadarache Center, 13108 Saint-Paul-lez-Durance, France

5.9 Large Component Technology II / 217

The development of a computer code for predicting fast reactor oxide fuel element thermal and mechanical behavior (FIBER-Oxide)

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The in-core behaviors of fast reactor oxide fuel element are highly complicated and coupled due to the overall irradiation, thermal and mechanical effects. The structural integrity of fuel cladding in steady and transient conditions is important for the safe operation of reactors. A computer code FIBER-Oxide is developed to simulate fuel element behavior and further to predict life. First, based on international open literatures, the main adopted material property models are presented. Second, according to solid heat transfer equation and mechanical equilibrium equation, numerical resolving equations for fuel pellet and cladding are derived and established. Finally, the general computation procedures are given and a sample is tested. The successful operation demonstrates the feasibility of adopted modeling methods. The future further development priorities for the code are also proposed. The development of FIBER-Oxide lays the foundation for the independent design of China fast reactor fuel element.

Country/Int. Organization:

China Institute of Atomic Energy

6.5 Uncertainty Analysis and Tools / 218

Evaluation of βeff measurements from BERENICE programme with TRIPOLI4® and uncertainties quantification

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The use of the Iterated Fission Probability method in the Monte Carlo code Tripoli4® gives credit to deterministic codes such as ERANOS for calculating βeff. The asset of Tripoli4® is the possibility to get a better representation of experimental cores, especially the R2 experimental core which exhibit more experimental canals for hosting large fission chambers. The BERENICE measurements campaign took place in the experimental facility MASURCA at CEA Cadarache with the two cores R2 reference and R2 experimental using enriched uranium fuel and one core ZONA2 using MOX fuel.
For JEFF3.2, the revised C/E ratios are of 1.2% ± 2.0% for the ZONA2 core and -1.2% ± 2.9 % for the R2 experimental core when using the Noise measurement technique.
The nuclear data uncertainty propagation has been leading to a 2.6% uncertainty for U-Pu core and 2.8% for enriched uranium cores with main contributors being the delayed neutron fission yield and the fission cross section of U238.

Country/Int. Organization:

CEA, Cadarache Center, France

6.5 Uncertainty Analysis and Tools / 220

Objectives and Status of the OECD/NEA sub-group on Uncertainty Analysis in Modelling (UAM) for Design, Operation and Safety Analysis of SFRs (SFR-UAM)

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An OECD/NEA sub-group on Uncertainty Analysis in Modelling (UAM) for Design, Operation and Safety Analysis of Sodium-cooled Fast Reactors (SFR-UAM) has been formed under the NSC/WPRS/EGUAM and is currently undertaking preliminary studies after having specified a series of benchmarks.
The incentive for launching the SFR-UAM task force comes from the desire to utilize current understanding of important phenomena to define and quantify the main core characteristics affecting safety and performance of SFRs. Best-estimate codes and data together with an evaluation of the uncertainties are required for that purpose, which challenges existing calculation methods. The group benefits from the results of a previous Sodium Fast Reactor core Feed-back and Transient response (SFR-FT) Task Force work under the NSC/WPRS/EGRPANS.
Two SFR cores have been selected for the SFR-UAM benchmark, a 3600MWth oxide core and a 1000MWth metallic core. Their neutronic feedback coefficients are being calculated for transient analyses. The SFR-UAM sub-group is currently defining the grace period or the margin to melting available in the different accident scenarios and this within uncertainty margins.
Recently, the work of the sub-group has been updated to incorporate new exercises, namely, the depletion benchmark, the control rod withdrawal benchmark, and the SUPER-PHENIX start up transient. Experimental evidence in support of the studies is also being developed.

Country/Int. Organization:

CEA, Cadarache Center
ANL, USA
GRS, Germany
KfKI, Hungary
IJS, Slovenia
OECD, NEA, France

5.10 Fuel Modeling and Simulation / 222

Current status and progression of GERMINAL fuel performance code for SFR oxide fuel pins

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A fuel performance code for SFR oxide fuel pins, GERMINAL, is developed by CEA within the PLEIADES simulation framework. The present main goal of GERMINAL is to meet the needs of the design studies of ASTRID, the future Advanced Sodium Technological Reactor for Industrial Demonstration in France.

Recent works have been conducted to improve the modelling of different physical mechanisms having a strong influence on the design criteria evaluation. Thus, the formulation of the fuel pellet fragments relocation model has been revisited, by introducing a dependence to the thermal gradient inside the pellet. The description of this mechanism represents a key point to evaluate the pellet-to-cladding gap closure and the margin to melting at beginning of life. Another evolution concerns the pellet-clad mechanical interaction. The ability to simulate a stronger interaction for fuel pins with a higher filling fraction has been acquired with a focused work on fuel mechanical behavior. A stronger mechanical interaction may also happen with lower power operating conditions and a cladding material remaining stable under irradiation. Moreover, the description of the thermo-chemistry of oxide fuel is currently being improved by coupling GERMINAL with the OpenCalphad thermodynamic calculation software. In doing this, the goal is to obtain a better prediction of the amount of volatile fission products being transported outside the fuel pellet, and then contributing to the "Joint Oxyde-Gaine"formation. With refined estimations of JOG volume and composition, we expect further to improve the evaluation of heat transfer through pellet-to-cladding gap at high burn-up, and also a more mechanistic description of cladding corrosion due to released fission products.

These works are based on a systematic comparison of calculation results to post-irradiation measures, by integrating progressively additional objects to our validation base. This process leads to a wider validity range targeting ASTRID design, and brings out new working perspectives.

Country/Int. Organization:

FRANCE / French Alternative Energies and Atomic Energy Commission (CEA)

5.10 Fuel Modeling and Simulation / **223**

3D SIMULATION IN THE PLEIADES SOFTWARE ENVIRONMENT FOR SODIUM FAST REACTOR FUEL PIN BEHAVIOR UNDER IRRADIATION

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In the framework of the basic design of ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) the GERMINAL fuel performance code is developed in the PLEIADES software environment. In order to improve one dimensional modelling of GERMINAL, a 3D simulation for the SFR fuel pin behavior under irradiation has been proposed.
The 3D model represents a single pellet fragment and its associated piece of cladding. The scale transfer between this single fragment model and the fuel pin scale is achieved through appropriate boundary conditions given by GERMINAL results. The 3D thermo-mechanical computation scheme is implemented in the LICOS code of the PLEIADES platform. In this approach, chemo-physical aspects are still computed by the GERMINAL code and are introduced in the 3D computation scheme as some input data in a two-step procedure. First studies have been achieved in order to analyze pellet-to-cladding gap closure mechanisms at the beginning of irradiation. Two mechanisms of fuel

relocation are simulated through the 3D simulation. The first one is linked to the hourglass shape of the fragmented fuel pellet under thermal gradient, and the second one is induced by the mass transfer due the central hole formation and fuel restructuration. According our results, the gap closure rate given by the GERMINAL empirical model can be understood. The 3D coupling formulation has now to be extended to the mass transfer equations in order to improve the results.

Country/Int. Organization:

FRANCE/CEA

5.2 Advanced Fast Reactor Fuel Development II / 224

Effects of Oxygen Partial Pressure During Sintering at Laboratory and Industrial Scales on FR MOX Fuels

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In order to prepare the industrial deployment of Sodium Fast Reactor (SFR), a French prototype is envisaged for 2025 (ASTRID: Advanced Sodium Technological Reactor for Innovative Demonstration). Its MOX (mixed oxide) fuels will be produced by a new industrial facility, currently under development and named AFC (for "Atelier de Fabrication des Coeurs", core fabrication facility).
The fabrication process of MOX fuel is based on powder metallurgy processes. The UO_2 and PuO_2 mixture is pelletized and then sintered at about 1700°C under reducing atmosphere of Ar/4%H_2/H_2O. Fuel has to be in compliance with specifications. In particular, the O/M (atomic oxygen to metal ratio) has to be hypostochiometric and close to 1.97 and the microstructure has to be dense, around 95 %ThD and free of cracks. The O/M and microstructure can affect numerous properties of the fuel during operation including thermal conductivity, mechanical properties and fuel-cladding interactions. To comply with these specifications, better knowledge of sintering at laboratory and industrial scale is needed.
An original analysis method has been therefore developed for a better understanding of the O/M ratio evolution and of densification mechanisms during the sintering step. By coupling a dilatometer with an oxygen zirconia probe, it is possible to identify the different redox phenomena and to plot the evolution of the O/M of the oxides versus time during the densification process. This innovative method helps overcoming the obstacles in reaching the thermodynamic equilibrium between gas and fuel. Whereas it was difficult to predict a precise final O/M, this new method produces the expected ratio every time.
This paper highlights the different final O/M values and microstructure, particularly in terms of microcracking, obtained during sintering in a continuous industrial or laboratory kiln. The impact of the evolution of moisture content in the gas is explained. Based on these results, recommendations can be made about the sintering atmosphere to improve industrial cycles and optimize fuel characteristics in order to obtain an O/M as close as possible to the target value and the right microstructure.

Country/Int. Organization:

CEA is the French national research center on nuclear fuel.

Poster Session 1 / 225

Impact of the irradiation of an ASTRID-type core during an ULOF with SIMMER-III

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Innovative Sodium-cooled Fast Reactors (SFRs) are currently investigated in the ESNII+ European project. The goal of the WP6 "Core safety"of this project is to support the development of the ESNII roadmap as well as the implementation of the ESNII deployment strategy and licensing of the ESNII systems by identifying the experimental and theoretical R&D activities which are necessary for improving the present designs, as well as the existing methods, tools and databases for static and transient safety analysis of the ESNII critical reactor cores.

One of the main issues of the WP6 "Core safety"of the ESNII+ project is to assess the behavior of the ESNII core (ASTRID-like core) in severe accidents at a representative stage, ie. the end of equivalent cycle (EOC), as the sodium voiding effect is less favorable at this moment. Consequently, the SIMMER-III code system (coupled thermohydraulics, pin mechanics and neutronics) is used as it can represent the accident up to an advanced core degradation. However, it has been developed to perform neutronics calculations at the beginning of life (BOL, without irradiation), and a new methodology needs to be implemented to perform neutronics calculations at the EOC.

The aim of this paper is to present the difference of behaviors of the ESNII+ core at BOL and EOC, so as to highlight the importance of the irradiation in the accident scenario. Thus, a new methodology developed in the framework of the ESNII+ project to perform neutronics calculations at EOC is presented. Then, Unprotected Loss Of Flow (ULOF) calculations, with a 30s primary flow-rate halving time, are performed at BOL and EOC. The sodium boiling and the pin degradation happen earlier at EOC, but the core degradation is slow in both calculations and there are no power excursions.
Despite less favorable feedback coefficients at EOC, and thanks to its heterogeneous geometry, the ESNII+ core in ULOF with a 30s halving time, does not lead, with the given hypotheses, to a power excursion.

Country/Int. Organization:

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Poster Session 1 / 226

The optimization of core characteristics of fast molten salt reactor based on neutron-physical and thermal-hydraulic calculations and the analysis of fuel cycle closure options

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The growing importance of nuclear energy in the overall balance calls for the further improvement of existing nuclear reactors. However, it is equally important that other reactor types are also considered. The report represents the concept of fast molten salt reactor, neutron-physical and thermo-hydraulic calculations for several core models. It also presents the analysis of different core configurations including cylindrical, elliptical, and the block model with partition of energy generation and energy transmission functions. For further study the choice of geometry has been made based on criteria developed. After calculation of neutron-physical characteristics the optimization of core geometry has been carried out. The result of calculation of effective fraction of delayed neutrons is given.
The modeling of nuclide composition evolution till steady-state operations has been carried out taking into account partial recycling of soluble and non-soluble fission products. Change in nuclide compossition of uranium, plutonium and basic minor actinides is given. Time to reach steady-state has been determined.
Based on obtained power density the thermo-hydraulic calculation was carried out. There have been determined maximal temperature of core structures, density profile and fuel salt velocity.

Country/Int. Organization:

RFNC-VNIITF named after academ. E.I.Zababakhin, Russia

Poster Session 1 / **227**

Towards a new approach for structural materials of Lead Fast Reactors

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The development of Pb and LBE cooled fast reactors represents a unique challenge for the materials that are subject, at the same time, to degradation mechanisms related to neutron radiation damage, exposure to high temperatures and exposure to the HLM. The protection towards HLM corrosion, in the past, has been based on the control of the oxygen concentration in the HLM by keeping the oxygen concentration between 10-5 and 10-6wt.% in the liquid metal and prevent the onset of severe oxidation. This approach presents technical difficulties in its application and poses the risk that the accumulation of oxide particles can prevent the heat exchange of refrigerant in critical parts of the reactor. The new approach bases the protection of structural materials on oxides more stable than chromium oxide in HLM and namely alumina. Two paths are under study to perform stable oxides protections on LFR structures and are thoroughly discussed: 1) steel coating by anti-corrosion barriers, 2) advanced austenitics. The first approach is feasible in the short term since several coating techniques already qualified in other contexts exist to protect the steels. Alumina coatings are a promising option for the protection of the fuel claddings and the R&D to demonstrate the feasibility of this option is in progress. The second approach require a long path to tune and qualify self passivating alloys for core applications where the features for swelling resistance have to be associated with those for HLM corrosion.
The development of radiation resistant steels has been based on the optimization of composition and thermo-mechanical treatments, lasted several decades, to improve the resistance to irradiation swelling and maintain suitable mechanical properties. However the advanced austenitics for reactor internals which are less exposed than the core to fast neutrons require less R&D and could be produced in a short time.

Country/Int. Organization:

Italy/ENEA (Agenzia Nazionale per le Nuove Tecnologie, l'Energia e lo Sviluppo Economico Sostenibile)

4.3 Partitioning and Sustainability / 228

DYNAMIC TEST OF EXTRACTION PROCESS FOR AMERICIUM PARTITIONING FROM THE PUREX RAFFINATE

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The fast reactors and related fuel cycle technologies are extensively development at present. It is possible to transmutate the radiotoxic nuclides, contained in PUREX raffinates (primarily minor actinides), in fast reactors. This raises the challenge of minor actinides recovery from PUREX raffinats.
The process for actinides (III) partition with solvent based on N,N,N′,N′-tetraocthyl-diglycolamide (TODGA) in meta-nitrobenzotrifluoride (F-3) was proposed. The process includes actinides (III) and REE co-extraction, Zr and Pd scrubbing, HNO3 scrubbing, selective actinides stripping with buffered DTPA solution and REE stripping. The dynamic test using mixer-settlers set-up was carried out. The feed solution contained about 4.5 g/L REE and trace amounts of 241Am. Not less 99,97 % of americium were recovered. Decontamination factor for the removal of REE from Am product was about 100. Most of the zirconium, molybdenum and palladium were in the raffinate. The testing and improving of the process will continue.

Country/Int. Organization:

Russia, Rosatom

Poster Session 2 / 231

Evaluation of data and model uncertainties and their effect on the fuel assembly temperature field of the ALFRED Lead-cooled Fast Reactor

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One of the crucial objectives for the Advanced Lead-cooled Fast Reactor European Demonstrator (ALFRED) is proving the viability of the general concept adopted in the design. This proof passes through the successful operation of ALFRED, demonstrating that the design assumptions provide not only the foreseen performances, but also the aimed reliability. The demonstration of the reliability can be rephrased stating that the margins assumed for the design must be proven to be well suited, in the sense they accommodate the uncertainties on the main technological constraints. This, indeed, was the aim of the task "ALFRED core safety parameters and influence of model uncertainties on transients" in the collaborative project "Preparing ESNII for Horizon 2020"(ESNII+), co-funded by the EU within the 7th EURATOM Framework Programme, where the first step, object of this work, was the evaluation of data, model, fabrication and measurement uncertainties and their effect on the fuel assembly temperature field.

The paper present first the identification of the various factors contributing to the overall uncertainty on the temperature field; then, the propagation of each effect, by means of the heat equations, so to retrieve the actual uncertainties on the parameters of interest (the temperatures themselves) and finally, a hot spot analysis to quantify the uncertainty-distorted temperature field. The hot spot analysis has been performed by means of a semi-statistical vertical approach –characterized by an optimal degree of conservatism among the classical approaches –targeting a 3σ (99.73%) confidence interval.

The performed analysis has highlighted the importance of fabrication tolerances, especially on the assessment of the coolant bulk temperature, and of data/models especially on the clad outer, gap and fuel temperatures evaluation, pinpointing the research areas where more efforts are needed.

The further step is the application of the present analysis to unprotected transients, combined with the uncertainties on the reactivity coefficients, so to gain a real insight on the safety performances and degree of forgiveness to be reckoned to ALFRED.

Country/Int. Organization:

Italy

6.2 Thermal Hydraulics Calculations and Experiments / 232

Extension to Heavy Liquid Metal coolants of the validation database of the ANTEO+ sub-channel code

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Among the numerous numerical methods available for preliminary design verification purposes, the sub-channel one has historically been the reference, thanks to its ability to cover the scale between CFD and system codes which is the one of particular interest for the core designer. Recently, the sub-channel code for liquid metal applications ANTEO+ has been developed by ENEA and a comprehensive validation performed, covering all the salient aspects of the fuel assembly thermal-hydraulic analysis like pressure drops, sub-channel and outer clad temperatures. Due to the database available at the time, the focus for the sub-channel temperatures validation was mainly related to sodium and sodium eutectics coolants. Thanks to the increasing interest in heavy liquid metal coolants for fast reactor applications, numerous experimental activities have been very recently performed (CLEAR-S, SEARCH) and many are still ongoing, enabling the extension of the previous ANTEO+ validation database so to make this tool even more persuasive for Generation IV reactor concepts applications

and mostly to estimate the uncertainty to recon to the code results. In the present work, ANTEO+ validation against the most recent experiments with heavy liquid metal coolant is presented: several tens of new experimental points have been considered in this campaign, covering a broad range of configurations which spans over the one of anticipated interest. The results of this validation activity have confirmed the good predictive ability of the code, notably when compared to other state of the art tools. Some criticalities have also emerged, especially to what concerns the sub-channels and pins close to the wrapper, which significantly modifies their thermal field; this has a particular impact on the Nusselt number, highlighting the lack, in the open literature, of a reliable correlation for the outer row of pins.

Country/Int. Organization:

Italy

Poster Session 1 / 233

ROUZ CODE: CFD APPROACH FOR ASSESSMENT OF RADIATION SITUATION DURING ATMOSPHERE RADIOACTIVITY RELEASES WITHIN AN INDUSTRIAL SITE

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In order to perform calculations for determination of zones of radioactive contamination within a particular urban area one needs to take into account the inhomogeneous wind field and essentially anisotropic turbulence.
According to the state-of-the-art trends in applied computational meteorology a robust CFD model has been developed in the IBRAE RAN in frame of "Codes of new generation"project included into the "BREAKTHROUGH"(or "PRORYV") project. The model, based on the numerical solution of Navier-Stokes equations via RANS approach, yields the realistic estimations of the levels of concentration depending on the geometry complexity, atmosphere stability, inhomogeneous turbulence and the actual wind profile.
The model has been realized in ROUZ, in which calculations of the doses received by population is performed taking into account the arbitrary geometry of a plume and the effect of buildings shielding. Application of parallel programming techniques allows increasing the performance of the dose calculation module by a few orders of magnitude.
Such codes are commonly used in such important spheres as safety assessment, estimation of consequences of terror attacks and city planning. A necessary requirement of applicability of such codes is their validation against experimental data.
The V&V matrix of the ROUZ code contains various data obtained from real emergency cases as well as gathered from experiments conducted in urban conditions and at industrial sites, for example, experiment in Oklahoma City. In addition the V&V matrix includes experiments conducted in laboratory conditions in wind tunnels on the flows around obstacles with geometry of various degrees of complexity, simulating different elements of urban area, and provided by the environmental wind tunnel laboratory of the University of Hamburg.
The ROUZ code calculation results have been compared with those obtained by the foreign analogues. Quality assessment of the results is based on quantitative criteria recommended by the international expert community (project COST 732). It has been obtained via statistical analysis of the results that the empirical probability of a more than twofold exceedance of level of concentration from a measured value is less than 0.22 and for all velocity components it is less than 0.13. Therefore, ROUZ code satisfies the quality acceptance criteria defined by the expert community for models of the same class.

Country/Int. Organization:

Russia, Nuclear Safety Institute (IBRAE RAN), Moscow

Poster Session 1 / 234

Multiscale computer modeling of nuclear fuel properties at radiation and thermal impacts

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Description and prediction of behavior of nuclear engineering materials under operating conditions is one of the challenging goals in actual materials science. To solve this problem, when the restricted experimental information is only available (e.g. for the new kinds of nuclear fuel), the most perspective method seems to be theoretical description based on multiscale approach. In this case the various subtasks are jointly solved on different various time and spatial scales using theoretical physics and computer modeling. The cooperation of different techniques (such as quantum calculations, atomistic simulation, dislocation dynamics, phase field modeling, kinetic equations and continuum mechanics) allows predicting behavior of the nuclear materials in the absence of experimental data in the analyzed range of temperature, fission rate and other external conditions.
In this work, we developed multiscale computer models for various types of nuclear fuel: UN, U-Mo, UO2. The work includes several stages of model development: development of a set of novel interatomic potentials; study of primary irradiation damage (collision cascades and radiation track); calculation of basic properties of matter (diffusion coefficients, dislocation and grain boundaries properties, phase transitions); mesoscale model for evolution of phase-structural composition and change of mechanical/thermodynamics properties under operating conditions.

Country/Int. Organization:

Russia/Nuclear Safety Institute of the Russian Academy of Sciences

Poster Session 1 / 235

The lead-cooled fast reactor transition to equilibrium operating conditions

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Lead-cooled fast reactors are promising direction in the development of nuclear energy because of opportunities to ensure their safety and security. Efforts are underway to develop lead-cooled fast reactors BREST-OD-300 and BREST-1200 in Russia. Nitride fuel is expected to use at the core of these reactors. High density of this fuel gives an opportunity to create an core with the breeding ratio close to unity. This makes it possible to realize the equilibrium condition with feeding of regenerated fuel only depleted uranium. The reactivity change between the refueling does not exceed the effective fraction of delayed neutrons in this case.
The use of earlier accumulated plutonium and newly emitted plutonium in the processing of thermal reactors irradiated fuel is assumed as initial core charge. In addition the initial charge of enriched uranium is considered. Stage reactor operating, in which the fuel composition of the transition from start-up to equilibrium may be accompanied by a number of problems associated, in particular, with the need to ensure a minimum change of the reactivity between the refueling. The report summarizes the features of work in the reactor at this stage. It is shown that the minimum time-to-equilibrium is achieved by using the initial charge of plutonium obtained from VVER reactor spent fuel.

Country/Int. Organization:

Russia/Russian Federal Nuclear Center - All-Russia Research Institute of Technical Physics

4.3 Partitioning and Sustainability / **237**

Hot test of technique separation of americium and curium

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Modern technology for reprocessing of spent nuclear fuel (SNF), corresponding to a closed fuel cycle concept is developed in the framework of the «PRORYV» PROJECT». This technology require separation of Am from Cm and REE. Americium can be transmutated in fast reactors.
Two-stage technology of Cm and Am separation from REE-TPE concentrate was tested on Mayak Production Association. Used concentrate was produced from reprocesing SNF WWER-440.
Tokem-308 cation-exchange resin was used in the final separation stage. The resin grain size was 0.2 mm. About 14 g of Cm was separate. 9 g of pure Cm fraction comtained 6 activity % of Am. The Cm-Am fraction contenined about 4,6 g 244Cm and about 40 g 241,243Am.
Content of Cm lower than 0.8 mass. % and 154,155Eu lower than 0.1 activity % in pure fraction of americium.
Keywords: separation Am-Cm, sorption, cation resin, reprocessing SNF, hydrometallurgy

Country/Int. Organization:

INSTITUTION «ITC «PRORYV» PROJECT», Moscow, Russia

5.6 Liquid Metal Technologies / **238**

Stainless Steels Corrosion in Sodium Fast Reactor: Feedback from Risks during Maintenance Operations (SCC in Caustic Solution

and Intergranular Corrosion by Acid Solution)

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Stainless steels are widely used in Sodium Fast Reactor and exhibit a very satisfactory feedback regarding their behavior in contact with high purity sodium.
The French past experience with Phenix and Superphenix confirmed this trend but it also highlighted that utilities have to take care at material susceptibility to different corrosion mechanisms during maintenance operations: Stress Corrosion Cracking (SCC) induced by caustic solution and Intergranular corrosion induced by acid solution used during maintenance operation.
In this paper, more than an overview of these mechanisms, the feedback and lessons learned from Phenix operation and maintenance experience will be presented as well as the present opportunity of materials investigations on components with Phenix dismantling. Finally, the precautions for ASTRID design and future operation will be highlighted.

Country/Int. Organization:

France EDF, CEA and Areva NP

Poster Session 2 / 240

Comparative analysis of nuclear energy lexicon

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The results of the analysis of terminological subsets of the characteristic lexicon, that is used for semantic identification of scientific and technical information in the field of fast neutron reactors is discussed. Describes the procedure of automatic formation of ordered dictionaries reference lexicon based on the full text processing of documents –scientific reports, dissertations, articles, technical documentation. The methods of linguo-statistical analysis of full texts stand out characteristic terms are represented by word expressions. Preliminary analysis showed that the vocabulary of documents, generated and used at different stages of the nuclear knowledge life cycle, differently correlated with the INIS thesaurus lexicon. Largest intersection (over 50%) has a vocabulary of documents relating to the research and educational activities. Minimum intersection (less than 10%) - the terminology of the operational documentation.

Country/Int. Organization:

National Research Nuclear University MEPhI (NRNU MEPhI)

Poster Session 1 / 241

A Conceptual design of engineering-scale plant applied the simplified MA-bearing fuel fabrication process

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Researchers at Japan Atomic Energy Agency (JAEA) have proposed the transmutation of minor actinides (MAs) by both fast reactors (FRs) and accelerator driven system (ADSs) as a way to contribute significantly to the reduction of the volume and the potential radiotoxicity of radioactive wastes. In order to achieve this goal, it is important to introduce a fully automated and remote operation fuel fabrication plant with shielded hot cells and manipulators to deal with extremely strong radiation dose and heat generation from MAs. JAEA's facilities including Plutonium Fuel Production Facility (PFPF) have fabricated MOX fuel. On the basis of the operational and technical experience obtained in above facilities, the conceptual design of engineering-scale plant applied the simplified MA-bearing fuel fabrication process with shielded hot cells and manipulator was done. It will be able to fabricate high MA-bearing fuel and to perform the maintenance and repairing of each equipment with manipulators. This plant will be constructed based on this concept and development plan.

Country/Int. Organization:

Japan/Japan Atomic Enerergy Agency

7.4 Fuel Cycle Analysis / 242

Performance Analysis of Various Thorium Fuel Options for the Sodium Cooled Fast Reactor

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Thorium is an attractive fuel option due to its inherent characteristics compared with U-Pu fuels. Less production of the long-lived transuranics with thorium fuel increases the proliferation-resistance and also reduces the high level radioactive waste in concern. Purpose of this study is to test and evaluate fuel performance and TRU transmutation performance of various design options with thorium fuel loaded in the core of sodium cooled fast reactor.
Evaluation was done for the modified core design concept of the Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR) which is a R&D reference model developed by Korea Atomic Energy Research Institute in Korea. Calculation was done with code package, TRANSX/DANTSTS/REBUS-3. Analyses was conducted on three fuel type categories (1) Oxide fuel; UO2, (Th,U)O2, (U,Pu)O2, (Th,Pu)O2, (U,TRU)O2, (Th,TRU)O2, (2) Metal fuel; U-Zr, Th-U-Zr, U-TRU-Zr, Th-TRU-Zr, and (3) Nitride fuel; (U,TRU)N. For reasonable comparison, all geometry and structure material, except smear density, had same size and same composition. Thorium and Uranium fuels were compared in each fuel type.

Because of the low conversion ratio, more than 20% enrichment was required in case of UO2 core. Therefore the fuel cycle length was decreased from 290 days to 190 days. Only UO2 fuel and (Th,U)O2 cycle length was changed.
Results showed that TRU fraction charged in Th-TRU-Zr fuel was higher than U-TRU-Zr fuel resulting in higher TRU consumption rate. As more TRU was charged in the core, BOC excess reactivity was increased. This influenced safety parameters for the Unprotected Transient Over Power accident. Neutron spectrum in all cores using oxide fuel was softened compared with metal fuels. In comparison of (Th,TRU)O2 and (U,TRU)O2, thorium oxide makes sodium void worth less positive and TRU consumption rate much larger than uranium oxide.

Country/Int. Organization:

Rep. of Korea

Poster Session 2 / 243

Preliminary Design of Zero Power Reactor for CEFR MOX Core

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Research on MOX fuel has lasted for a long time in China Institute of Atomic Energy (CIAE). Right now the focused topics are the manufacture and irradiation performance test for the China Experimental Fast Reactor (CEFR) MOX assembly, large batch production for CEFR MOX assembly and the CEFR core transition from Uranium fuel to MOX fuel. In order to determine the uncertainty of the CEFR MOX core design and improve the design codes and nuclear data, a zero power reactor using MOX fuel will be built based on an existed fast zero power reactor DF 6# in CIAE. The preliminary design is carried out by UK Monte Carlo code MONK, using MOX fuel rods replacing part of 90% uranium fuel rods. New DF 6# will be the first MOX ZPR and experimental research platform in China. Important neutronic parameters will be measured and validations will be done for the design methods of Fast Reactor design in China.

Country/Int. Organization:

China Institute of Atomic Energy

Poster Session 1 / 244

Experience on MOX fuel fabrication for fast reactor at PFPF

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Japan Atomic Energy Agency has developed mixed plutonium-uranium oxide (MOX) fuel fabrication technologies in large-scale and fabricated MOX fuel assemblies for experimental fast reactor "JOYO"

and prototype fast reactor "MONJU"at Plutonium Fuel Production Facility (PFPF) since 1988. Low density pellet is adopted as MONJU fuel. For the low density pellet fabrication in large-scale, various challenges were encountered. Typical examples of the challenges are as shown below;

1. Thermal degradation of organic compound used as pore former
2. Large standard deviation of pellet density due to inhomogeneous dispersion of pore former in granulated MOX powder

In order to resolve these challenges, countermeasures such as new pore former with high softening temperature and improved granulation method for MOX powder were considered.
In this presentation, accumulated MOX fuel fabrication technologies as mentioned above and recent R&D activity for low-decontaminated TRU fuel fabrication such as new pelletizing method, or die wall lubrication pelletizing, will be discussed.

Country/Int. Organization:

JAPAN/Japan Atomic Energy Agency

Poster Session 1 / 245

Thermal conductivity of non-stoichiometric (Pu0.928Am0.072)O2-x

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Am-bearing oxide fuel is considered as the fuel candidate for Fast Reactor to reduce the amount of high level radioactive waste. Thermo-physical properties of oxide fuel such as thermal conductivity and diffusion coefficient are very sensitive to the change of O/M ratio. Am is one of high produced minor actinide in irradiated nuclear fuel and its dioxide (AmO_{2-x}) has higher oxygen potential than PuO_2. Therefore, in the development of Am-bearing MOX fuel, it is very important to evaluate the effect of Am on thermo-physical properties. In this study, thermal diffusivity of $(Pu_{0.928}Am_{0.072})O_{2-x}$ was measured to evaluate the Am effect on thermal conductivity in the hypo-stoichiometric region.

Thermal conductivity of non-stoichiometric $(Pu_{0.928}Am_{0.072})O_{2-x}$ (x = 0.000 –0.058) was evaluated using experimentally measured thermal diffusivity by laser flash method, the bulk density and the heat capacity.

The obtained thermal conductivity was analyzed using oxygen potential data of $(Pu_{0.928}Am_{0.072})O_{2-x}$ to evaluate the effect of Am on thermal conductivity in the hypo-stoichiometric region. The oxygen potential data shows that Am and Pu in $(Pu_{0.928}Am_{0.072})O_{2-x}$ is reduced to trivalent over O/M = 1.964 (x = 0.036), and below O/M = 1.964, respectively. Therefore, it is considered that the effect of anion on thermal conductivity switches from Am to Pu, and the thermal conductivity discontinuously decrease at O/M = 1.964. However, the experimental results showed that thermal conductivity continuously decreased with reduction of the O/M ratio. This suggests that Am has almost comparable effect on the thermal conductivity with Pu in the hypo-stoichiometric region.

The O/M ratio dependence of thermal conductivity was well explained by the evaluation with the phonon transport model and slack model. This evaluation showed that the decrease of thermal conductivity was caused by the increase of oxygen vacancy and ionic radius differences between Pu^{4+} and Pu^{3+} or Am^{4+} and Am^{3+}. In addition, it is considered that Am has almost identical

effect on the thermal conductivity with Pu due to the similar ionic radius of Pu^{4+} and Am^{4+} or Pu^{3+} and Pu^{3+}.

Country/Int. Organization:

Japan Atomic Energy Agency

Poster Session 2 / 246

Investigation of the homogenization effect in sodium void reactivity in PGSFR

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Korea Atomic Energy Research Institute (KAERI) has been developing an SFR to aim at specific design approval of a Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR). In the PGSFR, a metal fueled, blanket-free, pool type SFR concept is adopted to acquire the inherent safety characteristics and high proliferation-resistance.
The metal-fueled SFR such as the PGSFR is known to be inherently safe at unprotected events due to the low operating fuel temperature and negative reactivity feedback mechanism. Several important tests of the EBR-II reactor support these characteristics based on a measurement of the integral reactivity. However, these inherent safety characteristics of a metal-fueled SFR depend on the uncertainties of various reactivity worth and reactor design. Hence, validation of the each reactivity worth, generated by the core neutronics design code system, is an essential work for specific design approval.
Validation of various reactivity worth, or in other words, validation of the core neutronics design code system can be divided by two parts: 1) validation of the cross-section and 2) validation of a modeling error. Validation of the cross-section will be finalized at 2017 based on the several physics experiments. In this paper, validation of the modeling error in a SVR (Sodium Void Reactivity) of the PGSFR core was examined by comparing PGSFR core design procedure (multi-group homogeneous MC2-3/TWODANT/DIF3D-VARIANT) and explicit Monte Carlo modeling (continuous-energy heterogeneous MCNP6) based on the ENDF-B/VII.0 library. SVRs were obtained by direct calculation in both of the MC2-3/TWODANT/DIF3D-VARIANT and MCNP6 calculations for core central and peripheral regions.

Country/Int. Organization:

Republic of Korea / Korea Atomic Energy Research Institute (KAERI)

Poster Session 2 / 247

Benchmark Analysis of EBR-II SHRT45R using MARS-LMR

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KAERI has joined the International Atomic Energy Agency (IAEA) coordinated research project (CRP) on Benchmark Analysis of an EBR-II Shutdown Heat Removal Test (SHRT). The major goal for this program is to validate MARS-LMR, which is a newly developed safety analysis code for PGSFR. One of benchmark tests is a SHRT-45R, which is an unprotected loss of flow test in an EBR-II. Thus, sodium natural circulation and reactivity feedbacks are major phenomena of interest. The EBR-II SHRT45R is analyzed using MARS-LMR. Overall prediction of the EBR-II SHRT45R by MARS-LMR shows good agreement with experimental results. Except the results of the XX10, the temperature and flow in the XX09 agreed well with the experiments. In addition, sensitivity tests are carried out for a decay heat model, reactivity feedback model, inter-subassembly heat transfer, internal heat structures and so on. The decay heat model of ANS-94 shows better results of fission power, however, the fission power is still over-estimated in the long-term transient region by the reactivity feedbacks. The inter-subassembly heat transfer is the most influential parameter, especially for the non-fueled XX10, which has a low flow and power subassembly. In addition, the appropriate internal heat structure model can be an influential parameter. Finally, the corrected results are proposed with reasonably conjectured parameters. This study can give the validation data for the MARS-LMR and better understanding of the EBR-II SHRT-45R.

Country/Int. Organization:

South Korea / Korea Atomic Energy Research Institute

6.9 Research Reactors / **248**

Development of Flow Identification Technology for the PGSFR Thermal Fluidic Design Validation

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Korea Atomic Energy Research Institute (KAERI) is currently developing the prototype SFR under the program of mid and long term project of the Korean government. Various experimental programs were selected for the validation of thermal fluidic design of the reactor system and design codes as the results of the PIRT. For a core thermal design, the assessment of the thermal margin is very important for the reactor safety where a friction coefficient, mixing factor, and pressure drop are important parameters having significant uncertainties in the model and correlations of the core thermal design code. The experimental database for core inlet flow and outlet pressure distribution are also essential for the evaluation of the core thermal margin. The identification of the pool side flow distribution including the pressure drop of the major flow path of the PHTS is important for the validation of the fluidic design of the reactor. The current validation program of the PGSFR design of our study includes a core subchannel flow experiment, a reactor flow distribution test and an IHX flow characteristic test. To identify the core subchannel flow rate and mixing characteristic, an iso-kinetic method, wire mesh and LIF technique were developed and optimized for the purpose of our experiment. For the identification of the reactor flow, the PHTS of the prototype plant was reduced to a 1/5 length scale in our test facilities with a preservation of the internal structures affecting the flow characteristics. The experimental techniques for each fuel assembly inlet flow rate and outlet pressure were developed. To validate the pressure drop correlations used in the design code for the IHX, a test facility was design for the flow characteristics of the shell side of the IHX with proper scaling approach. In this paper, a brief experimental technology for the identification of the reactor flow behaviors were described including the design feature of the test facilities and results of the finished experiment. The experimental database constructed in the current project will contribute to acquire the license of the core thermal and reactor fluidic design of the PGSFR.

Country/Int. Organization:

KOREA/Korea Atomic Energy Research Institute

6.2 Thermal Hydraulics Calculations and Experiments / 249

Thermal Hydraulic Study of Steam Generator of PGSFR

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In Prototype Gen-IV Sodium cooled Fast Reactor (PGSFR), integral once-through type, counter-flow shell-and-tube heat exchanger with straight vertical tubes was adopted for steam generators. Reliable operation of the steam generators has been a key issue through operating experience of foreign sodium cooled fast reactors because it is one of the most important components deciding the plant availability and reliability. Non-uniformity in sodium flow and temperature distributions might cause mechanical integrity problems such as tube buckling and tube-to-tube sheet junction failure in straight tubes. This work reports thermal hydraulic study on the sodium flow at the inlet plenum and the temperature distribution in the sodium-side of the PGSFR steam generator based on multidimensional numerical analysis. Optimization of porosity of distributers for achieving circumferentially uniform flow at the inlet plenum was carried out with the STAR-CCM+ CFD package. Then, the multidimensional sodium temperature distribution at tube bundle region was also calculated by the STAR-CCM+ CFD package. The heat flux from the sodium-side to the water-side was estimated using 1-D in-house code and supplied as boundary conditions at tube walls in the multidimensional CFD simulation. Iterative calculations between the STAR-CCM+ and 1-D in-house code were successfully conducted to acquire the radial and axial sodium temperature distributions under normal operation condition. The thermal hydraulic analysis results would be provided as input data to evaluate the mechanical structure integrity of the steam generator of the PGSFR.

Country/Int. Organization:

Replubic of Korea/Korea Atomic Energy Research Institute (KAERI)

5.8 Structural Materials / 250

Performance evaluation of ferroboron shielding material after irradiation in FBTR

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Ferroboron has been identified as a candidate material for in–vessel radiation shielding application in future Fast Breeder Reactors (FBRs) in India that can result in significant cost savings. Out-of-pile physical and chemical characterisation studies have established its neutron shielding property and long-term compatibility with 304L SS clad under sodium at the operating temperatures. An irradiation experiment was designed and carried out with the aim of establishing in-reactor performance of ferroboron shielding material over a target life-time of 60 years. Performance parameters such as

slumping of the Ferroboron column, generation of helium gas and extent of ferroboron –clad chemical interaction have been evaluated as a part of Post Irradiation Examinations.
The irradiation capsule was designed to cover a arange of temperature and flux combinations. The ferroboron powder was packed to a known density under high purity argon atmosphere. The capsule was subjected to detailed pre-irradiation checks. The irradiation was carried out in FBTR core to a total fluence corresponding to expected lite-time. After discharge from FBTR the ferroboron capsule was subjected to neutron radiography, released helium measurements and metallography.
Neutron radiography was carried out using an indirect imaging technique. The radiographs indicated that the slumping of the ferroboron stack is limited to a maximum of 1mm in a 100mm pre-irradiation stack height.
The quantity of helium released due to (n,α) reaction is an important paramter of interest since it results in pressure increase in the cladding. The quantity of helium gas released in three sub-capsules was measured. These tests revealed that the maximum internal pressure developed after irradiation is 0.16 Mpa. A few samples of ferroboron clad were extracted for cross-sectional metallography and microstructural examination was carried out to evaluate ferroboron-clad chemical interaction.
PIE of the ferroboron capsule irradiated in FBTR has indicated that this material has performed very well and is a suitable for deployment in future fast reactors. This paper will describe the various examinations carried out and the salient results obtained.

Country/Int. Organization:

India / Indira Gandhi Centre for Atomic Research

Poster Session 1 / 251

The influence of porosity on thermal conductivity of low-density uranium oxide.

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In uranium-plutonium mixed oxide (MOX) fuel fabrication, the mitigation of specifications on fuel design is considered from a viewpoint of improvement of economic efficiency. The pellet density of the MOX fuel for fast reactors is one of the important specifications in the fuel design and it fluctuates with changes in the properties of raw material powders. As the one of the mitigations of specifications, the tolerance expansion of the density specification is considered. In this consideration, it is necessary to confirm the applicability of the porosity correction equation to the thermal conductivity in a low density region. The relation among a porosity (p), a density (d) and a theoretical density (dth) is described as follows: p=1-(d/dth).
In this study, UO2 pellet was adopted as test specimen for the following reasons. The same porosity correction equation can be applied to the thermal conductivities of UO2 and MOX. The stability of the oxygen to metal (O/M) ratio of UO2 pellets in the thermal conductivity measurement is superior to that of MOX pellets. The specimens were prepared by a conventional powder metallurgy process and the densities of the specimens were adjusted in the wide range by using crystalline cellulose. The thermal conductivities of these specimens were measured and the applicability of the porosity correction equation to the thermal conductivity in a low density region was evaluated.

Country/Int. Organization:

JAPAN/Japan Atomic Energy Agency

5.2 Advanced Fast Reactor Fuel Development II / 252

Fission product and swelling behaviour in FBTR mixed carbide fuel

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The advantages of a fast reactor, especially one that uses Uranium-Plutonium Carbide as its fuel is well documented. Irradiation performance assessment of carbide fuels began with experimental irradiations in EBR-II, FFTF, HFR, Rapsodie and Phenix etc. India has the extensive experience with this type of fuel at the Fast Breeder Test Reactor at Kalpakkam that has been operating for over 25 years. The fuel has attained a peak burn-up of 155 GWd/t at linear heat rating of 400 W/cm, in a large number of fuel pins. Comprehensive post-irradiation examinations (PIE) at various stages up to this high burn-up have yielded a wealth of information on behaviour of mixed-carbide fuel under steady state operations. In this paper, selected recent results on fission product migration, gas release, fuel swelling behaviour and microstructural evolution of the mixed-carbide fuel will be presented. Results of the PIE towards analysing the cause of failure in a fuel pin are also discussed.
Axial distribution of fission products such as 137Cs and 106Ru in the fuel pins was assessed by gamma scanning. A steep increase in the fuel stack length was observed beyond 100GWd/t burn-up indicating onset of FCMI. Fission gas release in fuel pins after a burn-up of 155 GWd/t indicated relatively low gas release of 16%. Systematic change in the fuel-clad gap and cracking pattern was observed with increasing burn-up. Fabrication porosities present in the fuel was found to decrease with increasing burn-up indicating that the fuel swelling is being accommodated in the porosities.
Caesium axial distribution in the failed pin and some of the fuel pins adjacent to it in a failed sub-assembly irradiated at a lower linear power of 260 W/cm. Gas release behaviour showed contrasting trends with higher gas release in the pins adjacent to the failed pin and lower gas release in pins located farther away from the failed pin. The micrograph of the failed pin cross-section at the location of failure showed highly densified fuel region. Asymmetric circumferential cracking was observed, indicating non-uniform temperature around the pin resulting from the diameter increase and local bowing in the fuel pins. Clad carburisation was not observed. The performance assessment through PIE has provided valuable insights into the behaviour of the mixed carbide fuel and cause of failure in a fuel pin.

Country/Int. Organization:

India / Indira Gandhi Centre for Atomic Research

Poster Session 1 / 253

Development of Ultra Sub-size Tensile Specimen for Evaluation of Tensile Properties of Irradiated Materials

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The idea of using small specimens for mechanical testing had actually originated in the nuclear industry to cater to the irradiation material testing and reactor surveillance programs. Small or

miniaturized specimens ensure efficient use of available irradiation volume in nuclear reactors, reduce uncertainty in irradiation parameters due to flux and temperature variations and also reduce radiological hazard during testing.

A procedure for tensile testing employing a miniature tensile specimen called ultra sub-size (USS) tensile specimen carved out of a 10.0 mm diameter and 0.5mm thick disc sample has been developed and standardised. The geometry of the specimen was optimized using Finite Element Analysis (FEA) with the purpose of maintaining stress concentration in the fillet radius and gripping area equivalent to that in standard ASTM and sub-size tensile specimen. FEA was also employed to evaluate the allowable fabrication tolerances for gage width and thickness of USS by examining its effects on the stress strain curves obtained and comparing with that for standard ASTM and sub-size tensile specimens.

USS specimens along with ASTM standard and sub-size specimens were tested on a range of fast reactor structural materials for comparison of mechanical properties. Due to difficulty of employing extensometer on a small gage length of 3.0mm, digital image correlation (DIC) was employed for strain measurement. The strain obtained through DIC was co-related with that obtained from the cross-head displacement of the UTM. Online strain distribution was extracted from DIC images to study nature of strain distribution over the USS specimen during tensile test and was compared with that obtained from standard specimens. The results obtained from tensile testing of ultra sub-size specimen at ambient and elevated temperatures were found to be consistent and comparable with those obtained from standard specimens for a wide range of alloys examined. The scatter obtained in the UTS, yield strength and uniform strain values evaluated from USS specimen is comparable to that obtained from standard and sub-size samples. The uncertainty in the YS and UTS values from USS specimen were evaluated and compared with that of the ASTM standard and sub-size specimens. The results of this study show that USS tensile specimen geometry can be reliably employed for mechanical property evaluation of irradiated structural materials. Further efforts are required to formulate standards through round robin testing, before the technique can be deployed in the field.

Country/Int. Organization:

India / Indira Gandhi Centre for Atomic Research

Poster Session 1 / 254

Evaluation of irradiation-induced point defects migration during neutron irradiation in modified 316 stainless steel

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For the development of nuclear core materials, especially fuel cladding tube, in sodium-cooling type fast breeder reactor, void swelling suppression is one of the most important issues to keep the dimensional stability in reactor. A large number of theoretical and experimental investigations on void swelling behavior have been carried out, and void swelling directly depends on the diffusion of point defects induced by neutron irradiation as well as the strength of point defect sinks such as dislocations and precipitates. Evaluations of the point defects diffusion in metal during neutron irradiation have been qualitatively done through various researches, however the quantitative estimation is hardly performed due to the difficulty of in-situ experiments during neutron irradiation. Instead of that, the indirect estimations from the temperature dependence measurements of dislocation loop

densities and growth rates using electron in-situ observation are often carried out, but the irradiation correlation between electron and neutron irradiations, such as the differences of irradiation dose rate and damage morphology, should be discussed with accuracy.

Therefore, the evaluation of point defects diffusion, especially vacancy migration, during neutron irradiation by the other method was tried in this study. In detail, from already neutron-irradiated microstructures, vacancy migration energy was estimated using the knowledge that void denuded zone (VDZ) widths formed near random grain boundaries depend on temperatures. The test material was PNC316 steel, which is the modified 316 stainless steel with cold-working and additives to improve the void swelling resistance. The fuels assemblies composed of PNC316 steel were irradiated in the experimental fast reactor JOYO. For the PNC316 specimens cut from these assemblies, which were irradiated at temperatures from 722 K to 821 K and doses of 74.5–87.5 dpa, VDZ widths were analyzed from the transmission electron microscope observations and the temperature dependence was investigated.

As the result, VDZ widths increased with increasing temperature. From the Arrhenius plots of VDZ widths and the reciprocal temperatures, the vacancy migration energy during neutron irradiation in PNC316 steel was quantitatively estimated to be about 1.4-1.5 eV. As vacancy migration energy in Fe-Cr-Ni model alloy is about 1.05 eV, the value of PNC316 steel implies that the vacancy mobility is low as a result of interaction of vacancies with minor alloying elements.

Country/Int. Organization:

Japan/Japan Atomic Energy Agency

6.5 Uncertainty Analysis and Tools / 255

Sensitivity and Uncertainty Analysis in Best-Estimate modeling for PGSFR Under ULOF Transient

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In this research, uncertainty analyses for multiple safety parameters were performed for Unprotected Loss of Flow (ULOF) for the Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR) by using the PArallel Computing Platform IntegRated for Uncertainty and Sensitivity analysis (PAPIRUS). The objective of the global uncertainty analysis is to evaluate all safety parameters of the system in the combined phase space formed by the parameters and dependent variables. The uncertainty propagation was performed by mapping the uncertainty bands of the model parameters through the MARS-LMR to determine the distributions for the fuel centerline, cladding, and coolant temperatures. The Best Estimate Plus Uncertainty (BEPU) analysis adopted for uncertainty quantification of the code predictions has been performed through a statistical approach where the Figure of Merit (FOM) is evaluated multiple times by using several combinations of parameters that are randomly generated according to their distributions. The statistical approach of uncertainty quantification is known to be very powerful for estimating response distributions, but sometimes inapplicable owing to demanding calculation requirements. In this research, Wilks'formula was used to estimate the 95% probability value of the FOM from a limited number of code calculations. This paper also introduces the application of data assimilation in best-estimate modeling to improve the prediction of the reactor system performance by refining various sources of uncertainties through model calibration technique. An inverse problem was formulated based upon Bayes theorem and solved to estimate the posteriori distributions of parameters.

Country/Int. Organization:

South Korea/Korea Atomic Energy Research Institute

Poster Session 1 / 256

Mechanical and Thermal Properties of (U,Pu)O2-x

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Designing nuclear fuels and simulating their irradiation behaviors in a reactor require modeling and formulation of a variety of fundamental properties. The property study of uranium-plutonium mixed oxide (MOX) as fast reactor fuels still requires further investigation because of the diverse parameters and the technical obstacles of plutonium operation.
Young's modulus of MOX pellets was evaluated by measuring the sound velocities of longitudinal and transverse waves in the pellets as functions of porosity, oxygen-to-metal ratio (O/M) and plutonium content. The effect of each was fitted to give a single equation, which is important in designing nuclear fuels and simulating their irradiation behaviors in a reactor. The results showed that porosity was the most important factor that 20% of the porosity decreased Young's modulus by neatly 100GPa while O/M and plutonium content could change the Young's modulus by ~20GPa.
From the measured sound velocities, temperature dependence on Young's modulus and specific heat capacity were calculated on the Debye model by leveraging the thermal expansion data. The temperature dependence that Young's modulus decreases with increasing temperature is in good agreement with literature data. The specific heat capacity also agrees with that of calculated value by Kopp's method, taken the Schottky term and the excited term into account. The relationship between mechanical and thermal properties was well described.

Country/Int. Organization:

Japan Atomic Energy Agency

4.3 Partitioning and Sustainability / 259

Pyrochemical recycling of the nitride SNF of fast neutron reactors in molten salts as a part of the short-circuited nuclear fuel cycle

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The scientific and technological aspects of the pyrochemical recycling of uranium-plutonium nitride nuclear fuel used in the circuited nuclear fuel cycle on the basis of nuclear power plant with fast neutron reactors were considered (in the framework of "Breakthrough" Project). It was expected that

the developed pyrochemical technology allows recycling spent nuclear fuel with high energy release (1 year of cooling) and returning of uranium, plutonium and some minor actinides into a reactor.
A special attention was paid to the initial process stages, including the nitride SNF component dissolution and ionization of actinide and lanthanide nitrides using the method of "soft chlorination" by transition metal salts.
The conditions to prevent uranium and plutonium nitride chloride formation were determined, and they were checked and proved by the laboratory and industrial tests.
The corrosion studies of the constructional chromium-nickel alloys in chloride electrolytes containing uranium and lanthanides salts were performed. The studies revealed that the alloys are suitable for production of electrolytic cells and other supporting apparatus operating with molten salts.
The complex investigation of physicochemical and electrochemical properties of salt compositions imitating practicable electrolytes for SNF recycling as well as actinide and lanthanide separation with the acceptable separation factor was carried out.
The conclusion was drawn on the technological feasibility of the pyrochemical technology realization in a closed nuclear fuel cycle on the basis of nuclear power plant with fast neutron reactors using mixed uranium-plutonium nitride nuclear fuel.

Country/Int. Organization:

Russia / Institute of High Temperature Electrochemistry

Poster Session 1 / 260

ELECTRICAL CONDUCTIVITY OF MOLTEN LiCl-KCl EUTECTIC WITH COMPONENTS OF SPENT NUCLEAR FUEL

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Pyroelectrochemistry is one of the most prospective approaches for spent nitride nuclear fuel (nitride SNF) reprocessing. It includes the anodic dissolution of the SNF pellets into the molten LiCl-KCl eutectic with the subsequent electrochemical separation of U, Pu, Np, Am from other constituents. Physical-chemical properties of such complex melts are still insufficiently studied.
The electrical conductivity of a number of quasi binary melts (LiCl-KCl)eut., containing CeCl3, NdCl3, UCl3, as well as CsCl and CdCl2, are studied in detail in the present work. The majority of measurements were performed in the whole concentration range and in the wide temperature range from the liquidus point to 900 - 920 0C. The electrical conductivity of a number of 3-4 component (LiCl-KCl)eut. - CeCl3 - NdCl3 - UCl3 mixtures was measured.
The density of the melts under study was evaluated and their molar conductivity was calculated. The liquidus line of these salt systems was built using the polytherm breakpoints. These data are required to create a general model for the electrolyzer operation and to develop complied technological equipment.

This research was partially supported by the Russian Ministry of Education and Science through Targeted Federal Program (project number: 14.607.21.0084).

Country/Int. Organization:

Russia / Institute of High Temperature Electrochemistry

Poster Session 1 / 261

CORROSION OF 12X18H10T STEEL IN Ce-, Nd- AND U-CONTAINING MOLTEN LiCl-KCl EUTECTIC

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The present work is aimed at the study of the 12X18H10T steel corrosion in the molten LiCl-KCl eutectic, which contains different proportions of CeCl3, NdCl3 and UCl3.
The CeCl3 and NdCl3 concentrations varied within the interval of 0.2-5.0 mol.%, and the UCl3concentration varied within the interval of 1.0-2.5 mol.%. The temperature of the experiments was 500 0C. The composition of the melts under study was close to the composition of real electrolytes, which appear at the nitride SNF processing.
The basic method of study is the gravimetric method with the exposure time from 24 to 100 hours. Atomic-adsorption, micro X-ray spectral and X-ray structural methods were used for samples analysis.
The first component of selective dissolution in the steel under study was Fe. Chromium and manganese dissolution degrees were the smallest. The presence of UCl3 in the melt was found to have the largest impact on corrosion. The corrosion rate is rather small. For example, in the melt containing 1 mol.% of NdCl3 the corrosion is equal to 1.93 g/(m2·h), and in the (LiCl-KCl)eut. + 1%CeCl3 +1%NdCl3 + 1%UCl3 melt it is 2.61 g/(m2·h).
The corrosion mechanism was found to be electrochemical.

Country/Int. Organization:

Russia / Institute of High Temperature Electrochemistru

Poster Session 2 / 263

CFD investigation of thermal-hydraulic characteristics in a SFR fuel assembly

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The wire effect on three-dimensional flow field and heat transfer characteristics in a helically wrapped fuel assembly mock-up of an SFR (Sodium-cooled Fast Reactor) have been investigated through a numerical analysis using the commercial CFD (Computational Fluid Dynamics) code, CFX. The SFR system has a tight package of the fuel bundle and a high power density. The sodium material has a high thermal conductivity and boiling temperature than the water. That can make core design to be more compact than LWR (Light Water Reactor). The fuel assembly of the SFR system consists of long and thin wire-wrapped fuel bundles and a hexagonal duct, in which wire-wrapped fuel bundles in the hexagonal duct has triangular array. The main purpose of a wire spacer is to avoid collisions between adjacent rods. Furthermore, a vortex induced vibration can be mitigated by wire spacers. The wire spacer can enhances a convective heat transfer due to the secondary flow by helically wrapped wires.

In this study, complicated and separated flow phenomena in the fuel assembly without wire spacer and with wire spacer were captured by a RANS (Reynolds-Averaged Navier-Stokes) flow simulation

with the SST (Shear Stress Transport) turbulence model, and by the vortex structure identification technique based on the critical point theory.

It is concluded that the wire spacers locally induce a tangential flow by up to about 13 % of the axial velocity. The tangential flow in the corner and edge sub-channels is much stronger than that in the interior sub-channels. The flow with a high tangential velocity is periodically rotating in a period of wire lead pitch. The cross flow due to the wire spacer can achieve to enhance heat transfer characteristics up to about 50 %.

Country/Int. Organization:

Korea / Korea Atomic Energy Research Institute

Poster Session 1 / 264

STATISTICAL INVESTIGATION OF RADIATION-INDUCED POROSITY IN BN FUEL CLADDINGS USING SCANNING ELECTRON MICROSCOPY

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Radiation-induced swelling of claddings is one of the factors limiting service life of fast reactor fuel assemblies. Hydrostatic weighing and transmission electron microscopy are conventional porosity investigation methods.
Hydrostatic weighing is advantageous in terms of the determination radiation porosity integral characteristics, specifically cladding dimensional change after operation in the reactor. However, at low swelling levels (tenths of a percent) the method accuracy is comparable to the measurement error, and its application at the swelling initial stage is ineffective.
Transmission electron microscopy is used to determine quantitative characteristics of radiation porosity, such as size and concentration of radiation-induced voids. One of the main disadvantages of the method is its locality and complexity of sample preparation. High resolution of transmission electron microscope allows to observe even small voids (1 nm) in the foil up to 150 nm thick. Nevertheless, observation of voids exceeding the foil thickness in diameter is complicated, and the reliability of quantitative assessment of large void concentration is quite low.
The paper aims to apply scanning electron microscopy having both method advantages in the radiation porosity investigation. Significant areas of the examined surface together with detection of voids from 10 nm give statistically representative data. Therefore it is possible to obtain information on radiation porosity macroscopic nonuniformity in structural elements subjected to the examination.
The paper presents methodological aspects of radiation porosity investigation with scanning electron microscopy: different modes of sample surface preparation and identification of electron beam optimal parameters. SEM and TEM quantitative results are compared, and radiation porosity nonuniformity in claddings is demonstrated.

Country/Int. Organization:

Joint Stock Company "Institute of Nuclear Materials"

Poster Session 1 / 265

Change in Mechanical Properties of Spent Fast Reactor Claddings

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During fuel element operation changes in structure and physical and mechanical properties of the claddings are induced by irradiation and other factors. In particular, there are such changes as swelling, embrittlement, softening and corrosion damages. To predict cladding limit state it is necessary to know the changes occurred. In particular, it is important that cladding mechanical properties after operation are properly determined. In this respect there are different mechanical tests. Tensile test of annular cladding specimens is a conventional method. Test stress strain state essentially differs from that occurring in claddings during operation when they are subjected to gas pressure and deformation from the swelling fuel. Mechanical properties determined with annular sample tensile test are too conservative and cannot be used for a proper description of the cladding behaviour during operation.
At JSC "INM" a technique for mechanical testing with tough plastic aggregate internal pressure has been developed. Aggregate compression leads to its plastic deformation exerting internal pressure on the cladding tubular sample. Mechanical properties of the cladding material are calculated according to the recorded 'movement of aggregate compressing plungers - compression force' curve. During the test a loading pattern and a stress strain state of the cladding simulate its loading under irradiation in the reactor. Characteristics of tubular samples tested with internal pressure clearly demonstrate operated cladding behaviour.
The paper shows the results of the short-term mechanical properties change after irradiation in fast reactors obtained for the mentioned techniques. Mechanical characteristics after testing in different loading patterns are compared. Advantages and disadvantages of each technique and possibility of their integration to predict cladding behaviour during operation are pointed out.

Country/Int. Organization:

Russian Federation

6.8 Experimental Facilities / 266

Sodium testing of fast reactor components

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Fast reactor components are required to work in hostile atmosphere of sodium, high temperature and radiation. Sodium testing of components before installation in reactor is essential for design validation and safe operation of the reactor. Components like Transfer arm, Inclined fuel transfer machine, Control and Safety Rod Drive Mechanism (CSRDM) and Diverse Safety Rod Drive mechanism (DSRDM), Under sodium ultrasonic scanner (USUSS) and Air Heat Exchanger of Safety Grade Decay Heat Removal system etc were sodium tested.

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In vessel purification for the primary sodium in future Indian fast breeder reactors is envisaged du
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An Under Sodium Ultrasonic Scanner (USUSS) has been developed for scanning the above core plenum to 
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Integral testing of Transfer Arm was carried out in sodium at the fuel handling temperature o
Leak collection trays are provided below the secondary sodium pipe of PFBR. Functionality of c
Two new sodium test facilities have been designed and are being erected for conduction of vari
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Country/Int. Organization:

India / Indira Gandhi Centre For Atomic Research, Department of Atomic Energy, Kalpakkam 603102 Tamilnadu India

2.1 Commissioning and Operating Experience of Fast Reactors I / 267

Development of under sodium viewer for next generation sodium-cooled fast reactor

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Inspection technique in opaque liquid metal coolant is one of important issue for sodium-cooled fast reactor. To facilitate operations and maintenance activities, various under sodium viewers (USV) has been developed in several research institutes and countries. For example, a horizontal USV, which detects obstacles on the long distance and an imaging USVs, which make images from a short distance and to a middle distance were developed. In this study, the imaging USV from a middle distance, approximately 1 m, was developed. The USV of this study adopts the optical receiving system which measures the vibration of displacement diaphragm by the laser as the receiving sensor. This study mainly focused on the improvement sensitivity in the transmission sensor and the receiving sensor. In addition, the imaging experiment in the water was conducted by using the developed transmission sensor and receiving sensors. From the experimental results, it was confirmed that the developed USV sensors can make imaging with high resolution from 800 mm distance.

Country/Int. Organization:

Japan / Japan Atomic Energy Agency

Poster Session 1 / 268

Neutronic Self-sustainability of a Breed-and-Burn Fast Reactor Using Super-Simple Fuel Recycling

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The breed-and-burn fast reactor (B&BR) is a unique concept of fast reactor, which can breed the fissile fuels and use the bred fuels in situ. Thanks to this characteristic, the fuel utilization in a B&BR can be extremely high and even the spent nuclear fuel of a B&BR can be re-used as a fuel to spawn the next generation B&BR after appropriate reprocessing or reconditioning. In this paper, a super-simplified melt and treatment (SSMT) process, which removes volatile elements only from the spent fuel, is suggested to enhance the proliferation-resistance and economy of the reprocessing. The neutronic feasibility of B&BR self-sustainability with SSMT is studied in terms of the burnup reactivity change, conversion ratio, core lifetime, power profiles and safety parameters. The fuel and core design was optimized to maximize the self-sustainability while preserving the inherent proliferation-resistance of the core.

Country/Int. Organization:

Republic of Korea/Department of Nuclear & Quantum Engineering, KAIST

7.1 Sustainability of Fast Reactors / 269

Current Status of Next Generation Fast Reactor Core & Fuel Design and Related R&D in Japan

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The next generation fast reactor is being investigated in Japan, aiming at several targets such as "safety", "reduction of environmental burden"and "economic competitiveness". As for the safety aspect, FAIDUS (fuel assembly with inner duct structure) concept is adopted to avoid re-criticality in core destructive accidents. The uranium-plutonium mixed oxide (MOX) fuel, in which minor actinide (MA) elements are included, will be applied to reduce the amount and potential radio-toxicity of radioactive wastes. The high burn-up fuel is pursued to reduce fuel cycle cost. The candidate concept of the core and fuel design, which could satisfy various design criteria by design devisals, has been established. In addition, JAEA is investigating material properties and irradiation behavior of MA-MOX fuel. JAEA is developing the fuel design code especially for the fuel pin with annular pellets. Furthermore, JAEA is developing oxide dispersion strengthened (ODS) ferritic steel cladding for the high burnup fuel.

Country/Int. Organization:

Japan/ Japan Atomic Energy Agency

Poster Session 2 / 270

Computational Analysis Code Development for Emergency Heat Removal of Pool-style Fast Reactors

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For the lacking of applicable code to analyze emergency heat removal capacity of pool-style fast reactors, it is developed according to the design requirement of demonstration fast reactors. The code builds a consolidated platform developing modules of reactor core, sodium pool, relevant components and so on individually from one-dimension to three-dimension, forming program module packages (in which the reactor core contains an inter-wrapper flow model, and relevant verification experiments are also arranged); in order to meet application requirements at different stages of the engineering design, the corresponding modules can be selected to perform coupling calculation. The code can be used for computational analysis of various means of emergency heat removal including inter-wrapper cooling.

Country/Int. Organization:

China/China Institute of Atomic Energy

Poster Session 2 / 271

Concept of multifunctional fast neutron research reactor (MBIR) core with metal (U-Pu-Zr)-fuel

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Multifunctional fast neutron research reactor MBIR is intended to provide the basis for broad scope of research and experimental activities due to achieving 5E15 n/cm2/s integral neutron flux in central loop channel, i.e. MBIR is considered as neutron source of high intensity.
This work focuses on the possibilities of the use of different fuel in MBIR to provide required neutron flux in loop channels and experimental cells without perturbation of the core design. Three options has been investigated: 6 mm diameter fuel pin with vibrocompacted MOX-fuel, with metal (U-Pu-Zr)-fuel and gaseous sub-layer and fuel pin of decreased diameter (5.5 mm) with metal (U-Pu-Zr)-fuel and sodium sub-layer. In the last option the fuel pin structure has been slightly changed: gap filled with sodium resulted in gas plenum arrangement in top part of the fuel pin. In all options considered fuel assemblies contain the same amount of fuel pins and keep their sizes across-flat and height.
It is shown that metal fuel with gaseous sub-layer option enables to decrease reactivity drop over reactor lifetime and keeps acceptable neutron flux in central loop channel. Metal fuel with sodium sub-layer option provides reactivity drop and neutron flux in central loop channel at the level of as-designed vibrocompacted MOX-fuel core option.
Thermo-hydraulic simulation shows that pin cladding temperatures in all three core options meet criteria of reliable heat removal provided design coolant flow rate. Moreover, 5.5 mm fuel pin option provides secure fuel assembly cooling for wide range of power density distribution in the core in case of the linear heat load is limited by 48 kW/m.
Stress-strain behavior analysis shows that metal fuel with sodium sub-layer and MOX-fuel options demonstrate appropriate stress-strain conditions of the fuel pin during burnup. Taking into account irradiation embrittlement of austenitic steels under high fluence and circumferential strain exceeding 2%, tangential stresses developed in cladding of MOX and metallic fuel pins don't exceed maximum permissible value of 200 MPa up to damage dose of 80 dpa.
The studies have shown the use of metallic fuel in the pins of decreased diameter in the MBIR core meets design requirements on operability and extends the reactor experimental performance for end user due to improved neutron balance and enlarged thermal-hydraulic margins.

Country/Int. Organization:

Russian Federation, National Research Center "Kurchatov Institute"

Poster Session 1 / 272

Use of ion irradiations to help design of advanced austenitic steels

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CEA develops new austenitic and ODS alloys to limit the swelling and withstand very high doses. In this study, 10 Austenitic Stainless Steels with different content of phosphorus, nickel, silicon, titanium and niobium were elaborated and irradiated with iron ions at 600°C in several metallurgical conditions. Different effects on void swelling were observed by Transmission Electron Microscopy (TEM).
The presence of dislocations, nano-precipitates and solutes in the matrix modify the size and the density of the cavities which appear during the irradiation. Results help to design new alloys optimized regarding the swelling resistance.

Country/Int. Organization:

CEA

Poster Session 2 / 273

Development of innovating Na leak detector on pipes

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Within the ASTRID reactor project, CEA, EDF and AREVA, have launched a R&D program focused on the low leak rates detection of sodium on pipes. This program is focused on the development of innovating detectors, multilayer-type and Optic Fiber, involving tests in the FUTUNA-2 sodium loop. This loop is designed to produce very accurate sodium leak rates within a range around 1 cc/min, the tests being performed at different temperature (up to 550°C) on large-diameter pipe mock-ups (DN 800) at ambient atmosphere.
This paper presents the first series of tests carried out with various materials of the first and second layer of the detector. The results are compared and discussed as well as the observations made after removing the mock-ups. The most interesting result of the overall tests is a detection time less than 2 hours for the two types of detectors.

Country/Int. Organization:

FRANCE/ CEA (Commissariat à l'Energie Atomique et aux Energies alternatives)

6.1 CFD and 3D Modeling / 274

Applications of the DNS CONV-3D Code for Simulations of Liquid Metal Flows

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For the simulation of the thermalhydraulic processes in fast reactors with liquid metal coolant DNS CFD code CONV-3D has been developed.
This code has ideal scalability and is very effective for calculations on high performance cluster computers.
The code has been validated on the set of analytical tests and experiments in a wide range of Rayleigh and Reynolds numbers, in particular, at extremely small Prandtl numbers. The paper presents the results of the application of CONV-3D code for simulation of sodium natural convection in the upper plenum of the MONJU (Japan) and BN-600 (Russia) reactor vessel. A satisfactory agreement of the numerical predictions with experiments is demonstrated. The calculation results of the experiment conducted on the Phenix facility (France) with sodium coolant are demonstrated. The experiment focuses on the mixing of two fluxes at different temperatures in the secondary circuit of reactor facility with liquid metal coolant in the presence of a bending tube. A small pipe is connected via T-connection to the main pipe and unloads of sodium in the main pipe at a temperature which is higher than in the main pipe. A satisfactory agreement of the numerical predictions with experiments and commercial codes is demonstrated, in particular for the temperature distribution vs the coordinates.
The results of simulation of heavy-liquid metal (LBE) flow and heat transfer along a hexagonal 19-rod bundle with wire spacers (KALLA, Germany) are presented. A convergence on a sequence of grids and convergence with the experiment is demonstrated.
The results obtained allow to conclude that using of CONV-3D code with high predictive power can be recommended for reactor applications.

Country/Int. Organization:

Russia/State Atomic Energy Corporation "ROSATOM"

1.4 CORE AND DESIGN FEATURES - 1 / 275

Physics Investigation of a Supercritical CO2–cooled Micro-Modular Reactor (MMR) for Autonomous Load-Follow Operation

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This paper presents a physics study for a passive autonomous load-follow operation in a supercritical CO2-cooled micro-modular reactor (MMR). The proposed long-life 36.2 MWth MMR is a super-compact, fully-integrated, and truck-transportable fast reactor module in which all components are integrated in a single pressure vessel. UC fuel is considered to maximize the fuel inventory and to enhance neutron economy. The core lifetime is designed to be over 20 years without any refueling. To minimize the excess reactivity, a replaceable fixed absorber (RFA) is used and the resulting excess reactivity is found to be less than 1 $ during the whole lifetime of the core. For demonstration of a passive autonomous operation of the MMR, analyses for the reactor startup from initial CZP to HFP and daily load-follow operation are performed. In a passive autonomous load-follow operation, the reactor power is automatically controlled by the feedback reactivity only, which mainly depends on the fuel and coolant temperatures. For favorable passive autonomous load-follower operations, the core expansion feedback reactivity is also taken into account in this paper. All neutronics calculations are performed using the continuous energy Monte-Carlo Serpent code with the ENDF/B-VII.1 library.

Country/Int. Organization:

KOREA / Department of Nuclear and Quantum Engineering,
Korean Advanced Institute of Science and Technology (KAIST)

Poster Session 1 / 276

Possibility studies of a boiling water cooled traveling wave reactor

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This paper investigates the possibility of a boiling water cooled traveling wave reactor, which can improve the natural uranium utilization of a Light Water Reactor (LWR) by umpteen times. The high density variation of boiling water through the core is favourable to fission at the lower part of the core and to the fuel breeding at the upper part of the core. This is the case in a low pressure Boiling Water Reactor (BWR). A serial axial fuel shuffling, which makes fuel moving, is considered. The natural uranium oxide fuel is fed in from the top of the core and discharged from the bottom of the core, as the water at the saturation point is fed in from the bottom of the core as in a boiling water reactor. The asymptotic state of the breeding/burning wave is searched theoretically and numerically, where the power (neutron flux) and the water density are fitted to each other to form a fission-breeding mixed reactor configuration. The major parameters of power, coolant mass flow rate, and the fuel shuffling speed are coupled to each other and determined by numerical solutions. A theoretical model for the water boiling is established based on a slip ratio model of two phase flow. The critical heat flux limit has been taken into account. The neutronics and burn-up calculations are performed with the ERANOS2.2 code, where models of axial fuel shuffling and coolant density change have been implemented. The 1-D preliminary numerical results are encouraging and show that the breeding is sufficient to make the core be critical and the maximum burn-up can reach up to 40%. The more detailed analyses and larger benchmarking efforts should be aimed further as discussed in the paper,. Also safety concerns will be addressed, in particular related to the sign and magnitude of the coolant density coefficient. Moreover a 1-D diffusion model is set up based on numerical neutron fluence and macroscopic neutron cross-sections. The solitary breeding-fission wave is obtained. The wave length of the neutron flux wave length can be understood. This implies that only a sufficient long core is needed for operating such a reactor instead of the fuel shuffling. Further discussions on clad irradiation damage and coolant feedback coefficients will be done.

Country/Int. Organization:

Germany, Karlsruhe Institute of Technology

Poster Session 2 / 277

Inspection specifications leading to extended ASTRID Design rules

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CEA initiated a study in 2008 for improving design rules of Fast Reactors, with French utilities (EDF), French Designers (AREVA) and Non Destructive Examination (NDE) specialists (Aix Marseille University), and dealing with the specific aspect of in-service inspection (ISI).
Thus, at the end of 2012, RCC-MRx specifications for NDE code could be enlarged, extending those performed at manufacturing stage to periodic inspection.
Due to the complexity of the links between design, material, access, inspection techniques and tools, these rules cannot be strict instructions but rather lead to a fruitful dialogue between Designers and Controllers.
Further work on the links between in service inspection (NDE) and manufacturing processes and specifications is now in progress. This article will present the approach and R&D program in support of this specific work.
This initiative should lead to a better connection and compromise between design work, material specification and in service inspection.

Country/Int. Organization:

France/CEA

2.2 Commissioning and Operating Experience of Fast Reactors II / 278

OPERATING EXPERIENCE OF FBTR

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The Fast Breeder Test Reactor (FBTR) is sodium cooled, loop type fast reactor commissioned in the year 1985. It is a research reactor with an evolving core with a unique Pu/U carbide fuel. The reactor has continued to provide valuable experience to improve its performance figure year after year. FBTR has been operated at different power level up to 27.3MWt/5.8MWe over the last thirty years, realizing its objectives viz mastering sodium cooled fast reactor technology and testing future reactor materials.
Based on the performance of the Mark-1 fuel, its burn up limit has been increased in steps and attained maximum burn up level of 155GWd/t completed 25 irradiation campaigns. Primary sodium temperature nearer to rated design value with reduced reactor power level was achieved by operating the Steam Generator with three out of seven tubes blanked.
This paper describes the operating experiences of FBTR starting from commissioning and successful

operation and various problems encountered in different systems / areas during the last thirty years viz; Spurious reactor trips due to noise pickup, Experience with failed fuel localization, Fuel handling incident, Spurious trips from Steam generator Leak detection System, Leaks from Biological concrete shield cooling system, Positive Reactivity transients, Problems with Control Rod drive Mechanism & Core cover plate Mechanism, Dropping of orifices from once- through -Steam Generator, Detection and management of sodium leak incidents.

Country/Int. Organization:

INDIA/INDIRA GANDHI CENTRE FOR ATOMIC RESEARCH

2.2 Commissioning and Operating Experience of Fast Reactors II / 279

R&D status on in-sodium ultrasonic transducers for ASTRID inspection

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In Service Inspection of the sodium cooled fast reactor prototype ASTRID leads to a large R&D effort for selecting, developing and qualifying ultrasonic techniques and tools.
Several ultrasonic transducers are developed and tested: TUSHT model (CEA), TUCSS model (AREVA) both based on piezoelectric material, and Electro Magnetic Acoustic Transducers (EMAT, CEA). Each type of transducer presents advantages and drawbacks regarding the various ultrasonic applications considered for ASTRID: sensor location, ultrasonic techniques ("contact"or "immersion", bulk or guide waves…), temperature, and the nature of measure (telemetry/vision or defect detection).
This article describes development and qualification programs that are currently performed. This program aims at ultimately selecting the most appropriate transducer for each ASTRID ultrasonic application. They involve simple targets and elaborate ones (with defects to be detected or specific shapes to be measured/seen), in-water tests and 200℃ in-sodium tests. Associated simulation is also performed, using CIVA software platform.

Country/Int. Organization:

France / CEA, AREVA, EDF

Poster Session 2 / 280

the simulation of reactor physics for China Experimental Fast Reactor

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China Experimental Fast Reactor (CEFR) is the first fast neutron breed reactor in China, which is different with PWR. In order to research the operational performance of CEFR, the real-time simulator was developed. The simulation of core physics is an important part of the simulator.
The neutron dynamic model used in the simulator is three dimensions and four groups neutron diffusion model, which was solved by the improved quasi-static approximation node method. The neutron flux was divided into shape function and amplitude function. The shape function changes slowly with time, so a large time step is adopted. And the amplitude function changes quickly with time, so a small step is adopted. The calculation time can be saved, it is important for real-time simulation. According the character of CEFR, the core was divided into many nodes. The homogenization parameters of each node were calculated by HELIOS. Considering influences of fuel burnup, fuel temperature and coolant temperature on fuel assembly cross section, four-order polynomial is adopted for fitting.
Because there are hexagonal fuel assemblies in CEFR core, the calculation of leakage term was modified based on the pressurized water reactor calculation program. The improved alternative direction implicit (ADI) algorithm is used to solve diffusion equations. The simulation result indicates that the improved algorithm is able to meet requirements for the real-time simulation.
Two steady states (BOL and EOL) were simulated. And some dynamic operation cases were simulated, including reactor star-up and a control rod drawing out of core without control. Compared with the Final Safety Analysis Report for CEFR, the three-dimensional power distribution and control rod value are in good agreement. The core physics simulation program is able to use the operation research of CEFR.

Country/Int. Organization:

Harbin Engineering Universtiy,China

3.6 Safety Analysis / 281

SOCRAT-BN integral code for safety analysis of NPP with sodium cooled fast reactors: development and plant applications

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The SOCRAT-BN integral code has been developed in frame of Federal Target Program «New-Generation Nuclear Power Technologies for the Period 2010–2015 and up to 2020». The code is intended for safety justification of NPPs with sodium-cooled reactors under DBA and BDBA including those that lead to core melting.
The modules integrated into SOCRAT-BN code allow to perform coupled simulation of thermal-hydraulics, neutron physics, thermal-mechanics state of fuel and cladding, melting and displacement of core materials, accumulation and transport of fission products in the coolant loops and environment. The report represents description of the basic code modules and its V&V results on the base of out-of-pile and in-pile experiments.
The SOCRAT-BN code is used for the safety justification of NPP with sodium cooled reactors. In particular, the calculations of the following hypothetical accidents have been performed: unprotected loss of flow through the core (ULOF), unprotected positive reactivity insertion (UTOP) and total inlet flow area blockage of single subassembly (TIB). The report presents the analysis of the simulation results.

Country/Int. Organization:

Russia

1.7 ADS AND OTHER REACTOR DESIGNS / 282

Study for Accelerator-driven System in J-PARC/JAEA

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The management of radioactive waste is one of the critical issues for sustainable nuclear energy application especially after the Fukushima accident. In the latest strategic energy policy of Japan express to enhance a research and development to reduce the burden of long-lived nuclides in spent nuclear fuel using both fast reactors and accelerator-driven systems (ADS). Japan Atomic Energy Agency (JAEA) proposes a transmutation of minor actinides (MA) by ADS. A lead-bismuth eutectic alloy (LBE) is used as a spallation target and a coolant of subcritical core because LBE has a good spallation neutron production performance and a chemically inert characteristic. However, the compatibility with steels is unfavourable for the typical structural materials such as a 316 stainless steel. To obtain the data for ADS design, JAEA plans to construct the Transmutation Experimental Facility (TEF) within the framework of the J-PARC project, which consists of two buildings, an ADS Target Test Facility (TEF-T) and a Transmutation Physics Experimental Facility (TEF-P).
A 250kW LBE spallation target will be installed in TEF-T to prepare the irradiation database for candidate ADS structural materials in flowing LBE environment. Engineering tests for LBE loop operation and experiments to determine the effective lifetime of proton beam window will be also performed. Spallation neutrons from LBE target will be used for multi-purpose applications. A critical/subcritical assembly with a certain amount of MA fuel will be set up in TEF-P to perform the neutronic experiments for MA-loaded core.
To realize both TEF-T and TEF-P, various studies are being carried out. Test loops for the TEF-T LBE target were manufactured and are ready for operation. One is a loop for TEF-T target mock-up and the other is that for collection of material corrosion characteristics in flowing oxygen controlled LBE environment. Sensor systems for LBE flow and oxygen potential have been also developed. Remote handling tests for TEF-T LBE target loop maintenance are underway to fix a design of the loop and the spallation target trolley. Basic tests to handle MA-bearing fuel have been performed and low power proton beam injection is under preparation.
The activities to realize the TEF, a roadmap to establish the ADS transmutor and latest elemental test results for TEF construction will be introduced.

Country/Int. Organization:

J-PARC Center, Japan Atomic Energy Agency, Japan

6.7 Experimental Thermal Hydraulics / 283

Thermal-hydraulic experiments supporting the MYRRHA fuel assembly

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The development of the LBE-cooled MYRRHA fuel assemblies is supported by an extensive thermal-hydraulic experimental program, as well as numerical studies. For the safety assessment of the reactor, several experimental campaigns considering fuel assembly mockups in representative operating conditions have been completed, and others are ongoing and planned at KIT (Germany) and SCK•CEN (Belgium). These are individually focused on specific issues, such as the heat transfer and pressure drop in nominal conditions, effects of local blockages and their formation, and influences of inter-wrapper flow between neighboring fuel assemblies. Heated tests using LBE, as well as isothermal studied with water as a model fluid, are considered.
This article summarizes the main results of completed projects, , highlighting the accuracy of existing correlations, and the relevance of hot spots based on local temperature distribution both at the wall and in the fluid. Moreover, the status of ongoing work is presented and the main open thermal-hydraulic issues for supporting the development of the MYRRHA fuel assembly are identified.

Country/Int. Organization:

Germany and Belgium

3.7 Core Disruptive Accident Prevention / 284

Optimization of Passive Safety Devices FAST and SAFE for Sodium-cooled Fast Reactors

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This paper presents two novel passive safety devices for Sodium-cooled Fast Reactors (SFR): SAFE (Static Absorber Feedback Equipment) and FAST (Floating Absorber for Safety at Transient) to deal with the positive coolant void reactivity (CVR) and coolant temperature coefficient (CTC). It is well-known that the positive CVR and CTC limit the maximum performance of a SFR. Especially, CVR and CTC become more positive as the core average burnup of U-loaded SFR increases. Both FAST and SAFE can be easily introduced into an SFR core by replacing some fuel pins or control rods without any complicated core design changes. In this study, the optimum configurations of FAST and SAFE devices in an innovative Sodium-cooled Fast Reactor (iSFR), which is a small (393 MWth) and long-life (>20 years) SFR, are studied in terms of safety parameters, transient responses, and core lifetime.

Country/Int. Organization:

Republic of Korea/Korea Advanced Institute of Science and Technology

1.3 SYSTEM DESIGN AND VALIDATION / 285

Progress in the ASTRID Gas Power Conversion System development

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Within the framework of the French 600 MWe Advanced Sodium Technological Reactor for Industrial Demonstration project (ASTRID), two options of Power Conversion System (PCS) were studied during the conceptual design phase (2010-2015):
- the use of a classical Rankine water-steam cycle, similar to the solution implemented in France in Phenix and Superphenix , but with the goal of greatly reducing the probability of occurrence and limiting the potential consequences of a sodium-water reaction; chosen as the reference for the ASTRID Plant Model during the conceptual design phase due its high level of maturity,
- an approach which has never been implemented in any Sodium Fast Reactor using a Brayton gas cycle. Its application is mainly justified by safety and acceptance considerations in inherently eliminating the sodium-water and sodium-water-air reaction risk existing with a Rankine cycle.
The ASTRID conceptual design phase period allowed to greatly increase the maturity level of a standalone Gas Power Conversion System option. It has been thus decided to lay during the 2016-2017 phase the ASTRID Gas PCS integration studies at the same level as that achieved by ASTRID Water based PCS at the end of 2015.
The 2016-2017 period, in which the Gas PCS is integrated in the overall layout of the reactor, will allow to better specify the technical and economic implications of the selection of gas PCS taking into account all the aspects of the integration of such an option. A well-documented comparison between the two systems will be therefore facilitated.
This paper resumes progress in the integration of the Gas Power Conversion System in the Astrid Reactor Plant Model. It describes the main characteristics defined particularly on the Balance of Plant (BOP), the turbomachinery, the Sodium Gas Heat Exchangers (SGHE) as well as expected performances, operability and safety analysis.

Country/Int. Organization:

CEA CADARACHE / FRANCE

5.5 Large Component Technology I / 286

ASTRID French SFR: Progress in Sodium Gas Heat Exchanger development

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Within the framework of the French 600MWe Advanced Sodium Technological Reactor for Industrial Demonstration project (ASTRID), a Gas Power Conversion System (PCS) based on a Brayton cycle is studied. This innovative option has never been implemented in any Sodium Fast Reactor and is mainly justified by safety and acceptance considerations in inherently eliminating the sodium-water and sodium-water-air reaction risk existing in Steam Generators with a Rankine cycle.
The present work describes the current status of the design of an innovative compact Sodium Gas Heat Exchanger (SGHE) and highlights the industrial challenges this technology raises.
This paper presents the details of the design of the SGHE which allows a high thermal compactness. The main studies supporting the development are described whether on the external pressure vessel or on the compact internal heat exchanger modules; the thermal hydraulic program demonstrates the potential of the technology used whereas the thermo mechanical analyses show the good behavior of this exchanger under the ASTRID operating conditions.
The manufacturing welding process optimization for the heat exchanger modules is ongoing in order to produce a component with nuclear specifications. Specific sensors and control techniques are also being developed in order to assess the manufacturing process quality and to allow future in-service inspections.
At last, the qualification program is presented and the results obtained on an operating small scale SGHE mock up (DIADEMO) working under ASTRID conditions are described.

Country/Int. Organization:

CEA CADARACHE / FRANCE

Poster Session 1 / **287**

Thermal and elastic properties of CexTh1-xO2 mixed oxides: a self-consistent thermodynamic approach

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Nuclear fuel based on mixture of thorium dioxide ThO2 with uranium or plutonium is perspective for many types of breeder reactors, e.g. liquid metal cooled fast breeder reactors (LMFBR), advanced heavy water reactors (AHWRs), gas cooled reactors (HTGR), etc. [1]. Effective and safe usage of these fuels requires information on its thermal and mechanical properties. In the case of PuxTh1-xO the available experimental data on these properties is very scarce, probably due to high radioactivity of plutonium. Therefore, in place of PuO2, its surrogate CeO2 is often used since the physicochemical properties of these two compounds are similar. In the present study we investigate temperature dependencies of the heat capacity, volumetric coefficient of thermal expansion, bulk modulus and thermal conductivity of CexTh1-xO2 systems by means of a self-consistent thermodynamic approach. This approach incorporates the impact of anharmonicity of both the acoustical and optical phonon modes.
[1] International Atomic Energy Agency, Thorium Fuel Cycle-potential Benefits
and Challenges, IAEA-TECDOC-1450, IAEA, Vienna, 2005.

Country/Int. Organization:

Russian Federation/Ural Federal University

1.6 CORE AND DESIGN FEATURES - 2 / **288**

The ASTRID core at the end of the conceptual design phase

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Within the framework of the French ASTRID project, core design studies are being conducted by the CEA with support from AREVA and EDF. The design studies include the GEN IV reactor objectives, particularly in terms of improving safety.

Options selection was performed at the conclusion of the pre-conceptual design phase. The CFV core was confirmed as the reference core for the ASTRID project. The design routes of the core has be reoriented for the conceptual design phase of the ASTRID project :

- Limitation of the core diameter,
- Innovative options of control and shutdown architecture : control and safety absorber rods used tomanage the core reactivity during the cycle,
- Introduction of complementary safety device for prevention and mitigation of severe accidents,
- Choice of S/A internal storage instead of external storage, Neutron shielding on the Inner vessel components.

At the end of the ASTRID conceptual design phase (2015), a new evolution of the CFV core (CFV V4) which integrated these above options was designed. This paper will describe the CFV V4 focusing core performances, behavior during unprotected transients and experimental validation programs.

Country/Int. Organization:

France, CEA

Poster Session 1 / 289

The study of U-232 accumulation in reprocessed uranium for fast reactor fuel cycle

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One of the main objects of fast reactor nuclear fuel cycle radiation safety is fuel assembly handling. In case of closed nuclear fuel cycle fresh fuel assemblies will be produced from regenerated uranium and plutonium.

Uranium-232 is produced and accumulated in fuel assemblies during the irradiation. One of the U-232 decay products is Tl-208 which emits high energy gamma radiation. In addition, uranium-232 can't be chemically separated from reprocessed uranium. Thereby, the uranium-232 content in reprocessed fuel is very important for fuel cycle radiation safety.

The main ways of uranium-232 production are (n,2n) and (n,3n) reactions on several nuclides. Their contribution to U-232 production depends on their initial content in the fuel. These reactions have neutron energy threshold about 1 MeV.

The difficulty of calculating uranium-232 accumulation is caused by threshold reactions cross sections uncertainties. The evaluation of these cross sections in different libraries can vary by an order or even more.

The paper presents the results of the study into the effect of reaction cross section uncertainties in some modern nuclear data libraries on uranium-232 content and dose rate for reprocessed uranium in fuel assemblies. Fuel cycle scenarios with different fuel compositions, irradiated fuel cooling and fresh fuel storage before irradiation time are considered.

Country/Int. Organization:

JSC "SSC RF –IPPE", Obninsk, Russia

3.4 Sodium leak/fire and other safety issues / 290

Evaluation of multiple primary coolant leakages accidents in Monju with consideration of passive safety features

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To maintain sodium level inside reactor vessels above cores is essential to keep core-cooling for sodium leakage accidents in loop-type sodium-cooled fast reactors. In the loop-type prototype fast reactor Monju which has three primary heat transport systems (PHTSs), a single coolant leakage accident in a PHTS has been taken into account as a design basis accident (DBA). On the other hand, it is important to investigate that another primary coolant leakage would occur after the first coolant leakage accident as a design extension condition (DEC).
In this presentation, we evaluate multiple primary coolant leakages accidents in Monju with consideration of passive safety features. Concretely speaking, the flow rate and the amount of leakages can be reduced by the effect of the decrease of the cover gas pressure due to lowering reactor coolant level (negative pressure effect). The sodium coolant level necessary for the decay heat removal can be maintained, taking account of the negative pressure effect and other measures.

Country/Int. Organization:

JAPAN/Japan Atomic Energy Agency

Poster Session 1 / 291

The way of nitride fuel producing by high voltage electrodischarge compaction

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Modern development of nuclear power industry is closing of nuclear fuel cycle. That is possible when fast neutron reactors is used. An attractive fuel for this type of reactor is nitride nuclear fuel. However, the widespread use of a nitride nuclear fuel encounters by problems of its manufacturing and, in particular, during the sintering of the finished powder material. So for the manufacture of tablets of uranium nitride traditionally use free sintering technology. To achieve the desired density (85-95%, etc.) tablets require exposure to high temperatures (1900 –20000 C) for several hours, as well as the use of additional preliminary steps: powder granulation and pressing at high pressures (up to 1, 5 GPa). The whole process is quite time, energy and labor-intensive. To solve the problem the method of high-voltage electric pulse consolidation is proposed. This method consists of passing a short (up to 1 ms) high-voltage discharge (up to tens of kV) power of several kW directly through the powder. The result is an instantaneous sintering of powder materials. The main advantage of this method is the sintering times and lower temperature processes, which affect both the quality of the final product (smaller grain size and preservation of the original phase composition) and the overall unit energy consumption during the manufacture of the product. General regularities of the influence of the parameters of high-voltage electric pulse consolidation density of uranium nitride compacts are obtained. It is shown that the density of the compact is linearly dependent on the voltage on the capacitor bank (pulse energy) in the investigated range of energies. Depending on the application of pressure is increasing with the saturation curve. Compacts with densities of 85 - 96% are obtained. The peculiarities of formation of macro and microstructure, as well as the phase composition of the samples are studied.

Country/Int. Organization:

National Research Nuclear University MEPhI (Moscow Engineering Physics Institute)

7.3 Non Proliferation Aspects of Fast Reactors / 294

ASTRID - An original and efficient project organization

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CEA is the contracting authority and industrial architect of ASTRID Project, an industrial prototype of 4th generation Sodium Fast Reactor. This reactor of 600 eMW is integrating French and international SFRs feedback, especially in domains of safety, operability and ultimate wastes transmutation. The project is funded for basic design phase (2016-2019) through France Future Investments Program. The industrial network is made of bilateral agreements between CEA and fourteen industrial partners. Main Keys of the success are the followings:

- Industrial companies chosen in their core area of excellence,
- Partnerships with co-funding and involvement in strategic decisions rather than commercial contracts,
- Strong, flexible and efficient R&D program in support of ASTRID design international agreements with Europe, Japan, Russia, India, USA, China,

- Early discussions with regulatory authorities,
- Strategic and Operational managements,Technical control with Engineering System tools and 3D mock-up consolidation.
- CEA has created a specific entity: Astrid Project Cell (CPA for Cellule Projet ASTRID in French): in charge of creating and ruling an efficient project management. It acts as the industrial architect of the project.

Country/Int. Organization:

French Atomic Energy Commission,
Nuclear Energy Division,
Reactor Studies Department,
ASTRID Project Team.

7.2 Economics of Fast Reactors / 296

How to take into account the fleet composition in order to evaluate Fast Breeder Competitiveness

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Fast reactor competitiveness is usually examined by comparing fast reactors and light water reactors (LWRs) LCOE (levelized cost of electricity).
As fast reactors have an investment cost higher than LWRs, their kWh production cost is higher than that of LWRs (with the natural uranium current price) and their competitiveness will thus take place when the increase of the natural uranium cost will be enough to counterbalance their additional investment cost.

In fact, the interest of the fast reactors is to allow the implementation of sustainable nuclear power fleet not consuming natural uranium, but only depleted uranium which we have, unlike the natural uranium, considerable resources.
The real objective is not to build a reactor not consuming natural uranium, but to have a sustainable nuclear fleet consuming depleted uranium only.

For this purpose, fast reactors are necessary, but it doesn't imply that the whole fleet will be made of fast reactors only. It is a possibility, but not the only one.

Indeed, these break-even(isogenerator) reactors can become fast-breeder reactors (FBRs) by using blankets and available plutonium surplus can be used in other reactors consuming plutonium, but not consuming natural uranium. For example, we can build a fleet including fast-breeder reactors but also LWRs like EPR with a 100 % MOX load. The different shares of the two reactor types will be defined by the balance between plutonium produced in FBRs and plutonium consumed in LWRs.

By doing this, the additional investment cost of FBR is diluted because it does not concern more than a part of the sustainable fleet (as a matter of fact the cost of the kWh produced by a100 % MOX EPR is not very different from that of the UOX EPR with a current natural uranium cost).

Considering the effect of such a fleet including both FBR and 100% MOX LWR this study suggests that increasing the breeding ratio for FBR and increasing the conversion ratio of LWR by considering high conversion ratio LWR could be economically efficient.

This study shows that a corrective factor depending on the fleet composition should be taken into account in the FBR overcosts in order to examine its competitiveness.

Country/Int. Organization:

France

4.1 Fuel Cycle Overview / 297

Concurrent Trends in Indian Fast Reactor Fuel Reprocessing Programme

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The Indian fast reactor program, which began with the construction of the mixed carbide fuelled Fast Breeder Test Reactor (FBTR) at Kalpakkam, has reached a level of maturity with three decades of operating experience and is stepping into the realm of commercial operations with the construction of Prototype Fast Breeder Reactor (PFBR). The necessary technology for closing the fuel cycle, which is vital for the success of the fast reactor program, has been concurrently developed facing the unique challenges posed by the fast reactor fuel. CORAL (COmpact Reprocessing of Advanced fuels in Lead cells), a pilot facility, has been operating successfully since the year 2003, reprocessing the spent fuel discharged from FBTR, with burnup upto 155 GWd/t and very short cooling periods as low as 18 days. This facility has served the purpose of validation of the process as well as the equipments that were developed for fast reactor fuel reprocessing. Operating the facility has given valuable feedback for the Demonstration fast reactor Fuel Reprocessing Plant (DFRP), which will be a regular reprocessing plant for FBTR and also serve as a demonstration for the reprocessing of mixed oxide fuel from PFBR. The CORAL experience was also vital in designing the Fuel Reprocessing Plant (FRP) of Fast Reactor Fuel Cycle Facility (FRFCF), which would be a regular reprocessing plant for spent fuel discharged from PFBR. Considerable experiences gained and feedback obtained in design and operation of the reprocessing facilities provided vital inputs for achieving the required robustness in the fast reactor fuel reprocessing program. With the construction of FRFCF, the Indian fast reactor fuel reprocessing program would step in to the realms of commercial reprocessing. R&D efforts are also concurrently under progress to develop efficient processes and equipments, aqueous processes for reprocessing of the U-Pu-Zr metallic fuel as a fall-back for the pyro-chemical process.

Country/Int. Organization:

INDIRA GANDHI CENTRE FOR ATOMIC RESEARCH,
DEPARMENT OF ATOMIC ENERGY,
INDIA

1.2 SFR DESIGN & DEVELOPMENT - 2 / 298

Progress of Design and related Researches of Sodium-cooled Fast Reactor in Japan

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In Japan, we have implemented the development of a sodium-cooled fast reactor from the viewpoint of severe accident measures in order to strengthen safety of a fast reactor since the Great East Japan Earthquake. This paper describes the progress of design study and research and development (R&D) related to safety enhancement and severe accident measures. For the purpose of strengthening of decay heat removal function, we are performing R&D and development of test facilities on the decay heat removal after core disruptive accident (CDA), the application of a variety of heat removal system, and the evaluation methods for thermal hydraulics. In order to elucidate the behavior of molten fuel during CDA, we are conducting the in-pile and out-of-pile tests by international collaboration, the basic experiments, and the development of evaluation methods for CDA. Also, we have promoted the improvement of core design from the viewpoint of preventing the occurrence of severe accident.

Country/Int. Organization:

JAPAN

6.6 Coupled Calculations / 299

Neutronics Experimental Verification for ADS with China Lead-based Zero Power Reactor

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Abstract:
An Accelerator Driven System (ADS) project for nuclear waste transmutation has been launched by Chinese Academy of Sciences (CAS) since 2011. China LEAd-based Reactor (CLEAR) was selected as the reference reactor for the CAS ADS project and was designed and developed by Institute of Nuclear Energy Safety Technology (INEST), CAS. According to the research and development roadmap of CLEAR, a 10MWth lead-bismuth cooled pool-type research reactor named CLEAR-I coupled with a proton accelerator will be constructed at the first stage.
In order to verify the nuclear physics performances and the coupling techniques for the ADS system, a multifunctional lead-based zero power reactor (CLEAR-0) has been built. A brief introduction on the design objective, experiment functions and system description for CLEAR-0 is given in this paper. The recent R&D progress on core design and coupling system is also presented. The first stage of CLEAR-0 is scheduled to be finished for construction and will be commissioning in the end of 2016, series of core characteristic experiments will be carried out in CLEAR-0. The testing data will be used to validate the calculation method, program and database used in the nuclear design, and also to support the safety analysis and license application for CLEAR-I.
Keywords: Accelerator Driven System (ADS); Zero Power Reactor; China Lead-based Research Reactor; CLEAR-0

Country/Int. Organization:

China/Institute of Nuclear Energy Safety Technology, CAS ·FDS Team

1.1 SFR DESIGN & DEVELOPMENT - 1 / 300

Advanced Design Features of MOX Fuelled Future Indian SFRs

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India has been operating a Fast Breeder Test Reactor (FBTR) successfully since 1985. Currently, a 500 MWe MOX fuelled pool type Sodium cooled Fast Reactor called Prototype Fast Breeder Reactor (PFBR) is under advanced stage of commissioning. The design, R&D, safety review, construction and commissioning experience from PFBR has motivated the commercial exploitation of MOX fuelled Sodium cooled Fast Reactors (SFR) with closed fuel cycle. Accordingly, six FBRs are planned in which, the first two units (FBR 1&2) will be located at Kalpakkam. These reactors are incorporated with advanced design features towards improved economy and enhanced safety.
FBR 1&2 will be of MOX fuelled to be deployed ahead of metal fuelled reactors in order to capitalize on the experience gained in all the domains of SFR technology and to sustain the program. These future reactors need to have improved economy, enhanced safety and possible higher performance parameters. Economy is achieved by design optimization, reduction of material quantities, adoption of twin unit concept with sharing of facilities, design enabling integrated manufacture and erection leading to reduced construction time. Based on detailed studies, reactor power is enhanced with a slightly larger core and by way of design optimization and exploiting the improved manufacturing technologies, the sizes of major large size components are kept close to the industrial capacity that have been built in the country. This approach has led to raising of reactor power to 600 MWe leading to economic gains.
With regard to safety, the important aspects taken into consideration are the internationally evolving Gen-IV safety criteria especially after Fukushima. The enhanced safety level seek to prevent severe core damage and large radioactivity release to the public and practical elimination of severe accident scenarios involving energy release and public evacuation. The major safety enhancements envisaged are (i) improved core inherent safety characteristics with sodium void coefficient less than 1 $, (ii) passive shutdown features and additional shutdown systems employing alternative working principles to prevent events leading to accident situations and (iii) passive & augmented decay heat removal capacity. This paper presents the advanced design features envisaged, towards enhancing safety and improving economy in the future MOX fuelled Indian SFRs.

Country/Int. Organization:

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1.5 LFR DESIGN & DEVELOPMENT / 301

Strategy and R&D status of China Lead-based Reactor

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Lead-based reactor is one of the most promising nuclear energy systems for Generation-IV Accelerator Driven subcritical System (ADS) and reactors. Chinese Academy of Sciences (CAS) had launched a project to develop ADS and lead-based fast reactor technology since 2011. China LEAd-based

Reactor (CLEAR) was selected as the reference reactor, which was performed by Institute of Nuclear Energy Safety Technology (INEST/FDS Team), CAS. The program consists of three stages with the goal of developing 10MWth lead-based research reactor (CLEAR-I), 100MWth lead-based engineering demonstration reactor (CLEAR-II) and 1000MWth lead-based commercial prototype reactor (CLEAR-III) on each stage. To promote the CLEAR project successfully, INEST places more emphases on reactor design, reactor safety assessment, design and analysis software development, lead alloy experiment loop, key technologies and components R&D activities.

Detailed conceptual design of CLEAR-I has been completed and the engineering design is underway, which has subcritical and critical dual-mode operation capability for validation of lead cooled fast reactor (LFR) and ADS transmutation system and technology. KYLIN series Lead-Bismuth Eutectic (LBE) experimental loops have been constructed which is a large multi-functional lead-bismuth experiment loop platform. It has three independent loops, including material test loop, thermal hydraulic loop and safety loop. The objective of KYLIN is to perform structural material corrosion experiments, thermal-hydraulics tests and safety experiments. The key components including the control rod drive mechanism, refueling system, fuel assembly, and simulator for principle verification have been fabricated and tested. In order to integrated test the technologies of lead-based reactor, three integrated test facilities have been built, including the lead alloy cooled engineering validation reactor CLEAR-S, the lead-based zero power critical/subcritica reactor CLEAR-0, the lead-based virtual reactor CLEAR-V.

In addition, series of innovative concepts for different purpose are being developed to enlarge the application perspective of lead-based reactors, which are not only for ADS and fast reactors but also for other innovative applications, such as CLEAR-SFB for spent fuel burning, CLEAR-Th for thorium utilization, CLEAR-H for hydrogen production, etc..

Country/Int. Organization:

Institute of Nuclear Energy Safety Technology (INEST), Chinese Academy of Sciences

Poster Session 1 / 302

Application of Heterogeneous Fuel Assemblies in the Core of Modular Fast Sodium Reactor

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The work contains results of calculation studies of neutron physics characteristics of the fast modular sodium reactor core, in which fuel assemblies without casing with heterogeneity inside fuel assemblies are used. Metal fuel (U-Pu-Zr) is the most advantageous fuel of all known challenging fuel types for a fast sodium reactor regarding neutron characteristics. It enables obtaining the maximal mass of heavy nuclei in the core and a harder neutron spectrum due to absence of light nuclei in comparison with other fuel types. However, experience of metal fuel application is extremely little, and this fuel has not been in commercial operation yet. Application of a heterogeneous fuel assembly consisting of fuel elements with highly enriched (<30%) mixed oxide fuel combined with fuel elements of metallic uranium (or alloy) enables increasing concentration of fissionable and fertile nuclides in comparison with homogeneous fuel assemblies with MOX fuel and obtain similar indices to ones of homogeneous fuel assemblies with metal U-Pu-Zr fuel. A heterogeneous fuel assembly consisting of fuel elements with MOX fuel and fuel elements with metallic uranium of natural composition or U-Zr alloy and a homogeneous fuel assemblies were compared in the course of research. Use of U-Zr alloy without plutonium at the beginning of the campaign and its relatively low average burnup reduces requirements to metal fuel and enables using it already in the nearest future. A heterogeneous fuel assemblies can become an intermediate variant during conversion to

the metal fuel core or a final variant if it has better indices than fuel assemblies with metal U-Pu-Zr fuel.

Country/Int. Organization:

National Research Center "Kurchatov Institute"(NRC KI)
Moscow Russia

Poster Session 1 / 303

Improving inherent safety BN-800 by the use of fuel assembly with (U, Pu)C microfuel.

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The task undertaken in the report is to increase inherent safety of the fast reactor with a sodium coolant of type BN-800 due to considering the possibility of using an innovation fuel assemblies with mixed uranium-plutonium carbide fuel in form of coated particles. Fuel assemblies with pellet MOX fuel and fuel rods are directly replaced by microspherical mixed (U,Pu)C-fuel. Calculation evaluations of characteristics of fuel assemblies with microspherical fuel are realized.
A calculation comparison of neutron physics and thermal hydraulics characteristics of the innovation fuel assemblies with microspherical mixed (U,Pu)C-fuel and the traditional fuel assemblies with pellet MOX fuel and fuel rods was conducted.
The chosen calculation model was BN-800 reactor core with MOX fuel, where a three-zone radial power density field flattening due to plutonium content change in fuel was used.
Thanks to microspherical carbide fuel, inherent safety of the reactor increases in accidents with loss of coolant flow and introduction of positive reactivity because the coated particles have developed heat-exchange surface and their coats are able to keep fission products at higher temperatures than the steel cladding of traditional fuel rods.

Country/Int. Organization:

NRNU MEPHI, Moscow, Russia
NRC «Kurchatov Institute», Moscow, Russia

2.1 Commissioning and Operating Experience of Fast Reactors I / 307

Safety Upgradation of Fast Breeder Test Reactor

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Fast Breeder Test Reactor (FBTR) has completed 30 years of operation and is relicensed for further operation up to 2018. FBTR has undertaken major upgradation of systems, components and structures to enhance the safety level, based on the operational feedback, maintenance difficulties and obsolescence. Further, post Fukushima, an extensive retrofitting programme is underway to protect the plant against external events such as flood, Tsunami and seismicity. As per the upgradation programme, several major components have been replaced. These include the Neutronic channels, UPS, computers of the Central Data Processing System, main boiler feed pumps, five control rod drive mechanisms, two control rods, central canal plug, deaerator lift pumps, reheaters of the steam water system, station batteries, DM plant, Nitrogen plant, starting air system of the emergency diesel generators, entire fire water system including pumps and isolation dampers of the reactor containment building. Due to obsolescence, 6.6kV MOCB were replaced with VCB and 415V electro-mechanical relays were replaced with numerical relays. Residual life assessment has been carried out for the nonreplaceable components based on the operational history, the design limits for each component by which their capability for continued operation has been ensured.
As a part of seismic retrofitting programme, the adequacy of the systems to withstand SSE for safe shutdown, decay heat removal and containment integrity have been assessed. Inparticular plant buildings, anchoring of electrical & instrumentation panels and sodium tanks and other capacities were verified and wooden battery stands of UPS and control power supply were replaced with seismically qualified metallic stands.
A new seismically qualified service building is under construction for housing two seismically qualified DG sets and emergency switch gears. Seismic Instrumentation to measure seismic activity in safety structures as well as free-field close to the reactor, is being procured. Supplementary control panel for monitoring the reactor during non-availability of main control room is being implemented. This paper details the various measures implemented for enhancing the safety of FBTR which includes post Fukushima retrofits also.

Country/Int. Organization:

India/Department of Atomic Energy/Indira Gandhi Centre for Atomic Research, Kalpakkam, India

Poster Session 2 / 308

Hydraulic Design and Evaluation of the PHTS Mechanical Pump of PGSFR

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The Prototype Generation IV Sodium-Cooled Fast Reactor (PGSFR) has been developed by Korea Atomic Energy Research Institute (KAERI). The hydraulic part such as the impellar and diffuser of the PHTS pump has been designed to satisfy the requirement of the hydraulic performance. The essential geometric parameters of the impellar and diffuser were determined through the optimal design methodology. The hydraulic performance and cavitation of the prototype pump were confirmed using CFD simulation. To verify performance of the pump and produce safety analysis input data, the scaled-down model pump and test facility were designed and fabricated based on the scaling law. The performance curve, NPSH curve, coastdown curve, pressure pulse curve, homologous curve and flow resistance curve were obtained from the model pump test facility. The hydraulic performance with rational margin were verified from the model pump test.

Country/Int. Organization:

Republic of Korea/Korea Atomic Energy Research Institute

7.3 Non Proliferation Aspects of Fast Reactors / 309

Closing Up Nuclear Fuel Cycle In a Two-Component System with Thermal And Fast Neutron Reactors

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The modern open fuel cycle nuclear power has created a number of obstacles for its further development due to enormously low effectiveness of natural uranium use, accumulation of nuclear wastes in thermal reactors, its increasing environmental unacceptability, limited explored natural uranium stockpiles and instability of natural uranium market price originated from the reasons similar to those of fossil fuel market, the most vague and critical reasons being the political ones.

The major part of the above issues are related to the weaknesses of open fuel cycle and can be resolved by the operation of fast reactors which use plutonium from the nuclear wastes as a nuclear fuel, thus making processing nuclear waste economically desirable. Fast reactors turn uranium 238 waste into plutonium with the breeding ratio being more than one due to the excessive number of neutrons and neutron energy as compared to uranium 235.

Plutonium bred in fast reactors may and should be used in thermal reactors during the active operational life. The combined operation of thermal and fast reactors makes a highly efficient and sustainable two-component power plant. In addition, fast reactors "burn"minor actinides which are the major contributors to an overburdened radioactive wastes handling in a long-term perspective.

Two industrial power units with fast reactors are currently operated in Russia, BN600 and BN800 with total electric power of 1500 MW. In addition to power generation, the power units are designed to implement the initial stage of closed nuclear fuel cycle. The implementation stage covers creating the economically favorable environment for thermal reactor waste processing, plutonium breeding for further utilization in both fast and thermal rectors, gaining practical experience in minor actinides disposal. The implementation stage will be accomplished in 2026. The practical outcome of the above program presents an apparent commercial interest to the countries which are currently developing efficient and environmentally friendly nuclear power and is likely to be received in Russia much earlier than in the countries which do not own the industrial fast reactor technologies.

Country/Int. Organization:

Russian Federation
State Atomic Energy Corporation "Rosatom"

6.2 Thermal Hydraulics Calculations and Experiments / 310

EFFECT OF INLET TEMPERATURE AND OPERATING LINEAR HEAT RATING (LHR) ON THE MAXIMUM ACHIEVABLE BURNUP OF MK-1 CARBIDE FUEL IN FBTR

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India had constructed and is operating Fast Breeder Test Reactor (FBTR) with Mixed Carbide Fuel, a first of its kind in the world, as driver fuel. Mixed Carbide was chosen as fuel due to its high stability

with Pu rich fuel, compatibility with coolant and for its better thermal performance. Being a unique fuel of its kind without any irradiation data, it was decided to use the reactor itself as the test bed for this driver fuel. The fuel has performed extremely well, with the peak burn-up reaching 165 GWd/t. The Linear Heat Rating (LHR) and burnup of the fuel was initially set at 250 W/cm and 25 GWd/t respectively. Based on rigorous theoretical analysis and Post Irradiation Examination (PIE) done at 25 GWd/t, 50 GWd/t and 100 GWd/t burnup intervals, the LHR limit was raised to 400 W/cm and allowable burn-up was raised to 155 GWd/t.
The burnup limit of the fuel SA comes from the following factors: Wrapper dilation; Wrapper residual ductility; Fission gas pressure and Fuel Clad Mechanical Interaction (FCMI) induced stress in pin; Clad strains; Clad residual ductility; Clad Cumulative Damage Fraction (CDF); Subassembly flow reduction, etc. Presently, the operating parameters like inlet temperature and the peak LHR of the FBTR of MK-1 fuel SA are 400℃ and 400 W/cm respectively which may result in different limits on the achievable burnups. In this work, the effect of LHR & inlet temperature have been comprehensively studied on the achievable burnup of the MK- 1 fuel SA. From the analysis, it is observed that the two enveloping parameters that govern the SA life are wrapper dilation and pin CDF. The maximum burnup achievable with an operating LHR of 400 W/cm is 85 GWd/t and 114 GWd/t for inlet temperatures of 400℃ and 380 ° C respectively. The reduction in the inlet temperature by 20 ℃ not only decreases the fuel swelling but also helps in increasing the free swelling phase without FCMI. Thus, this study gives an insight on the behaviour of the MK-1 carbide fuel in FBTR for the present operating conditions of the FBTR and the influence of inlet temperature and operating LHR on the achievable burnup.

Country/Int. Organization:

Indira Gandhi Centre for Atomic Research, Kalpakkam, India

Poster Session 1 / 311

On-site nuclear fuel cycle of "BREST"reactors

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Dynamic developing of modern nuclear industry demands meeting the following requirements: improved safety, reduced capital costs, radioactive waste management issues, independence of limited resources.

Efficiency of uranium resources used in "Brest" reactors based on a closed fuel cycle is about 160 times higher than for VVER, RBMK reactors, which allows to stop searching for new deposits and uranium mining.

The need for periodical fuel regeneration and fabrication in a closed cycle includes:

- Reproduction of plutonium in the core without the uranium containing screens. Breeding ratio is approximately 1,05, ensuring a high level of safety and support of the non-proliferation regime;
- Transmutation of the most dangerous long-lived actinides and high refining of radioactive waste, achieving the radiation balance of buried radioactive waste and extracted uranium ore.

The manufacturing is located directly at the NPP to avoid transportation of fissile materials. This approach provides economic efficiency of the entire complex.

Country/Int. Organization:

Institution "Innovation and Technology Center by "PRORYV"Project", State Atomic Energy Corporation "Rosatom"

6.8 Experimental Facilities / 312

CLEAR-S: A Large Pool-type Components and Thermo-hydraulic Integrated Test Facility for China Lead based reactor

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Liquid lead-alloy is a potential candidate coolant for fast reactor and Accelerator Driven System (ADS) subcritical system because of its many unique nuclear, thermophysical and chemical attributes. Chinese Academy of Sciences (CAS) had launched a project to develop ADS and lead-based fast reactors technology since 2011. China LEAd-based Reactor (CLEAR) was selected as the reference reactor. China LEAd-based Research Reactor CLEAR-I is a 10MW lead-bismuth cooled integrated pool-type reactor proposed by Institute of Nuclear Energy Safety Technology (INEST), CAS•FDS Team. In order to verify the key components and investigate the thermal-hydraulics phenomena for CLEAR-I and even for pool type lead-based reactor, an integrated multifunctional non-nuclear test facility named CLEAR-S is being built and commissioning in the end of 2016.

CLEAR-S is a pool type test facility with electrically heating core simulator as 2.5 MW. It would be used to test the 1:1 prototype components for CLEAR-I, such as primary pump, heat exchanger, control rod driven system, in-vessel refueling system, and to verify the design and safety analysis codes, and could verify the specific thermal and security characteristics for liquid heavy metal pool-type reactor. In addition, CLEAR-S could provide the integrated test platform with international advanced level for engineering verification and basic research of liquid heavy metal cooled reactor technology.

CLEAR-S will be the largest full-scale integrated lead-based pool-type experimental facility in the world, which has some advantages for the key components and structure materials verification, thermal hydraulics phenomena investigation, instrumentation and chemistry control technology development, and will become an integral facility for the design and licensing of CLEAR-I and R&D work of ADS and lead-based reactor system. In this contribution, the design and latest progress has been presented for CLEAR-S.

Country/Int. Organization:

Institute of Nuclear Energy Safety Technology , Chinese Academy of Sciences

6.8 Experimental Facilities / 313

PLINIUS-2: a new corium facility and programs to support the safety demonstration of the ASTRID mitigation provisions under Severe Accident Conditions

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The ASTRID reactor (Advanced Sodium Technological Reactor for Industrial Demonstration) is a technological demonstrator of sodium-cooled fast reactor (SFR) designed by the CEA with its industrial partners, with very high levels of requirements. Innovative options have been integrated to enhance the safety, to reduce the capital cost and improve the efficiency, reliability and operability, making the Generation IV SFR an attractive option for electricity production.
In the ASTRID project, the safety objectives are first to prevent the core melting, in particular by the development of an innovative core (named CFV core) with heterogeneous pins and complementary safety prevention devices, and second, to enhance the reactor resistance to severe accident by design. In order to mitigate the consequences of hypothetical core melting situations, specific dispositions or mitigation devices are added to the core and to the reactor: some corium Transfer Tubes are implemented, allowing molten corium discharge outside the core region toward a core catcher which insures sub-criticality, cooling and confinement of the relocated materials.
For a robust safety demonstration, CEA with its partners is improving or developing codes (SIMMER, SCONE and EUROPLEXUS) to simulate de Severe Accidents progression. These codes must be assessed, and the mitigation devices qualified against experiments. Since no facility is worldwide available allowing tests with Sodium and large masses of prototypic corium (about 500kg) to study corium discharge through full-scale Corium Transfer Tube, Fuel-Sodium-Interactions and subsequent Sodium vapor explosion, and Corium Interactions with the sacrificial material which protects the Core-Catcher tray, CEA has decided to build a new versatile facility, called PLINIUS-2; this new experimental platform will extend the PLINIUS capabilities where the handled corium mass was limited to 50kg of UO2, and only Fuel-Water-Interaction where studied.
After describing the ASTRID design options related to Severe Accidents and the main features of the PLINIUS-2, the paper will describe the analytical and global experimental programs planned in PLINIUS-2, supporting the ASTRID development; the used molten mass of UO2 will range from few grams to 500kg. The programs will be devoted to the study of Fuel-Sodium Interactions (Droplet fragmentation, Corium Jet Fragmentation, Sodium Vapor Explosion), of Corium-Sacrificial Material Interactions (corium jet impingement, long term sacrificial material ablation by the corium), and to the qualification of the corium Transfer Tubes.

Country/Int. Organization:

Commissariat à l'Energie Atomique et aux Energies Alternatives (CEA)

1.3 SYSTEM DESIGN AND VALIDATION / 314

COMPONENT HANDLING SYSTEM : PFBR AND BEYOND

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Component handling system deals with the handling of fresh and spent subassemblies (fuel handling) and irradiated primary system components using special flasks (special handling). In FBRs, design of fuel handling machines is very important considering the fact that in-vessel handling is a blind operation due to opacity of sodium and most of the fuel handling operations are carried out remotely. Special features are provided in the design of hoisting system of fuel handling machines like single failure proof design features in order to avoid fall of subassembly during handling. Incidents on

component handling system have a serious impact on plant availability and hence utmost care is taken in the design to avoid wrong operations of fuel handling machines.

PFBR in-vessel handling utilises two rotatable plugs and an offset arm type machine (Transfer Arm). For ex-vessel handling, an A-frame type machine called Inclined fuel transfer machine (IFTM) is used. Several other machines are used as part of the fresh and spent fuel handling chain. A water pool type storage is provided for ex-vessel storage before the subassemblies are transferred to the reprocessing plant. Critical primary fuel handling machines namely Transfer arm and IFTM were qualified by cyclic testing in air and in sodium in dedicated test facilities. The design of PFBR fuel handling system and the design validation of the critical fuel handling machines are described in this paper.

The design, manufacturing and testing of fuel handling machines of PFBR have given valuable feedback for future FBRs. Beyond PFBR, six more oxide fuelled FBRs are planned as twin units. Refuelling in fast reactors being done off-line, gives opportunity to evolve a fuel handling system shared between multiple units for improved economy. The design of fuel handling system for the twin unit 600 MWe future FBRs is described. The rationale behind the changes proposed with respect to PFBR is brought out. Most of the fuel handling equipment is shared between the twin units and a unique twin unit layout has been evolved which is also covered in this paper.

In the future, it is planned to deploy metal fuelled based reactors for achieving faster growth through rapid deployment of FBRs. The details of fuel handling system conceived for future metal fuelled FBRs is also brought out.

Country/Int. Organization:

Indira Gandhi Centre for Atomic Research, Kalpakkam 603 102, India

4.3 Partitioning and Sustainability / 315

ADVANCED FLOW-SHEET FOR PARTITIONING OF TRIVALENT ACTINIDES FROM FAST REACTOR HIGH ACTIVE WASTE

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Partitioning of radiotoxic elements present in the high-level liquid waste (HLLW) followed by transmutation of them (P&T strategy) into stable or short-lived products in accelerated driven systems or fast-reactors is a viable option for the safe management and minimizing the radiotoxicity of HLLW. In this context, a typical high-active waste (HAW) arising from reprocessing of spent carbide fuel, (U0.3Pu0.7)C irradiated to a burn-up of 155 GWd/Te in the Fast Breeder Test Reactor (FBTR) was characterized by various analytical techniques. The composition of this fast reactor high-active waste (FR-HAW) differed significantly from the HAW, arising from thermal reactor fuel reprocessing in terms of radioactivity and elemental composition of various metal ions, which are likely to pose several challenges in the handling, treatment, management and disposal of FR-HAW. A method has been developed for partitioning of minor actinides from the FR-HAW using a solvent system composed of 0.2 M n-octyl(phenyl)-N,N-diisobutylcarbamoylmethylphosphine oxide (CMPO) –1.2 M tri-n-butylphosphate (TBP) in n-dodecane (n-DD), and subsequently demonstrated with the actual (FR-HAW) (155 GWd/Te) using a 16-stage ejector mixer settler in hot cells. The results established the recovery of >99% of trivalents (Am(III) + Ln(III)) using citric acid-nitric acid formulation, developed for back extraction. About 5 - 10% of 106Ru was found in the product and nearly 20% of radioruthenium was carried to the lean organic phase after the first cycle requiring cleanup of the solvent.
However, the demonstrated method and the other the existing methods available in various countries

for partitioning of trivalent actinides from high level liquid waste (HLLW) use organic phase modifiers in significant concentrations to maneuver the undesirable third phase formation encountered during trivalent actinide partitioning, even though the solvent system without any phase modifier was desirable. To avoid these complications novel unsymmetrical diglycolamides (UDGAs) and diglycolamic acids were developed in our laboratory and systematically studied for the group separation of Ln(III)-An(III) as well as for Ln-An separation from fast reactor simulated high-level liquid waste (SHLLW). In this paper, it is proposed to provide the summary of our research and development activities carried out at Chemistry Group, IGCAR towards the development of advanced flow-sheet for trivalent actinide group separation, lanthanide-actinide separation and demonstrations with real high-active waste.

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Poster Session 2 / 317

PERFORMANCE EVALUATION OF TIN OXIDE BASED SENSOR FOR MONIOTORING TRACE LEVELS OF H2 IN ARGON COVER GAS PLENUM OF FBTR

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Large volume of liquid sodium is being handled in primary and secondary coolant circuits in Fast Breeder Reactors (FBRs). In the steam generator section, sodium is separated from high pressure steam/water by a thin wall of ferritic steel. In the event of any sudden leak, high pressure steam/water comes in contact with liquid sodium resulting into sodium-water reaction. Such an eventuality needs to be detected in the incipient stage itself, in order to avert major sodium-water reaction that can otherwise cause excessive pressure-built up in the steam generator, apart from affecting reactor operation, since sodium-water reactions lead to the formation of hydrogen, NaOH, Na2O and NaH .

During the start-up and low-power operation of the reactor, the temperature of sodium is about 473 K, at which the dissolution of H2 in sodium is kinetically hindered. Thus, the hydrogen formed, will evolve in the argon cover gas over sodium. Hence, monitoring hydrogen concentration in argon cover gas will help detection of steam leak into liquid sodium at its inception. Thermal conductivity detector (TCD)-based detectors are reported to be the most promising on-line monitors for hydrogen especially in inert streams such as argon cover gas space of Fast Breeder Test Reactor (FBTR), Kalpakkam. However, their lower detection limit is reported to be about 30 ppm only. A sensor, which can sense below 30 ppm is preferable for identifying the release of trace levels of H2 in argon cover gas. Among various sensing materials, semi-conducting metal-oxides like SnO2, ZnO, etc., are promising materials for the detection of trace levels of hydrocarbons, hydrogen, carbon monoxide, etc. The working principle is the measurement of change in surface-conductivity of the metal oxide during the interaction with the analyte, which is directly related to its concentration. A thin-film based tin-oxide (SnO2) sensor was developed in our laboratory, which can sense between 1 and 100 ppm of hydrogen. This sensor was interfaced at the outlet of TCD based Hydrogen-In-Argon (HAD) system in the secondary-sodium circuit of FBTR; its performance was evaluated both during reactor shut-down condition as well as during power campaigns. This paper presents the details of these experiments and the results obtained.

Country/Int. Organization:

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2.2 Commissioning and Operating Experience of Fast Reactors II / 318

Design modifications of Instrumentation & Control System of future FBRs

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Abstract: The purposes of Instrumentation and Control systems are to assist the operator in controlling the plant at the specified power level, monitor the plant and warn of deviations from normal conditions, prevent accidents by carrying out independent automatic safety and control actions, and mitigate consequences of an accident automatically. Design of I&C for PFBR is done in line with 'AERB Safety Criteria for design of Fast Reactors'and to suit operational and environmental conditions.
I & C of PFBR uses hybrid system consisting of computer based control system and hard wired system. Safety critical systems are built using triple redundant computer systems and / or hardwired systems. Safety related systems use dual redundant VME based real time computers and non safety class I & C systems are built using pre-developed systems. Sensors used are either indigenously developed or imported. The core temperature monitoring probe uses dual thermocouples for monitoring the temperature of each sub-assembly.

For future FBR designs, thrust is given to align with Gen IV requirements and our experiences with safety review and commissioning also demands design enhancements. Indigenisation along with design enhancements has resulted in enhanced safety and economic benefits.

The paper details the design enhancements in the architecture of computer based systems, computer hardware, Human-Machine Interface and sensors like core temperature probe, neutron detectors, sodium instruments etc.

The core temperature monitoring probe is indigenously designed and manufactured with three thermocouples for each fuel sub-assembly. Also the thermocouple channels will have indigenously developed diverse hardware. Triple redundant computer systems will have diverse computers running software developed by different agencies. The non-safety systems will use wireless interface for signal and command transfer.

Country/Int. Organization:

Indira Gandhi Centre for Atomic Research , Kalpakkam , India

3.2 Core Disruptive Accident / 320

Computational modeling of flow blockage in fuel subassemblies and molten material relocation in sodium cooled fast reactors

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The heat generating fuel pins in Sodium cooled Fast Reactors (SFR) are arranged in a tightly packed triangular pitch within a hexagonal sheath forming a fuel subassembly (SA). Due to the compact design, formation of local flow blockage inside the SA is possible. Such blockages are expected to grow gradually and the core monitoring thermocouples which are located at the top of the SA are capable of detecting these blockages at their infancy. But, large size blockages may not be detected by the thermocouples due to low velocity of sodium issuing from the blocked subassembly eventually leading to core damage. The extent of damage propagation before reactor shuts down depends on the size of the blockage and its rate of growth. The thermal hydraulics phenomena involved during damage progression are very complex, involving phase change heat transfer with moving solid-liquid interfaces. To investigate (i) the sequence of damage progression, (ii) possibility of Total Instantaneous Blockage (TIB) detection by online monitoring of the sodium outlet temperature from the neighboring SA and iii) determination of number of SA that are likely to get damaged severely before reactor shutdown etc. a transient a enthalpy based thermal hydraulic model has been developed. The transient model considering multi-material and multi-phase flow features adopts an explicit finite difference method employing Voller's algorithm for interface tracking. The model has been validated against published benchmark data. The SA that are likely to get damaged during a TIB event is determined to be seven.

During a CDA, a significant fraction of the hot molten fuel moves downwards and gets relocated to the core catcher. The core catcher design requires prior knowledge of core-melt relocation time which is the time taken for the molten fuel to reach the lower plenum from the active core region. The initial thermal load on the core catcher is primarily dictated by the core melt relocation time. By mathematical models, upper and lower bounds for core-melt relocation time for postulated accident conditions of Protected Loss of Heat Sink (PLOHS) accident and Unprotected Loss of Flow Accident (ULOFA) have been determined. The potential of a multi layer core catcher in handling the debris generated from a whole core melt down accident has been assessed by CFD studies.

Country/Int. Organization:

INDIA / Indira Gandhi Centre for Atomic Research

Poster Session 1 / 322

Optimization of the thermomechanical treatment to achieve a homogeneous microstructure in a 14Cr ODS steel

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Oxide dispersion strengthened (ODS) ferritic steels are promising candidates for high burn up fuel pins of Sodium Fast Reactors. They are elaborated by powder metallurgy and their manufacturing route is complex and specific, including hot forming and cold working. Different heat treatments are necessary to recrystallize these materials and to relieve internal stresses. The presence of highly stable nano-oxydes and the limited stored energy after elaboration can make the recrystallization temperature extremely high (> 0.9 Tmelting). The aim of this paper is to present the results of a generic study conducted on the recrystallization mechanisms in ODS materials. Model alloys with specific deformations by cold working were studied. It comes out that the direction along which the samples are deformed is determining to increase the driving force for recrystallization and obtain homogeneous microstructures, more than the rate of cold-working applied.

Country/Int. Organization:

France/Commissariat à l'énergie atomique

2.1 Commissioning and Operating Experience of Fast Reactors I / 323

Testing and Qualification of Trailing Cable system for Prototype Fast Breeder Reactor

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Small and large rotatable plugs (SRP & LRP) are provided to facilitate in-vessel handling of core subassemblies using transfe arm. These plugs are rotated during refuelling of the reactor. The control & instrumentation signals and power to various systems / components located on the rotatable plugs are carried by cables and are connectied to their respective control panels located outside. Among large number of signals / power supply, some are needed during rotation of the plugs also. Trailing Cable System is conceived and designed to carry power/control cables whose continuity is to be ensured during rotations of SRP & LRP. The design requirement for trailing cable system is to accommodate twist of cables between the stationary roof slab and SRP by 540 deg. while maintaining their continuity, which otherwise is not possible.

The system is designed with a set of posts mounted one each on SRP & LRP and set of overhanging arms through which, cables are routed in a predetermined way. The overhanging arm bring the cables to the centre of SRP / LRP as the case may be and hence avoids pulling and bending of cables, instead results in twisting. The bunch of cables are freely suspended in the form of 'S'shape between roof slab centre and SRP centre with a vertical separation of 3 m between clamping points. The free length of cables is designed to accommodate the twist without causing entanglement of cables. Provisions are made to swing the system away for facilitating handling of components over control plug. Free from interference of trailing cable system with other components, particularly fuel handling machine during plug rotation is also ensured in the design.

To carry out functional testing and qualification of the important system, a prototype arrangement between SRP and roofslab was manufactured and erected at full scale Top Shield Layout Model. In line with the approved testing program, the rotatable plugs were rotated by designated angles and twist in cables was critically studied. From the detailed tests, it is observed that the configuration conceived for the supporting structure and the cable routing between SRP and roof slab satisfy the design intent and is capable of maintaining electrical continuity of the cables, meeting functional requirements. Subsequently, the system was qualified through systematic cyclic testing.

Country/Int. Organization:

INDIA

Poster Session 1 / 324

Main outcomes from the JASMIN project: development of ASTEC-Na for severe accident simulation in Na cooled fast reactors

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The JASMIN project was launched in the frame of the 7th Framework Programme (FP) of the European Commission (EC). It was inspired by the Gen-IV target of designing innovative reactors that intrinsically prevent severe accidents from occurring or drastically reduce their consequences. One of the main objectives of the 7th FP was the enhancement of the current capability of analysis of severe accidents in Na-cooled fast reactors notably by developing new simulation tools able to evaluate the consequences of unprotected accidents leading to fuel pin failure, fuel and cladding relocation, primary system loads, fission product and aerosols releases. To do so, the ASTEC platform originally developed for LWRs, was chosen to be adapted and extended to the environment of Na-cooled fast reactors, the result being called ASTEC-Na. The main advantage was to simulate all phenomena of interest that are today generally simulated by separate codes focusing on specific aspects (i.e., SAS-SFR, CATHARE, RELAP, CONTAIN-LMR, etc.) using only a single code. In fact, this integrated approach is not complex to be implemented due to the high modularity of ASTEC-Na which allows developing, validating and maintaining separately each of its modules that represent a macro-phenomenon. In addition, the flexibility in defining the core geometry, materials composition and reactor components makes ASTEC-Na able to study new SFR designs, e.g. with fertile layers in outer radial or inner axial core regions (such as in ASTRID design), subassemblies with an inner duct channel to induce fast fuel axial relocation (such as FAIDUS design), or new safety systems to shut-down the core power. The JASMIN project has addressed four main areas: thermal-hydraulics, pin thermal-mechanical behavior, source term and core neutronics. In each area, model development and assessment have been performed. In addition to the experimental test matrix built within the frame of the project and used as references for the model validation, the adequacy of ASTEC-Na models have been evaluated through the comparison with results of other suitable and referenced codes used for benchmarking purposes. The main outcomes from the assessment and validation work have been summarized in the form of a SWOT analysis (Strengths, Weaknesses, Opportunities and Threats) that clearly allows identifying the main needs for future model developments.

Country/Int. Organization:

France/Institut de Radioprotection et de Sûreté Nucléaire

5.9 Large Component Technology II / 325

Design of Sleeve Valve mechanism for Primary Sodium Pump of future FBR

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Future FBRs in India with 600MWe capacity are designed with three Primary Sodium Pumps (PSP). As the PSP are operating in parallel, failure of one PSP will result in a significant reverse flow through it, thereby reducing the flow through the core. Minimizing or arresting the reverse flow will in turn increase the flow to the core and the power operation of reactor can be resumed with two pumps. Hence, an active system called sleeve valve mechanism is conceptualized in PSP, to facilitate the power operation of reactor in 2/3 mode. The mechanism contains a sleeve shell (outside the PSP shell) for closing the suction passage. It is designed to withstand the pressure surge by taking accidental closure into account, under all operating conditions. The sleeve shell will not perfectly seal the suction passage due to provision of gap for movement of sleeve shell outside the PSP shell, thereby leaving an annular gap resulting in a leakage. The leak flow rate and leak flow velocity are reduced by increasing the leak path resistance with a labyrinth, which is optimized to give the maximum possible pressure drop.

The sleeve shell can be raised or lowered using three tie-rods, which are designed with galling resistant screw threads for converting the rotary motion from drive motor to linear motion of sleeve shell. A universal coupling is provided in the tie-rods to accommodate the tilting of the PSP. The synchronous motion of the tie-rods is ensured by a planetary gear drive type arrangement provided above the roof slab. The drive arrangement is designed with manual and electric drives for diversity. Safety interlock systems are designed, which prevents any unwarranted operation of the mechanism. The primary cover gas is sealed with dedicated seal systems for each tie-rod, with provisions for monitoring interspace argon. The tie-rods along with the sleeve shell are designed to be an integral part of pump, thus facilitating handling them as a single unit. It is planned to validate the design of the sleeve valve mechanism by experimental simulation and testing.

Country/Int. Organization:

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3.4 Sodium leak/fire and other safety issues / 326

Learning from 1970 and 1980-Era Sodium Fire Experiments

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The original sodium fast reactor concepts date back to the 1950s and 1960s, and there were large programs in France, Japan, Russia, and the United States in the 1970s and 1980s to design and build commercial-scale SFRs. Many of these programs were abandoned in the 1990s, but a considerable amount of work was done prior to that in order to demonstrate the concepts, and to support the safety cases for the commercial prototypes. There is renewed interest in the sodium fast reactor technology with the advent of Gen-IV concepts developed though the GIF initiative, and safety standards for these designs are higher than they were for the original Gen-III designs. Evaluating the effects of sodium fires in the containment vessel would be an essential part of any modern safety evaluation because beyond the risk of thermal load of the structures and containment overpressure, and the fact that they would be a source of airborne fission products. Validated computational tools able to simulate the in-containment phenomenology are then necessary for a reliable estimation of the

source term to the environment in the case of an accident. Many of the fundamental safety questions have already been explored in the past, however, and there is a huge amount of value in re-visiting the experimental and theoretical work that has already been done.

This paper will discuss an effort to retrieve and re-examine experimental work that was carried out in the 1970s and 1980s at the Cadarache research center in France. Different sodium fire programs will be outlined, and will be linked to new theoretical analyses and modeling efforts. As examples, studies on theoretical developments for sodium spray fire combustion, pool fire combustion, combustion product aerosolization, and fission product emission phenomena have been enabled from access to this data. The experimental results have also been used in studies to validate sodium fire modules in the severe accident code ASTEC-Na. This paper will underline how important it is to preserve the knowledge that was generated in the past, and will outline some of the ways that it can still be applied today.

Country/Int. Organization:

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Poster Session 1 / 327

Design and Development of Stroke Limiting Device for Control & Safety Rod Drive Mechanisms (CSRDMs) of future FBRs

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The core degradation due to Anticipated Transients Without Scram (ATWS) has to be practically eliminated as the fast reactor core is not in its most reactive configuration during normal operation. The ATWS events can lead to early and large release of radioactivity. Deterministic demonstration of dispersal of fuel to avoid further core compaction after initiation of large scale core damage is almost impractical. Hence the following are the important decisions related to shutdown systems for next generation SFRs to facilitate practical elimination of core degradation due to anticipated Transients Without Scram. i) Strengthen the first two shutdown systems by addition of passive/active features. ii) Introduce an additional shutdown system, which is completely diverse, independent, passive & confined within core sub-assembly. This shall come into action on failure of first two systems. In this paper design augmentation of first Active Shutdown system by addition of a safety device is discussed.
The first shutdown mechanism of future FBR is Control and Safety Rod Drive Mechanism (CSRDM). A Stroke Limiting Device (SLD) is provided in CSRDM of future FBR to prevent inadvertent withdrawal of Control & Safety Rods beyond a pre-set level and thereby limit the consequences of inadvertent control rod withdrawal event well within limits even with the failure of other safety actions. SLD limits the consequences of inadvertent withdrawal of CSR by physically limiting the withdrawal stroke length of CSR to 20 mm.
Two different concepts of SLD have been conceived, manufactured and standalone endurance tested. One of them has also been integrated with CSRDM and tested. Based on the above, Mark-II design of SLD with some additional design improvements is being developed. The final design of SLD (Mark - II) will be adopted in CSRDM for future Fast Breeder Reactors. The details of two concepts developed, testing carried out as part of design validation and design improvements being incorporated in Mark-II design are presented in this paper.

Country/Int. Organization:

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6.1 CFD and 3D Modeling / 332

Steady State Modelling and Validation of Once Through Steam Generator

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One dimensional steady state model is developed for counter current shell and tube once through steam generator for PFBR with water flowing through tubes and sodium through shell. All the heat transfer processes i.e. preheating, evaporation of water and superheating of the steam happen along the length of the tube. The length of the steam generator is divided into a number of control volumes. For modelling water side, continuity, momentum and energy equations are solved for single phase water, two phase steam and super heated vapour over the length of the tube using homogeneous model. The momentum and continuity equation on the sodium side are not solved as the density variation of sodium with pressure is negligible. The differential equations are discretised to linear algebraic equations using finite difference scheme. All three terms (advection, frictional and gravitational) are considered in momentum equation of water, while only advection and source term are considered in energy equation of water and sodium. The discretisation of the advection term is carried out using upwind scheme and of frictional term by backward differencing. The continuity equation for water is discretised using backward differencing. The linear algebraic equations are then solved simultaneously using numerical methods to get the temperature and pressure profiles in the tube and shell side.
Water side heat transfer coefficient is calculated using Steiner and Taborek asymptotic model and Subbotin correlation is used for sodium side heat transfer coefficient. The code developed can be applied to simulate similar steam generators of any length with any number of tubes. Towards validation of the mathematical model and the solution method, 19 tube steam generator tested in-house steam generator test facility (SGTF) has been simulated and the predicted results are compared with the measured data. Results from the code match well with experimentally observed data from the facility. In addition, grid sensitivity studies have been carried out to establish consistency in the solution.

Country/Int. Organization:

India/ Indira Gandhi Centre for Atomic Research

5.2 Advanced Fast Reactor Fuel Development II / 333

Fuel Melting Margin Assessment of Fast Reactor Oxide Fuel Pins using a Statistical Approach

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In the framework of the Basic Design of the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) project, the design margins have to be defined with accuracy. The design criterion considered here is the fuel melting margin during nominal operation conditions, which is given by the melting probability.
Oxide fuel temperature and melting temperature being calculated with the CEA fuel performance code, GERMINAL, results could be depend on parameters from manufacturing processes, irradiation conditions and fuel behavior laws. The aim of this paper is to take into account uncertainties associated to these parameters in the melting margin evaluation and to quantify the sensitivity to these parameters.
In a first approach, GERMINAL calculations of the temperature distribution given by analytic method is compared with direct Monte-Carlo sampling and with a reliability dedicated method by using the CEA uncertainty and sensitivity simulation platform, URANIE. First sensitivity analysis shows that linear heat rate, fuel stoichiometry and fuel clad gap are the first order parameters.
Afterwards, in a detailed approach, pellet geometrical defects in fuel pellets are taken into account using a 3D finite element model based on CEA LICOS code. The maximum temperature being considerably dependent of these defects, a simple meta-model is built, and is linked with GERMINAL in order to build an artificial neural network. Using this global meta-model, first and second order reliability methods are finally compared with a large number of direct Monte-Carlo simulations in order to determine the fuel melting probability.

Country/Int. Organization:

French Alternative Energies and Atomic Energy Commission, DEN/CAD/DEC/SESC/LC2I

Poster Session 1 / 334

Thermal hydraulic investigation of sodium fire and hydrogen production in top shield enclosure of an FBR following a core disruptive accident

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During Core Disruptive Accident (CDA) in a pool type SFR, primary sodium from hot pool is expelled into the top enclosure, which is vented to reactor containment building (RCB). Sodium coming in contact with oxygen and moisture undergoes combustion and forms sodium oxide / sodium peroxide and hydrogen releasing large amount of heat. The phenomenon of sodium combustion in the form of pool fire and reaction of sodium with moisture present in the air producing hydrogen has been mathematically modeled. The rate of reaction is taken as a function of the oxygen concentration and moisture content in the air. Various parameters, viz., temperature, pressure, density, inlet and outlet air flow rates in the enclosure are estimated by numerically solving the appropriate governing equations. A rapid increase in enclosure pressure is observed initially at the start of combustion but it falls afterwards when natural circulation of RCB air is established. Concentration of various constituents of air in the enclosure, viz., oxygen, nitrogen, moisture and hydrogen are estimated during the combustion period. It is found that the hydrogen concentration in the enclosure is within auto ignition limit. As the combustion starts, hydrogen concentration increases, remains fairly constant during stable combustion period and comes down after the reaction is over. Heat absorbed by structural materials within the enclosure is calculated and it is found that the rise in temperature of the structural materials is within 10℃. Sensitivity of the results to the values of emissivity and humidity has been assessed.
Considering the formation of sodium peroxide during combustion, it is found that the duration of combustion is longer as compared to the reaction forming only sodium oxide. Hydrogen concentration in the enclosure increases in this reaction. The maximum temperature of air is found to increase

by ~ 25 K due to the formation of sodium peroxide. By considering the reaction rate as a function of available sodium, the duration of combustion increases and the maximum temperature in the enclosure decreases with the peak hydrogen concentration remaining nearly same. Full paper would address the modeling approach and evolution of various parameters and sensitivity analysis.

Country/Int. Organization:

India/Indira Gandhi Centre for Atomic Research, Kalpakkam

3.2 Core Disruptive Accident / 335

Source Term Estimation for Radioactivity Release under Severe Accident Scenarios in Sodium cooled Fast Reactors

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Due to the inherent characteristics and robust design of Sodium cooled Fast Reactors (SFR), the core disruptive accident (CDA) is considered a very unlikely event. Nevertheless, to confirm the safety of the reactor, one of the hypothetical scenarios arising from the loss of coolant flow coupled with the failure of the shutdown system, referred to as the Unprotected Loss of Flow (ULOF) accident is postulated to serve as a basis for containment design and severe accident management measures. Determination of the corresponding radioactive source term released into the containment is an important initial condition for the assessment of the adequacy of the containment and subsequently the radiological impact at the site. Estimation of the source term for the sodium cooled fast reactors requires computational tools similar to those developed for the assessment of thermal reactor source term. Towards improving the current state of the art for modeling the in-vessel and in-containment source terms IAEA launched the CRP in which participants from nine countries are doing benchmark simulations for the source term estimation with different models and tools. The technical aspects to be addressed are divided into three main parts. First is the in-vessel source term estimation, consisting of risk important fission product distribution in the fuel pins, their release mechanisms into the coolant and subsequent reaction and transport in the coolant and release to the cover gas. Second is the primary system/containment interface source term estimation consisting of models for the cover gas, sodium ejection and radionuclide chemical composition and distribution in the containment. The third part is the estimation of the fission product evolution within the containment considering sodium burning scenarios, aerosol behavior and physical boundary conditions. Towards this a benchmark SFR model has been defined and developed. The input data required for the simulations have been calculated and boundary conditions identified and specified. The paper presents the problem definition, approach and results obtained from the preliminary modeling. The resulting models are expected to provide a more realistic than existing conservative estimates and would further help to identify areas for experimental investigations through sensitivity and uncertainty analysis of the improved integrated models.

Country/Int. Organization:

IAEA, Vienna

Poster Session 1 / 336

Numerical and Experimental Investigations of Tube-to-Tube Interaction of Air Heat Exchangers of PFBR under Seismic Excitations

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Numerical and experimental investigations of seismic response behavior of the air heat exchanger (AHX) of prototype fast breeder reactor (PFBR) were carried out for operating basis earthquake (OBE), safe shutdown earthquake (SSE) and beyond design basis earthquake conditions. For the numerical study, a finite element model consisting of AHX header and connecting tubes were developed using general purpose finite element code CAST3M and time history analyses were performed for all the three earthquake loading conditions. To perform the analyses, spectrum compatible time histories were generated from the floor response spectrums at the support location of the AHX. Studies predicted the possibility of tube-tube interaction between the middle and outer tubes due to the presence of circumferential fines provided along the tube length. To confirm the analyses findings, shake table experiments were performed using 100 t multi axial shake table. The test set up consists of five AHX tubes along with fines arranged in triangular pitch with tube to tube spacing same as the AHX in the reactor. The tubes were supported simulating the actual supporting conditions in the reactor. To simulate the fluid effects under dynamic conditions, tubes were filled with water and pressurized up to 7 bars. Prior to the seismic studies, free vibration characteristics of the tube bundle were estimated by performing resonance search tests and compared the results with numerical predictions. The responses were captured using accelerometers, strain gauges and no contact type displacement sensors. Tube responses are assessed for OBE, SSE and beyond SSE conditions by performing the tri axial excitations as per IEEE-344 guidelines using spectrum compatible time histories and responses are captured using a 96 channel data acquisition system. Tube to tube interactions at fin locations were observed under SSE and beyond SSE conditions as evidenced from the response spikes in accelerometers and strain gauge readings and from the relative displacements measured using non contact type displacement sensors. However, the structural integrity of tube bundle is demonstrated by repeating the experiments many times for SSE and beyond SSE conditions. From these experiments it is confirmed that, the local impacts at the fin locations are not a concern for the structural integrity of AHX.

Country/Int. Organization:

INDIA

Poster Session 1 / 338

The Effect of Proton Irradiation on the Corrosion Behaviors of Ferritic/Martensitic Steel

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Ferritic-martensitic steel (FMS), which contains 9-12 wt.% of chromium, is being considered as an attractive candidate material for a fuel cladding of a sodium-cooled fast reactor (SFR) due to their high thermal conductivities, low expansion coefficients and excellent irradiation resistances to a void swelling compared with austenitic stainless steels [1, 2]. The cladding should have excellent features such as a good corrosion resistance and low dissolution at high temperatures. Through evaluation of the compatibility of the cladding materials, it was also observed that the degradation, caused by corrosion and dissolution, decreases the mechanical property.

Gr. 92 steel (9Cr-0.5Mo-1.8W-VNb) is one of the advanced versions of high chromium steels that was developed in early 1990. It has higher creep-rupture strength than its predecessors, 9Cr-1Mo steel and Gr. 91 steel (9Cr-1Mo-VNb) [4]. Thus, it can be used even above 650℃, making it suitable for high temperature applications in advanced reactors.

While nuclear reactor is under operation, the structural materials of a cladding materials will be irradiated by neutron and multiple types of ion beams, which will induce the damages of the structural materials. Proton irradiation has been used in the past decades to reproduce neutron damages because protons have a scattering cross-section higher than fast neutrons and it results in the formation of displacements [5, 6].

This study is aimed to investigate the irradiation effects on the corrosion behavior of FMS using an accelerated technique, namely proton irradiation, to understand the irradiation situation of cladding material during its life time in the core. The non-irradiated and irradiated FMS materials were used for corrosion and dissolution experiments. Through the experiments, it was observed that the corrosion and dissolution were accelerated with the irradiation by proton.

Country/Int. Organization:

Republic of Korea/Ulsan National Institute of Science and Technology (UNIST)

6.7 Experimental Thermal Hydraulics / 340

Heat transfer and temperature non-uniformities in pin bundles with heavy liquid metal coolant at various spacing ways

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In the report heat transfer and temperature non-uniformities over the perimeter of the pins in the free-packed pin bundles (s/d=1,33, s/d=1,28) with heavy liquid-metal coolant are considered at various spacing ways. Experimental data for three pin bundles are analyzed: a bundle of smooth fuel pins, a bundle of fuel pins spaced by bilifar-helix wire wrapper of «wire to wire» type and a bundle with spacer grids. Data were obtained for model fuel subassemblies of a reactor with liquid-metal coolant. In the free-packed bundles of smooth fuel pins temperature non-uniformities are absent over the perimeter of fuel pins as opposed to high general temperature non-uniformities over the perimeter of fuel pins in the bundles with wire wrapping. Such high non-uniformities results in considerable decrease of heat transfer coefficients. In the bundles with transverse spacer grids heat transfer increases only in the region of grids, but between them it is approximately equal to heat transfer coefficients in bundles of smooth fuel pins. The correlations recommended for calculations of Nusselt numbers and temperature non-uniformities over the perimeter of fuel pins are given for the above described ways of spacing of fuel pins.

Country/Int. Organization:

SSC RF-IPPE, Russia

Plenary Session 26 June / 342

Closed fuel cycle technologies based on fast reactors as the corner stone for sustainable development of nuclear power

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This article analyzes problems and approaches to modern nuclear power development using closed nuclear fuel cycle and fast reactors. It describes specified technical requirements for nuclear power systems in large-scale nuclear power industry. Targets and scientific problems solved by Rosatom' s "PRORYV"Project which is a part of the Federal State Program "Nuclear Power Technologies of New Generation in the Period of 2010-2015 and up to 2020"are examined.

Country/Int. Organization:

Russia/INSTITUTION "ITC "PRORYV"PROJECT

7.4 Fuel Cycle Analysis / 343

Analysis of the SVBR-100 nuclear fuel cycle by means of the advanced nuclear fuel cycle assessment methodology (ATTR)

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Commissioning of small modular reactors (SMR) seems to be the most reasonable option for the majority of countries embarking on nuclear energy. According to a number of IAEA publications, more than 45 different projects of small nuclear power plants are currently under development and implementation. Russian technological experience with nuclear propulsion systems for navy and ice-breaking fleet is currently implemented for development of SMRs for electricity and heat supply. One of such SMR is SVBR-100 with chemically inert coolant based on Pb-Bi, which can reduce the probability of severe accidents. It guarantees the increased resistance to certain equipment failures, human errors and voluntary actions. However, high initial enrichment of SVBR-100 nuclear fuel (average enrichment is 16.5% for current SVBR option) and weapon-grade quality of plutonium produced reduce the proliferation resistance characteristics of its fuel cycle. This is particularly relevant for the promotion of such reactors to the markets of the embarking states.
Enhancement of SVBR-100 nuclear fuel cycle proliferation resistance is possible by means of involvement of nuclear fuels based on reprocessed uranium. The following table shows the dependence of changes in the plutonium isotopic vector at the moment of fuel discharge. As one can see, the initial content of 236U even isotope leads to the increase in 238Pu (reduces the attractiveness of fissile materials from the perspective of non-proliferation, source of neutrons with spontaneous fission and high energy release).

Involvement of reprocessed uranium in the SVBR-100 nuclear fuel cycle aside from mitigation of risks of proliferation leads to the formation of synergistic effect. This effect is expressed in minimizing of spent nuclear fuel of thermal reactors, saving of uranium resources as well as boosting the competitiveness of such reactors in organizing supplies abroad.
The present report provides the analysis of nuclear fuel cycle of NPP's with SVBR-100 reactor type using the advanced nuclear fuel cycle assessment methodology (ATTR) to ensure non-proliferation of fissile materials and to assess the consumption and saving of uranium recourses by means of IAEA software.

Country/Int. Organization:

Russian Federation

1.3 SYSTEM DESIGN AND VALIDATION / 344

EXPERIMENTAL SEISMIC QUALIFICATION OF DIVERSE SAFETY ROD AND ITS DRIVE MECHANISM OF PROTOTYPE FAST BREEDER REACTOR

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Prototype Fast Breeder Reactor (PFBR) has two independent and diverse fast acting shutdown systems. The mechanisms that handle Control & Safety Rods (CSR) are called Control and Safety Rod Drive Mechanism (CSRDM). CSRDM & CSR are for start up, control of reactor power, controlled shutdown and SCRAM of the reactor. The mechanisms that handle Diverse Safety Rods (DSR) are called Diverse Safety Rod Drive Mechanism (DSRDM). There are 9 CSRDMs and 3 DSRDMs provided. DSR serves to shutdown the reactor on demand and are in fully raised position during normal reactor operation.

CSRDM/DSRDM consists of independent sets of sensors connected to two reactor protection logics of different designs. The output of either of the reactor protection logic system is capable of ordering safety actions through SCRAM signal by de-energizing electromagnet of CSRDM & DSRDM.

As part of seismic qualification, full scale DSRDM along with DSR was extensively tested at room temperature in water for two earthquake levels, namely Operation Base Earthquake (OBE) and Safe Shutdown Earthquake (SSE). Drop time of DSR and its mobile assembly at different instant of dropping from the beginning of shaking were obtained. Full insertion of DSR within the stipulated time and healthy functioning of DSRDM during and after seismic testing have been demonstrated. The details of seismic testing carried out for DSRDM is presented in this paper.

Country/Int. Organization:

India/ IGCAR

Poster Session 1 / 345

Computational modelling of inter-wrapper flow and primary system temperature evolution in FBTR under extended Station Blackout

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To handle a station black-out (SBO) event, sodium cooled fast reactors are equipped with passive systems to remove decay heat, which do not require external power source. The duration of SBO can extend up to a week. Decay heat removal in FBTR, depends on natural convection, driving flow through the core subassemblies and inter-wrapper spaces. The relatively low decay heat and the extended duration of SBO presents a risk of coolant freezing in the primary circuit inlet pipes. This impacts coolability of core, as flow through subassemblies would be impeded and major heat removal path would become ineffective. It becomes essential to understand heat transfer to reactor vault, temperature distribution in the primary sodium pool and evolution of inter-wrapper flow and clad temperature.
The present study aims to develop mathematical models for this scenario and investigates thermal hydraulics of the reactor. The study assumes that internal flow through subassemblies is impeded and only inter wrapper space is available for heat removal from core. Heat sink is provided by primary vessel sodium plenum and surrounding structural components. Ultimate heat sink is atmosphere. A two dimensional axisymmetric Computational Fluid Dynamics model of primary plenum is developed. Influence of reactor subassemblies on inter-wrapper flow, which act as primary heat removal path is modeled with the aid of appropriate momentum and heat sinks. Decay heat generation in core is obtained from available reactor physics calculations. Thermal effects of structures surrounding the reactor vessel (thermal capacity and resistance) are modeled using a one dimensional lumped parameter model, which supplies boundary conditions to the reactor vessel model.
Flow in primary reactor plenum is seen to be controlled by natural convection in the inter-wrapper space, carrying heat from reactor core to main plenum (above core region), which dissipates heat to the surrounding structures through reactor vessel. Flow and temperature in reactor plenum during duration of the transient is predicted and compared against available safety limits. These predicted temperatures are extended using a two dimensional model, to predict fuel and clad temperatures inside fuel subassemblies. Results reveal that fuel clad temperatures reach design safety limit(s) after two days. It is concluded that adequate time is available for deploying Emergency Diesel Generator(s) and initiation of double envelope cooling. Full paper would present the sequentially coupled thermal hydraulic model, evolution of inter-wrapper flow and finally clad temperature as a function of time.

Country/Int. Organization:

Reactor Design Group, Indira Gandhi Center for Atomic Research, India

Poster Session 2 / 346

Development and Applications of Nuclear Design and Safety Assessment Program SuperMC for Fast Reactor

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Abstract:
Compared with the other reactor types, the neutron spectrum of fast reactors is hard, affecting the neutronics and safety performance, for which advanced nuclear simulation methods such as Monte Carlo method is necessary for the nuclear design. Super Monte Carlo Simulation Program for Nuclear and Radiation Process (SuperMC) is a general, intelligent, accurate and precise program for the design and safety evaluation of nuclear system including fast reactors. The latest version of SuperMC can accomplish the n, γ transport calculation and depletion calculation, and is integrated with CAD-based automatic modeling, visualization and cloud computing framework. More than 2000 benchmark models and experiments have been duly verified and validated, including the nuclear analyses of fast reactor benchmark BN600, BFS2, etc. Moreover, SuperMC has been applied in the core and shielding design and analysis of a fast reactor, the China Lead-based Research Reactor (CLEAR).

Keywords: SuperMC; Nuclear Design and Safety Evaluation Program; Fast Reactors; CLEAR

Country/Int. Organization:

China/Institute of Nuclear Energy Safety Technology, CAS ·FDS Team

5.1 Advanced Fast Reactor Fuel Development I / 347

Metal fuel for fast reactors, a new concept

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Choice of the fuel composition is important question to improve the competitiveness of fast reactors. It should have a high density and thermal conductivity, a high concentration of fissile nuclide, and high manufacturability. The best fuel composition of fast reactors remains a metallic nuclear fuel, based on uranium and plutonium alloys. The undeniable advantages of a metal fuel composition is high density of 15-18 g/cm3; high thermal conductivity λ = 30-40 W/m·K; the ability to achieve ultra-deep burnup; simplicity of recycling spent nuclear fuel, based on conventional metallurgical methods.
Significant disadvantages of metallic fuels are a large swelling by the gas and the possibility of irradiation growth in the case of injection-molded parts with a pronounced texture, as well as the opportunity to interact with the fuel cladding above 700 C.
As a solution to these problems, connected channel creation in the fuel core to the output of the gas fission products for the entire fuel campaign is proposed. It can be realized by creation of open porosity 15-25% of the entire volume of the fuel pellet by applying the methods of powder metallurgy. Due to the deformation of the porous structure of the fuel core reduces the risk of cladding damage, and close contact "fuel-clad" provides minimal contact resistance, which improves heat dissipation. Usage of metallic fuel tablet simplifies the fuel element technology and equipment, it makes possible to use fuel elements with gas fuel-cladding gap (as evidenced by the thermophysical calculations). Compatibility metal fuel with cladding can be increased by usage of vanadium alloys or ferritic steels as cladding materials.
Preparation of porous structure in fuel is impossible without usage the powder metallurgy techniques. Wherein a feature of uranium alloys is the difficulty of obtaining compacts by conventional pressing and sintering, it is associated with high oxidative capacity of uranium powder. We use advance methods of compaction. High-voltage electrodischarge consolidation is based on passing an electric current through the powder compact, with the simultaneous application of pressure. The advantages of this method is the short time of compaction (milliseconds), high density products. Due to the short sintering time consolidation comes with minimal changes in the microstructure (grain

growth, recrystallization). Combining technological stages of sintering and pressing has a positive impact on performance. The final density of the product is achieved by selecting parameters such as pressure, voltage, current density.

Country/Int. Organization:

Russian Federation

4.2 Reprocessing and Partitioning / 348

Pu recycling capabilities of ASTRID reactor

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Amid the ASTRID's main goals, one of them is to demonstrate the full fuel cycle closing at the industrial scale. In particular with the recycling of plutonium coming from the treatment of UOX and MOX fuels from PWRs and also the MOX coming from ASTRID itself. Associated with the fuel cycle facilities, fabrication and reprocessing, the lessons learned from this industrial demonstration will be transposable to commercial Sodium cooled- Fast Reactors (SFR) and their associated fuel cycle. The paper presents the capability of the ASTRID reactor with its innovative CFV core (low void sodium worth), to recycle Pu from the treatment of MOX fuels from PWR, during its operation. The safety and performances goals assigned to the CFV core by the ASTRID project are maintained. Physic impacts linked to various aspects initial content of Pu, decay heat, radiation sources), fuel subassemblies type (fresh and used) has been evaluated to identify the plutonium needs and the impacts on ASTRID fuel management (interim storage, handling) and in its associated fuel cycle (transport, facilities).

Country/Int. Organization:

France / Commissariat à l'Energie Atomique et aux Energies Alternatives

5.6 Liquid Metal Technologies / 349

Chemical compatibility with liquid sodium after in service solicitations: feedback on stainless steel in French sodium Fast reactor after 35 years of operation

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French fast breeder nuclear plant PHENIX was definitively shut down in 2009. The dismantling preparations operations of some of PHENIX components are underway. Since 2008, CEA, EDF and AREVA have identified more than twenty relevant components of PHENIX, to get the most from the feedback of the components behavior.
Feedback regarding PHENIX components materials like austenitic stainless steels is large and valuable because some service sollicitations of base metal or welds are difficult to reproduce in laboratory.
In 2012, first examination have been performed on a rod made of SS 304L and SS 316L grades and exposed for about 70 000h to sodium at high temperature.
In this paper, we proposed to highlight the chemical compatibility with liquid sodium of the rod materials and compare it with the state of art on corrosion of austenitic SS (for structural components) in liquid sodium. Examinations (SEM, EDX, WDS) of the rod from PHENIX are the opportunity to improve the overall understanding of the coolant chemistry and its interactions with the materials, enabling easier operation of such reactor.
Finally, relevant data are obtained from this feedback for corrosion modeling for ASTRID fast reactor.

Country/Int. Organization:

FRANCE / CEA (COMMISSARIAT A L'ENERGIE ATOMIQUE ET AUX ENERGIES ALTERNATIVES)EDF, Areva NP

Poster Session 1 / 350

Modeling of Lanthanide Transport in Metallic Fuels: Recent Progresses

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fission products as well as lanthanide impurities in recycled feedstock are known to migrate to the periphery of metal fuels and initiate FCCI that weakens the cladding material. We will present here the most recent developments regarding efforts to implement reliable lanthanide transport models into the 3D fuel performance analysis code, BISON, and experimental and analytical efforts to determine the solubility of certain lanthanides in liquid metals that may be the mechanism for liquid-like transport.

Using ab-initio MD, we found that the solubility of cerium in liquid sodium at 1000K was less than 0.78 at. %, and the diffusion coefficient of cerium in liquid sodium was calculated to be 5.57 10-5 cm2/s. We extended the MD work to two temperatures 723 K and 1000 K for Cerium (Ce), Praseodymium (Pr), and Neodymium (Nd) diffusion in Sodium (Na) and Cesium (Cs) three abundant Ln fission products diffusion coefficients in liquid Na at multiple temperatures. The Ln diffusivities are found to be in the magnitude order of liquid diffusion (10-5cm2/s) and the temperature dependence of diffusivity is developed according to Arrhenius equation.

Experiments have been performed to measure the solubility of lanthanides in liquid sodium.. Using ICP-MS to measure the concentration of lanthanide in the sodium sample, the solubility at that testing temperature is calculated. The experimental results indicated that the solubility of cerium, praseodymium, and neodymium varied from 1x10-6 to 3x10-5 at. % between 723 to 823 K with considerable scatter. The time dependence of solubility is also obtained from experiments conducted at different equilibration times.

To better describe the Ln transport behavior, a model of Ln transport through porous media is being developed from pore-scale to a continuum description through three main steps: 1) Ln dissolution at an isothermal fuel-pore interface, 2) Ln migration within a single 1D tubular pore along a temperature gradient, 3) Ln transport through porous media with specific porosity. Temperature effects are considered with inclusion of Soret term. Finally, an integrated model with regarding to an effective diffusivity, porosity and Soret effect is obtained. Initial results of modeling are discussed.

Country/Int. Organization:

USA Los Alamos National Laboratory

8.1 Professional Development and Knowledge Management - I / 351

'EURATOM SUCCESS STORIES'IN FACILITATING PAN-EUROPEAN E&T COLLABORATIVE EFFORTS

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The European Atomic Energy Community (Euratom) Research and Training framework programmes are benefitting from a consistent success in pursuing excellence in research and facilitating Pan European collaborative efforts across a broad range of nuclear science and technologies, nuclear fission and radiation protection.
To fulfil Euratom R&D programmes keys objectives of maintaining high levels of nuclear knowledge and building a more dynamic and competitive European industry, promotion of Pan-European mobility of researchers are implemented by co-financing transnational access to research infrastructures and joint research activities through to Research and Innovation and Coordination and Support Actions funding schemes.
Establishment by the research community of European technology platforms are being capitalised. Mapping of research infrastructures and E&T capabilities is allowing a closer cooperation within the European Union and beyond, benefiting from multilateral international agreements and from closer cooperation between Euratom, OECD/NEA and IAEA and international fora.
'Euratom success stories' in facilitating Pan-European E&T collaborative efforts through Research and Training framework programs show the benefits of research efforts in key fields, of building an effective 'critical mass', of promoting the creation of 'centres of excellence'with an increased support for 'open access to key research infrastructures', exploitation of research results, management of knowledge, dissemination and sharing of learning outcomes.

Country/Int. Organization:

European Commission, DG Research and Innovation, Euratom Fission, Brussels, Belgium

3.6 Safety Analysis / 354

Current Thermal Hydraulic Activities on Sodium-cooled Fast Reactors in Japan

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The safety design criteria (SDC) has been established in the framework of the Generation-IV International Forum by incorporating safety-related R&D results on innovative technologies and lessons-learned from Fukushima Dai-ichi nuclear power plants accident in order to provide the set of general criteria for the safety designs of structures, systems and components of Generation-IV sodium-cooled fast reactors. To meet the concept of the criteria, several design studies and related researches are carried out for the development of Japan sodium-cooled fast reactor, which should ensure high levels of safety and reliability along with achieving economic competitiveness with future contemporary light water reactors. In this paper, the authors focus on the thermal hydraulic issues related to the SDC, e.g., natural circulation decay heat removal, thermal striping, sodium combustion, and sodium-water chemical reaction. Progress of quantitative evaluation methods on these issues are reviewed along with activities on validation studies.

Country/Int. Organization:

Japan/The University of Tokyo

3.4 Sodium leak/fire and other safety issues / 355

SEISMIC SLOSHING EFFECTS IN LEAD-COOLED FAST REACTORS

Author: Marti Jeltsov[1]

Co-authors: Pavel Kudinov [1]; Walter Villanueva [1]

[1] *KTH Royal Institute of Technology*

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Pool-type configuration of LFR (lead-cooled fast reactor) primary systems allows for simple and economic reactor design solutions. However, partially-filled heavy liquid pools pose seismic safety concerns related to sloshing. Violent sloshing can lead to structural failures, gas entrapment and potential core voiding. Seismic isolation systems are used to reduce the mechanical stresses in structures, but its effect on sloshing must be clarified.
This paper describes a numerical CFD (computational fluid dynamics) study of lead sloshing in ELSY reactor. The motion of free surface is modeled using a VOF (volume of fluid) method. A fixed base reactor and seismically isolated reactor cases are modeled using synthetic earthquake data produced in SILER project. Verification and validation of the numerical model is presented.
The adverse effects of seismic isolation system in terms of sloshing-induced hydrodynamic loads and gas entrapment are demonstrated. Furthermore, influence of geometry on sloshing behavior has been discussed. A mitigation solution using flow restrictions is proposed and analyzed.

Country/Int. Organization:

Estonian at KTH Royal Institute of Technology in Stockholm, Sweden

Poster Session 1 / 356

VOIDING OF ELSY PRIMARY SYSTEM DURING STEAM GENERATOR LEAKAGE

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Lead-cooled fast reactor (LFR) is one of most attractive innovative reactor design considered for research and development by the Generation IV International Forum (GIF). The major advantages of LFRs are related to the use of heavy liquid metal coolant (lead or lead-alloy). However, there are still pending safety issues that need resolution. Steam generation tube leakage and/or rupture (SGTL/R) is one of them. During SGTL/R, water from high-pressure secondary side enters the low-pressure primary side. This can potentially lead to void ingression to the core, which has adverse effects on the reactor performance including heat transfer deterioration and reactivity feedbacks.
The main objective of this study is to analyze the steam bubble entrainment phenomena and the extent of primary system and core voiding during an SGTL accident in ELSY reactor. A CFD (computational fluid dynamics) model of ELSY primary system nominal operation flow field is prepared and verified. Assuming small leaks, a Eulerian-Lagrangian approach with one-way coupling between the phases with no bubble-bubble interaction models is used. Validation of the drag model representing the main forces acting on the gas-liquid interface has been carried out.
A probabilistic methodology to estimate the core and primary system voiding rates is presented. The results include identification of the most important factors governing the entrainment phenomena and quantification of voiding rates.

Country/Int. Organization:

Estonian at KTH Royal Institute of Technology, Stockholm, Sweden

1.1 SFR DESIGN & DEVELOPMENT - 1 / 357

Overview of U.S. Fast Reactor Technology R&D Program

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This paper provides an overview of fast reactor research and development efforts in the United States. Fast reactors are envisioned for a wide variety of actinide management strategies ranging from actinide destruction in closed fuel cycles to enhanced uranium utilization. With successful technology development, fast reactors are also intended for electricity and heat production, as being pursued through the Generation-IV International Forum collaborations. Several new initiatives for industry-led R&D, advanced reactor licensing framework, and discussions on advanced test/demonstration reactors are indicative of rising national interest in advanced nuclear technologies.

Because capital investment in reactors is the dominant cost of any nuclear fuel cycle, R&D efforts to improve fast reactor performance are the primary focus. A variety of innovative features that hold the promise for significant cost reduction are being pursued; the diverse R&D activities are funded by several Programs in the DOE nuclear energy portfolio. Innovative technology options that may yield significant cost reduction benefits have been identified through concept development studies: high strength structural materials, a supercritical CO2 Brayton energy conversion cycle, advanced modeling and simulation tools, and in-service inspection techniques. In addition, technology development efforts for safety and licensing, and improved transmutation fuels are ongoing. For each technical area, recent accomplishments and key facilities will be identified to provide an indication of current status.

Country/Int. Organization:

United States of America

Argonne National Laboratory

1.7 ADS AND OTHER REACTOR DESIGNS / 358

The evolution of the primary system design of the MYRRHA facility

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MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) is a multipurpose research facility being developed since 1998 at SCK•CEN, based on the Accelerator Driven System (ADS) concept where a proton accelerator, a spallation target and a lead-bismuth cooled subcritical reactor are coupled. MYRRHA will demonstrate the ADS full concept by coupling these three components at a reasonable power level to allow operation feedback, scalable to an industrial demonstrator, and allow the study of efficient transmutation of high-level nuclear waste.

The MYRRHA research facility will be able to work in both critical as subcritical modes and will allow fuel developments for innovative reactor systems, material developments for GEN IV and fusion reactors, and radioisotope production for medical and industrial applications. MYRRHA will contribute to the development of Lead Fast Reactor (LFR) technology and in critical mode it will play the role of European Technology Pilot Plant in the roadmap for LFR.

In the beginning of 2014, SCK•CEN has consolidated a coherent version of the primary system. This version 1.6 of the primary system forms the basis for the pre-licensing activities. In this paper, the evolution of design of MYRRHA is presented in detail with regard to the primary system and also the current investigations towards a new release will be mentioned.

Country/Int. Organization:

Belgian Nuclear Research Centre (SCK•CEN)
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8.2 Professional Development and Knowledge Management - II / 359

Overview of the IAEA Activities in the Field of Fast Reactor Technology Development: Current State and Future Vision

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Under the project on advanced technology for fast and gas-cooled reactors, the IAEA is carrying out several activities on fast reactor technology development including i) Modelling and Simulations, ii)

Technical Support, iii) Fast Reactors Safety, iv) International Cooperation and Information Exchange, v) Education and Training, and vi) Knowledge Preservation.

Technical Working Group on Fast Reactors (TWG-FR) is a one of the main driving forces directing IAEA activities in the field. Eighteen countries and two international organizations are the members of the TWG-FR while six other countries and Generation-IV International Forum (GIF) are observers.

Coordinated Research Projects (CRP) are important IAEA instruments for organizing international research work to achieve specific research objectives consistent with the IAEA programme goals. Several CRPs have been conducted since last FR13 Conference in Paris. IAEA have completed CRPS on Phenix End-of-Life Tests, Monju Natural Convection, and Benchmark on Accelerator-Driven Systems (ADS). A special session on this conference is devoted to the recently completed CRP on benchmark analysis of EBR-II shutdown heat removal tests. CRP on radioactive release from the prototype sodium cooled fast reactor under severe accident conditions is ongoing now. Another in-progress NAPPRO CRP is on sodium properties and design and safe operation. A new benchmark CRP on CEFR Physics Start-Up Experiments has been initiated. In addition IAEA is conducting study on passive shutdown systems for fast neutron reactors, an initiative on knowledge preservation, joint IAEA-GIF initiative on safety of sodium cooled fast reactor. There are also a number of activities in the field of fast reactor education and training including annual schools and workshops on innovative nuclear systems and development of the SFR simulator for educational purposes.

The details of the IAEA activities will be presented on the special IAEA corner at the FR17 poster session.

Country/Int. Organization:

International Atomic Energy Agency

Poster Session 2 / 360

Overview of the international cooperation and collaboration activities initiated and performed under the Technical Working Group on Fast Reactors in last 50 years

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The Technical Working Group on Fast Reactors (TWG-FR) of the International Atomic Energy Agency (IAEA) was established in 1967 and since then it has been a foundation of the agency's activities in the field of fast reactor. For last five decades the group of experts under the umbrella of TWG-FR have provided advice and supported the implementation of the programme. The TWG-FR assists in defining and carrying out the Agency's activities in the field of fast reactor technology development, promotes the exchange of information on national and multi-national programmes, fosters new developments and experience, with the goal to identify and review problems of importance and to stimulate and facilitate cooperation, development and practical application of fast reactors and sub-critical hybrid systems. There have been numerous technical meetings, consultants'meetings, coordinated research projects producing a significant number of technical documents, nuclear energy series documents etc. supporting member states in their pursuit of research and development in this field. With a current membership of more than 25 countries, TWG-FR plays a significant role in addressing major issues, finding coordinated solutions to overcome technological/research barriers and effectively communicating and transferring knowledge amongst its members. A combined vision of all the members helps in overcoming challenges and increasing effectiveness.

The paper discusses and highlights the activities proposed by the TWG-FR, its major achievements, ongoing events and future plans.

Country/Int. Organization:

International Atomic Energy Agency

6.11 IAEA Benchmark on EBR-II Shutdown Heat Removal Tests / 361

IAEA's Coordinated Research Project on EBR-II Shutdown Heat Removal Tests: An Overview

Author: Vladimir Kriventsev[1]

Co-authors: ANDONG SHIN [2]; Barbara Vezzoni [3]; Chirayu Batra [1]; Chiwoong CHOI [4]; Danting SUI [5]; Ethan Bates [6]; Guanghui SU [7]; Hiroaki Ohira [8]; Koji Morita [9]; Konstantin Mikityuk [10]; Laural Briggs [11]; Ludovic Maas [12]; Marek Stempniewicz [13]; Nikita Rtishchev [14]; PARTHA SARATHY UPPALA [15]; Petruzzi Alessandro [16]; Stefano Monti [1]; Wenjun Hu [17]; Willem Frederik Geert Van Rooijen [18]; del Nevo Alessandro [19]; roberto zanino [20]

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A Coordinated Research Project (CRP) on "Benchmark analysis of EBR-II Shutdown Heat Removal Tests (SHRT)"was launched by the International Atomic Energy Agency (IAEA) in 2012. A series of transient tests were conducted on the EBR-II reactor at Argonne National Laboratory (ANL) to improve the understanding of thermal hydraulics and neutronics of fast reactors. Shutdown heat removal tests conducted in 1984 and 1986 demonstrated mechanisms by which fast reactors can survive severe accident initiators with no core damage. Two SHRT tests, SHRT-17 representing Protected Loss of Flow (PLOF) transient and SHRT-45R representing Unprotected Loss of Flow (ULOF) transients have been studied in the IAEA CRP.

The objectives of the CRP were to improve design and simulation capabilities in fast reactor thermal hydraulics, neutronics and safety analyses through benchmark analysis of these two important tests.

At the first stage of the benchmark, ANL provided the input data on EBR-II geometry, as well as initial and boundary conditions for the SHRT-17 and SHRT-45R tests to perform "blind"calculations. At the second stage, ANL released the experimental observations and participants had the chance to discuss the difference and refine the models. Nineteen organizations from eleven countries participated in the CRP making it one of the largest CRP coordinated by the IAEA fast reactor team.

The papers provides a general CRP overview while the companion papers presented both on this session and at the poster session give the details of the EBR-II reactor design, describe the shutdown heat removal tests, the benchmark setup, results of numerical simulations, and the detailed discussion on this CRP.

Country/Int. Organization:

International Atomic Energy Agency

5.10 Fuel Modeling and Simulation / 363

BERKUT –Best Estimate Code for Modelling of Fast Reactor Fuel Rod Behavior under Normal and Accidental Conditions

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The advanced version of code BERKUT designed for mechanistic modelling of oxide and nitride single fuel rod behavior under normal and accidental conditions of liquid metal cooled fast reactor operation, is under development at IBRAE RAN during the last five years in frame of "Codes of new generation"project included into the "BREAKTHROUGH"(or "PRORYV") project. The code models are grounded on the contemporary understanding of mechanisms governing the most important processes in fuel rods under irradiation, which substantially enhances the predictive ability of the code in comparison with the engineering analogs. The code is the multi-scale one simulating the processes characterized by the range varied from 1nm to 1 m.
At the micro-level the code describes evolution of fuel micro-structure in the fuel grain scale:
• vacancy/interstitial field, nucleation and development of dislocation network and gas filled porosity,
• fission product generation, their radioactive transformations, transport and release out of fuel grains,
• formation of chemical compounds, the fuel phase composition.
At the meso-level the code simulates the processes in the fuel pellet scale:
• mass transfer of fission products, oxygen or nitrogen within pellets,
• evolution of as-fabricated porosity and formation of columnar grains,
• fission product release by recoil and knockout mechanisms.
At the macro-level the code describes thermomechanical behavior of the fuel rod as a whole:
• heat transfer within the rods and heat exchange with the coolant,
• temperature distribution in fuel, fuel-cladding gap and cladding,
• evolution of the stress-strain state of fuel and cladding,
• fission gas composition and gas pressure within the cladding.
The code models describing the processes in oxide fuel, which are common for thermal and fast neutron reactors, have been validated against extensive experimental data set found in the literature. Some particular microscopic parameters have been defined through the theoretical estimates.
The calculations have been performed simulating oxide and nitride fuel rod behavior in BN-600 and BOR-60 reactors. Analysis of the calculation results and their comparison with the data of the post-reactor fuel rod examination has demonstrated that BERKUT describes satisfactorily the fuel and cladding geometry changes, fission gas release as well as porosity profiles and fission product concentration profiles within fuel pallets.

The calculation results obtained allow to make a conclusion that mechanistic fuel rod codes can be used both for safety justification and to predict ways of achieving the specified fuel rod characteristic.

Country/Int. Organization:

Russia, Nuclear Safety Institute (IBRAE RAS)

3.6 Safety Analysis / 364

Design Safety Limits for Transients in a Metal Fuelled Reactor

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To provide rapid growth and faster deployment of fast breeder reactors in future, metal fuelled reactors are planned. The Design safety limits (DSL) for normal and anticipated transient events are arrived at for a typical 500 MWe metal fuelled reactor, which is addressed in this paper.
All the design basis events occurring during the lifetime of the components concerned are classified into four categories based on the frequency of occurrences. The design approach followed in determining the DSL limits for clad is that the pin is deemed to have failed if Cumulative Damage Fraction (CDF), based on creep rupture data under operating pressure & temperature conditions, reaches 1.0. This CDF limit is applied uniformly across all the categories, viz., CDF of 0.25 is to be respected for each of the category –1, 2 & 3 event and the rest 0.25 is allocated for SA during handling & in internal storage. In order to estimate the temperature limits for clad, the time of transient events for which the clad has to withstand the given temperature is considered as 30 minutes for category-2 event and 2 minutes for catergory-3 events which is based on the occurrence of events.
The failure mode in a metal fuel pin is primarily due to swelling, fission gas pressure loading and due to the thinning of clad (because of eutectic formation between fuel and clad at high temperature). In order to arrive at the temperature limits, the temperature dependent liquid penetration rate into the cladding and hence the thinning effect are considered along with the end-of life fission gas pressure loading on the clad. The creep rupture properties of T91 are taken from RCC-MR code after accommodating for irradiation effects.
Based on the analysis, for category-2 event, the clad inside hotspot temperature is limited to 993 K and for catergory-3 events, it is restricted to 1043 K and category-4 event, it is restricted to 1243 K. Similarly, the limits for fuel and coolant are also arrived at. These limits will be fine tuned based on the out-pile and in-pile experiments planned in the future.

Country/Int. Organization:

INDIRA GANDHI CENTRE FOR ATOMIC RESEARCH, INDIA

Poster Session 1 / 366

BISON for Metallic Fuels Modeling

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The fuel performance code BISON has recently been extended to simulate U-Zr and U-Pu-Zr metallic fuel rods irradiated in the US sodium cooled fast reactor EBR-II. By introducing fuel and clad specific material models, the backbone provided by the MOOSE/BISON software architecture has allowed rapid development of metallic fuel capabilities. Zirconium based fuels present unique challenges due to the different phases that exist at irradiation temperatures. Each phase possess differing thermo-mechanical properties, necessitating explicit tracking of the relative concentration of phases throughout the fuel rod in order to capture the integral behavior. In addition, the transition temperatures between phases change over the course of irradiation due to zirconium diffusion in the fuel, necessitating coupling of the thermo-mechanical simulation to the Fickian and Soret diffusion of Zr.

Along with a robust U-Zr and U-Pu-Zr zirconium redistribution model, newly formulated thermodynamic properties such as thermal conductivity, and phase-dependent mechanical properties such as swelling, allow BISON to capture the behavior of zirconium based metallic fuel at a variety of operating temperatures and irradiation histories. These models have been integrated into BISON, verified using standard practices, and validated against full 2D-RZ simulations of fuel irradiated in EBR-II. The incorporation of phase-dependent models allows BISON to be extended to test other novel fuel designs, some of which show promising characteristics depending on fabrication feasibility.

Country/Int. Organization:

US/Los Alamos National Laboratory

Poster Session 1 / 367

IMPLEMENTATION STATUS OF CONTAIN-LMR SODIUM CHEMISTRY MODELS INTO MELCOR 2.1

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This paper describes the progress of the CONTAIN-LMR sodium physics and chemistry models to be implemented into MELCOR 2.1. It also describes the progress to implement these models into CONTAIN2. In the past three years, the implementation included the addition of sodium equations of state and sodium properties from two different sources. The first source is based on the previous work done by Idaho National Laboratory by modifying MELCOR to include liquid lithium equation of state as a working fluid to model the nuclear fusion safety research. The second source uses properties generated for the SIMMER code. Testing and results from this implementation of sodium properties are given. Many of CONTAIN2's physical models were developed since CONTAIN-LMR. Therefore, CONTAIN 2 is being updated with the sodium models in CONTAIN-LMR in order to facilitate verification of these models with the MELCOR code.

Country/Int. Organization:

USA/Sandia National Laboratories

Poster Session 1 / 368

Development of the U.S. Sodium Component Reliability Database

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With the advent of the use of Probabilistic Risk Assessments (PRAs) for safety analysis of Light Water Reactors (LWRs) in the 1970s, the SFR community used PRA as a tool which can demonstrate the safety of SFR designs while avoiding the pitfalls associated with an over-reliance on highly conservative safety requirements. Throughout the 1970s, 80s, and 90s, the US compiled sodium reactor specific PRA information into the Centralized Reliability Database Organization (CREDO) database, maintained by Oak Ridge National Laboratories in collaboration with the Japanese Atomic Energy Agency (JAEA). Unfortunately, the funding for the CREDO database was cut in the 1990s and the database was lost and was regained in August of 2016. This paper will describe three databases being developed at Sandia National Laboratories(SNL):
1. CREDO-I –A summary of the state of the original CREDO database;
2. CREDO-II –Early attempts by Argonne National Laboratory (ANL) and SNL to recreate the CREDO database from operational documents;
3. The future combination of the CREDO-I and CREDO-II databases into a unified database.

Country/Int. Organization:

Sandia National Laboratories

Poster Session 1 / 370

Americium Retention During Metallic Fuel Fabrication

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Under the US Fuel Cycle Technologies program Advanced Fuels Campaign metallic fuel has been chosen as a leading candidate for fast reactor transmutation fuels. Significant losses were seen in an earlier attempt to incorporate americium into a metallic fuel, although the majority of the losses were likely not caused by the volatility of americium, these concerns have persisted. A furnace has been installed in a transuranic qualified glovebox in order to verify that americium losses can be controlled during the fuel casting process through atmospheric pressure. A charge of 81U-7.5Pu-1.5Am-10Zr was melted under an argon atmosphere a total of three times. Each time the charge was held at 1450℃ for approximately 10 minutes under flowing argon. After each melting cycle, the resulting fuel ingot was sampled for chemical analysis to verify americium content. Resulting analysis showed americium content remained stable throughout the heating cycles.

Country/Int. Organization:

United States/Idaho National Laboratory

Poster Session 1 / 371

3-D Core Design of the TRU-Incinerating Thorium RBWR Using Accident Tolerant Cladding

Author: Sandra Bogetic[1]

Co-authors: Ehud Greenspan [1]; Jasmina Vujic [1]; Phillip Gorman [1]

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This project investigates the safety of the optimal core design for the RBWR-TR –a reduced moderation BWR with a high transuranic (TRU) consumption rate. This design is a variant of the Hitachi RBWR-TB2, which arranges its fuel in a hexagonal lattice, axially segregated seed and blanket regions, and fits within a standard ABWR pressure vessel. The RBWR-TR eliminates the internal axial blanket and absorbers from the upper reflector, and uses thorium instead of depleted uranium for the axial blankets. Its coolant flow rate is higher than of the TB2 design. Both designs are initially presented with Zircaloy-2 cladding.
The softer neutron spectrum of the RBWR-TR core, along with its lower peak linear heat generation rate, results in a lower cladding fast neutron fluence than of the RBWR-TB2 core. However, the peak fluence of fast neutrons (E >= 0.1 MeV) the cladding is exposed to exceeds the limits for Zircaloy-2, making this material not acceptable even for the RBWR-TR design. At such high fast fluence levels, zirconium-based cladding experiences faster rates of corrosion and hydrogen pickup, which embrittles the cladding and eliminates any margin for accident scenarios. Alternative cladding materials to Zr-based alloys are being investigated for accident tolerant fuels such as stainless steel based materials that are not limited by hydrogen pickup phenomena. However, the steel cladding penalizes the neutron economy and limits the discharge burnup. The design variables of the parametric studies include the cladding material type, cladding thickness, gap between fuel and cladding, fuel smear density and fuel-to-moderator volume ratio. The changes of the void feedback, cycle length, burnup, shutdown margin, and critical power ratio to variation in each of the design variables are calculated to determine the optimal design. A design that meets all the design constraints will be presented.

Country/Int. Organization:

University of California at Berkeley, USA

6.11 IAEA Benchmark on EBR-II Shutdown Heat Removal Tests / 372

Conclusions of a Benchmark Study on the EBR-II SHRT-45R Experiment

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This paper presents the conclusions of a 4 year benchmark study on the simulation of the EBR-II SHRT-45R experiment. The SHRT-45R experiment was an unprotected loss of flow transient where pump dynamics, natural convection, core and mechanical behavior played a large role in passively and safely limiting the power and temperature rise of the fuel assemblies. Participants from China, Germany, Japan, Korea, Netherlands, Russia and the U.S. presented transient reactor system modeling results for a variety of instrumented parameters, including core outlet temperatures, pump flow rates, and fission power. Detailed pin-level experimental data of the instrumented XX10 (non-fueled) and XX09 (fueled) subassemblies were also assessed. Code-to-code comparisons were made for other non-measured parameters, such as decay heat and peak cladding and fuel temperatures. A small subset of participants presented code predictions of the negative expansion feedbacks (coolant, axial, radial) and Doppler feedback inherent to the EBR-II core.
The final meeting held April 2016 in Vienna summarized key findings and sensitivity studies completed after the experimental data was released and the benchmark study converted from blind to open. The fidelity and methodology of core and system models varied greatly between participants. It was found that accurate simulation of the pump coastdown, system pressure drop, and coolant and radial expansion feedbacks strongly influenced the fission power and temperatures in the core during the transient. Relatively simple models for radial expansion were sufficient to capture the behavior during the transient, in part due to the simpler mechanical dynamics of EBR-II's core and the applicability of the point-kinetics model. Reactor core outlet (Z-pipe and IHX) temperatures were somewhat difficult to match due to the high fidelity required to capture the temperature at the specific thermocouple location. Faulty subassembly flow meter data from XX09 and XX10 prevented a more accurate study of the core flow redistribution occurring during the pump coastdown. Uncertainties and variations in heat transfer and subassembly pressure drop correlations, and fuel expansion assumptions were found to have little effect on the prediction of fission power and temperature. Overall, the benchmark of the SHRT-45R was a valuable exercise that facilitated the development of state-of-the-art models for sodium fast reactor system and neutronic reactivity feedbacks.

"Note: EBR-II Benchmarks Invited Session".

Country/Int. Organization:

USA (TerraPower LLC) and China (Xin Jiang Technical University)

Panel 1: Development and Standardization of Safety Design Criteria (SDC) and Guidelines (SDG) for Sodium Cooled Fast Reactors / 374

Considerations on GEN IV safety goals and how to implement them in future Sodium-cooled Fast Reactors (France)

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From a general perspective, generation IV (GEN IV) reactors should excel in safety, and as part of a continuous improvement process, provide safety enhancements with respect to GENIII reactors.
GENIII safety objectives are already very ambitious, notably regarding:
- Prevention of the severe accident;
- Mitigation of the severe accident in the frame of the fourth level of defense in depth;
- Response to external hazards, including natural hazards of extreme intensity.
Concerning GENIV sodium-cooled fast reactors (SFR), the achievement of these ambitious safety objectives and the reinforcement of the robustness of the safety demonstration, will be ensured:
- Firstly, by mastering the sensitive points of the SFR such as neutron reactivity potential of the core, chemical reactivity of sodium, inspection of structures under sodium.
- Secondly, by taking full benefit in the design of the favorable characteristics of the SFR such as large thermal inertia, large margin to boiling, natural convection capabilities and by providing high diversification and independence between safety systems associated to different levels of defense in depth.
The paper presents some of these possible ways of safety improvement for the future SFR.

Country/Int. Organization:

FRANCE CEA/AREVA/EDF

Poster Session 1 / 375

Study on the sensitivity analysis of the installed capacity and the high-level waste generation based on closed nuclear fuel cycle

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The sustainable development of nuclear energy calls for the maximization of uranium utilization and, meanwhile, the minimization of the waste produced. The fuel cycle mode and its key parameters have a great influence to the deployment of nuclear energy and the generation of high-level waste. In this paper, a sensitivity analysis is made on the influence of out-of-core residence time and the recovery ratio of the actinides; a series of recommended value are given to the parameters mentioned above; the effect of the closed fuel cycle on the reduction of high-level waste is analyzed. The following conclusions are drawn: 1) in the multi-cycle of industrial Pu or transuranics(TRU) in fast reactors, the contents will reach an equilibrium state; in typical SFR, the fissile Pu takes an fraction of 70% in the equilibrium state, and the minor actinides(Mas) take an fraction of 3% in the TRU recycle; 2) the installed capacity of fast reactor is very sensitive to the out-of-core residence time and recovery ratio; the generation of high-level waste is sensitive to the recovery ratio; the recommendation is the out-of-core residence time be no more than 5 years and the recovery ratio of actinides be no less than 99.9%; the reasons are that in order to avoid the decreasing of the nuclear installed capacity during the transition from PWRs to FRs, and that in order to reduce the recycling loss and decrease the generation of the waste, the recovery ratio is proposed to be no less than 99.9%; the benefit of further improvement of the ratio beyond 99.9% is insignificant either for the installed capacity or for the high-level waste; 3) the synergistic development of the PWRs and FRs in closed fuel cycle can not only improve the utilization of uranium but also effectively reduce the generation of the high-level waste; compared with the once-through method in PWRs, closed-fuel-cycle can reduce the long-term radioactive toxicity of high-level waste to 1/5~1/6 with a recovery ratio of 99.9%;

the TRU whole cycle can effectively reduce the amount of MAs and further reduce the long-term radioactive toxicity of high-level waste to 1/7~1/8.

Country/Int. Organization:

China Institute of Atomic Energy

Poster Session 1 / **376**

Full-fledged affination extractive-crystallizng platform for technology validation of the fast reactor spent fuel reprocessing on fast neutrons –the results of first experiments

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Further effective development of fast energy engineering could not be realized without strategy of nuclear fuel cycle closure. Throughout the realization of this strategy combined PH-technology of mixed nitride uranium-plutonium fast reactor spent fuel is proposed.
Full-fledged affination extractive-crystallizng platform was created for hydrometallurgical pro-cesses adjustment and above mentioned technology functional test. The platform guarantees the compliance with the radiation and nuclear safety requirements for working processes with hot spent fuel simulator, which main component is U-Pu-Np mixture, including Am, Tc, stable elements and radioactive tracers.
Scientific and engineering solutions provide the conduction of spent fuel reprocessing technology adjustment research on low capacity, which appears to be the boundary between laboratory test equipment and industrial grade equipment, also provide the performance of equipment rerouting for different alternate layouts verification, the performance of computer-assisted control and operating procedure monitoring.
The platform provides the research results verification along with mathematical model prediction results, the testing of different reagents, the performance procedure adjustment of analytical measurements; the platform can be also used as training complex.
Platform equipment functional test operations on uranium simulator with stable fission products were accomplished, the results of first experiments were obtained.
Keywords: affination platform, spent fuel reprocessing, spent fuel simulator

Country/Int. Organization:

JOIN STOCK COMPANY «SIBERIAN GROUP OF CHEMICAL ENTERPRISES»,SEVERSK, RUSSIAN FEDERATION

Poster Session 1 / **377**

THE CODE ROM FOR ASSESSMENT OF RADIATION SITUATION ON A REGIONAL SCALE DURING ATMOSPHERE RADIOACTIVITY RELEASES

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A new approach for assessment of radiation situation outside industrial sites of objects at radiation risk has been being developed in the last few years. This approach is based on performing of multiple calculations using real time series of weather condition parameters. A standard conservative estimation assumes that the meteorological parameters are constant. This assumption increases the conservatism of estimates because in practice the weather conditions during a long-term release (BDBA) are changing.
According to the new approach a series of calculations with shifted dates of the release beginning are performed. Each calculation realizes real weather conditions taking into account their changeability. A realistic dose distribution is obtained on the base of statistical analyses of the results of all separate calculations.
An example of successful application of such approach is safety analyses of a new designed NPP in Lithuania (Visaginas) with regard to assessment of emergency radioactivity releases into the atmosphere.
The code ROM intended for realistic assessment of doses received by population outside the site of an object at radiation risk during incidents of arbitrary duration has been developed in the Nuclear Safety Institute of the Russian Academy of Sciences in frame of «BREAKTHROUGH» (or «PRORYV» project). This code is a next stage of the development of NOSTRADAMUS computer system, which is based on a Lagrangian model of atmosphere dispersion. Models for the estimation of deposition rates of aerosols formed by polonium and sodium burning products, that are specific for fast reactors, are developed and realized in the code. The models of atmosphere dispersion and of deposition rates of aerosols are validated against experimental data.
The code ROM utilizes time series of meteorological parameters measured at the WMO stations close to the object for a sufficiently long time period (at least, for a few years). In doing that, the code ROM employs an algorithm of reduction of overlapping parts of calculations, which has no analogues within the considered class of models. This algorithm allows significantly reduce the time of multiple calculations of long-term releases (by a few orders of magnitude) due to omitting the parts of calculations performed under the same weather conditions.
The code was applied for analysis of various hypothetical emergency scenarios typical for fast reactors. It was demonstrated that the algorithm of reduction of overlapping parts of calculations is efficient.

Country/Int. Organization:

Nuclear Safety Institute of the Russian Academy of Sciences

6.1 CFD and 3D Modeling / 378

3D Modeling of Fuel Handling System for PFBR Operator Training Simulator

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Recent advances in computer science have paved way to increased ease in design of nuclear power plants and imparting efficient operator training thereby improving the safety of nuclear plants. As nuclear power plant is considered safety critical, need to improve the skill set of the control room operators by getting the operator trained in simulator is becoming a mandatory requirement before commissioning the plant. Theoretical models, response and actions in control room panels/consoles, human reasoning etc. are simply not enough to impart full understanding of complex systems in reactor to the operators. Over the last few years, addition of 3D models along with the simulated control systems has gained wide attention in training simulators. Additional 3D models aid the operator in understanding the system and helps in quicker and more efficient training. One of the most important systems in Prototype Fast Breeder Reactor (PFBR) is its fuel handling system. It requires complex predefined sequential blind operations to be performed by the operator. Many of the operations are needed to be carried out in handling control room through panels and consoles. The skill set of the operator needs to be enhanced as the operations are critical and some of them are directly linked with the safety of the nuclear reactor. This paper covers the details of PFBR Full Scope Replica Simulator, various 3D models developed for Fuel handling System along with their associated process and logic models. The sequential procedure based on interlocks and the responses of the 3D fuel handling component models to the operator actions in simulator are also elaborated.

Country/Int. Organization:

India/ Indira Gandhi Centre for Atomic Research, Kalpakkam

Poster Session 1 / 379

The UO2–MeO2 (Me = Th, Pu, Zr) cathode crystalline deposits formation during the melts electrolysis.

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The cathode crystalline UO_2–ThO_2 (30-50 mol/ %), UO_2-PuO_2 (6, 42 and 72 mol. %), UO_2-ZrO_2 (0.1 –98 mol. %) deposits, were formed. The electrolyte - melt: (NaCl-KCl)-UO_2Cl_2-$MeCl_4$, where $MeCl_4$ = $ThCl_4$, $PuCl_4$, $ZrCl_4$. The influence of the $MeCl_4$ concentration in the (NaCl–KCl)–UO_2Cl_2–$MeCl_4$ melt, the initial electrolysis current density, the temperature, and the electrolysis duration on the average MeO_2 concentration in the UO_2–MeO_2 deposits was studied. The MeO_2 fraction in the cathode UO_2–MeO_2 deposits regularly decreased as the $MeCl_4$ concentration decreased and the current density and electrolysis duration increased. The electrolytic of UO_2–MeO_2 cathode crystalline deposit are formed through simultaneous electrolytic reduction of UO_2^{2+} ions to UO_2 and the exchange between the UO_2 and Me^{4+} ions present in the molten (NaCl-KCl)-UO_2Cl_2-$MeCl_4$ electrolyte.

Country/Int. Organization:

Russia/Institute of High-Temperature Electrochemistry UB RAS

Poster Session 1 / 380

OSCAR-Na validation against sodium loop experiments

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The OSCAR-Na code has been developed to calculate the mass transfer of corrosion products and related contamination in the primary circuit of sodium fast reactors (SFR). Indeed, even if fuel cladding corrosion appears to be very limited, the contamination of the reactor components plays an important role in defining the design, the maintenance and the decommissioning operations for SFR.

The modeling is based on the solution/precipitation of the different elements of the steel. These elements dissolve mainly at the hot surfaces, and precipitate on the cold surfaces, and then induce the shifting of the metal/sodium interface (bulk corrosion or bulk deposit). The diffusion in the steel is also taken into account and allows calculating the preferential release of the most soluble elements (nickel, chromium, and manganese).

The code uses a numerical method for solving the diffusion equation in the steel and the complete mass balance in sodium for all elements, allowing the calculation of the metal/sodium interface shifting and of the flux of each element through this interface.

Code validation has already been carried out against PHENIX contamination on heat exchanger surfaces for the main radionuclides. This paper presents the continuation of the validation process against experimental results obtained on sodium loops, namely STCL and TIGIBUS, with well controlled experimental conditions. Different parameters of the model are adjusted to match concentration profiles in the metal and elementary releases measured at 600 ℃. These parameters are the solubility in the sodium and the diffusion coefficient in the steel for each element, as well as the oxygen-enhanced iron dissolution rate. The new values are compared to those published in the literature and discussed. Moreover the modeling allows reproducing the effects of temperature, sodium velocity, and oxygen concentration which were varied in the experiments. The effect of the length of an isothermal pipe (downstream effect) observed in the loops is also considered.

Country/Int. Organization:

France / C.E.A, DEN

8.1 Professional Development and Knowledge Management - I / 382

Development and Deployment of Knowledge Management Portal for Fast Breeder Reactors

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Knowledge Management is the process of creating value from an organization's tangible and intangible assets and regarded as a significant contributing tool to enhance the performance of organization. Knowledge accumulated over decades of nuclear research, development & operation (organizational memory) have to be preserved and used for the future design, innovations and continued safe operation of nuclear plants. IGCAR's IT-enabled nuclear knowledge management system is designed as a generic, customizable framework and developed in-house fully using open-source platform and APIs. This paper describes the development and implementation of web-enabled, taxonomy based, advanced knowledge management system for effective management and utilization of the Prototype Fast Breeder Reactor(PFBR) records available in the form of Control Notes, Design Notes, Operation Notes, Experiments Notes, Specifications, Project Reports, Commissioning Documents, Test Procedures & Reports, Manuals, Drawings etc. The portal deployed acts as a gateway to FR

Knowledge repository and enables collection, retrieval, preservation and presentation of knowledge assets in different forms. It also highlights the capabilities with which the system has been designed like controlled-vocabulary based organization of documents, multi-format document upload facility with meta-data, enhanced authentication and multi-level access control, advanced search and retrieval mechanism, online viewing and print requests management, dynamic reports generation facility.

Country/Int. Organization:

Indira Gandhi Centre for Atomic Research, Kalpakkam, India

3.1 Safety Program / 385

Safety Assurance for BN-1200 Power Unit During Accidents

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Safety analysis of BN-1200 RP Unit is performed with account of requirements specified for innovative nuclear technologies including IAEA recommendations.
The following BDBAs are considered within BN-1200 project:
 Heat removal accidents with loss of energy supply, failures of active systems for reactor shutdown, failures of normal cooldown system and active components of emergency cooldown system;
 Accidents with insertion of positive reactivity and failure of active reactor shutdown systems;
 Accidents with plugging up of FSA flow area.
Computational validation of safety in accidents is performed using verified computational codes of new generation.
To validate safety, an integrated approach is applied that analyses main processes and phenomena occurring in the RP and Unit's rooms during accident.
Computational results showed that there is no need to evacuate or move out population under projected population radiation doses for population beyond the NPP site.

Country/Int. Organization:

Russia
Joint Stock Company "Afrikantov Experimental Design Bureau for Mechanical Engineering"

2.3 Decommissioning of Fast Reactors and Radioactive Waste Management / 386

ARRANGEMENT OF THE BN-600 REACTOR CORE REFUELING AT TRANSITION TO THE INCREASED FUEL BURN-UP

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The further fuel burn-up increase in the BN-600 reactor is believed to be connected with transition to a new more radiation-resistant construction material of fuel rod claddings that is steel EK164-ID c.d. The core is planned to be switched to the increased fuel life time of 752 eff. days with the fivefold refueling scheme of base FSA quantity.
To switch the reactor operation mode from the existing fourfold scheme to the fivefold one, a special order of FSA refueling in transient period is needed. The main special feature of the period is that

five groups to refuel FSAs are formed differing in their operating time.
To arrange FSA refueling, the following approach is adopted. Each of the four existing FSA groups is divided into two subgroups: the main group and the additional one. FSAs of the four main groups make four groups of a new refueling scheme. Refueling of these FSAs is performed as per usual scheme replacing life-expired FSAs with "fresh"ones.
To form the fifth group, premature unloading of the four additional FSA subgroups is needed which need to operate 1-2 intervals to achieve their design service life. To decrease losses caused by the premature unloading, one of the four additional FSA groups returns to the core for reburning after its temporary conditioning in in-vessel storage.

Country/Int. Organization:

Russia
JSC "Afrikantov OKBM"

Poster Session 1 / 387

A Demand Driven Way of Thinking Nuclear Development –Neutron Physical Feasibility of a Reactor Directly Operating SNF from LWR

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Invention and innovation with regards to nuclear reactor development can be described with the concept of developments in s-curves. This view is taken to identify the problems of nuclear development. Following this, the request are formed for the definition of demand driven objectives. The future objectives should follow the UN sustainable development goals. The key words to form the vision of electric energy production are: very limited request for resources and production of waste, affordable economics and safe, secure and reliable operation which can be assembled to the dream solution –the perpetuum mobile. It is concluded that a reactor operating in closed fuel cycle using spent nuclear fuel from Light Water reactors would come close to this vision. Additionally, the technology fully fulfils the UN sustainability request of using technologies which provide future generation with solutions to increase the amount of available resources.
Following this demand driven theoretical discussion of objectives, a new innovative proposal is presented. A proposed reactor which is operated directly on SNF from LWRs as main fuel resource. The simulation tools and the limitation of the simulation are discussed. A proof of feasibility is given from neutron physical point of view. The major challenge is to establish a breeding process which provides enough new fissile material from the inserted SNF. For the start-up of the system a support of fissile material in the initial core and the transition phase is required. The feasibility of sufficient breeding is demonstrated, a first estimation of the resources in a possible fuel cycle is given, and the consequences on the back end of the fuel cycle are discussed. Finally, the challenges of the proposed technology are highlighted to stimulated future R&D to make a sustainable innovative nuclear reactor possible. This could form attractive major innovation challenge for a wide variety of engineers to form the basis for the long term success of nuclear reactors as a major carbon free, sustainable, and applied highly reliable energy source.

Country/Int. Organization:

United Kingdom

Poster Session 1 / 388

On the feasibility of Breed-and-Burn fuel cycles in Molten Salt Reactors

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Reactors able to operate in a Breed-and-Burn fuel cycle can operate at equilibrium with minimum fissile input and fuel processing without actinides separation, making them particularly attractive. Historically, the concept has essentially been considered for solid-fuel fast reactors, in particular to sodium-cooled fast reactors. Their performance in breed-and-burn is however limited by the neutron fluences necessary to reach break-even neutron generation in the fertile feed and their inhomogeneous burn-up. However, molten salt reactors do not suffer from such limitations, but have other constraints due to their homogeneous nature. In this paper, the constraints in term of fuel selection and utilization and reactor dimensions are studied and reported.

Country/Int. Organization:

Paul Scherrer Institut (PSI), Switzerland
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5.7 Chemistry Related Technology / 390

Helium Recovery from Guard Vessel Atmosphere of the ALLEGRO Reactor

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ALLEGRO is a helium-cooled experimental fast reactor, which is under development by the consortium „V4G4 Centre of Excellence"(Czech Republic, Hungary, Poland and Slovakia) associated with France. The current pre-conceptual stage of development by the V4G4 CoE is based on the 75 MWt concept presented by CEA in 2009. The main purpose of ALLEGRO is 1) Demonstration of viability of the gas-cooled fast reactor (GFR) technology in pilot scale, 2) Testing of innovative carbide-based refractory GFR fuels in the start-up oxide core driver, 3) Qualification of other GFR-specific technologies such as components of the primary circuit, helium-related systems, fuel handling etc.

One of the GFR-related helium technologies is the recovery system for the helium leaked from the pressurized boundaries such as primary circuit into the gas-tight guard vessel (GV - a metallic pressure boundary around the primary circuit filled with 0.1 MPa nitrogen, whose main function is to provide backpressure of at least cca 0.4 MPa to make the emergency cooling system functioning

properly in loss-of-coolant accident conditions). The helium recovery system will be able to separate the leaked helium from the nitrogen-helium GV atmosphere and return it into the helium storage system. This feature will make the future GFRs (including ALLEGRO) much less dependent on the helium market.

The paper describes a helium recovery system with emphasis on both the separation method and the proposed (pre-conceptual) technical solution. The system is based on multi-cycle semipermeable membrane separation and is expected to be operated not continuously, only when the helium concentration in the GV exceeds a certain limit. Up to 99% helium purity can be reached by this process, while the final purification is planned to be performed by other methods, e.g. by using a pressure swing adsorption (PSA) technology, which can achieve high purity helium. For good yield of helium a membrane with high selectivity of nitrogen/helium should be used. A membrane-based recovery system was developed, tested on laboratory scale and a first concept of the recovery system was proposed for the conditions of ALLEGRO. The work has been so far supported mainly by the Technology Agency of Czech Republic.

Country/Int. Organization:

UJV Řež, a. s.
Czech republic

5.7 Chemistry Related Technology / 392

Methods of controlling concentration of oxygen dissolved in heavy liquid metal coolants (lead and lead-bismuth) of nuclear reactors and test facilities

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Heavy liquid metal coolants (HLMC) including lead-bismuth and lead make significant corrosion and erosion impact on structural steels. The only way to assure the long-term reliable operation of steels of the primary system components operating in contact with HLMC is to provide protective coating on the steel surface. By now, oxygen-based technology of structural steel surface passivation has been chosen for this purpose. This technology implies formation of protective oxide films on the steel surface and assurance of their integrity during plant operation by maintaining specified oxygen potential of coolant.
In case of HLMC circuits operation without purposeful supply of dissolved oxygen to the coolant spontaneous deoxidization of coolant takes place down to the level, at which corrosion protection of structural steels cannot be provided. Therefore, stable and reliable protection of steels contacting with HLMC requires adding of dissolved oxygen to the coolant on a regular basis during specified plant lifetime.
SSC RF –IPPE specialists proposed various methods of continued maintaining of desired HLMC oxygen potential in both nuclear power plants and experimental facilities. These methods have been developed using test facilities. Currently, the efforts are focused on the development of systems and equipment for implementation of technology of maintaining specified oxygen potential of HLMC in the advanced reactor designs (BREST-OD-300, SVBR-100, etc.).
Solid-phase method of oxygen content control for maintaining specified oxygen potential in HLMC is the most promising. This method developed by the SSC RF –IPPE is based on the use of mass exchangers with solid-phase source of oxygen.
For the moment, over 50 various mass exchanger designs have been developed by the SSC RF –IPPE specialists. They differ from each other in the arrangement of solid-phase lead oxides dissolving process. Large experience of their operation in test facilities with HLMC has been gained.
In the paper presented are the various methods of maintaining specified HLMC oxygen potential and

approaches to their implementation based on the experience gained in operating various experimental facilities and power plants located in the research institutes of the Russian Federation.

Country/Int. Organization:

Russian Federation, JSC "SSC RF –IPPE"

5.7 Chemistry Related Technology / 393

Strategies of maintaining appropriate technology of heavy liquid metal coolants in advanced nuclear power plants

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Up to date, the following three main problems can be emphasized concerning technology of Pb-Bi and Pb coolants used in the civilian reactors:
1) maintaining high purity of coolant and cleanness of the surfaces of NPP circuits in order to assure design thermal-hydraulic performance of the plant operating on power up to 100 % during several decades;
2) long-term prevention of corrosion and erosion of structural materials (plant operating on power up to 100 % during several decades);
3) meeting of up-to-date safety requirements in the various stages of reactor operation (coolant preparation, reactor start-up, day-to-day operation, repair and refueling operations, loss of integrity, and abnormal operating conditions).
As follows from the above problems, up-to-date set of measures concerning Pb and Pb-Bi coolants technology should assure implementation of the following procedures:
1) preparation of coolant (Pb-Bi or Pb) and filling of NPP circuits;
2) preliminary passivation of reactor plant elements before their installation;
3) passivation of the inner surface of the primary circuit of the reactor plant;
4) coolant technology concerning repair and refueling procedures;
5) coolant purification and removal of impurities from the circuit surfaces during day-to-day operation;
6) control of coolant oxidizing potential during operation of NPP with HLMC;
7) purification of cover gas in the liquid metal circuit;
8) technological procedures under abnormal operating conditions;
9) technological procedures aimed at coolant reuse.
All necessary technological measures can be implemented using special equipment created on the basis of the modern realities, which would be the essential part of the plant safety system in all stages of its operation.

Country/Int. Organization:

Russian Federation, JSC "SSC RF –IPPE"

Poster Session 1 / 394

SELECTION OF CARRIER SALT FOR MOLTEN SALT FAST REACTOR

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The salt compositions based on the fluorides of alkaline elements (Li, Na, K) have all the necessary properties for the reliable molten-salt reactor operation. Currently available information on the molten salts properties allows to make some recommendations for the choice of the carrier salt for the fast MSR of the different types. The high solubility of the actinide and lanthanide fluorides in FLiNaK combined with its other physical and chemical characteristics allows to consider it as the perspective one for the fast MSR with U-Pu fuel cycle
The choice of the carrier salt for the molten salt reactor (MSR) is discussed. The special attention is paid for the solubility of PuF3 in these salts which is necessary for the fast molten salt reactor with U-Pu fuel cycle (U-Pu FMSR) operation. It is shown that the PuF3 solubility, UF4 and AmF3 in the eutectic LiF-NaF-KF (FLiNaK) at 700°C is ~ 30, 45 and 40 mole%, respectively. This result opens the way for the development U-Pu FMSR with closed nuclear fuel cycle as well as the effective reactor-burner of minor actinides. The viscosity, corrosive activity and other properties of the molten fluoride salts are outlined.

Country/Int. Organization:

Rosatom State Corporation Enterprise "State Scientific Center –Research Institute of Atomic Reactors"

1.3 SYSTEM DESIGN AND VALIDATION / 395

ASTRID FUEL HANDLING ROUTE FOR THE BASIC DESIGN

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At the beginning of the Basic Design phases of ASTRID starting in 2016, the entire fuel handling route has been challenged in order to improve some aspects like availability and investment cost. Especially, studies performed at the end of preliminary design (2010 –2015) phase show that the availability target will be difficult to obtain with a significant risk of malfunction because of multiple handling operations in series. Two main changes have been then decided: the implementation of an in-sodium external buffer zone, similar with an in-sodium external storage but associated with an in-primary vessel storage to reduce its size and its allowable residual power, and the merging of the fresh assembly storage with the spent fuel assembly storage that allows both reduction of size and equipment. These new options have the first advantages to reduce drastically the number of operation that has to be done during scheduled outage and minimizing the overall Balance Of Plant. After description of the entire fuel handling route adopted for ASTRID, this paper aims at identifying the advantages of this new option and points the remaining issues or questions that will be studied in details during the current Basic Design phase.

Country/Int. Organization:

FRANCE/COMMISSARIAT A L'ENERGIE ATOMIQUE

5.10 Fuel Modeling and Simulation / 396

EXPERIENCE AND APPLICABILITY OF HIGH DENSE METAL URANIUM IN ADVANCED BN-REACTORS

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To guarantee an inherent safety in advanced BN-reactors the breeding ratio of its active core (BRC) must have the meaning of BRC ≥ 1,0. It could be reached in heterogeneous oxide-metal cores of various types in use of metal uranium as the fertile components in the proportion of MOX:U ≈ 2:1. We obtained the experience of manufacturing the fertile columns of various types from the metal uranium, the experience of manufacturing and irradiation in fast reactors BOR-60 and BN-350 of full-size elements (FE) and fuel assemblies (FA) that have such columns (4010 elements in part of 108 fuel assemblies).
Besides the obtaining of the inherent safety in advanced BN-reactors with the heterogeneous oxide-metal cores of various types (by FA-heterogenization of the core, IFAH –by intra FA-heterogenization, IFEH –by intra fuel elements heterogenization) we could achieve considerable additional economic and ecological preferences. Among them there are the increase of the admissible average burnup of MOX-fuel by ≈20%, the decrease of the mass of manufactured and consumable Pu-containing MOX-fuel by ≈30%, the decrease of consumable Pu-containing FE or Pu-containing FA by ≈30%, etc.

Country/Int. Organization:

Russian Federation, Joint Stock Company
"State Scientific Center –Research Institute of Atomic Reactors"

6.3 Neutronics - 1 / 397

Verification of the neutron diffusion code AZNHEX by means of the Serpent-DYN3D and Serpent-PARCS solution of the OECD/NEA SFR Benchmark

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AZNHEX is a neutron diffusion code for hexagonal-z geometry currently under development as part of the AZTLAN project in which a Mexican platform for nuclear core simulations is being developed. The diffusion solver is based on the RTN0 (Raviart-Thomas-Nédélec of index 0) nodal finite element method together with the Gordon-Hall transfinite interpolation which is used to convert, in the radial plane, each one of the four trapezoids in a hexagon to squares. The main objective of this work is to test the AZNHEX code capabilities against two well-known diffusion codes DYN3D and PARCS. In a previous work, the Serpent Monte Carlo code was used as a tool for preparation of homogenized group constants for the nodal diffusion analysis of a large U-Pu MOX fueled Sodium-cooled Fast Reactor (SFR) core specified in the OECD/WPRS neutronic SFR benchmark. The group constants generated by Serpent were employed by DYN3D and PARCS nodal diffusion codes in 3D full core calculations. A good agreement between the reference Monte Carlo and nodal diffusion results was reported demonstrating the feasibility of using Serpent as a group constant generator for the deterministic SFR analysis. In order to verify the under development solver inside AZNHEX, the same Serpent generated cross sections sets for each material were exported to AZNHEX format for four different states (as in DYN3D and PARCS): a) a reference case in which the multiplication factor (keff) is the compared value, b) the Doppler constant (KD), c) the sodium void worth, and d) the total control rod worth. Additionally, the radial power distribution was also calculated. The results calculated with AZNHEX showed also a quite good agreement in the direct comparison with DYN3D (-66 pcm in keff) and PARCS (-109 pcm in keff) and therefore against the Serpent reference solution (-194 pcm in keff). As AZNHEX is still under development further improvements will be implemented and new tests will be carried out, but so far the results presented here give confidence in the development.

Country/Int. Organization:

Mexico/ININ,IPN
Germany/HZDR

7.1 Sustainability of Fast Reactors / 399

Evaluation results of BN-1200 compliance with the requirements of GENERATION IV and INPRO

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The project of a power unit with BN-1200 reactor is designed using advanced technical solutions, which define evolution of the fast breeder technology in the field of safety parameters and in the field of technical and economical indicators.
At present there was completed estimation of the BN-1200 project from the point of view of its compliance with the requirements of nuclear energy systems of Generation IV in the frames of International forum Generation IV, and comparison of the BN-1200 project with other fast breeder projects using NESA INPRO procedure, developed and verified for comparison of nuclear energy systems with PWR.
The paper presents the results of preliminary estimation by the INPRO procedure, which showed that BN-1200 has good margin of safety and economical characteristics in comparison with the previous projects; and BN-1200 meets all the basic principles in the fields of 'safety'and 'economics'; and BN-1200 can ensure sustainable development of the nuclear energy system.
Estimation results of the BN-1200 concept for compliance with the requirements to Generation IV plants testify that BN-1200 project, as a whole, has good potential from the point of view of compliance with the stated requirements.

Country/Int. Organization:

Russia\
JSC "Afrikantov OKBM"

Poster Session 1 / 400

ASTRID reactor: design overview and main innovative options for Basic Design

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Abstract FR 17

Track 1. Innovative Fast Reactor Designs
ASTRID reactor: design overview and main innovative options for Basic Design

F. Chanteclair(a) in association with all the ASTRID industrial partners

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Sodium-cooled Fast Reactors (SFR) is one of the Generation IV reactor concepts selected to secure the nuclear fuel resources and to manage radioactive wastes. In the frame of the June 2006 French act on sustainable management of radioactive materials and wastes, French Government entrusted CEA (French Commission for Atomic Energy and Alternative Energy) to conduct design studies of ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) prototype in collaboration with industrial partners.
The ambitious objectives of ASTRID reactor are to fulfil the GEN IV requirements. It has led to the implementation of innovative technological solutions which go beyond the current feedbacks. Necessarily, these innovations will have to be consolidated within the framework of Research & Development actions and qualification programs.

In its Basic Design stage, ASTRID has built a coherent conceptual design configuration with innovative techniques and systems across all domains: core, fuel assembly technology, nuclear island, civil engineering, energy conversion system, plant layout, ISI&R, fabricability, ⋯and even in the project management.

The object of this document is to provide an overview of the significant innovations under consideration on ASTRID. It will also allow to present the partners contribution to this seek for innovations for better performances and/or enhanced safety.

Country/Int. Organization:

French Alternative Energies and Atomic Energy Commission (CEA)Cadarache center France

Poster Session 2 / 401

Development of Research Nuclear Power Facility with MBIR Multi-Purpose Fast Neutron Research Reactor

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This paper presents the technical parameters and the experimental capabilities of the MBIR reactor; it shows the main reactor engineering solutions of the project and describes the key events related to the implementation of the MBIR construction project.

Country/Int. Organization:

Russia, JSC “SSC RIAR”

1.1 SFR DESIGN & DEVELOPMENT - 1 / 402

Development of the new generation power unit with the BN-1200 reactor

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One of the most important stage of works for the BN-1200 power unit project, which are implemented since 2007 according to Rosenergoatom Concern JSC program and Target Federal program "Nuclear energy technologies of the new generation for 2010-2015 and for perspective to 2020", became the development of technical designs of the RP, turbine plant and materials of the power unit project in 2014.
Main requirements at the technical design of the RP project defined development of the technical solutions in comparison with the solutions implemented in the previous projects, and ensured not only complete integration of the primary circuit in the reactor pressure vessel, but also significantly reduced number of systems, equipment, valves and pipelines, and optimized architectural solutions of the main and auxiliary buildings and constructions of the power unit, and optimized general layout of the site. These improved the main technical and economical indicators of the BN-1200 power unit and ensured their comparability with VVER RP not only in the field of safety, but also in the field of specific capital costs and LCOE.
Further development of the project was defined with the design research of systems and equipment in the second half of 2015 and 2016, which indicated the following main directions of design work:
increase of the power of the unit without change of the equipment design;
change of design and layout solutions of the primary circulating pumps, emergency heat removal system, cold absorption trap filter of the secondary circuit, refueling box, and the secondary circuit.
Implementation of the proposed technical solutions defines further optimization of the architectural solutions for the power unit and improvement of the technical and economical indicators without reduction of the safety level.

Country/Int. Organization:

Russia/JSC "Afrikantov OKBM"

5.5 Large Component Technology I / 404

Development of the built-in primary sodium purification system for the

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To purify primary sodium in the advanced BN-1200 reactor plant, a purification system with cold traps has been used that are located in the reactor vessel (in-built purification system). Such decision has excluded external communications of the auxiliary system with radioactive sodium and respectively a possibility that sodium will outflow to compartments outside the reactor.
The sizes of cold traps located in the reactor are small that has limited the sodium flowrate through them, impurity storage capacity, and has made it necessary to replace traps in the course of reactor plant operation. Cold traps include such main components of the conventional external purification system as sodium communications, a portion of the cooling circuit, flow meter devices, and electromagnetic devices (a pump and throttle pump) to ensure sodium circulation and to control the sodium flowrate. In the course of development, options have been considered to cool traps with argon at the pressure of 1.5 MPa, liquid sodium, and gallium.
To validate operation of electromagnetic devices for the cold trap, a package of research activities and R&D activities has been done:

- Thermal irradiation studies have been done of sample electrotechnical materials intended for the electromagnetic pump and throttle pump.
- Mockups of the electromagnetic pump and throttle pump have been manufactured and tested.

Country/Int. Organization:

Russia/JSC "Afrikantov OKBM"

1.4 CORE AND DESIGN FEATURES - 1 / 405

BN-800 core with MOX fuel

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One of the main objectives for BN-800 development is to master the closed nuclear fuel cycle technologies using the mixed uranium-plutonium (MOX) fuel. The initial loading of BN-800 with the hybrid core mainly made of the uranium oxide fuel includes the limited quantity (16%) of MOX fuel subassemblies fabricated at experimental production facilities of Mayak Production Association and JSC SSC RIAR.
The core will be completely fueled with the MOX fuel fabricated at the Mining and Chemical Combine by step-by-step replacement (three refuelings) of fuel subassemblies in the hybrid core with MOX fuel subassemblies. To flatten power distribution, the core uses three types of fuel subassemblies with the different plutonium content in the fuel. A technique to adjust plutonium enrichment in fuel depending on the fuel isotope composition makes it possible to fabricate plutonium-based MOX fuel with a wide range of isotope compositions and retain core operation parameters within the design limits.
To reduce the sodium void reactivity effect, the fuel subassembly design has an upper sodium cavity and absorbing shield made of natural boron carbide.
In the BN-800 core, the ChS-68 steel is used for fuel rod claddings the same as for standard fuel rods in BN-600. Fuel rods with such cladding continue to operate up to the damaging dose of ⊠ 90 dpa, which as applied to BN-800 corresponds to the average burnup of 66 MW day/kg for the fuel unloaded.
Prospects to increase BN-800 fuel burnup are connected with a transition to the more radiation-resistant steel EK-164 for fuel rod cladding and later on, to ferritic-martensitic steels and oxide-dispersion-strengthened ferritic steels.

Country/Int. Organization:

Russia/JSC "Afrikantov OKBM"

1.4 CORE AND DESIGN FEATURES - 1 / 406

SELECTION OF A LAYOUT FOR THE BN-800 REACTOR HYBRID CORE

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The initial loading of BN-800 is mainly made of the uranium oxide fuel and partially of MOX fuel subassemblies (16% of the total quantity), which were fabricated using both the pellet technology and vibro-packing technology. With account of this specific completing process, such core is called the hybrid core.
The core layout was selected to simplify the future transition from the hybrid core to the core fully loaded with the MOX fuel and to maximally adapt BN-600 uranium fuel subassemblies fabrication to BN-800 uranium fuel subassemblies fabrication.
The hybrid core uses three types of fuel subassemblies with the different content of fissile material (degrees of enrichment) to retain fuel enrichment limits and fuel column height the same as in the MOX core. To ensure compatibility of uranium fuel subassemblies and MOX subassemblies, the plutonium content of the MOX fuel was defined to retain the same physical efficiency for respective types of fuel subassemblies. To minimize distortion of the power field, the MOX fuel subassemblies are arranged in the periphery of the hybrid core (within the high enrichment zone). Fuel subassemblies with the MOX pellet fuel are arranged in the first row, and fuel subassemblies with the vibro-packed MOX fuel are arranged in the peripheric row, under less severe operation conditions.
The report discusses the main prerequisites to develop the hybrid core, describes the core design, and gives information about main operation characteristics.

Country/Int. Organization:

Russia/JSC "Afrikantov OKBM"

Poster Session 1 / 407

ASTRID hot cells

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Christoph DÖDERLEIN (CEA)

The ASTRID reactor is the French demonstrator for Generation IV sodium cooled fast reactors and needs as such to respond to challenges in the qualification of innovative components and materials. Considering its role as R&D platform for the fast reactor line to come, ASTRID will be endowed with a set of hot cells. The French company SEIV, subsidiary of the ALCEN group, has been in charge since 2013 of the full preliminary design of this facility.
The main purpose of the ASTRID hot cells is to perform non-destructive examinations (NDE, such as visual inspection, 3D X-ray tomography, dimensional inspection, eddy current testing) on the spent core sub-assemblies and fuel pins. To extract the latter from the sub-assemblies, a dismantling unit is foreseen in the facility.
The proposed paper gives a description of the components and capabilities of the ASTRID hot cell facility. The facility consists of a main cell, where the NDE equipment are installed, the lower cells with the dismantling machine and 3D X-ray scanner device and finally the upper cell which serves as an airlock for handling functions.
The ASTRID hot cells will feature remote operations of NDE equipment, eg. with new generation manipulator with electrical master arm using haptic technology. This design aims to minimize of the use of expensive lead windows, increase handling capabilities and improve operator ergonomics.

Country/Int. Organization:

SEIV is a french company from ALCEN Group, CEA partner for the 4th generation reactor.

1.6 CORE AND DESIGN FEATURES - 2 / 408

Specific features of BN-1200 core in case of use of nitride or MOX fuel

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The core of a commercial reactor BN-1200 designed for operation with developed MOX-fuel or advanced nitride fuel. The common approach to the design of the specified cores was:
- reduction of the fuel rating, in comparison with BN-600 and BN-800, which permits to use larger fuel pins (Ø 9.3 mm) to reduce their consumption and minimizing reactivity margin for fuel burn-up;
- use of the fuel with the same plutonium enrichment for all FA to ensure stability of the core power density during operation between refueling and simplification of the fuel fabrication;
- use of FA design with top sodium cavity and born absorption shield for minimizing sodium void reactivity effect;
- an annual interval of operation between refueling is accepted.

In view of difference of neutron physical characteristics of the nitride and MOX-fuel the specified variants of the core have differences in the design:
- in case of operation with nitride fuel the plutonium breeding rate in the core is self-sufficient, and in comparison with the MOX-fuel core, it has no frontal and lateral breeding zones. To reduce reactivity margin and additionally equalize power density for peripheral FA there are used larger diameter (Ø 10.5 mm) fuel pins;
- configuration of the MOX-fuel core has axial breeding layer to reduce reactivity loss rate due to fuel burn-up and to ensure annual operating interval between refueling. Additional advantage of this configuration is the reduction of the accumulation rate of the damaging dose.

Country/Int. Organization:

Russia/JSC "Afrikantov OKBM"

5.4 Advanced Fast Reactor Cladding Development II / 409

OPERABILITY VALIDATION OF FUEL PINS WITH CLADDINGS MADE OF EK164-ID STEEL IN THE BN-600 REACTOR

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To ensure the increased fuel burn-up in the BN-600 and BN-800 reactors, EKA64-ID steel is going to be used as a fuel rod cladding material because it has bigger index of radiation resistance (swelling and creeping) in comparison with the used ChS68-ID steel. To introduce this steel, an irradiation examination of experimental FSAs is needed to be performed. Owing to the irradiation examination, experimental data will be obtained to validate FSA operability and a database on properties of the steel will be updated thanks to which computational codes will be verified. Tests are performed as per appropriate procedure in cooperation with operating organization and Rostechnadzor experts.

By now, reactor examination of 14 experimental FSAs has been successfully performed in the BN-600 reactor. The maximum achieved irradiation parameters are as follows: the fuel burn-up is ~ 14 % h.a., the damaging dose is ~ 100 dpa. The examination is planned to be continued for using the steel as fuel rod cladding with higher parameters: the fuel burn-up should be 14.8% h.a., the damaging dose should be ~ 112 dpa.

Activities aimed at improving the quality of cladding tubes both in the stage of fuel rod cladding manufacture and in the metallurgic stage of tubing stock manufacture are performed simultaneously with manufacture and irradiation of the experimental FSAs.

Results of these experimental activities will be used to validate operability of fuel rods made of this steel in the initial stage of the BN-1200 reactor operation.

Country/Int. Organization:

Russia/JSC "Afrikantov OKBM"

3.2 Core Disruptive Accident / 410

Status of severe accident studies at the end of the conceptual design: feedback on mitigation features

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The ASTRID reactor developed by the CEA with its industrial partners, will be used for demonstration of the safety and operability, at the industrial scale, of sodium fast reactors of the 4th generation. Among the goals assigned to ASTRID, one is to improve the safety and the reliability of such reactor (compared to previous sodium-cooled fast reactors). Among the innovations promoted in the ASTRID design, a low sodium worth core concept (CFV core) has been developed. By means of various design provisions enhancing the neutron leak in case of sodium draining, the overall sodium void effect of the ASTRID core is near zero and could even be negative. Additionally, mitigation devices should be implemented into the core in order to limit the calorific energy released in the fuel during the secondary phase of the accident.

This paper deals with a synthesis of severe accidents studies performed during the second period of the pre-conceptual design stage of the ASTRID project (2013-2015). The main insights of the studies in term of mitigation strategy and of mitigation device design are highlighted in the paper. The core transient behavior has been investigated in case of generalized core melting situations initiated by postulated reactivity insertion ramps (UTOP) and unprotected loss of flow (ULOF). In case of postulated reactivity insertion ramps, the mechanical energy release assumed to be released by molten fuel vapor expansion does not exceed several tenths of megajoule ULOF transient does not lead to energetic power excursions neither thanks to the mitigation provisions and to the core design. Moreover, the ULOF early boiling phase leads to core power decrease. Thus, the primary phase of the accident is not governed by a power excursion. The paper deals with the approach and the presentation of preliminary findings regarding mitigation provisions. Those provisions are investigated by considering a core degraded state representing the end of the transition phase. The scenario possible evolutions from this degraded state provide the following parameters: mass and temperature of molten materials, mass and flow rate of materials relocated on the core catcher and possible ejected material mass above the core. Those parameters are used for the determination of approximate loadings for the primary vessel and for the design of the core catcher. Finally, a methodology and first results dedicated to assess the efficiency of mitigation device design is presented as well as first preliminary results of checking process.

Country/Int. Organization:

CEA Cadarache

6.4 Neutronics –2 / 411

Solution of the OECD/NEA SFR Benchmark with the Mexican neutron diffusion code AZNHEX

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The AZTLAN Platform project is a Mexican national initiative led by the National Institute for Nuclear Research of Mexico, which brings together nuclear institutions of higher education in Mexico: the National Polytechnic Institute, the National Autonomous University of Mexico and the Autonomous Metropolitan University, in an effort to take a significant step towards positioning Mexico, in the medium term, in a competitive international level on nuclear reactors analysis and modeling software. The project is funded by the Sectorial Fund for Energy Sustainability CONACYT-SENER and one of its main goals is to build up as well as strengthen the national development of specialized nuclear knowledge and human resources. The AZTLAN platform consists of several neutronics and thermal-hydraulics modules. Among the neutronics tools, the AZNHEX code has been developed. AZNHEX is a 3D diffusion code that solves numerically the time dependent neutron diffusion equations in hexagonal-z geometry. The diffusion solver is based on the RTN0 (Raviart-Thomas-Nédélec of index 0) nodal finite element method together with the Gordon-Hall transfinite interpolation which is used to convert, in the radial plane, each one of the four trapezoids in a hexagon to squares. In order to support and provide reliability to the platform, a stringent verification and validation (V&V) process in which the use of international Benchmarks and Monte Carlo reference solutions has been started. As a part of this V&V activities, results obtained with AZNHEX for the full-core simulations of the two nuclear cores of the OECD/NEA SFR Benchmark (a 1000 MW metallic-fueled and a 3600 MW MOX-fueled) are shown and compared with the ones obtained with the reference Monte Carlo code Serpent. The cross sections sets used in AZNHEX were also generated in a previous step with the Serpent code to maintain consistency between calculations. The obtained Results for keff, sodium void worth and control rods worth are within reasonable agreement; in the order of tens of pcms. The results presented are not only useful for the verification of AZNHEX, but also these ones help to define a well-tested methodology in order to generate cross section sets for future dynamic calculations with AZNHEX. Based on the results, the strengths and limitations of the AZNHEX code are discussed in the conclusions and a series of improvements have been identified and planned to be implemented.

Country/Int. Organization:

MEXICO/INSTITUTO NACIONAL DE INVESTIGACIONES NUCLEARES

7.4 Fuel Cycle Analysis / 412

Primary Analysis on The Nuclear Energy Development Scenario base on the U-Pu Multicycling with PWR, FR and CNFC in China

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As one of the largest developing country, China needs reliable energy supply. At the same time, China should improve the energy structure and reduce carbon dioxide emissions. Nuclear and renewable energy is the main solution to these problems. According to some studies, nuclear power

capacity will increase to 400GWe in 2050. Due to limitations of uranium resources, we must consider the development of fast reactor (FR) and closed nuclear fuel cycle. Development Strategy of China's FR is three-step model "Experimental Reactor - Demonstration Reactor - Commercial Reactor". The construction of the China Experimental Fast Reactor (CEFR) has completed, and obtain the necessary experience on FR. The design of the demonstration FR CFR-600 is ongoing, which is 600MWe power. After this step, the commercial FR with more large power will be constructed. Based on the development of nuclear energy and the constraints of uranium resource in China, this article presents and analyses some cases of nuclear power scenarios of PWR-FR matching development with closed nuclear fuel cycle (CNFC) including some indicators such as the matching capacity, the uranium resource consumption, reprocessing capabilities etc.

Country/Int. Organization:

China

1.6 CORE AND DESIGN FEATURES - 2 / 414

CORE CONDITON MONITORING IN ADVANCED COMMERCIAL SODIUM BN-1200

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One of the most important criteria of successful NPP functioning is ensuring personnel and population radiation safety by defense in depth safety barrier system. Primary coolant and reactor gas radioactivity depends on condition of fuel pins, claddings of which can lose their leaktightness during operation due to various reasons (factory fault, changing of irradiation conditions). That is why, failure detection of fuel pin claddings, i.e. the main safety barrier, is of primary importance during core condition monitoring in the process of reactor plant operation.
The task of fuel pin cladding failure detection (FCFD) at sodium reactor is carried out during reactor power operation (operational in-vessel systems of FCFD) and at a shutdown reactor (non-operational in-vessel and ex-vessel systems of FCFD).
Operational FCFD systems of BN reactor plants, both operating and prospective ones, comprise several systems:
 Gas system of FCFD registers appearance of leaky fuel pins in core analyzing activity of various fission gases in reactor gas blanket;
 Sector system of FCFD registers appearance of leaky (by fuel) fuel pins recording delayed neutrons from fission fragments brought in the primary coolant through fuel pin cladding fault;
 Sodium system of FCFD registers level of primary sodium contamination by various radio nuclides (mainly by 137Cs).
Non-operational systems of FCFD comprise the following ones:
 In-vessel failed fuel detection system (SODS-R) is intended to detect leaky FSAs at shutdown reactor based on the results of FCFD sector system data analysis;
 In-washing socket failed fuel detection system (SODS-GO) is intended to detect leaky FSAs measuring fission product activity in washing media (gas, water, steam).
Efficiency of FCFD system is confirmed by operating experience both in Russia and abroad. FCFD system application enables to improve operating safety, ensures monitoring of the spent fuel storage, and contributes to ensuring high operational indices of power unit.
The paper considers existing FCFD systems for fast reactors and discusses opportunities to apply experience of their design and operation to develop FCFD system for the prospective commercial sodium BN-1200 reactor.

Country/Int. Organization:

Russia/JSC "Afrikantov OKBM"

Poster Session 2 / 416

INTEGRATED R&D TO VALIDATE INNOVATIVE EMERGENCY HEAT REMOVAL SYSTEM FOR BN-1200 REACTOR

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Decay heat removal from BN-1200 to atmospheric air is performed using passive DHRS. Several innovative solutions are applied during the DHRS development:
- hydraulic connection of the decay heat exchanger (DHX) with reactor high pressure plenum, installation of check valve at DHX outlet;
- application of slide valve of air heat exchanger with passive elements;
- natural circulation in all DHRS circuits.

Set of experimental research is performed at facilities with water and sodium:
- hydraulic testing of several options of DHX check valve are performed using scaled-down models;
- main characteristics of the check valve are determined using full-scale model;
- in the closed position of check valve diffusion like process was investigated (long-term testing);
- experimental studies of thermohydraulic characteristics are performed using scaled-down reactor model with one DHRS loop.

Integrated computational validation of DHRS effectiveness is performed using new generation code SOCRAT-BN.

Country/Int. Organization:

Russia/JSC "Afrikantov OKBM"

2.1 Commissioning and Operating Experience of Fast Reactors I / 417

Main R&D objectives and results for under-sodium inspection carriers –Example of the ASTRID matting exceptional inspection carrier.

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In Service Inspection (ISI) of sodium cooled fast reactor prototype ASTRID implies a large R&D effort for associated tools: among others, a specific articulated carrier is being designed to allow exceptional ultrasonic controls of under-sodium core support structure (strongback) at about 200°C.

This carrier has to reach deep in the main sodium vessel and yet adapt to the many different weld positions of the strongback, while being simple and robust. Its design thus includes a hollow rigid pole inside which a specific chain can deploy its ultrasonic transducers bearing head in several directions.

But first the specific components needed for this carrier have to be developed and tested for these harsh "sodium"conditions : small electrical motor (reducers, sensors), dry bearings, elastomers for leaktightness…

Consequently a large qualification program is starting involving tests to be performed with specific samples and prototypes, in air at 200℃, in water, then in sodium.

Country/Int. Organization:

FRANCE / AREVA NP / CEA /EDF

Poster Session 2 / 418

V&V STATUS OF CFD CODES APPLIED TO BN REACTORS

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Using CFD codes for numerical simulation of thermohydraulic processes occurring in fast sodium reactors, specific character of heat transfer in liquid metals and complicity of computational model development should be taken into account due to integral layout of reactor equipment.
Application of universal non-Russian CFD codes (CFX, Star-CD, Fluent, etc.) does not enable to take into account specific character of sodium coolant because Reynolds analogy is taken as basis for parameter determination.
To solve the problem, the Russian code FlowVision implements an original model of turbulent heat transfer. Such problem is set during implementation of Project "New generation codes"within which LOGOS code is developed.
One more way to solve the problem is to apply thermohydraulic codes of DNS category, and particularly CONV-3D code.
To verify CFD codes with regard to BN reactors, the verification matrix is developed which includes:
- analytical tests;
- benchmarks of the basis of experimental studies;
- task on the basis of data obtained during BN-600 operation;
- tasks with regard to newly performed experimental studies.

To obtain missing data, sodium facility is designed, constructed, and commissioned at RAS UB ICMM (Perm). The following has been experimentally studied:
- convective current of sodium in pipes with various aspect relations and grade angles;
- mixing of sodium flows of different temperatures using various models.

The paper contains results of experimental studies and performed verification for codes FlowVision, CONV-3D, and LOGOS with regard to BN reactors.

Country/Int. Organization:

Russia/JSC "Afrikantov OKBM"

3.3 Probabilistic Safety Assessment / 419

PROBABILISTIC SAFETY ANALYSIS RESULTS FOR BN REACTOR POWER UNITS

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Probabilistic safety analysis (PSA) is a constituent part of range of works aimed at BN power units safety assessment during operation (BN-600 and BN-800 reactors power units), lifetime extension (BN-600 reactor power unit), designing (BN-1200 reactor power unit). PSA reports are part of the document sets required to obtain appropriate Rostechnadzor licenses.
JSC "Afrikantov OKBM"together with General Designer and Scientific Supervisor has performed the following PSA Level 1 (PSA-1):
a) BN-600 and BN-800 reactors power units:
- PSA-1 for internal initiating events for power operation mode,
- PSA-1 for internal initiating events for shutdown reactor modes,
- PSA-1 for internal fires,
- PSA-1 for internal floods,
- PSA-1 for external hazards.
б) BN-1200 reactor power unit:
- PSA-1 for internal initiating events for power operation mode (preliminary).
General PSA goals are the following ones:
–power unit safety level assessment;
–recommendation development for power unit safety measures improvement.
First of all, PSA-1 was developed for internal initiating events for power operation mode. All following studies are based on models prepared within that PSA-1.
Within each of the PSA-1 studies, systems reliability analysis was performed, accident sequences were developed, human reliability analysis was implemented, database on initiating event frequencies and system component reliability indices was developed, integral probabilistic reactor power unit model was formed, quantitative analysis was performed including of importance, sensitivity and uncertainty analysis.
The PSA database on initiating event frequencies and component reliability indices is developed and updated based on the analyzed experience of BN-600 reactor power unit. For BN-800 and BN-1200 reactors power units PSA data analysis takes into account power units design distinctions.
At the present time JSC "Afrikantov OKBM"is continuing to improve all PSA models. Among other actions BN-600 and BN-800 reactors power units safety measures improvement based on Fukushima Daichi accident lessons learned and power units operating experience feedback update are taken into account. Probabilistic safety analysis Level 2 is being performed.

Country/Int. Organization:

Russia/JSC "Afrikantov OKBM"

Plenary Session 27 June / 420

Indian Fast Reactor Programme : Status and R&D Achievements

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Fast Breeder Reactors form the second stage of India's three stage Nuclear Power Programme based on the domestic nuclear resources. Indira Gandhi Centre for Atomic Research (IGCAR) is primarily dedicated for the broad based research & development of sodium cooled fast reactors, fuel cycle and associated technologies.
India is operating a Fast Breeder Test Reactor (FBTR) since 1985, fuelled with a unique Pu rich mixed carbide fuel (70% PuC + 30% UC). It has so far completed 24 irradiation campaigns in its successful operation over thirty years. Fuels of all types viz. carbide, oxide as well as metal fuels (both binary and ternary) are currently under irradiation. FBTR has served as a test bed for various experiments, fuel and structural material irradiation, isotope generation programs. The mixed carbide fuel has demonstrated a record burnup of 165 GWd/t and it has been operated at 400 W/cm peak LHR and at higher operating temperatures. Currently, a 500 MWe Prototype Fast Breeder Reactor (PFBR)

designed and developed by IGCAR, is in an advanced stage of commissioning. The design of PFBR incorporates several state-of-art features and is foreseen as an industrial scale techno-economic viability demonstrator for India's FBR program. IGCAR is presently engaged in the design of 600 MWe oxide fuelled FBRs incorporating many advanced features.
CORAL (COmpact Reprocessing of Advanced fuels in Lead cell) facility has reprocessed spent fuel discharged from FBTR with burnup up to 155 GWd/t and adequate decontamination has been demonstrated. Currently, a Demonstration fast reactor Fuel Reprocessing Plant (DFRP) is being established to process both MOX and mixed carbide fuels. A dedicated co-located Fast Reactor Fuel Cycle Facility (FRFCF) for PFBR is under construction. For the future, IGCAR has initiated development program on metallic fuel. Demonstration of fuel fabrication and pyroprocessing / aqueous technologies for metal fuels on an engineering scale is being pursued.
The R & D areas address all domains of fast reactor science and technology, including sodium technology, safety, materials development, fuel cycle, chemistry, sensors, advanced instrumentation and inspection. This paper presents an overview of the broad based R&D carried out by IGCAR in the domain of reactor technology, fuel cycle technology, materials development, basic sciences in support of fast reactor program, fuel chemistry, sodium technology, engineering development etc.

Country/Int. Organization:

INDIRA GANDHI CENTRE FOR ATOMIC RESEARCH, KALPAKKAM, INDIA

Poster Session 2 / **422**

Development experience for experimental reactor facility cooled vith evaporating liquid metals

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Development experience for experimental reactor facility cooled with evaporating liquid metals
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SSC RF-IPPE developed technical project of experimental reactor facility cooled with evaporating liquid metals at 1988-97. Evaporating sodium used as a coolant of the core of facility. As a coolant for second loop evaporating sodium-potassium eutectic alloy used. The third loop fulfilled with gas –working fluid of Sterling or Bryton cycle. Nuclear and thermo-hydraulic calculations performed for nuclear facility1.2 Mw of heat. Many thermo-hydraulic experiments with single-element, three-element and seven-element electrical imitators of fuel roads performed.
So, calculated parameters of core elements validated. Two fuel roads cooled with evaporating sodium tested in nuclear fast reactor.
Three-loops sector mock up including 72 imitators of fuel roads created and tested as well. It corresponds to 1/6 part of the core. Beside it, the mock up includes imitators of neutron reflector, shield and natural size dome. Outside of the dome interlope heat exchanger and the third lope of the facility replaced. Many thermo-hydraulic experiments performed but not finished, unfortunately.

Country/Int. Organization:

Russia. State Scientific Center of the Russian Federation - Institute of Physics and Power Engineering

Poster Session 2 / **423**

Russian Companies'involvement in CEFR RP (China) construction

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PRC –Russia cooperation in CEFR RP construction commenced in 1992.
In 1992 –1995, specialists of RF from "Experimental Design Bureau of Mechanical Engineering"developed "CEFR unit concept"and technical requirements for the reactor and its main components.
In 1995 –1996, Russian companies'specialists developed detailed design of CEFR nuclear power plant.
Designing solutions implemented in CEFR design were studied using Russian companies'testing facilities.
In all, nearly 30 various tests and studies were carried out.
In March 1995, "RF Ministry of Atomic Energy-CNEIC Inter-Agency Agreement of Cooperation in the Field of Developing the Experimental Sodium-Cooled Fast Reactor in the PRC"was signed.
Since 2002, the cooperation in CEFR has been exercising the granted Status of International one. "Agreement between the Government of the Russian Federation and the Government of People's Republic China on Cooperation in Construction and Operation Experimental Fast Reactor in China" was signed in Beijing on July 18, 2002.
Early in 1999, CIAE obtained permission of the Chinese government and National Nuclear Security Administration (NNSA) to construct CEFR. That enabled to prepare manufacture and testing of prototypes, RP equipment supply by Chinese and Russian companies, as well as supply of fuel (JSC "TVEL", Elemash, OKBM), components for reactor vessel and rotating plugs, primary and secondary pumps, CRDMs, intermediate heat exchangers, refueling mechanism, fuel loading/unloading elevators (OKBM), steam generators (JSC "Ziomar", Podolsk), devices (JSC "NII Teplopribor"), FSA flow meter (OKBM), devices of steam generator emergency protection (RF SSC-IPPE), level meters (OKBM, NII Teplopribor), electromagnetic pumps (OKBM, NIIEFA), and ionization chamber suspensions (OKBM, STC Elegiya). In all, Russian companies supplied 100 units of equipment.
In that stage, consulting services were rendered by specialists from OKBM, IPPE, and SPb AEP, lectures on BN reactors were delivered, practical trainings for CIAE specialists were prepared using OKBM and IPPE facilities (BFS large physical test facility), training for CNI-23 installation company' s specialists and technical briefing for operating repair personnel were conducted in OKBM, CIAE operating personnel (operators) were trained in RIAR Training center using BOR-60.

Currently, Russian companies are continuing to cooperate with China in the following areas:
JSC "Afrikantov OKBM"–SPTA supply for CEFR equipment and provision of consulting services for repair technology development for equipment of Russian origin;
JSC "RF SSC RIAR"–training of CEFR operating personnel;
JSC "Rusatom Service"–provision of consulting cervices in the RP operation stage
JSC "TVEL"–fuel supply.

Country/Int. Organization:

Russia/JSC "Afrikantov OKBM"

2.1 Commissioning and Operating Experience of Fast Reactors I / 425

Manufacture, Installation and Adjustment of the BN-800 Reactor Plant Equipment

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The construction of Power Unit No. 4 with the BN-800 reactor plant at the Beloyarsk NPP is the crucial stage in the industrial-scale development of the sodium-cooled fast neutron reactors(SFR).
The activities on the development of the BN-800 reactor plant commenced in1980.
During the period from 1993 to 2005 and later on until the equipment deliveries started, the activities were being in progress to check the engineering solutions of the design in test facilities.
To substantiate the reactor plant safety, more than 150 unique R&D activities were accomplished mainly in full-size test facilities.
Also, individual assemblies and elements of equipment were tested on mockups. The processes were being developed to fabricate individual assemblies of equipment, assemble and install articles, ensure interactions of sets of devices and mechanisms.
Contracts between Rosenergoatom Concern and OKBM provided for fabrication and delivery of more than 150 items of equipment and systems:
reactor vessel;
heat exchange equipment;
CRDMs;
fuel handling equipment;
purification system equipment;
primary and secondary sodium pumps, electromagnetic pumps and their control systems;
secondary pipelines and ECDS;
sodium tanks, 10–150 m3;
sodium valves;
metal structures of the reactor compartment;
non-standard equipment of the reactor plant;
dummy fuel subassemblies (FSA);hot cell equipment;
sodium technology instrumentation;
ionization chambers;
According to the adopted process, the equipment is installed in the Power Unit using two basic methods:
1.Installation work conjoined with the erection of the building
2.Modular installation work on large-size equipment
The modular installation work is basically done on the reactor vessel. In a specially erected Reactor Vessel Assembly Building(RVAB), more than 230 supplied units were pre-assembled into the 6 mounting modules that were later transported to the construction site and installed into the reactor pit.
The preoperational adjustment activities on the BN-800 reactor plant before the in-house electricity is generated-were performed according to an individual work schedule with the reactor plant equipment attributed to a startup complex. The startup complex ensured that the equipment and systems of the nuclear power station were ready for the gas heatup of the reactor, sodium filling of the reactor and FSA loading into the core with building up the minimum critical mass.
The completed deliveries, installation and adjustment work made it possible to accomplish the following in 2013-2016:
preoperational adjustment activities;
first criticality;
pilot industrial operation

Country/Int. Organization:

Russia/JSC "Afrikantov OKBM"

1.8 INNOVATIVE REACTOR DESIGNS / 426

Design of a nitride-fueled lead fast reactor for MA transmutation

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Research and development of fast reactors have been carried out in several European nations with mixed oxide fuel and sodium coolant as reference materials. In addition, the transmutation of Minor Actinides (MAs) in fast reactors has been investigated extensively from the viewpoint of reducing the environmental burden of long-lived radioisotopes.

Nitride fuel and lead as coolant represent possible alternative options for reactor design, which can bring substantial advantages as compared to oxides and sodium, respectively. Nitride fuel exhibits a higher thermal conductivity and a higher heavy nuclides density than oxide fuel, which would both enhance the core safety performance and improve the neutron balance in favor of MA burning due to the harder neutron spectrum. As a consequence, the amount of MAs that could be loaded in the core could be increased.

In this work, the core design of a 600 MWe lead-cooled, nitride-fueled fast reactor aimed at transmuting MAs is presented. The core design was performed aiming at accomplishing the following major goals: (i) obtaining a unitary conversion ratio; (ii) achieving a 6 kg/TWeh specific Am consumption after 6 years of cooling in a homogeneous transmutation scenario, while (iii) respecting the fuel cycle constraints for fuel maximum thermal load of 7.5 kW per assembly after 5 years cooling and of 3 kW per fresh assembly.

In addition to the core transmutation performance, safety analysis was performed in order to predict the core transient behavior following postulated accident initiators. Three reference accidents were considered in this study: an inadvertent control rod ejection leading to an Unprotected Transient OverPower (UTOP), a pump coast-down causing an Unprotected Loss Of Flow (ULOF), and a steam generator failure, resulting in an Unprotected Loss of Heat Sink (ULOHS). Reference safety criteria, such as margins against cladding failure, fuel melting and nitride dissociation were assessed.

Neutronics parameters were calculated by means of the Monte Carlo code Serpent, whereas transient simulations were performed using BELLA, an in-house code for the safety analysis of Generation-IV innovative lead fast reactor systems, currently under development at KTH.

Country/Int. Organization:

Sweden, Royal Institute of Technology (KTH)

8.1 Professional Development and Knowledge Management - I / 427

A proposal for a pan-European E&T programme supporting the development and deployment of ALFRED

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The entire process of implementation of a nuclear program relies on the availability of qualified expertise and of national infrastructures providing the general framework for the smooth execution of regulated activities. Building an innovative reactor, besides the challenges related to the advanced nuclear technology and the important aspects of costs and financing, implies also the availability of: reactor design theoretical and experimental tools, communication methods and tools, adequate regulatory approaches, building techniques and, of course, connection to the past and current reactors operational experience.

The EU ARCADIA project was conceived so as to promote the further development of nuclear research programs in Europe, including providing support for the ALFRED project towards its realization in Romania.
Consequently, crucial focus was put both on the identification of a comprehensive list of primary needs for the ALFRED project, mainly for what concerns E&T, supporting Infrastructures and regulatory aspects, and on the investigation of the existing national and regional supporting structures –with a particular attention to the ones in Romania and in all the participating New Member States –for defining a map of competences potentially eligible to satisfy the previously identified needs.

According to the output of this analysis and to the definition of Competence as "a holistic notion, consisting of cognitive, technical and behavioural aspects, each of them necessary for the complete definition of the job requirements", which is found in the nuclear field Job Taxonomy formulated by the EHRO-N working group, an approach was proposed to fill the gaps in competences and infrastructures required for a country to develop and pursue a Gen-IV nuclear programme, which were identified based on the overall picture of the competences required for the implementation of ALFRED demonstrator as innovative reactor in Romania.
In particular, two building blocks were identified and discussed as essential for developing and implementing such an E&T progamme, the latter consisting in applying an outcome-based pedagogical approach to lifelong learning, and harmonizing with the European Credit System for Vocational Education and Training (ECVET) principles.

Country/Int. Organization:

Sweden/Royal Institute of technology (KTH)

Poster Session 1 / 428

Preliminary Safety Performance Assessment of ESFR CONF-2 Sphere-pac-Fueled Core

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Within the European FP-7 project PELGRIMM, oxide fuel forms for Minor Actinide (MA) transmutation were researched for both homogeneous and heterogeneous recycling in Sodium-cooled Fast Reactors (SFRs).

Among the investigated fuels, sphere-pac fuels were given priority emphasis, as they offer great advantages compared to pellet fuels when MAs are to be embedded, while presenting, however, one major drawback, the latter being their low thermal conductivity.

In order to determine the actual suitability of sphere-pac fuels for use in SFRs, safety analyses were planned within the PELGRIMM Work Package 4, so as to provide a first assessment of both the transient behavior of sphere-pac-loaded cores as compared to the reference ones incorporating classical pellet MOX fuel, and the relative safety margins.
In a broader perspective, such a study was expected to help identify potential hindrances preventing the use of MOX sphere-pac fuels, as well as needs for further code development and validation.

A preliminary safety assessment of the CONF-2 version of the European Sodium Fast Reactor core (ESFR) at Beginning of Life, loaded with both pellet and sphere-pac fuels was performed by using the BELLA code and the SAS4A/SASSYS-1 code, with core neutronic characteristics and safety parameters being calculated with the Serpent code.
Both the reference Unprotected Loss Of Flow (ULOF) and an Unprotected Transient Over-Power (UTOP) accidents were simulated, along with a set of sensitivity studies.

As major outcomes of this study, it could be preliminarily concluded that the use of sphere-pac fuel may bring some disadvantages from the safety point of view in the event of a UTOP accident, whereas no concerns are raised for a ULOF scenario.
In particular, the dynamic response of the ESFR CONF-2 core loaded with sphere-pac fuel to a ULOF accident resulted to be essentially not affected by the characteristics and properties of this innovative fuel; conversely, in the event of a UTOP, safety margins would be reduced, due its low thermal conductivity, leading to larger magnitudes of the fuel temperature gradients ensuing from positive reactivity insertions.
Consistently with the previous conclusions, uncertainties in the determination of the fuel thermal conductivity and Doppler constant were found to have no significant impact on the ULOF transient predictions, but to influence the UTOP simulations, making their accurate determination critical for the system safety assessment.

Country/Int. Organization:

Sweden/Royal Institute of Technology (KTH)

Poster Session 1 / 430

On the possibility of using various types of fuel in the MBIR reactor core

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MBIR is a 150 MWt multipurpose research sodium cooled fast reactor, designed for a broad range of experimental studies in various lines of research. Vibropacked MOX-fuel with relatively high plutonium weight content (under 40%) is accepted as the standard fuel for MBIR reactor facility. At the same time, there is a principal possibility of using alternative types of fuel in this reactor: both uranium fuel (based on enriched uranium dioxide) and highly dense uranium-plutonium fuel (mixed nitride and mixed metal), which are of interest for innovative fast reactors. Moreover, at the initial stage of MBIR operating, it is possible to use combined vibropacked oxide fuel (based on plutonium and enriched uranium), which is accepted as the standard for reactor BOR-60 (content of plutonium under 24%).
Types of fuel under consideration differ not only by density, but also by other characteristics that are important for neutron physics of the reactor. Particularly, they have different nuclear properties of fission materials (plutonium or uranium-235), different quantity of diluent (oxygen etc.) nuclei per heavy nucleus etc. All this factors define the neutron spectrum and critical parameters of the core.

One of the important requirements for this reactor is high maximum neutron flux density (not less than 5*1015n/cm2sec). Special emphasis in this report is placed on the analysis of the dependence of neutron flux density and the rate of damaging dose accumulation from type of fuel, as well as the analysis of MBIR neutron flux distinctive features compared to energy reactors.

Country/Int. Organization:

Russia/JSC "SSC RF –IPPE"

1.8 INNOVATIVE REACTOR DESIGNS / 431

SEALER: a small lead-cooled reactor for power production in the Canadian Arctic

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SEALER (Swedish Advanced Lead Reactor) is a 3-10 MWe lead-cooled fast reactor operating on 19.9 % enriched UO2 fuel. It is designed for commercial production of electricity in communities and mining operations in the Canadian Arctic.
The reactor is capable of passive decay heat removal by radiation through the primary vessel and will make use of novel, highly corrosion resistant aluminum-alloyed steels developed by LeadCold engineers. Passive shut-down is accomplished by gravity-assisted insertion of tungsten-rhenium boride absorber elements.
In this paper, the general technical concept of SEALER is presented, together with the plan for licensing SEALER in Canada, including the pre-licensing design review process with CNSC, the R&D program necessary to qualify the design and associated materials, and the siting of a demo-plant in southern Canada.
Furthermore, the business plan for producing and selling up to 100 SEALER units on the Canadian market is outlined.

Country/Int. Organization:

Sweden/LeadCold Reactors

Poster Session 1 / 433

Preliminary transient analyses of SEALER

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SEALER (SwEdish Advanced LEad Reactor) is a small (8 MWt) lead-cooled fast reactor operating on 19.9 % enriched UO2 fuel. It is designed for commercial production of electricity and heat for off-grid consumers and is intended to function as a nuclear battery for up to 30 years. The reactor features a very small temperature gradient over the core to reduce the degrading process of structural materials. The safety-based design approach for SEALER relies essentially on passive characteristics, such as a negative temperature reactivity feedback, natural convection and heat radiation, the primary safety objective being that under no circumstances shall sheltering or evacuation of the public be necessary.

In this contribution, the present configuration of SEALER is discussed, along with the results of preliminary transient analyses including the coupled primary and secondary systems. The analyzed set of transients covers the consequences of most initiating faults. Based on the current system design, the latter were simulated in un-protected mode, corresponding to unsuccessful insertion of control elements to achieve sub-criticality in the event of failure. Therefore, the system dynamics following (i) reactivity insertions leading to transient overpower (UTOP), (ii) pump failures resulting in a loss of flow (ULOF), (iii) steam generator malfunctions causing a loss of heat sink (ULOHS), and (iv) loss of off-site power resulting in a station blackout (SB), was studied. In addition, changes of boundary conditions on the secondary side, such as decrease/increase of feed-water flow and temperature, and change in steam demand were simulated in order to investigate typical operational transients. Calculations were carried out using BELLA, an in-house code ad hoc developed for dynamic simulation of lead-cooled fast reactors, based on a lumped parameter approach to solve the coupled-physics governing equations.

As major outcomes of this study, it was concluded that, under the postulated accident conditions, adequate safety margins are provided against fuel melting and cladding failure, favored by an overall negative power feedback coefficient. Moreover, due to the negative temperature feedbacks of fuel and coolant, it is ensured that lead freezing will not take place in case of ULOF and ULOHS.
Finally, SEALER's load following capabilities were confirmed.

Country/Int. Organization:

Sweden/LeadCold Reactors

7.1 Sustainability of Fast Reactors / 434

Comparison of Innovative Nuclear Energy Systems Based on Selected Key Indicators and Their Weighing Factors

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The paper presents a methodological study on comparison of nuclear energy systems whose commissioning and commercial-scale operations are in the planning stage. These studies are carried out within the framework of the joint project INPRO.
Systems with numerous technical and economic uncertainties are poorly amenable to technology assessment using the INPRO methodology (for all areas, basic principles and criteria). Furthermore, it seems unreasonable to assess reactor systems in isolation from the system they were designed for. Some indicators from among those that should be referred to the key ones are system-related indicators. They directly affect not only the assessment of reactor facility, but also the characteristics of nuclear energy system as a whole. At the same time, there are indicators, which are slightly

related to the system or are generally unrelated to it. Therefore, the study compares particularly the systems with their inherent key indicators, rather than the technologies taken separately.
For the comparison of innovations, a set of key indicators was selected and aggregated into a single function by assigning weights to each indicator. At this stage, there is also an uncertainty both in key indicator assessments, and in determining the importance of each indicator among those selected (weighing factors of indicators).
The paper discusses several key indicators of sustainability for innovative nuclear energy systems from different areas of assessment (economics, technology readiness, waste management). As an example, consideration is given to several types of countries with different nuclear capacity scales. Key indicator weights were selected based on the intrinsic features of the countries; the comparison of innovations is presented. The paper presents the sensitivity of the weights of selected key indicators to the result of innovations comparison.
In addition, the paper identifies the ways of key indicators development. The study of assessment uncertainties using the key indicators could be one of the methodological improvements. Together with the sensitivity analysis of key indicators'weighing factors, it can significantly promote the study of the comparison of innovations in the context of the methodology described in the paper.

Country/Int. Organization:

Russia, State Scientific Centre of the Russian Federation –Institute for Physics and Power Engineering

7.2 Economics of Fast Reactors / 435

COMPARATIVE ANALYSIS OF ELECTRICITY GENERATION FUEL COST COMPONENT AT NPPs WITH WWER AND BN-TYPE REACTOR FACILITIES

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The fuel cost component (FCC) of electricity generation is defined as a specific indicator - the cost of 1 kWh of electricity produced. This value is obtained as the levelized (discounted) nuclear fuel cost value, generally beginning with natural uranium procurement and ending with spent fuel management, normalized to the total electric energy generated over the nuclear power plant lifetime. I.e. the result is the FCC average value over the entire lifetime.

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The methodology of levelized FCC calculation is based on the concept of taking into account t

    The paper describes the basic essential methodological and factual materials for the fue
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Country/Int. Organization:

Russia

Poster Session 1 / 436

Investigations in a substantiation of high-temperature nuclear energy technology with fast-neutron reactor cooled by sodium for manufacture of hydrogen and other innovative applications

Author: Aleksandr Sorokin[1]

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As results of neutronics and thermal physical investigations of reactor installation BN-HT type with heat rating 600 MW have shown that there is a principal possibility to provide demanded parameters of a high-temperature fast reactor for production of a considerable quantity of hydrogen, for example, on the basis of one of thermochemical cycles or a high-temperature electrolysis with high factor of thermal use of the electric power. Safety requirements will be thus observed. The relative small sizes, the coolant type, the fissionable substance and structural materials allow to create a reactor with immanent to it properties (exclusion of reactor runaway by instantaneous neutrons, passive system of decay heat removal), providing the raised nuclear and radiation safety.
By calculations BN-VT for production of electric power and hydrogen on basis of solid oxide electrolysis mass transfer hydrogen and tritium taking into account principal new method of clearing by pumping out through special membranes it is shown, that efficiency such system is ~40%, volume of maded hydrogen is 2.8104 l/s (under normal conditions). Danger from tritium in a finished stock originates after hydrogen combustion in an aerosphere. Therefore at calculation of parameters of the secondary circuit it was accepted, that maximum permissible tritium concentration in maded hydrogen should not exceed 3.26 Bk/l. Maximum concentration of permissible tritium in air is 2.44•103 Bk/l almost in 1000 times above. Clearing of sodium from tritium to the concentration providing in maded hydrogen maximum permissible concentration equal 3.26 Bk/l makes additional demands to system of clearing from hydrogen: the coefficient of permeability of system of clearing of the secondary circuit from tritium should exceed 140 kg/s.
Taking into account high temperature experiments in which high efficiency of deduction of suspended matters of products of corrosion on the filters installed in низкотемпературной to a zone is shown, it is offered to use a principle of work of a cold trap: to chill sodium to necessary temperature with simultaneous deduction of products of corrosion on mass transfer surfaces, including filters. Working out of a necessary high temperature material and its studying under radiation demands the further investigations.

Country/Int. Organization:

State Scientific Center of the Russian Federation –Institute for Physics and Power Engineering

Poster Session 2 / 437

Features of the physics of the MBIR reactor core

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Cores of research reactor facilities (RRF) as opposed to those of power reactors are designed taking into account their research function. This is their unique peculiarity reflected, among other features, in the flexibility (i.e. transformation in safe and reasonable limits) of the core arrangement according to the changing goals of specific experiments. Power generation for research reactor is a secondary function, significant as it is, therefore for RRF it is quite acceptable to change the power level and transform the arrangement of the core according to experimental requirements rather than to the

plan of power generation. Flexibility of the MBIR core makes it possible to address many different tasks simultaneously. For example, the starting core can be considerably smaller than the design one, yet it can provide technologically acceptable transition to core arrangements that are more research and production intensive without the necessity of reloading unburnt assemblies.
The MBIR core has a small size and a very large (up to 25%) neutron leakage outside the core with fast neutron spectrum. As a result, there is no positive sodium void reactivity effect in the MBIR core whatever isotopic composition of plutonium. Neutron leakage also ascertains the stability of neutron flux and energy release distribution in the core.
Due to the high leakage, MBIR reactor features relatively high plutonium enrichment, hence even at very high power density of the core, the neutron flux density and the rate of the damaging dose accumulation in MBIR are quite moderate and lower than in BN-600 and BN-800-type power reactors. For the same reason (due to high enrichment), MBIR demonstrates significant increase of neutron flux density per micro-campaign, reduced energy release in the fuel assemblies per campaign, and significant loss of reactivity with fuel burnup.
However, despite the fact that MBIR reactor is inferior to power reactors in terms of the neutron flux and the rate of damaging dose accumulation (which are critically important characteristics for research reactors), it provides conditions for a wide range of various experiments or isotope production.
High experimental volumes of the MBIR reactor and high sensitivity of fast neutron core to the location of research and irradiation subjects, as well as high thermal power of the reactor require gradual and careful testing of reactor power taking into account its physical characteristics.

Country/Int. Organization:

Russia/JSC "SSC RF –IPPE"

6.9 Research Reactors / 438

Justification of arrangement, parameters, and irradiation capabilities of the MBIR reactor core at the initial stage of operation

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Reactor MBIR will start operating in the absence of loop channels, which represent technologically complex equipment. At the initial stage of operation this will make it possible, on the one hand, to develop technological modes of reactor facility operation with small amount of experimental devices, and on the other hand, with this development, to gradually fill the cells intended for the loop channels additional irradiation assemblies. For example, central loop channel occupies 7 cells in the center of the core. It is unreasonable to use all the seven cells for irradiation assemblies, because of their strong influence on each other. To eliminate this impact, it is possible to fill the central loop channel cells with three irradiation assemblies and four fuel assemblies.
To level the energy release, it is reasonable to install one more irradiation assembly in the core and compensate the critically loss by installing 4 additional fuel assemblies on the core periphery. Consequently, core at the initial stage of operation will consist of 85 fuel assemblies (93 assemblies in the design), 8 cells with CPS control rods and 21 irradiation assemblies.
Thus, the number of the irradiation cells can be increased from 17 to 21 at the initial stage of operation by eliminating central loop channel. Reactor power at this stage must be reduce from 150 to 137 MWt.
With this modification of the core, fuel with the design plutonium weight content is used, enabling

transition to a design version of the core without changing the fuel assemblies design. For the same reason, the neutron flux and damaging dose in the irradiation assemblies at the initial stage of operation are almost the same as the design values.
The rate of damaging dose accumulation in the irradiation assemblies located in the core is from 16 to 16.7 dpa per micro-campaign (100 eff. days). Inner volume of one irradiation assembly is 2.28 l. General rate of damaging dose accumulation in the MBIR reactor at the initial stage of operation is 1370 dpa*l/year. In the BOR-60 reactor this parameter equals 300 dpa*l/year.

Country/Int. Organization:

Russia/JSC "SSC RF –IPPE"

6.2 Thermal Hydraulics Calculations and Experiments / 439

Experimental investigations of velocity and temperature fields, stratification phenomena in a integral water model of fast reactor in the steady state forced circulation

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The results of experimental investigations of local velocities for height and radius of the top of the camera in the plane in the direction from the core center to the intermediate heat exchanger and the temperature of the coolant in the upper (hot) chamber, and other elements of the circulation circuit on the integrated water model of the reactor on fast neutrons (scale ~ 1:10) for the stationary forced circulation mode, simulating a nominal operation regime. The data obtained on stand V-200 using a specially designed and implemented system of measurement that provides high measurement accuracy and speed of registration. The results show that the structure of nonisothermal motion of the coolant in the top chamber model is defined by the action of lifting forces: hot coolant from the core rises up through the Central column to the surface section and forms a vast vortex nearly isothermal zone in the upper region of the chamber from which flows into the intermediate heat exchangers. Above the side screens formed of insulated cold zone of the heat carrier, the size of which increase with the overall consumption increase. On stratified horizontal boundary insulated zones across the cross-section model of the reactor tank there are internal waves which cause temperature pulsations in the material of the walls of the equipment. There is a significant and stable thermal stratification of the coolant not only in the peripheral area of the top chamber of the reactor above the side screens, but in the cold and the pressure chambers, elevating the enclosure, the cooling system of the reactor, at the outlet of the intermediate heat exchangers. At the boundaries of stratified and recycling entities recorded strong gradients and temperature pulsations, allowing to judge about the amplitude and frequency characteristics of temperature pulsations in these potentially hazardous areas. The data obtained indicate the necessity of taking into consideration the stratification phenomena in justifying reliability management, security, design terms of operation of fast reactors.

Country/Int. Organization:

SSC RF –IPPE, MPEI,IVT RAN

6.2 Thermal Hydraulics Calculations and Experiments / 440

Density of sodium along the Liquid-Vapor Coexistence Curve, including the Critical Point

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Sodium densities along the whole liquid-vapor coexistence curve are reanalyzed using the equation proposed by Apfelbaum and Vorob'ev(1). The formulation has built-in the correct behavior for liquid and vapor densities, both at low temperature and in the near-critical region. Thus, it satisfactorily represents the available experimental data in the low and intermediate temperature range, while providing a sound density extrapolation to the critical point: in reduced units, the calculated values for sodium are consistent with those measured for Rubidium and Cesium(2), as required by the principle of Corresponding States. The enthalpy of vaporization, calculated via Clausius-Clapeyron relation, is also correctly described.

The main differences between our results and those from the previous formulation by Finck and Leibowitz(3) are found in the high-temperature region (2300 K –Tc), where the coexistence curve predicted by the latter exhibits an unusual shape.

Our results indicate that the value for the critical density, (180 ± 10) kg/m3, is 20 % lower than the one recommended before (219 ± 20) kg/m3.

(1) E. M. Apfelbaum and V. S. Vorob'ev, The Wide-Range Method to Construct the Entire Coexistence Liquid–Gas Curve and to Determine the Critical Parameters of Metals, J. Phys. Chem. B, 2015, 119, 11825–11832.

(2) S. Jüngst, B. Knuth and F. Hensel, Observation of Singular Diameters in the Coexistence Curves of Metals, Phys. Rev. Lett. 1985, 55, 2160-2163.

(3) J. K. Fink and L. Leibowitz, Thermodynamic and Transport Properties of Sodium Liquid and Vapor, ANL/RE-95/2 (1995).

Country/Int. Organization:

Argentina - Comision Nacional de Energia Atomica (CNEA)

Poster Session 1 / 442

Precipitate phases in a weldment of P92 steel

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The microstructural stability of ferritic-martensitic heat resistance steels is mainly controlled by the thermal stability of precipitates that are able to retard the growth rate of subgrains. The welded zones appear to be a weak region compared to the base material (BM) and creep rupture takes place in the intercritical or the fine-grained heat affected zones (ICHAZ and FGHAZ).
In the present work, precipitates of a single-pass weld performed by the FCAW (flux-cored arc welding) process were characterized by means of a transmission electron microscope on carbon replicas extracted from the different regions generated during welding. In the as received condition, the BM of the weld, both the M23C6 carbides (M = Cr, Fe, Mo, W) and MX carbonitrides (M = V, Nb, Cr)

were present. Particle size number frequency histograms, the percentage of covered area by precipitates and ternary composition diagrams were obtained for each HAZ sub-zones. The size of the precipitates and the percentage of covered area diminished from the BM to the fusion zone, where original precipitates are replaced by inclusions and (Fe,Cr)3C. A change in the distribution of solutes in M of both types of precipitates was observed between the BM and fine-grained HAZ (FGHAZ) that attained the higher temperature. A preponderance of M23C6 carbides with high W+Mo and low Cr and an enrichment of Nb at the expense of V and Nb in the MX carbonitrides were observed in the FGHAZ near the coarse-grained HAZ.

Country/Int. Organization:

Argentina/ Argentina Atomic Energy Commission

Poster Session 1 / 443

Isothermal transformation austenite-ferrite in a P92 steel

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The Time-Temperature-Transformation (TTT) diagram of an ASTM A335 P92 steel (9CrMoWVNNb) has been established starting from an austenitization temperature of 1050 ℃. Isothermal transformation was carried out at temperatures from 625 up to 775 ℃ taking 25 ℃ intervals, using a high resolution dilatometer. Only two state fields (i.e., austenite and ferrite + carbides) were observed, in full agreement with previous results on similar steels. A subset of large austenite grains, with sizes significantly exceeding the mean, was observed in all of the tested samples. At temperatures below the nose of the TTT diagram, prior austenite grain boundaries were made visible by decorating them with carbides precipitated at the early stages of the transformation. Carbide decoration allowed to have an accurate picture of the size distribution of austenite grains under the prescribed conditions of thermal cycle. Above the nose, prior austenite grain boundaries are hardly seen due to a drastic change in carbide precipitation mechanisms. At the same time, the ferrite nucleation and growth is markedly different in these two temperature regions; there is a gradual transition between these two extreme behaviors.
The dilatometric curves obtained at each temperature were fitted to the Kolmogorov-Johnson-Mehl-Avrami expression in order to extract kinetic information about the austenite-ferrite transformation. Fitting was accomplished so as to take into account the presence of the large austenite grains.
At the same time, a thorough examination of the transformed samples was carried out by using optical and electron (FEG-SEM and TEM) microscopy. Carbon replicas were extracted from the surfaces of selected specimens and a detailed study of the carbides present in each case was added to the former information.

Country/Int. Organization:

Argentina/Argentina Atomic Energy Commission

Poster Session 1 / 444

Study of the austenitization process in a P91 steel

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In this contribution we address the behavior of precipitated second phases during the first minutes of holding in the austenite phase field for an ASTM A335 P91 steel.
It is well known that two main families of precipitated particles are present in this steel after a standard manufacturing process that includes, as final stages, normalizing and tempering, i.e., M23C6 carbides (M = Cr, Fe, Mo) and MX carbonitrides (M = V, Nb, Cr). In addition, some extra, very low volume fraction phases could be present as a result of tiny variations of the chemical composition within the specified ranges.
The austenitizing thermal cycles were carried out in a high resolution dilatometer Bähr DIL 805 A. The heating rate was fixed at a value of 50 ℃/s up to the austenite holding temperature (1050 ℃) and austenite holding time was varied between 0 and 5 minutes in steps of 1 minute. After that, samples were quenched using an Ar jet at 50 ℃/s.
The precipitated phases were characterized by high-resolution field emission gun scanning electron microscopy and transmission electron microscopy. For each austenite holding time, the sizes of approximately one thousand particles were measured using carbon replicas; the chemical composition was also determined for a selected subset of particles by energy-dispersive X-ray spectroscopy.. The histograms of the size number frequency were analyzed in each case and statistical parameters were extracted from the corresponding size distributions.
The experimental results indicate a rapid dissolution of the M23C6 major carbidic phase and a progressive enrichment in Nb of the MX phase in the specified interval of time. Some suggested trends in the size distribution behavior of the remaining particles are also presented.

Country/Int. Organization:

Argentina/ Argentina Atomic Energy Commission

Poster Session 1 / 445

New results on the continuous cooling behavior of an ASTM A335 P92 steel

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This work introduces new results on the transformation behavior and microstructural evolution of ASTM A335 P92 steel under continuous cooling conditions (CCT). The first results were already reported and stored in the INIS database under the report number INIS-AR-C−1704.
The material was austenized at 1050 °C and afterwards cooled down at controlled rates (300, 200, 140, 120, 100, 90, 70, 50, 25 and 15 °C/h). The transformation behavior of the steel samples was followed by dilatometry.
The determination of the phases present in the samples after the thermal cycles was performed by optical and field emission scanning electron microscopy for the eleven tested values of cooling rate. Additionally, a full characterization was performed for selected samples by Mössbauer spectroscopy and X-ray diffraction.
The phase domains identified according to the cooling rate were completely martensitic, completely ferritic and mixed martensitic-ferritic. Second-phase precipitation has been observed in all of the samples, and indications of the presence of retained austenite after some of the cooling cycles were also detected. The experimental results were collected in the form of a continuous cooling transformation diagram.

Country/Int. Organization:

Argentina/ Argentina Atomic Energy Commission

Poster Session 2 / 446

DESIGN VALIDATION OF PFBR FUEL SUBASSEMBLY TRANSPORTATION CASK WITH MOCKUP TRIAL RUN

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In the case of Indian Prototype Fast Breeder Reactor (PFBR), which is in an advanced stage of commissioning, the fuel pins along with other parts of fuel subassembly are stored in an Interim Fuel Storage Building (IFSB). The final assembling of subassembly is carried out in IFSB and the IFSB is located far away from PFBR site. It is essential to demonstrate safe transportation of the fuel assemblies to PFBR site. Hence, as a part of PFBR pre-commissioning activity, mock-up trial runs have been carried out for full scale representative dummy fuel subassemblies along with cask on transportation trailer from the IFSB to PFBR fuel building. The purpose of this trial runs was to demonstrate that the vibration and shock seen by the cask during transportation are lower than the values considered for the design of the cask as per the NUREG/CR-0128 guidelines.
The maximum acceleration measured at the top of the cask during the transportation in the onward trip from IFSB to reactor site was 8.9 m/s2, which was due local disturbance. Neglecting this driving scenario, the peak acceleration found at the top of the trailer due to the unevenness in the road condition was 5.3 m/s2 (along the transverse direction). The same measured during the return trip was 3.3 (along the transverse direction). Similarly, the peak acceleration along the longitudinal direction measured in the return trip at the top of the trailer was 4.3 m/s2 at the top of the cask. These values are traced to be due to the unevenness in the road conditions. All the above values are lower than the acceleration values used for the cask design. As per the NUREG guideline, the permissible values of acceleration are 29 m/s2 along the longitudinal direction and 13 m/s2 along the transverse direction. Thus, the measurements carried out during trial runs have demonstrated the safe transportation of fresh fuel subassemblies from IFSB to reactor site. The full paper would present the details of instrumentation adopted, measurement locations, spectrum of spectrum of shock and vibration data and their acceptability

Country/Int. Organization:

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Poster Session 1 / 447

Study of isolation valve for Sodium Fast Reactor

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In the framework of the ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) project, the VELAN Company is involved to propose a concept of isolation valve to guaranty the confinement of the reactor block in case of severe accident scenario. An innovative and compact design for the isolation sodium valve was developed. Following the basic design phase studies on the sodium secondary loops, a dedicated valve concept was study to evaluate the technical parameters. The large size of the valve ND 700 mm requires optimizing the mass and dimensions due to cost mastering and response to seismic spectrum. After a description of the service conditions, the paper presents the mains outcomes of the technical parameters (mechanical behavior, sealing performance, hydraulic performance) which led on several valve designs. Maintenance aspects are also considered and a proposal of a butterfly valve design is proposed for detailed studies.

Country/Int. Organization:

VELAN SAS; AREVA NP; CEA Cadarache.
FRANCE

Poster Session 1 / 448

Sensors of content of oxygen dissolved in heavy liquid metal coolants

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Up-to-date technology of lead-based coolants is now founded mainly on the paramount importance of oxygen addition to the coolant assuring corrosion resistance of structural steels.
In the early stage of HLMC technology development, oxygen content was monitored by coolant sampling and analysis. Later on, more efficient methods of control using electrochemical sensors with solid oxygen-conductive electrolyte were developed and tested.
An extensive work is now under way at the SSC RF –IPPE on solid electrolyte sensors for monitoring oxygen content in lead-based liquid metals. Significant R&D work has been done on the development of solid electrolyte sensor design and production technology. The results of sensor design optimization studies are as follows:
- solid electrolyte ceramic sensor (CS) made of the oxide ceramics, which is capable of operating during the long time in the liquid metals at high temperatures under thermal cycle conditions. It has stable conductivity and strength characteristics, as well as thermal stability and low gas permeability;
- various sensor designs devoted for oxygen control in the lead-based liquid metals in facilities with static coolant, experimental loops and full-scale pool reactor systems.
Among the latest developments there is a sensor consisting of several independent detecting elements with various reference electrodes located in one holder.

Country/Int. Organization:

Russian Federation, JSC "SSC RF –IPPE"

Poster Session 1 / 450

ESFR-SMART: new Horizon-2020 project on SFR safety

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To improve the public acceptance of the future nuclear power in Europe we have to demonstrate that the new reactors have significantly higher safety level compared to traditional reactors. The ESFR-SMART project (European Sodium Fast Reactor Safety Measures Assessment and Research Tools) aims at enhancing further the safety of Generation-IV SFRs and in particular of the commercial-size European Sodium Fast Reactor (ESFR) in accordance with the ESNII roadmap and in close cooperation with the ASTRID program. The project aims at 5 specific objectives:

1) Produce new experimental data in order to support calibration and validation of the computational tools for each defence-in-depth level.

2) Test and qualify new instrumentations in order to support their utilization in the reactor protection system.

3) Perform further calibration and validation of the computational tools for each defence-in-depth level in order to support safety assessments of Generation-IV SFRs, using the data produced in the project as well as selected legacy data.

4) Select, implement and assess new safety measures for the commercial-size ESFR, using the GIF methodologies, the FP7 CP-ESFR project legacy, the calibrated and validated codes and being in accordance with the update of the European and international safety frameworks taking into account the Fukushima accident.

5) Strengthen and link together new networks, in particular, the network of the European sodium facilities and the network of the European students working on the SFR technology.

Close interactions with the main European and international SFR stakeholders (GIF, ARDECo, ESNII and IAEA) via the Advisory Review Panel will enable reviews and recommendations on the project' s progress as well as dissemination of the new knowledge created by the project. By addressing the industry, policy makers and general public, the project is expected to make a meaningful impact on economics, environment, EU policy and society.

The full paper will present the project in details

Country/Int. Organization:

Switzerland

Poster Session 1 / 451

Chugging boiling in low-void SFR core: new phenomenology of unprotected loss of flow

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Calculational analysis of the unprotected loss of flow (ULOF) accident in a Generation-IV SFR, featuring a low-void core design, shows that the chugging sodium boiling regime in the core could last for several hundred seconds during the accident. While in the case of the traditional positive-void SFR core the sodium boiling onset
is almost immediately followed by the power run-away, fuel bundle overheating, melting and relocation (i.e. severe accident), the chugging boiling regime in the low-void SFR core could allow avoiding the power runaway and avoiding or at least significantly postponing the cladding overheating and melting caused by the permanent dryout. The low-void core design therefore could be classified as a new safety measure acting as a level of defence preventing the severe accidents. The state-of-the-art in the area of the chugging regime of the sodium boiling is very limited and very few corresponding experiments were performed. The paper will present the detailed transient analysis of the low-void core behaviour in unprotected loss of flow accident performed with the TRACE code (modified for the sodium boiling modeling) and discuss the physics of the predicted phenomena as well as the future research needed, including new experiments.

Country/Int. Organization:

Switzerland

6.1 CFD and 3D Modeling / 453

Numerical Simulation Method of Thermal Hydraulics in Wire-wrapped Fuel Pin Bundle of Sodium-cooled Fast Reactor

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A numerical simulation system, which consists of a deformation analysis program and three kinds of thermal-hydraulic analysis programs, is being developed in Japan Atomic Energy Agency in order to offer methodologies to clarify thermal-hydraulic phenomena in fuel assemblies of sodium-cooled fast reactors under various operating conditions including fuel deformation. In this paper, we focus on SPIRAL which is one component program of the numerical simulation system and plays the role to simulate detailed local flow and temperature fields in a wire-wrapped fuel pin bundle. SPIRAL adopts finite element method in order to treat complicated geometries and a hybrid turbulence model which has computation efficiency similar to high Re number models and high accuracy similar to low Re number models. As a validation study, SPIRAL was applied to several kinds of analyses of water/sodium experiments using wire-wrapped fuel pin bundles. Applicability of SPIRAL to the prediction of flow and temperature fields as well as pressure loss coefficients will be discussed.

Country/Int. Organization:

Japan

6.2 Thermal Hydraulics Calculations and Experiments / 455

Development and Validation of Multi-scale Thermal-Hydraulics Calculation Schemes for SFR Applications at CEA

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In the framework of the ASTRID Gen4 SFR project, extensive R&D efforts are under way to improve and better validate the SFR thermal-hydraulics codes available at CEA. These efforts include :

- The development and validation of SFR-specific models in CATHARE. Developed at CEA, CATHARE is the reference STH code for French LWR safety studies : SFR developments are being integrated and validated into the lastest version of the code, CATHARE3.
- The development and validation of TrioMC, a subchannel code specific to SFR core TH. Initially created for design studies (with the aim of computing the maximum cladding temperature of a given core flowrate), TrioMC has been upgraded in order to compute the local behavior of the core during accidental transients.
- The application and validation of TrioCFD, a 3D CFD code developed at CEA, to SFR studies. TrioCFD is being used to compute flow behavior in the large plena of pool-type SFRs (hot and cold pools), as well as in the IHX primary side and in the in-core inter-wrapper gap regions.

In most cases, these codes are used independently. However, in some cases, local phenomena may have a strong feedback effect on the global behavior of the reactor : for instance, during passive decay-heat removal by natural convection, inter-wrapper flows may contribute to up to 30% of the overall DHR if the heat sink is provided by DHXs in the hot pool. The strength of this contribution leads to a feed-back effect from a local (subchannel/CFD) phenomenon) to the global (system) scale. In order to model such effects, a coupling between CATHARE, TrioMC and TrioCFD has been developed at CEA and integrated into a new code : MATHYS (Multiscale ASTRID Thermal-HYdraulics Simulation). Within MATHYS, TrioMC and TrioCFD are coupled at their boundaries (core outlet and hex-can sides), using a domain-decomposition approach : then, the two codes are coupled with a CATHARE simulation of the complete system using a domain-overlapping method. The resulting multi-scale simulation is able to account for feedback effects between all three scales.
This paper first outlines the development and validation efforts related to CATHARE, TrioMC and TrioCFD; then, the coupling algorithm underlying MATHYS is described. The final section discusses the validation of MATHYS : overall approach, validation of coupled effects on existing experiments (TALL-3D for STH/CFD, PLANDTL-DHX for subchannel/CFD, PHENIX at the integral scale).

Country/Int. Organization:

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2.3 Decommissioning of Fast Reactors and Radioactive Waste Management / 456

Industrial Exploitation of Testing Ground for Treatment of Radwaste of Alkaline Coolants under Decommissioning of Fast Research Reactors

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Since 2002 Research Reactor BR-10 (BR-10) has a status of the decommissioning. Currently, BR-10 is implemented in transition period between final shutdown and decommissioning. These works are connecting with bringing of BR-10 in nuclear-safety and radiation-safety condition.

In accordance with the Programme of decommissioning of BR-10, before dismantling of equipment and systems of reactor, works must contain the remove all nuclear materials (last batch submitted for processing in 2016) and liquid radwaste, which also include radwaste of coolants. As a coolant at BR-10 was used a sodium and sodium-potassium alloy. Thus, after 46 years of operation, has accumulated 18 m3 of radwaste of alkaline coolants, including the sodium from 16 cold traps oxides from first circuit.

For implementation of activities in transition period at BR-10 was established the Testing Ground for treatment of total volume and residual of radwaste alkaline coolants, which are contaminated in individual equipments of reactor and in storage tanks [1]. The technologies, which are used for treatment, have had experimental and estimated studies for events with decommissioning project of BR-10 and protected by patents [2].

For treatment of total volume of radwaste alkaline coolants was organized Conditioning Site [3]. Ongoing site activities include: preparation, transporting of oxidant and alkaline coolant to the place of conditioning; reaching the stationary operating parameters of equipment and systems; portion holding of alkaline coolant; cooling and transporting of product of conditioning for temporary storage in the protective container. Now is carry out of works on conditioning radwaste of secondary sodium.

For removal of residual of radwaste alkaline coolant from inside surfaces of individual equipment was organized Neutralization Site [3]. Ongoing site activities include: transporting equipment (for example, cold trap oxides) from storage protective box; draining of radwaste alkaline coolants from equipment; neutralization of residues inside of amount of equipment; washing and decontamination of internal surfaces of equipment; transporting to the site for treatment of solid radwaste. Now is carry out of works on neutralization of another cold trap oxides from first circuit, was drained the volume of radwaste of sodium and sent to Conditioning Site.

In recent time, together with the conditioning radwaste of secondary sodium, is preparing for treatment of radwaste of primary sodium. Namely, tryout of activities to prevents of possible contamination of equipments and communications of Conditioning and Neutralization Sites.

Country/Int. Organization:

Russian Federation/State Atomic Energy Corporation "Rosatom"

6.4 Neutronics –2 / 457

Neutronic evaluation of a GFR of 100 MWt with reprocessed fuel and thorium using SCALE 6.0 and MCNPX

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A GFR core model with 100 MWt was evaluated using three different fuel compositions: conventional (U, Pu)C and two reprocessed fuels with transuranic (TRU) (Pu, Am, Np, Cm). One reprocessed by UREX+ technique and spiked with depleted uranium, (U,TRU)C, and the other reprocessed by the same technique but spiked with thorium, (Th,TRU)C. The reprocessed fuel came from a PWR standard fuel (33,000 MWd/T burned) with 3.1% of initial enrichment and left in the pool by 5 years. Some important nuclides were followed for burns and neutron absorption and kinf was evaluated 1400 days burning. Tests were also made for B4C absorber insertion and the temperature coefficient. The study concludes with an evaluation of power distribution in the core. The simulations were made comparing results of MCNPX and SCALE 6.0 programs. The goal is to validate the simulated model and evaluate the possibility to use TRU spiked with Th in a GFR conception.

Country/Int. Organization:

Brazil/Universidade Federal de Minas Gerais

5.2 Advanced Fast Reactor Fuel Development II / 458

The IAEA Coordinated Research Project on Sodium Properties and Safe Operation of Experimental Facilities in Support of the Development and Deployment of Sodium-cooled Fast Reactors (NAPRO)

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The International Atomic Energy Agency (IAEA) recently established a Coordinated Research Project (CRP) on "Sodium Properties and Safe Operation of Experimental Facilities in Support of the Development and Deployment of Sodium-cooled Fast Reactors - NAPRO", to be carried out in the period 2013 –2017.
Eleven institutions from ten Member States participate in this CRP.
The complete scope of this CRP is covered by three work packages.
A specific work package (WP1), under the coordination of the Argonne National Laboratory (USA),

is focused on the compilation and expert assessment of data sets of Na physical and chemical properties, as well as correlations for pressure drops and heat transfer in Na facilities. Identification of gaps in the data sets, and recommendations for their closure are included.
A second work package (WP2), under the coordination of the Institute of Physics and Power Engineering –IPPE (Russian Federation), addresses the compilation, evaluation and development of best practices and guidelines for the design, operation and maintenance of Na facilities.
Finally, Work Package 3 (WP3), coordinated by the French Alternative Energies and Atomic Energy Commission (CEA), concentrates in the compilation and development of guidelines and rules for the safe operation of Na facilities, including, among others, the prevention, detection and mitigation of Na leaks and fires.
This work presents an overview of the compiled data bases and correlations of WP1, including recommendations for their use, as well as a summary of the guidelines and rules evaluated and developed in WP2 and WP3.

Country/Int. Organization:

National Atomic Energy Commission of Argentina (CNEA)

Poster Session 1 / 459

The method of calculating tritium content in various technological media of BN-type reactors

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During the reactor operation a continuous generation of the radioactive isotope of hydrogen –tritium –takes place. Tritium is formed in nuclear reactions: fission of the fuel and interaction of neutrons with nuclei of some elements contained in the fuel, structural materials and coolant. Tritium has a great penetrating ability and migrates through the technological media. Thus it gets into the environment. Peculiarities of its transfer essentially depend on the reactor facility type.

Experience in operating domestic BN reactors as well as international experience suggests that only a small portion of tritium coming in the sodium coolant is released to the environment.

Data on the amount of tritium coming into the environment, along with the data on the content of tritium in technological environments, can be used in practical calculations of radiation exposure because of tritium for the population and the staff at NPPs with the BN reactor. This is required for the various operating modes of BN, including decommissioning.

As it has been shown in several studies, tritium transport and allocation in sodium circuits of BN reactors is directly connected with the content of protium (hereinafter referred to as hydrogen - H) in soduim. Therefore, there is a need for simultaneous determination of the mass transfer of hydrogen and tritium. Tritium mass transfer model in application to the three-loop reactor is based on a consideration of hydrogen and tritium balance in the first and second circuits, as well as in emergency heat removal system (EHRS).

In the present study it is noted that, in general, for plants with BN reactors isotopes of hydrogen (H and T) balanced equation must be written for sodium and for the gas phase in each circuit, and for the second circuit and EHRS equations must be written for each loop.

It is also noted that for the BN reactor these tasks are simplified due to the fact that with sufficient accuracy for practical purposes can be assumed that the gas phase has no affect on the mass transfer of hydrogen isotopes (H, T) in the sodium loops as their main mass is in the coolant.

These quantities of tritium content in different technological media of BN reactors provide the basis for determining the effects of tritium on the personnel and the environment.

Country/Int. Organization:

JSC "SSC RF –IPPE", Obninsk, Russia

Plenary Session 27 June / **460**

Status of Sodium Cooled Fast Reactor Development Program in Korea

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The Korea Atomic Energy Commission (KAEC) authorized the R&D action plan for the Advanced SFR (sodium-cooled fast reactor) and the pyro-process to provide a consistent direction to long-term R&D activities in December, 2008. This long-term advanced SFR R&D plan was revised by KAEC in November 2011 in order to refine the plan and to consider the available budget for SFR. The revised milestones include specific design of a prototype SFR by 2017, specific design approval by 2020, and construction of a prototype SFR by 2028.
The prototype SFR program includes the overall system engineering for SFR system (NSSS and BOP) design and optimization, integral V&V tests, and major components development. Based upon the experiences gained during the development of the conceptual designs for KALIMER, the conceptual design of SFR prototype plant (PGSFR) had been carried out in 2012 and has been performing a preliminary design since 2013.
The first phase of the development of PGSFR has been completed at the end of February 2016 and now going toward the second design phase in 2016. All the design concepts of systems, structures and components (SSCs) have been determined and incorporated into the preliminary safety information document (PSID), which includes basic design requirements, system and component descriptions, the results of safety analysis for the representative accident scenarios. The PSID will be a base material for a pre-review of the PGSFR safety.

The target of the second phase of PGSFR design is to prepare a specific design safety analysis report (SDSAR) by the end of 2017. The specific safety analysis report is equivalent to the conventional preliminary safety analysis report (PSAR) but without the specific site information of the plant. The design activities are being carried out to freeze the design details of PGSFR by the end of 2016.

To support the design, various R&D activities are being performed in parallel with design activities, including V&Vs of design codes and system performance tests. The details on the design status and plan will be presented in the conference

Country/Int. Organization:

Korea Atomic Energy Research Institute, Republic of Korea

Poster Session 2 / 461

The approaches to the radiation characteristics of structural elements of the core determination during operation and decommissioning for BN-type reactors

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In the analysis of nuclear and radiation safety of existing and designed BN reactors considerable attention is paid to the problems associated with the formation of radioactive waste (RW) during their operation and decommissioning.

This paper describes the approaches to determine radiation characteristics of non-fuel compositions and structural elements of fuel assemblies (FA) and non-fuel assemblies of the core. The latter include the control and protection system (CPS) assemblies and in-vessel storage shielding assemblies. During the operation of BN reactor CPS assemblies are replaced, with subsequent transfer to RW, and in-vessel storage shielding assemblies are transferred to RW on reactor decommissioning.

Development of the approaches to determine radiation characteristics of the structural elements of the BN reactor core assemblies is an actual problem and some developments in this direction are presented within the framework of this work.

The radiation characteristics of irradiated structural elements depend on:

• the value and the spectrum of neutron flux at the location of irradiated structural elements;

• the irradiation history (the irradiation time, the number of irradiation intervals (cycles between refueling));

• the type of assemblies (weight fractions of elements and isotopes in construction materials such as steel, boron carbide; impurities in the assembly).

The calculation of the radiation characteristics for any user-defined assemblies on the core load map is provided with modern nuclear data libraries and computer codes. For this purpose, the procedure was developed for automatic selection of all the necessary data on decay energy, quantities of isotopes, activity and gamma radiation spectrum in the axial layers and entire assembly. It is also possible to define RW categories in the assembly axial layers for the selected cooling times range.

The analysis of the present study results indicates the important aspects of the radiation characteristics of the considered assembly types that need to be taken into account at all stages of BN reactors lifecycle.

Country/Int. Organization:

JSC "SSC RF –IPPE", Obninsk, Russia

6.9 Research Reactors / 462

CALCULATION AND EXPERIMENTAL ANALYSIS OF NEUTRONIC PARAMETERS OF THE BN-800 REACTOR CORE AT THE STAGE OF REACHING FIRST CRITICALITY FOLLOWED BY RATED POWER TESTING

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The main task of the measurements at different stages of the reactor start-up (first criticality, first start and further testing of the rated power) is to obtain complete and accurate information on the monitored neutronic parameters of the core. It is essential for subsequent reactor operation and also helps to verify and improve the accuracy of calculating neutronic parameters.
Insertion of the start-up neutron source initiated BN-800 first criticality which continued till the start-up project core formed, including neutronic measurements carried out under conditions of both the minimum critical mass and start loading, at the minimum controllable power level. Then, measurements were performed at different power levels during the no-load stage (when the turbine-generator was connected to the grid, the reactor power reaching 25% of the design level, i.e. until the reactor power start-up) and the on-load stage (when the rated power was reached).
Analysis of the results of the performed measurements showed that experimental and calculated values agree well (within the declared design and experimental uncertainties):
- the minimum critical loading is determined very precisely and the start critical state is predicted with high accuracy;
- agreement between the calculated and experimental values of CR worths is proved;
- regulatory compliance for reactivity balances is confirmed;
- agreement is achieved between the calculated and measured values of fission reaction rate distribution (relative power density) in the core;
- calculated estimations of temperature and power reactivity coefficients, reactivity effect due to fuel burnup and neptunium reactivity effect are confirmed by the measurement results.
Calculation methods used for the experimental analysis are similar to those employed for design justification of the core neutronic parameters.
The obtained results of the measurements and of their calculation analysis will be used in cross-verification of the GEFEST-800 computer code designed for the calculation monitoring of BN-800 core operation.

Country/Int. Organization:

JSC "SSC RF-IPPE", Obninsk

3.5 General Safety Approach / 463

SFR INHERENT SAFETY FEATURES AND CRITERIA ANALYSIS

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This study presents the inherent safety performance of BN-type reactor 2800 MW of thermal power with MOX core during ATWS initiated by various accident initiating events with simultaneous failure of all shutdown systems in all cases under investigation.
The most severe cases leading to the pin cladding rupture and possible sodium boiling were found out.
The impact of various safety features on SFR inherent safety performance during ATWS was also analyzed. The decrease in hydraulic resistance of primary loop, increase in primary pump coastdown, the implementing of thermo-mechanical, leakage based and other self-actuated safety systems considered as additional natural feedbacks were considered. Performing analysis resulted in a set of recommendations to the characteristics of the features referred above for the purpose of enhancing the inherent safety performance of SFR under investigation.
In order to exclude the safety barrier rupture during ATWS the set of criteria defining the ATWS processes dynamics and requirements to them were recommended based on achieved results. These criteria are include the natural circulation onset level (must exceed 0.07 rel. units in most severe case), the coolant flow rate drop under the natural circulation onset level (must be missing 0.015 rel. units in most severe case) and time from coolant flow rate drop under the natural circulation onset level till natural circulation onsets (must be missing 101 s). The recommendations for way to implement the self-actuated safety systems are also elaborated.
The analysis of admitted assumptions and obtained results revealed that to develop the refined requirements for the proposed criteria it is necessary to couple the SFR performance analysis for ATWS with uncertainty analysis. It is also necessary to take into account heat removal through passive heat removal systems even in a failure mode by heat-conductivity through the HX walls and to refine the acceptable temperatures of critical components of reactor (fuel, cladding, coolant and reactor tank) with respect to reactor inherent safety. The suitability of chosen acceptable temperatures values of critical components of reactor is discussed.
The results of the inherent safety analysis presented in this study are obtained by using the one-dimensional DYANA code for inherent safety analysis of fast liquid metal cooled reactors. Estimated sodium temperature and mass flow obtained from LOHS+LOF analysis via DYANA code were in reasonable agreement with those obtained from PHENIX benchmark end-of-life test.

Country/Int. Organization:

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Poster Session 1 / 466

Remote detection of raised radioactivity in emission from Beloyarsk nuclear power plant

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In the paper, we consider the gas-aerosol radioactive emissions and theoretically justify the possibility of remote detection of increased radioactivity in air emissions from the nuclear reactor BN. The comparative analysis of injected radionuclides into the atmosphere from nuclear power plant with advanced fast neutron reactor is carried out. On example of Beloyarsk nuclear power plant, the problem of remote detection of radioactivity in the atmospheric pollution is examined. Considering the emissions of certain groups, we can conclude: inert gases in the extract tritium in gaseous and liquid emissions, 14C and 131I in the exhaust air, the radioactivity is adsorbed on the particles in the polluted air, and "other" contained in the liquid emissions.

Table 1. The average value of radionuclide emissions (1985-1989) of the nuclear reactor on fast neutrons.

https://docs.google.com/drawings/d/1W-xBLuTjKuIGlbia4aXA7Ok5sAiKwGDUn4fK6o4D3GU/edit?usp=sharing

Taking into account the total activity of radioactive noble gases and feasibility of remote detection of raised radioactivity in emission from nuclear power plant and radio-chemical plant, we make a conclusion that radiometric system able to detect radioactive emission from NPP with fast neutron reactor.

The reported study was funded by RFBR, according to the research project No. 16-38-60115 mol_a_dk.

Country/Int. Organization:

Russia, V.E. Zuev Institute of Atmospheric Optics SB RAS,

Poster Session 2 / 467

System of coordinated calculation benchmarks for a fast reactor with sodium coolant in closed fuel cycle

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System of benchmarks of the class "tests"with prototype for neutron-physical and thermal hydraulic calculations of the BN-type reactors with nitride uranium-plutonium fuel is presented. The system includes benchmarks for models: cell,fuel assembly with end elements, active core, protection of the reactor installation. The system is intended for verification of the codes including constant support, methods and algorithms of calculation, multiphysics scheme of calculations, scenariosof operation in closed fuel cycle. World experience of creation of the calculation benchmarks was used for the development of this system. Tasks of analysis of nonlinear deformations of the active core and transition of the reactor to the equilibrium mode of operation are included. Formulationof test tasks was based on the following principles: conformance of the benchmark model with the range of studied effects, foundedrejection of unnecessary detail in models and material compositions, uniform information for construction of geometric models and the agreed size.System combines 6 types of benchmarks. The results of benchmarkcalculations made by the authors using codes CONSYST (ABBN-RF), MCU, JARFR, MMK, SCALE, SERPENT are presented.

Country/Int. Organization:

Russia / National Research Nuclear University MEPhI (Moscow Engineering Physics Institute)

1.3 SYSTEM DESIGN AND VALIDATION / 468

Main operation procedures for ASTRID gas power conversion system

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Until the end of the first part of the basic design phase (2017), the ASTRID project has made the choice of studying a power conversion system (PCS) based on a Brayton cycle with nitrogen as coolant. The justification is related to a safety and public acceptance considerations in order to inherently eliminate the sodium-water and sodium-water-air reactions risks. The objective of the studies engaged is to enhance the level of maturity of the gas PCS as close as possible to the classical Rankine cycle. The choice of two PCS of 300 MWe each has been made in order to limit the gas inventory, the size and length of gas pipes as well as maintaining a high level of availability.
This paper presents the current architecture of the gas PCS, the layout of the tertiary circuits and will also deals with specific operating procedures as start-up of the plant, scram, normal shutdowns and grid frequency control. The current procedures in the three circuits of the plant and the expected regulation will be presented. A focus will be made on the nitrogen inventory control which takes part of the electric power regulation provided to the grid. When possible, the comparison with the vapor PCS will be shown in terms of impact of thermal transients on structures.
Finally some perspectives of the gas PCS use for the future of the sodium fast reactors will be drawn in terms of better cost-effectiveness of operation through optimization of its Brayton cycle.

Country/Int. Organization:

France/French Atomic and alternative energies Commission

Poster Session 2 / 469

LOGOS CFD software application for the analysis of liquid metal coolants in the fuel rod bundles geometries

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Liquid metals (LM), such as sodium, lead or lead-bismuth eutectic (LBE), are preferred candidate coolants for advanced fast nuclear reactors next generation. Despite the comprehensive amount of experimental and calculated data, obtained by Russian as well as EU scientists in previous 30-40 years, the investigation of hydraulic and heat transfer characteristics of the fuel pin bundles is one of the key issue under the reactor design.

With the development of computing technologies, the investigation of thermal-hydraulic behavior of these coolants in the fuel-rod bundle geometries can be computed using commercial of in-house CFD software. As a result, a number of expensive full-scaled experiments with LM flows can be reduced.

The LOGOS CFD software was developed at the Russian Federal Nuclear Center - VNIIEF (Sarov, Russia) in a framework of the development of supercomputers and grid-technologies project. At present, validation and verification of the LOGOS in a framework of the new generation computation codes project is been performed in application to the LM flow simulation in the fuel pin bundles.

In this paper, the SST (Shear Stress Transport) k-ω model as well as combination of this model with turbulent heat transfer models, such as AKN and TMBF, released in the LOGOS in a framework of PRORIV project, is applied. Results of numerical simulations of the LBE flow around a heated rod in an annular cavity (THESYS project), sodium flow in the tightly packed parallel rod-bundle (TEGENA experiments) and LBE flow in the 19-pin hexagonal rod bundle with support grids (THINS project) are described.

Comparative analysis of experimental and calculated data obtained using both LOGOS software and ANSYS Fluent is given.

Country/Int. Organization:

Russia / JSC N.A.Dollezhal Research and Development Institute of Power Engineering (JSC NIKIET)

6.4 Neutronics –2 / 470

Analysis of the BFS-115-1 experiments

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As part of a bilateral agreement on the study of large axially-heterogeneous oxide-fueled SFR cores, CEA and IPPE have recently performed neutron physics experiments in the BFS facility. The configurations of interest are pancake-shape cores with a split fissile column and a sodium plenum, designed to favor a high inner plutonium conversion ratio and a low sodium void worth. Separate effect tests, including local and global sodium void situations as well as various rodded cases, have been done. The measurements included reactivity effects, spectral indices, detailed reaction rate traverses, neutron importance, etc.

The analysis of the experiments with Monte Carlo codes and recent nuclear data files shows the following trends:
Core reactivity is predicted within 1.5 , $dependingonthenucleardatafileusedSodiumvoidinginthe91centraltubesispredict$
The calculated axial reaction rate traverses match the experimental ones
The weight of the simulated control rod is predicted within 10%

Country/Int. Organization:

CEA, CAD/DEN/SPRC

6.5 Uncertainty Analysis and Tools / 475

System of Codes and Nuclear Data for Neutronics Calculations of Fast Reactors and Uncertainty Estimation

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Designing of neutronics characteristics of fast reactor cores and fuel cycle requires to use certified and qualified sets of computer codes and nuclear data. The calculation codes should be related to the modern state of computational techniques. The used nuclear constants should be adequate to the most reliable evaluations, adopted in modern libraries of evaluated nuclear data.
The paper consider a modern state of Russian neutronics computer codes and nuclear data used in fast reactor applications for calculation of core and nuclear cycle parameters.
The ROSFOND evaluated nuclear data files and the ABBN group data set are used as the basis of nuclear input data. The ROSFOND library now contains about 650 files of data for most important

and not so important reactor materials. The selection of files was made based on BROND-3, ENDF/B-VI.8 and VII.0, JEF-2.2 and JEFF-3.1, JENDL-3.3 evaluations by comprehensive study of their quality. For treating the ABBN data the special code system CONSYST/ABBN was developed.
Three directions in developing of codes for fast reactor neutronics calculations can be stated: (1) discrete codes, (2) based on Monte-Carlo, (3) used synthesis methods. Codes, which are used in the design calculations, mostly solve the Boltzman transport equation in diffusion approximation, they are: TRIGEX, JARFR, GEFEST, FACT-BR, SYNTES. Codes, which are based on Monte-Carlo method, were developed during many years. Nowadays they have additional impulse in interest due to fast developing of the computational technique. Recently a code MMKK was developed. It now used in planning and analyzing of reactor-physics experiments as well as for precise calculations of fast reactors BN.
For the shielding calculations as well as for determining diffusion-transport corrections codes TWODANT and DORT-TORT are used. For the depletion and kinetic calculations CARE and ORIGEN codes are used.
The main feature of the all mentioned codes is that they use one, same and unique constants data base ABBN with the code CONSYST for generation of effective cross-sections.
The system INDECS of codes and archives is now used for uncertainty estimations which is based on usage of perturbation theory and covariance matrices of nuclear constants. The TRIUM code based on GRS method is now developed. It is a synthesis of TRIGEX, MMKK and INDECS codes.

Country/Int. Organization:

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3.1 Safety Program / 476

"ASTRID safety design: Radiological confinement improvements compared to previous SFRs"

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ASTRID is the Advanced Sodium Technological Reactor for Industrial Demonstration which is intended to prepare the Generation IV reactor, with improvements in safety and operability. In order to meet the objectives of the 4th generation reactors and comply with the related specifications, the ASTRID project integrates innovative options.
In the earlier phase of ASTRID project, a specific safety approach was set and its main guidelines were agreed by the French Nuclear Safety Authority. This basic safety design guide is currently applied as reference for the choices of the design options. The paper presents the safety approach, called "top-down"approach, relating to the "confinement"safety function. The confinement design of ASTRID has several safety objectives from both radiological point of view and sodium chemical risk, and its design is based on "plant state"approach.
As concerns potential radiological risk, main objectives are to postpone a hypothetical off-site release of radiological material coming from core degradation and also to decrease its health and environmental possible consequences.
As concerns the sodium chemical risk, main objectives are to prevent by design an overpressure of the containment, introducing drawbacks in terms of confinement, and also to cope with the risk of off-site release of soda aerosols with possible health effect.
In order to meet all these objectives, design provisions are taken, considering the different release ways inside the confinement. The paper presents the lessons learned from the previous SFR confinement and the method applied to choose for ASTRID consistent design options.
Major part of these design provisions has, in particular, an important function of severe accident mitigation. The design of these mitigation provisions takes into account the lessons learned from

Fukushima event, in order to prevent any cliff edge effect in terms of radiological consequences.

Country/Int. Organization:

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5.8 Structural Materials / 477

Recent suppling of 316L(N) stainless steel products for ASTRID

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CEA has been involved since 2006 in a substantial effort on the ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) conceptual design, mainly in cooperation with EDF, as experienced Sodium-cooled Fast Reactor (SFR) operator, and AREVA, as experienced SFR Nuclear Island and components engineering company. Whereas previous SFRs were designed for 30 or 40 years lifetime, ASTRID should meet the Generation IV systems requirements of 60 years life time. The priority of the ASTRID material program is then given to have robust time-dependent data to design the Demonstrator. 316L(N) stainless steel and its weldments are of prime interest, as they constitute the largest components of the primary circuit and secondary pipework, components which are difficult or impossible to replace
That's why an ambitious programm has been launched to supply typical products in 316L(N):

- Plates, as major products for the vessels,
- A seamless shell, for some internals, to reduce the numbers of welds,
- A thick forged part, representative of tubesheet for Intermediate Heat eXchangers.

It aims at re-activating the procurement of 316L(N) products and at feeding the material data base thanks to new manufactures.

In the full paper, the requirements of the 316L(N) products will be given according to RCC-MRx Code (AFCEN Code). The main acceptance tests results will then be presented and discussed.

Country/Int. Organization:

AREVA NP, CEA, EDF R&D - FRANCE

Poster Session 1 / 478

Fast Reactors - The Belgian Regulatory Approach

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In Belgium, the research center SCK•CEN is planning to build a lead alloy-cooled fast reactor as part of an ADS facility. And Whilst the Belgian Regulator, i.e. the Federal Agency for Nuclear Control - FANC, has experience with the licensing and control of PWRs, there was little to no experience with these less-common innovative reactor types. For this reason a prelicensing project was launched. The scope, the method and the goal of this prelicensing project is presented from the point of view of the regulator.

Country/Int. Organization:

Belgium - The Federal Agency for Nuclear Control

Poster Session 1 / 480

FEATURES OF THE NUCLEAR FUEL CYCLE SYSTEMS BASED ON JOINT OPERATION OF FAST AND THERMAL REACTORS

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In a time of the existence of the national nuclear program in the framework of the weapons and civil complex it has been accumulated and continues to accumulate a significant amount of plutonium.
The isotopic vector of plutonium produced in reactors varies strongly depending on the type of reactor, fuel burnup and the time elapsed since the moment of unloading it from the reactor before loading it as a fuel component in another reactor.
It is fundamental fact that in a two component nuclear power system based on thermal and fast reactors, there is plutonium exchange between these types of reactors in a joint closed nuclear fuel cycle (NFC). Plutonium vectors coming into fast and thermal reactors can vary within a wide range because they will not only depend on the reactor features, but also on NFC management.
The neutronic properties of plutonium isotopes differ also dramatically also. This leads to the fact that the physical characteristics (including safety features) of the reactor, in which the plutonium is used as fuel, will depend on the isotopic vector.
The aim of the paper is to determine the characteristics of stationary fuel cycles of nuclear power system based on VVER-TOI and BN-1200 loaded with oxide fuel of various compositions. Characteristics of reactor systems with a partial or complete recycling of spent nuclear fuel and plutonium recycled are compared with those of the reference system which consist of the VVER-TOI reactors with uranium fuel, operating in an open NFC.
The results of the computational researches of the transition of the two-component system of into the equilibrium mode in the closed NFC are presented.
A feature of the system which is balanced by plutonium is that both types of reactors spent fuel is completely reprocessed and the separated plutonium is used totally to make MOX - fuel. The MOX fuel is used not only in the BN-1200, but also as a partial load in reactors VVER-TOI. The optimization of fuel the reactor fuel performances is needed for its effective cooperation.
Complete closure by plutonium in the NFC consisting only of the VVER-TOI reactors using MOX - fuel is impossible.

Country/Int. Organization:

Russia,State Scientific Centre of the Russian Federation –Institute for Physics and Power Engineering

3.2 Core Disruptive Accident / 483

Quantitative Evaluation of the Post Disassembly Energetics of a Hypothetical Core Disruptive Accident in a Sodium Cooled Fast Reactor

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The analyses of Hypothetical Core Disruptive Accidents (HCDAs) play a fundamental role in the safety assessment of Sodium Fast Reactors (SFRs). The accident sequence is subdivided into different phases suggested by dominant key phenomena. The Initiation Phase (IP) describes the fatal deviation from nominal operation until the failure of single sub-assemblies (SAs), while the subsequent Transition Phase (TP) considers possible damage propagation up to the formation of a large fuel/steel pool. During the TP, a coherent movement of the liquid pool may result in a more compact fuel arrangement leading to recriticality events with consequent upward discharge of the pressurized hot fuel/steel mixture. The power peaks are considered as starting points for the Post-Disassembly Expansion Phase (PDE). The sodium vapor rapidly produced by Fuel-Coolant Interaction (FCI) in the upper plenum displaces and accelerates the surrounding liquid sodium. As a result, a significant mechanical energy may be released acting as a load on structures /vessel. The identification and evaluation of the main phenomena and event paths enhancing or mitigating the mechanical work potential during the PDE is essential to give evidence on the vessel/structures integrity with important design clues for the development of future SFRs. The present paper deals with PDE phenomenology and includes an overview of the quantitative evaluation of the work potential during the PDE of an Unprotected Loss of Flow (ULOF) on the basis of mechanistic SIMMER simulations. For assessing the important determining factors a large number of parametric analyses have been conducted at KIT for an SFR model case and, additionally, KIT simulations performed in previous years have been studied. A wide range of initial conditions and modelling options that may strongly impact the mechanical work potential has been investigated and are integrated in this work, i.e. different liquid fuel mass and temperature conditions, different steel contents in the melt pool, different structure conditions affecting the melt discharge into the sodium plenum, and different driving vapor pressures. The large amount of results has been also employed in the framework of the application of the Phenomenological Relationship Diagram (PRD) to perform a probabilistic evaluation of the work potential of an ULOF/PDE in a sodium small- to medium-sized reactor (SMR, 300 MWe). The present study has been conducted under the research contract between the KIT and the Regulatory Standard and Research Department of National Regulation Authority in Japan.

Country/Int. Organization:

Karlsruhe Institute of Technology - Germany

7.3 Non Proliferation Aspects of Fast Reactors / 485

FALCON advancements towards the implementation of the ALFRED Project

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In the European context, the implementation of the Generation-IV vision for safer, more sustainable, economic and proliferation resistant nuclear energy systems is being pursued as long-term action of the EU's strategies for a sustainable scenario securing energy availability. This action, envisaging innovative reactor concepts to implement the Gen-IV vision, is supported by the European Sustainable Nuclear Industrial Initiative: the Lead Fast Reactor is one of the concepts being pursued, and ALFRED –as its demonstration reactor –the cornerstone for its development.
Beyond this, the potentialities of the LFR, and the industrial interest that is continuously growing on this technology, have recently enlarged the scope of ALFRED. The evidences being collected in support to the claims of unparalleled safety and competitiveness of the LFR, opened the possibility for a shorter-term deployment, with lead-cooled Small Modular Fast Reactors as a promising option to address a market segment of increasing interest.
Accordingly, the efforts of the FALCON (Fostering ALFRED Construction) International Consortium are being oriented to fully exploit the potentialities of the LFR technology, extending the role of ALFRED also as a prototype of a lead-cooled SMFR, thereby addressing the objectives of industry both in the short- and long-term perspective.
To support such an ambitious mission, a pan-European effort is required at each layer composing the general picture of this programme: science and technics, industry, safety, management, financing, education and training and human resources. A first action in this sense is being pursued by setting up a Distributed Research Infrastructure, as a Centre of Excellence on the heavy liquid metal technology. The Infrastructure will gather the most relevant facilities already existing and planned, along with the new ones required to complement the present landscape for sustaining the realization of future LFRs. The vision for cooperation at a pan-European dimension will be achieved through the open access policy of the Infrastructure, that will permit students and senior experts to jointly program, design and perform experimental campaigns in the largest and most advanced technology park in the World.
The paper focuses on the phased Roadmap for the ALFRED Project implementation in its extended role, as well as on the vision for the Distributed Research Infrastructure, as an incubator for competences and ideas sustaining Europe in a leading position on the LFR technology. Finally, challenges and opportunities offered by this extended approach are outlined as a conclusion.

Country/Int. Organization:

Italy

7.3 Non Proliferation Aspects of Fast Reactors / 486

Status and perspectives of industrial supply chain for Fast Reactors

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Fast reactors, selected at European level as next generation Nuclear Energy Systems, pose undeniable challenges from a technological point of view. In order to support the foreseen deployment strategy, a survey of the existing industrial supply chain have been thoroughly carried out in terms of its capabilities and potentialities with respect to Fast Reactors needs.
The main challenges found to potentially affect the deployment strategy of Fast reactors have been found to be related to the maintaining the current supply chain capabilities, defining specifications of critical components, developing new materials and fabrication/inspection techniques, ensuring the necessary accreditation and quality.
The main critical components of Fast Reactor concepts have characteristics and requirements that will require further investments on R&D and qualification. This will represent a stimulus for the supply chain and, in perspective, considered a good market and a business opportunity for industry. Implementation of requirements for Fast Reactors into the nuclear codes and standards is still a key aspect. The nuclear industry is country-specific and different efforts aimed at international harmonization of codes and standards have not been very successful up to now. Top-level initiatives should be encouraged, as far as possible, for the ESNII concepts. A challenge for fast reactor development in the long term is to minimize or avoid code/country-related barriers, in order to assure the suppliers a larger, open and attractive market.
The analysis also covers the capacities and technologies that the EU industry will need to maintain in the medium to long terms to develop and build fast reactor projects. Any identified shortfall or weakness represents an opportunity for improvement, by strengthening the involvement of industry in the European sustainable nuclear program.

Country/Int. Organization:

Italy

7.4 Fuel Cycle Analysis / 487

Fast reactor systems in the German P&T and related studies

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A study on partitioning and transmutation (P&T) for nuclear waste management was done in Germany in 2012-2014 [1]. In particular, possible fuel cycle options and systems for burning/utilization of transuranic elements in the German and European frameworks were proposed and analyzed at Karlsruhe Institute of Technology (KIT), while exploring results of earlier projects, to which KIT contributed together with its partners. In one type of scenarios, oriented to the German framework, the objective is to burn almost all transuranic elements accumulated due to operation of German nuclear power plants which are scheduled to be shut-down by 2022. In alternative European scenarios the main attention in the short term is on burning of minor actinide (MA) inventories accumulated in Germany and other countries, which are phasing out of nuclear soon; while in the long term it is on management of MA inventories produced in countries relying on nuclear energy also in the future.

These analyses have been extended by studies performed at KIT more recently. In particular new ASTRID-like and ESFR-like sodium fast reactor models have been established and analyzed in addition to those studied earlier [2,3]. These models are based on proposed in European projects designs,

which are modified to allow a higher transuranic content, while avoiding deterioration of safety-related features. Their transmutation potential and safety performance are under investigation. In the paper these models are described and results of the investigations are reported. The preliminary conclusion is that the considered systems are suitable for all scenarios options considered in the German P&T study.

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[2] F. Gabrielli, et al. „ASTRID-like Fast Reactor Cores for Burning Plutonium and Minor Actinides ", Energy Procedia 71 (2015) 130 –139.
[3] B. Vezzoni, et al. "Plutonium and Minor Actinides Incineration Options using Innovative Na-Cooled Fast reactors: Impacting on Phasing-Out and On-Going Fuel Cycles", Progress In Nuclear Energy, 82 (2015), 58 –63.

Country/Int. Organization:

Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany

5.6 Liquid Metal Technologies / 489

Testing of electrochemical hydrogen meter in a sodium facility in Cadarache

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An electrochemical hydrogen meter (ECHM) developed in IGCAR, India has been tested in a sodium facility in CEA Cadarache, France as part of IGCAR-CEA collaboration of fast reactor safety. ECHM is basically a concentration cell which can be used for online monitoring of hydrogen concentration in ppb levels in liquid sodium in order to detect the eventuality of steam leak in the sodium circuit at its inception, during the operation of fast breeder reactors. A detailed description of ECHM is given in ref.[1]. These sensors were successfully tested in Fast Breeder Test Reactor (FBTR), large sodium facility of IGCAR and in Phenix, France [2]. ECHM provides an alternate technology to the conventional diffusion based hydrogen sensor for detecting steam leaks into sodium . It is a robust and inexpensive sensor and is simple to operate.

ECHM has been installed in the SUPERFENNEC sodium loop facility at Cadarache, France, along wi

Prior to the tests both ECHM and conventional hydrogen detection system (SPHYNX) were calibrated. ECHM has been calibrated at an operating temperature of 450oC in a bench top sodium loop with respect to cold trap temperature variations. SPHYNX system was calibrated in SUPERFENNEC loop for calibrated H_2 leak. Tests were carried out initially by introducing $NaOH$ in liquid sodium. Subsequently, NaH additions were also carried out. The performance of both ECHM and SPHYNX were monitored as a function of hydrogen concentration and temperature. Hydrogen concentrations in sodium were varied by adding NaH equivalent to 25 ppb to 150 ppb above the background value and temperature was varied from 424 to 453oC. The response, by both the sensors, was nearly identical at the operating temperature at which ECHM was calibrated in IGCAR. The study revealed that the performance of both the sensors were comparable.

Country/Int. Organization:

France/CEA
India/IGCAR

4.3 Partitioning and Sustainability / 492

External Assessment of the U.S. Sodium-Bonded Spent Fuel Treatment Program

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Some advocates of electrometallurgical reprocessing (or "pyroprocessing") of spent nuclear fuel argue that the technology has many advantages relative to aqueous reprocessing methods, including cost savings, safety benefits, and increased proliferation resistance. However, to date there has been very little actual operating experience with production-scale electrometallurgical processes, making it difficult to validate these claims. Since 1996, U.S. researchers have been implementing a program to pyroprocess 26 metric tons of sodium-bonded, metallic spent fuel (both driver and blanket) from the shutdown EBR-II and FFTF fast reactors at the Fuel Conditioning Facility at present-day Idaho National Laboratory. In 2000, the U.S. Department of Energy (DOE) asserted that the campaign would be completed within a decade. However, as of 2015, only about 15% of the inventory had been processed, and it appears likely that several more decades will be needed to finish the job. DOE has released very little information to the public on the program and the reasons why it is experiencing such severe delays. However, documents recently obtained by the Union of Concerned Scientists (UCS) under the U.S. Freedom of Information Act shed light on the operational issues of pyroprocessing technology that have contributed to the problems experienced during the campaign. It is apparent from this information that the technology is neither as efficient nor as clean as some claim. The Republic of Korea and other nations that have expressed a deep interest in the development of pyroprocessing technology would be well-advised to take note of the formidable challenges associated with its practical implementation.

Country/Int. Organization:

USA/Union of Concerned Scientists

Poster Session 1 / 495

A High Density Uranium Zirconium Carbonitride LEU Fuel for Application in Fast Reactors

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For many years, Russian researchers have developed and tested a high density, high temperature U-Zr-C-N fuel for potential application in different types of reactors, including fast reactors. As part of this effort, reactor tests have been performed to low burnup. However, reactor-testing data is still needed at high burnup to confirm the optimal performance of the fuel. The SM-3 reactor, which is a high-flux reactor located in Dmitrovgrad, Russia, will be used to test a U-Zr-C-N ($U_{0.9}Zr_{0.1}C_{0.5}N_{0.5}$) fuel to ~40% burnup. The fuel will then be examined to determine its performance during irradiation. The fuel that will be tested has a density of 11.9 g/cm3 and an enrichment of 19.75% (uranium-235), and the uranium density of this fuel material is 10.8 g/cm3. About 1000 effective days of irradiation will be required to achieve the targeted burnup. This presentation will discuss the details of the planned irradiation, along with results of out-of-pile research that has been performed on the as-fabricated fuel. The positive characteristics of the U-Zr-C-N fuel will be discussed, and comparisons will be made to other fuel types.

Country/Int. Organization:

Belarus/SOSNY of the National Academy of Sciences of Belarus

6.9 Research Reactors / 497

BOR-60 REACTOR OPERATIONAL EXPERIENCE AND EXPERIMENTAL CAPABILITIES

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The fast research reactor BOR-60 is one of the world's leading research reactors in large-scale testing of fuel, absorbing and structural materials to design new advanced fast reactors, pressurized water reactors, gas-cooled and fusion reactors, and demonstrate the feasibility of lifetime extension for VVER and BN-type reactors.
BOR-60 was commissioned in December 1969, and by early 2017 it will have been under operation for ~275000 hours. By this parameter BOR-60 is the world's leader, and it continues to show the potential for sodium fast reactor lifetime extension.
For nearly 48 years BOR-60 has been under reliable and efficient operation, being at present almost the only operating fast research reactor that has unique experimental capabilities for integrated research in different trends in combination with well-equipped materials testing labs and fuel manufacturing and reprocessing facilities. The scientific data obtained in this reactor made it possible to demonstrate the feasibility of using materials, fuel and absorbing elements for BN-350, BN-600, BN-800, and other reactor types.
In 2014 a range of activities was carried out including equipment condition survey, calculated and experimental data analysis of operational parameters and conditions for permanent reactor components, testing of structural materials samples after long-term operation under irradiation, etc. From these results the remaining lifetime has been evaluated, and the operator made a decision on reactor lifetime extension. After reviewing a set of reactor condition documents and obtaining a favorable conclusion from an independent expert company, the relevant regulatory authority of the Russian Federation has extended the BOR-60 operating license until 2020.

Country/Int. Organization:

Russia, JSC "SSC RIAR"

Poster Session 1 / 499

The concept of 50-300 MWe modular-transportable nuclear power plant with sodium coolant and a gas turbine

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Well-known vulnerability points for sodium-cooled reactors are:
–chemical interaction of sodium and water;
–three-circuits reactor design;
–increased material consumption and building cost compared to PWR and BWR;
–increased on-site amount of building and assembling operations.

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The most of these difficulties can be solved with replacing steam turbine by specially designed gas t
Since 2001 IPPE is working on development of so-called ``BN GT'' technology set, including:
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–the use for primary circuit BN-600's well-proven materials, elements and technologies of sodium reactor;
–the use for secondary circuit specially designed helium turbine without any intermediate circuits;
–design specification, intended to locate any reactor and turbine equipment in rail-car form-factor
–and some others.

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Currently achieved competitiveness estimates show significant commercial advantage of 300 MWe BN GT p
Commercial competitiveness over gas-burning power plants is also possible in case of nuclear fuel re
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Country/Int. Organization:

Russian Federation / Institute of Physics and Power Engineering

6.4 Neutronics –2 / 501

Physical start-up test of China Experimental Fast Reactor

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China Experimental Fast Reactor (abbr. CEFR) is a pool-type sodium-cooled fast reactor in China Institute of Atomic Energy (abbr. CIAE), with a thermal power of 65MW and an electric power of 20MW. The construction started in 2000 and the first criticality was reached in July 2010. On December 15th 2014, CEFR reached full power for the first time and was successfully operated for 72 hours.
During the physical start-up of CEFR, a series of tests were carried out in four aspects, i.e., fuel loading and first criticality, control rod worth measurements, reactivity coefficient measurements, and foil activation measurements. A large amount of experiment data was obtained in the process.
In order to compile and reserve the experimental data in a standard and refined form, and to benefit the worldwide fast reactor society on the validation of codes and nuclear data, China Institute of Atomic Energy proposed an IAEA Coordinated Research Project, and got approved preliminarily.
The specific objectives of the project lie in 4 aspects: firstly, to collect and evaluate experiment data obtained from CEFR physics start-up experiments mentioned above; secondly, to establish a simplified model of the CEFR core and give the correction factors and descriptions of associated

methods; thirdly, to share the experiment data and the simplified core model with CRP participants for joint calculations and analysis; fourthly, to gather and analyze the calculation results, and to publish a benchmark analysis report.
China Institute of Atomic Energy would like to take this great opportunity to express their welcome to all organizations to participate in this project!

Country/Int. Organization:

China P.R./China Institute of Atomic Energy

4.1 Fuel Cycle Overview / 506

Assessment of the anticipated improvement of the environmental footprint of future nuclear energy systems

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Environmental issues are nowadays a growing concern within most of the public opinion. It is therefore mandatory to propose relevant and qualified assessment of the overall environmental footprint of the different types of energy sources which are envisaged to be implemented. This question is specifically important for nuclear energy which suffers from a poor image in the public opinion due to the recent Fukushima accident. In this context, we developed a Life Cycle Assessment (LCA) tool, referred to as NELCAS, based on the current French nuclear energy system. Thanks to the Nuclear Safety and Transparency annual reports, detailed quantitative data were available for each of the fuel cycle plants. The whole fuel cycle from ore-mining to geological repository was considered as well as data for construction, deconstruction of any plants as well as the contribution of the transport. All the matter and energy fluxes were considered and normalised versus the electric production. Key environmental indicators, such as land use, water withdrawal and consumption, gaseous release, acidification, eutrophication, waste production …as well as potential impact indicators were hence assessed and validated with comparison with the few existing LCA results. This model was hence used to assess the respective figure of merits of the different generation of reactors and fuel cycles. In particular, it demonstrates that actinides recycling has a strong beneficial effect on the overall footprint due to the relative high impact of the front-end activities, specifically the ore mining.

In the framework of a joint CEA-EDF-AREVA group, reference deployment scenario for the 4th generation reactors were developed for the French case based on both technical and economic considerations. The NELCAS tool was therefore used to assess the impact on the overall environmental footprint of this reference scenario.

Country/Int. Organization:

CEA, French Nuclear and Alternative Energy Commission, Nuclear Energy Commission,

4.1 Fuel Cycle Overview / 507

1992-2017: 25 years of success story for the Development of Minor Actinides Partitioning Processes

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In the framework of the successive 1991 and 2006 Waste Management Act, French government supported a very significant R&D program on partitioning and transmutation in fast reactors of minor actinides (MA). This program aimed to study potential solutions for still minimizing the quantity and the hazardousness of final waste, by MA recycling. Indeed, MA 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.
Over the 20 years of development, different types of strategies were studied, from the early multi-stage DIAMEX-SANEX processes to the most recent innovative SANEX, from the grouped extraction of MA thanks to the GANEX process to the most recent sole Americium recycling thanks to the EXAm process. These developments were supported by a robust and long-standing approach allowing successively to screen the potential extractants, to quantify their extractive properties and develop a relevant chemical model to simulate it and to address their hydrolysis and radiolysis resistance. Finally, all these processes were qualified tested on a few kg of spent nuclear fuel within the Atalante CBP facility. This wide research program allows France to get in hand a flexible portfolio of MA recycling processes that could be implemented after industrial upscaling. More recently, CEA initiated a demonstration experiment, the so-called integral experiment, which aims to re-irradiate in a Material Testing Reactor some fuel pellet manufactured from recycled UAm. Most recent results on these key experiments will be presented.
Finally, several European Research Projects were funded in parallel by the European Commission and allow studying alternative separation processes. A general overview of this 20 years of successful and innovative research history will be synthesised in this presentation.

Country/Int. Organization:

CEA, French Nuclear And Alternative Energy Commission, Nuclear Energy Division

5.9 Large Component Technology II / 510

Challenges During Manufacture of Reactor Components of PFBR

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BHAVINI is constructing Prototype Fast Breeder Reactor (PFBR), forerunner of FBRs, a 500 MWe sodium cooled, pool type, mixed oxide (MOX) fueled reactor at Kalpakkam. Presently PFBR is in the commissioning phase. The reactor assembly consists of large dimensional vessels viz., Safety Vessel, Main Vessel and Inner Vessel made of Austenitic stainless steel. The top shield is a box type structure comprising of Roof Slab, Rotatable plugs viz., Large and Small plugs made of Carbon steel

A48P2 material and Control Plug at the center. The entire core is placed over the Grid Plate which in turn is supported by Core Support Structure equipped with Core Catcher at the bottom. The control and shutdown mechanisms are housed inside the Control Plug with necessary provisions for core instrumentation. The vessel houses the primary heat transport circuit which consists of Primary sodium pump, Primary Pipe and Intermediate heat exchanger transferring the primary heat from the core. The in-vessel and ex-vessel core handling are performed by Transfer Arm and Inclined Fuel Transfer machine. The decay heat is removed by passive systems consisting of Decay Heat Exchanger and associated components. The reactor is equipped with in-vessel and ex-vessel in-service inspection devices. These components had undergone many stages of manufacturing viz. forming, rolling, welding, machining etc. meeting the stringent specification requirements as neither repairs nor replacement can be possible at the later stages of reactor operation for major components. The dimensional tolerances were achieved at various stages of manufacture and the interfaces of these Over Dimensional Components were meticulously matched to avoid interferences during final assembly inside the reactor vessel. This paper presents the challenges faced during the manufacture of critical reactor components which serves as a vital input to future fast reactor program in India.

Country/Int. Organization:

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Poster Session 1 / **512**

Testing and Qualification of shielded flasks for handling sodium wetted large sized components of PFBR

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BHAVINI is constructing Prototype Fast Breeder Reactor (PFBR), forerunner of FBRs, a 500 MWe sodium cooled, pool type, mixed oxide (MOX) fueled reactor at Kalpakkam. Presently PFBR is in the commissioning phase. The reactor has three main heat transport circuits namely primary sodium, secondary sodium and steam-water system. The Primary sodium Pump and Intermediate Heat exchanger housed inside the Reactor vessel needs to be handled for repairs / replacements during reactor life. Due to the activity of primary sodium, corrosion products or induced activity in these components and to avoid chemical reaction of Sodium with air and moisture, PI Flask (Pump and IHX Flask) is used for handling these components. PI Flask is a 35m tall and ID 2250mm leak tight structure to handle 60T load in a leak tight manner and shielded environment. The total weight of the PI flask along with Pump or IHX is weighing 200 MT which is handled with 280MT EOT crane installed in Reactor Containment Building. Constructional feature includes hoist mechanism to lift 60T load designed with single failure proof system, mechanical stoppers to support the load at 30m height, 12 leak tight shells with bolted construction meeting the verticality, horizontality and leak tightness requirements and disc valve drive mechanism integrated with Airlock to facilitate opening and closing of separable discs for the movement of Pump or IHX. All the materials used were tested to ensure specification requirements are met and the fabricated joints were subjected to NDE and HLT. The shielded shells were subjected to radiometric testing to ensure shielding requirements. The mechanical stopper mechanism, hoist mechanism and disc valve drive mechanism were independently tested before final assembly and Performance tests under 'No'load, 'Full'load and 'Over' load conditions including verification of interlocks were conducted for the qualification of PI Flask using mobile Control Panel. This paper presents the various performance tests conducted under 'No load'and 'full load'conditions for the qualification of leak tight requirements, shielding adequacy and functionality checks of various in built mechanisms viz., Hoist, Disc Valve and Mechanical Stoppers using PLC of PI Flask

Country/Int. Organization:

BHAVINI (PFBR)
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Poster Session 2 / 514

Experiences during construction & Commissioning of electrical power Generation and Evacuation systems in PFBR

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Abstract

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Electric power supply comprising of both OFF site and ON site power supply systems is designed to fa
The emergency electric power supply (Class IE) system is generally categorized into three types base
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1. AC power supply to Auxiliaries, which can tolerate short interruption upto 3 minutes is classified as Class-III AC Emergency power supply.
2. No break AC power supply to auxiliaries derived from class-III buses through rectifier/charger and inverter with a battery backup is called class-II power supply.
3. No break DC power supply to auxiliaries derived from Class-III buses through rectifier/charger with a battery backup is called Class-I power supply. All Emergency power supply systems are designed to fulfill the safety criteria for Class-IE power supply system such as adequate redundancy by independent division having necessary capability & reliability, physical separation and functional isolation etc.

Variable speed AC drives are provided for the two each PSP & SSP pumps. An AC Pony motor is additionally provided for each of the primary sodium pumps. The electrical heating system for sodium circuits are designed to prevent arcing damage to the pipe and equipment that might be caused by the electrical heaters by adopting ungrounded power supply system. The insulation monitoring devices are provided in sodium circuits. The heaters on the primary sodium and Argon line are triplicated and heaters on the secondary sodium systems are duplicated.

Each section of the 6.6kV switchgear, 415V PCC, MCC, HCCs are provided with Switchgear Interface Panel to facilitate necessary interface/interlock for control, metering, indication, Annunciation and also provides galvanic isolation between electrical equipment and main control room.

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The installation of about 500 panels at various buildings are challenging due to layout constraints,
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Country/Int. Organization:

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Poster Session 1 / 516

CHALLENGES DURING CONSTRUCTION OF SODIUM PIPING SYSTEMS FOR 500MWe PROTOTYPE FAST BREEDER REACTOR

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Prototype Fast Breeder Reactor (PFBR) consists of Primary Sodium Circuit (PSC), Secondary Sodium Circuits (SSC), Safety Grade Heat Removal Circuits (SGDHRC) and Steam-Water circuit. The principal material of construction for sodium piping circuits is austenitic SS316LN/SS304LN.

Manufacturing of thin and big bore piping with tight tolerances along with the high distortion in stainless steels due to high thermal expansion and low thermal conductivity makes fabrication extremely challenging. With strict rules of sloping to be given to the piping to make conducive for full draining of the sodium loops, the fabrication challenges become multifold. All sodium pipelines inside Reactor Containment Building (RCB) are provided with hot guard pipe and are inerted with nitrogen. The guard piping and the containment penetrations require sequential welding. Limited space at site for the erection of sodium piping along with welding at inaccessible areas with confined space makes the work all the more challenging. Terminal joints hook-up to tanks having frozen sodium inventory needs to be done meticulously adhering to highest level of industrial safety standards.

The welding standards and acceptance criteria of PFBR sodium piping system is very stringent compared to conventional piping systems. Due to pyrophoric nature of sodium, the boundaries of various sodium piping systems must possess a high degree of reliability against failure. The welding of sodium piping systems are carried out by combination of Shielded Metal Arc Welding (SMAW) and Gas Tungsten Arc Welding (GTAW) process. Due to complex constructional features of the sodium piping systems, the argon gas purging, welding and non-destructive examinations are extremely difficult and challenging task. Apart from deployment of innovative purging methodologies, various special tools and fixtures were designed, developed and used for welding & fabrication. All the sodium pipe lines and components are provided with surface heaters, thermocouples, wire type leak detectors and insulation. Measurement of deflections of the sodium pipe lines during preheating and comparing with the analysis results is a vital step during the commissioning of sodium systems. This paper highlights on welding and fabrication aspects, challenges faced and innovations during construction of sodium piping circuits for 500MWe Prototype Fast Breeder Reactor.

Key Words: Sodium piping, welding, fabrication

Country/Int. Organization:

PFBR, Bharatiya Nabhikiya Vidyut Nigam Limited (BHAVINI)
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1.7 ADS AND OTHER REACTOR DESIGNS / 517

Physical and technical basics of the concept of a competitive gas cooled fast reactor facility with the core based on coated fuel microparticles

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At the turn of the century a new step in the development of gas cooled fast reactors (GCFR) in the world was made through international cooperation in the elaboration of innovative nuclear energy systems. In the mid of 2000s, in Russia rather active investigations in this direction were resumed under the scientific supervision of NRC "Kurchatov Institute", what happened largely due to the extensive development of technologies of high-temperature thermal reactors in this time.

The paper presents the results of the stage-by-stage development of a modern Russian proposal for the technical concept of a reactor facility with a high temperature fast helium cooled breeder reactor with expanded breeding working in a closed fuel cycle and having its own role in the nuclear energy system due to efficient electricity generation, and, in the longer term, possibility of industrial applications. It is expected that the unit will have the level of specific capital costs comparable to competitive nuclear energy sources.

The concept of a reactor facility with the BGR-1000 reactor of 1000 MWe capacity is based on the synthesis of technologies of high temperature and light water reactors. Gradual development of the concept is assumed in terms of the use of BGR-1000 for industrial and technological applications with a consistent increase of the core outlet coolant temperature.

The reactor design is based on the core with a fixed bed of coated microparticles made of different fuels and directly cooled by a longitudinally-cross flow of helium. The core design allows to have limited excess reactivity, exclude significant radiological consequences of accidents, provide the required level of fuel breeding and fuel reprocessing on the basis of best available practices, as well as, in the longer term, the closure of the fuel cycle with respect for all actinides.

Completed and planned studies are conducted in the following areas:

-choice of fundamental decisions for facility layout, design of the fuel and main equipment;

-optimization of the safety concept, including technical and economic analysis of the proposed safety solutions;

-formation of the in-reactor and external fuel cycle stages taking into account system requirements to a nuclear energy source;

-identification of critical technologies and design data needs;

-development of necessary R&D programs and experimental studies.

Country/Int. Organization:

Russian Federation, National Research Centre "Kurchatov Institute"

4.2 Reprocessing and Partitioning / 519

A comprehensive study of the dissolution of spent SFR MOX fuel in boiling nitric acid (the PHENIX NESTOR-3 case)

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Dissolution experiments undertaken on irradiated fuel pins from sodium fast reactors (SFR) date back to the 80s at the French Commissariat à l'Energie Atomique et aux Energies Alternatives (CEA). They were operated in different workshops and laboratories, using different experimental conditions.In order to regain then extend the knowledge on SFR MOX treatment and recycling for 4th generation systems, new dissolution studies were initiated two years ago on an irradiated pin stemming from a PHENIX NESTOR-3 assembly (initial Pu content of 22.5%, burn up of 12.9at%) that was characterized by destructive and non-destructive Post Irradiation Examinations at the CEA-Cadarache.
As previous dissolution experiments were always carried out on a whole irradiated pin (including the lower axial column), the observed dissolution behaviour was always averaged for a given pin, then sometimes difficult to interpret and to correlate to another fuel. It was therefore decided to carry out innovative dissolutions studies on perfectly known separate sections of the same fissile pin to better understand its dissolution behaviour. Three dissolution experiments were thus carried out at the CEA-Marcoule on 30 mm long pieces of irradiated materials after shearing three distinct 120 mm chosen sections of a (U,Pu)O2 fissile NESTOR-3 pin (bottom, medium i.e. full-flux zone, upper).
Dissolutions were carried out in boiling nitric acid (8 M) for 6 hours to produce a feed solution concentrated at about 180 g/L of U+Pu. Pu dissolution yield exceeded 99,8 % but varied with the zone studied, as did the mass of undissolved residues which increases with the local burn up within the pin.
The irradiated cladding, made of stainless steel 15-15Ti, is prone to corrosion in boiling concentrated nitric acid. Partial dissolution of the main constituents (Fe, Ni and Cr) proved to increase along the fissile pin toward the zone where sodium is the hottest during irradiation.

Keywords: dissolution, fast reactor, plutonium, residue

Country/Int. Organization:

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4.2 Reprocessing and Partitioning / 520

First assessment of a digestion method applied to recover plutonium from refractory residues after dissolving spent SFR MOX fuel in nitric acid

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In the scope of fast reactor spent fuel recycling, head-end research and development are performed at the Atalante facility in Marcoule. Considering the initial plutonium content, quantitative plutonium recovery is one of the main objectives for the dissolution process. In addition, the quantities of undissolved residue increase with the burn up and can impact the waste conditioning downstream. A silver (II) oxidizing digestion step was studied to assess its application to the treatment of undissolved residues containing eventually some plutonium. This process was first optimised on dioxide plutonium powders, and then tested on irradiated LWR MOX fuel residues.

More recently it was applied to the solid residue obtained after dissolving in nitric media a MOX fuel irradiated in the Phenix Sodium Fast Reactor (SFR). In light of past experimentations and in order to obtain new basic data, not the whole spin was dissolved but only separate sections linked to known local burn-ups (BU). The first objective was to better correlate the quantity/composition of the dissolution residues with the local BU and Pu content of the initial irradiated material. Then the digestion step was applied on each dissolution residue obtained from each fuel pin part (bottom, medium i.e. full-flux zone, upper) with a view to evaluating the complementary Pu recovery, studying key parameters and characterising secondary residues.
The digestion step permitted to recover up to 99% of residual plutonium with some slight differences depending on the position of the pin part the dissolution residue was obtained from. Oxidation conditions, local burn-up and chemical composition were found to be influential. Quantity of residues after digestion was significantly reduced thanks to this digestion treatment. The ultimate residues consisted mainly of metallic compound like ruthenium, molybdenum, rhodium or palladium.

Keywords: recycling, residues, digestion, plutonium, waste.

Country/Int. Organization:

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1.2 SFR DESIGN & DEVELOPMENT - 2 / 522

Lessons and strategies from PFBR to Future Fast Breeder Reactors

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BHAVINI, a public sector unit under Department of Atomic Energy, is responsible for construction, commissioning and operation of fast reactors in India. Prototype Fast Breeder Reactor (PFBR) which is in advanced stage of commissioning is the forerunner for the second stage of India's three stage nuclear programme. PFBR is a 1250 MWt (500 MWe), MOX fuelled, sodium cooled, pool type fast reactor. It is a first of its kind reactor with total indigenous technology.
Starting from civil construction, manufacturing of over-dimensional & precision machined components, installation, integration, till commissioning and operation of all the mechanical, electrical and control & instrumentation systems, there were many challenges and surprises which have been addressed one by one in a systematic manner.
The experiences gained during various phases of PFBR project have enriched the scientists and technologists to fine tune the specific aspects in design, sizing of layout, manufacturing & transportation methodologies, sequence of installation and commissioning of the plant and equipment. It is clear that special attention is needed for achieving leak-tightness, making provisions for pre-service and in-service inspections, appropriate routing of power & instrumentation cables and protecting nuclear & process instrumentations. The project management for the future fast breeder reactors, twin units of 2 x 600 MWe will be well established based on the feedback from PFBR. Concept of twin units will be beneficial for both economy and time schedule.
The site assembly shop can cater the need for fabrication of individual components of reactor assembly meeting the stringent tolerance limits and appropriate integration, so that handling and erection of the assembly will be cost effective and time beneficial. Advance planning is required for achieving leak-tightness of integrated assemblies. The well planned sequence of layout of sodium and associated piping, their interfaces with the equipment, provision of redundant heaters, thermocouples, and leak detectors will play key role in project schedule.
This paper details out the experiences gained and lessons learnt from PFBR and the strategies to be adopted for future fast reactors towards safety, economy and time schedule.

Country/Int. Organization:

BHAVINI, Department of Atomic Energy, Kalpakkam 603102, Tamil Nadu, India

6.10 Other issues of code development and application / 523

Review of Transient Testing of Fast Reactor Fuels in the Transient REActor Test Facility (TREAT)

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The restart of the Transient REActor Test (TREAT) facility provides a unique opportunity to engage the fast reactor fuels community to reinitiate in-pile experimental safety studies. Historically, the TREAT facility played a critical role in characterizing the behavior of both metal and oxide fast reactor fuels under off-normal conditions, irradiating hundreds of fuel pins to support fast reactor fuel development programs. The resulting test data has provided validation for a multitude of fuel performance and severe accident analysis computer codes. This paper will provide a review of the historical database of TREAT experiments including experiment design, instrumentation, test objectives, and salient findings. Additionally, the paper will provide an introduction to the current and future experiment plans of the U.S. transient testing program at TREAT.

Country/Int. Organization:

USA/Idaho National Laboratory (US DOE)

5.2 Advanced Fast Reactor Fuel Development II / 525

New catalog on (U,Pu)O2 properties for fast reactors and first measurements on irradiated and non-irradiated fuels within the ESNII+ project

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In order to develop the fast neutron systems, three prototypes of the Sodium Fast Reactor, the Gas-cooled Fast Reactor and a heavy liquid metal cooled Accelerator Driven System are studied in Europe: ASTRID (SFR prototype), ALLEGRO (GFR prototype) and MYRRHA (LBE-cooled ADS system related to the ALFRED LFR-demonstrator). The ESNII+ project with its workpackage 7-FUEL SAFETY aims to provide a set of oxide fuel properties needed for the fuel element design of each prototype. The improvement of fuel properties will also reduce uncertainties in safety behavior evaluations, in nominal conditions as well as during transients and will be achieved by the update of the European catalog on the MOX fuel properties.
The uncertainties on the fuel properties have to be rigorously determined; the two main driver criteria for fuel element evaluation are the margin to melt for the fuel and the risk of clad failure.
Property measurements are done on existing fresh and irradiated fuel samples, identified to cover the fuel characteristics for ESNII prototypes. The review of the state-of-the art has shown that the knowledge on the thermal conductivity of irradiated FBR MOX is currently very limited. Only one publication is available providing surprising experimental results: no degradation of thermal conductivity with burn-up was observed. The data and models available in the literature were reviewed and new experimental results are obtained in order to develop an updated recommendation. Fresh and irradiated fast reactor fuel was characterised and its thermal diffusivity was measured. The irradiated fuel has an average burn-up of 13.4 at.% and the thermal diffusivity was measured in 3 radial positions: 0.6 mm, 1 mm and 1.4 mm from the pellet cladding. The thermal diffusivity increases from the pellet periphery to the pellet centre and is significantly higher than for LWR UO2 or MOX fuels with similar burn-up. The impact of the main mechanisms is investigated in depth: radiation damage concentration as a function of the irradiation parameters, effect of the plutonium content, of microstructure, of fission gas atoms, of fission products and O/M. A new correlation for the conductivity is developed on the basis of the phenomena specific to FBR fuel: high irradiation temperature, restructuring, extensive fission gas release, diffusion of plutonium and fission products.

Country/Int. Organization:

CEA, JRC, NRG, SCK, UJV, MTA, ENEA, AREVA, PSI

7.3 Non Proliferation Aspects of Fast Reactors / 526

The GIF Proliferation Resistance and Physical Protection (PR&PP) Evaluation Methodology: Status, Applications and Outlook

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Methodologies have been developed within the Generation IV International Forum (GIF) to support the assessment and improvement of system performance in the areas of sustainability, safety and reliability, economics, proliferation resistance and physical protection (PR&PP). The last of these four areas was assigned to the GIF Working Group on Proliferation Resistance and Physical Protection (PRPPWG).

The PRPPWG developed the methodology through a series of development and demonstration case studies, by use of a hypothetical "Example Sodium Fast Reactor"(ESFR). This is a generic design of Generation IV reactor based on the US Advanced Fast Reactor (AFR) developed by Argonne National Laboratory.

The PR&PP ESFR assessment was the first opportunity to exercise the full methodology on a complete system, and many insights were gained from the process. In particular, the approach of breaking the assessment into subtasks, each focusing on a separate area of PR&PP (for PR: diversion, misuse, breakout; for PP: theft and sabotage) handled by a dedicated subgroup with diverse international membership, was useful in generating new insights and concept development.

In addition, over the past few years various national and international groups have applied the methodology to inform nuclear energy system designs, as well as to support the development of approaches to advanced safeguards. A number of international workshops have also been held which have introduced the methodology to design groups and other stakeholders.

In this paper we summarize the PR&PP methodology, its application to the ESFR case study, and other accomplishments of the PRPPWG. Current challenges with the efficient implementation of the methodology are outlined, along with the path forward for increasing its accessibility to a broader stakeholder audience - including supporting the next generation of skilled professionals in the field of nuclear non-proliferation and security.

Country/Int. Organization:

Generation IV International Forum (GIF) Proliferation Resistance and Physical Protection Working Group (PRPPWG)

4.1 Fuel Cycle Overview / 527

Overview of the Nuclear Energy Agency Scientific Activities on Advanced Fuel Cycles

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Activities related to the fast reactors systems at the Nuclear Energy Agency (NEA) mainly focus on scientific research and technology development needs and are carried out within the Nuclear Science Division. In particular, projects related to the advanced nuclear systems and fuel cycles are carried out through the Working Party of Scientific Issues of the Fuel Cycle (WPFC) and its five related experts groups covering all scientific aspects of the fuel cycle from front to back-end. Ongoing projects on advanced systems include fuel cycle scenarios, fuels, materials, physics and chemical separations. Members of the expert groups cooperate to share recent research advancements at an international level and help identify gaps and needs in the field.
Current activities focus on nuclear systems in particular on the challenges associated with the adoption of new materials and fuels such as for example cladding materials, fuels containing minor actinides, or the use of liquid metal as coolants.
The expert group on Innovative Fuels is conducting joint and comparative studies to support the development of innovative fuels in particular minor actinide bearing fuels. Ongoing work involves a benchmark study on fuel performance codes and experiments
The new expert group on Liquid Metal Technology includes projects on liquid Na, lead or lead-bismuth) to support: (1) the development of construction codes used for design (design rules), (2) identify the key technical issues for licensing, (3) give recommendations for operation, inspection and handling.
The report on the effects of uncertainties of input parameters prepared by the expert group on Advanced Fuel Cycle Scenarios will be published soon. The purpose of this study was to identify sources of uncertainty and use sensitivity studies to assess their impacts on system level results. Members

of the expert group are currently working on a scenario study on transuranic management in order to assess the quantity materials contained in spent fuel that could be burnt using various fast burner fleets. In addition, a benchmark study on dose rate calculation for irradiated fuel assembly is being undertaken.
At the back-end of the fuel cycle, separation technologies (aqueous and pyrochemical) are being assessed by the Expert Group on Fuel recycling Chemistry and a state-of-the art report on minor actinide separation chemistry is being finalised.

Country/Int. Organization:

Nuclear Energy Agency

1.2 SFR DESIGN & DEVELOPMENT - 2 / 528

Status of ASTRID Nuclear Island Design and Future Trends

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In the frame of the French ASTRID project led by CEA, AREVA NP is in charge of the design studies of the whole Nuclear Island. The conceptual design was completed end of December 2015 with the issuance of a large amount of engineering files (few thousands.

The design of ASTRID intends to cope with GEN IV objectives regarding the new reactor concepts and includes ambitious performances to deal with concerning safety and availability for instance.
Thus, numerous technical challenges were faced and successfully managed by AREVA NP during the conceptual design phase dealing with:

- Deployment of design process based on System Engineering standards,
- Selection of adequate architectures and design justification (at the conceptual level stage) for the various main systems and components: primary circuit, secondary loops, decay heat removal systems, fuel handling and component handling systems, I&C platforms, electrical systems, general layout of nuclear island building etc.

Throughout the design progress, AREVA NP experimented new approaches in terms of management of innovations, advanced numerical simulations, management of large CAD models and the related interfaces with the other industrial partners, introduction of Virtual Reality tools to enhance the complexity mastering of the layout.

In addition, this paper describes the main technical achievements regarding the NI and main systems or component definition at the end of the conceptual design phase.
Future trends for the design of the NI are presented in terms of evolution of the technical configuration and enhancement of the engineering tools.

Country/Int. Organization:

FRANCE - AREVA NP

Poster Session 2 / 529

Advanced Coupling Methodology for Thermal-hydraulic calculations

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The purpose of this paper is to describe in a first part a coupling methodology between two codes in order to describe global thermalhydraulic behavior inside a sodium-cooled fast reactor. A CFD code (STAR-CCM+) is used for the modelling of primary circuit, while a system code (CATHARE) is used for the modeling of specifics area in primary circuit (core structures and primary pumps) and the modeling of secondary circuits.

The main advantage of this method is the computation of the whole primary loop while representing accurately complex 3D phenomena like thermal stratification onset or unsymmetrical thermal mixture in plena.

The second part of the paper presents thermalhydraulics results obtained with this coupling tool in case of two reactor design sizing transients:

- a station blackout transient in which primary and secondary circuits are in natural circulation.
- a loss of one secondary loop in which there are thermal mixed phenomena in primary pools.

Country/Int. Organization:

FRANCE - AREVA NP

Poster Session 1 / 530

CFD Simulation of Corium / Materials Interaction for Severe Accidents

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In case of a severe accident inside a sodium-cooled fast reactor, corium interacts with many materials along its way towards the core catcher. Once deposited, corium can also interact with protective and / or sacrificial material. Among those materials, refractory ceramics like zirconia are credible candidates due to their high fusion temperature. Through CFD methodology, calculations have been made in order to reproduce the fusion mechanism and kinetics of the UO2-ZrO2 system, in order to reproduce ablation phenomena of ZrO2 by UO2. This process is not only thermal but also chemical, as eutectic material formation is expected, around 2500℃. That is why the eutectic diagram of the UO2-ZrO2 system has been linearized and put directly inside the CFD software in order to take into account the formation of this eutectic material. Comparisons have been made with experimental data: a layer of UO2 is deposited inside a cooled zirconia crucible. Results show good correspondence between calculated and experimental data: the onset and effective melting of the zirconia is

modelled, but also chemical saturation processes are identified, explaining the inhibition of the melting after a certain time.
A practical application of this development has been made in the frame of the AREVA research program on sodium reactor, demonstrating that in the case of a jet of corium flowing down on an internal core catcher, the shape of the molten sacrificial material enables the apparition of a so-called "pool effect"being very favorable with respect to the local ablation of the core catcher.
Finally, another application of this methodology could be in the frame of In-Vessel Retention study. A CFD simulation is made modelling the progression of the melting front inside the thickness of the bottom of a steel vessel due to the presence of molten UO2 emitting residual thermal power. Results show the formation of a floating steel layer at the surface of the UO2 molten pool, and the consecutive focusing effect occurring on the solid remaining parts of the steel vessel.
All these calculations show that some of the complex thermo-chemical phenomena occurring during a severe accident can be modelled and used in order to give better understanding of the main phenomena.

Country/Int. Organization:

FRANCE - AREVA NP

1.8 INNOVATIVE REACTOR DESIGNS / 531

Innovative TRU Burning Fast Reactor Cycle Using Uranium-free TRU Metal Fuel - Core Design Progress -

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Fast reactors have a capability to effectively burn TRU (transuranic) compared to LWR due to its higher fission-to-capture ratio of TRU and to reduce the burden of radioactive waste disposal. The most effective way to burn TRU is to use uranium-free TRU fuel since it does not produce any new TRU. In order to clarify the feasibility of uranium-free TRU burning fast reactor cycle with metal fuel, we have been investigating the related key technical issues not only on fuel cycle area but also reactor area since October, 2014 under the contract with Ministry of Education, Culture, Sports, Science and Technology (MEXT) in Japan.
In this paper, among the various investigation items in this study, the progress of the core design study will be presented, which will shows the promising core to simultaneously achieve enhanced Doppler feedback and low sodium void reactivity for the uranium-free TRU metal fuel fast reactor, considering the evaluated fundamental properties of the fuel and secured core safety.

Country/Int. Organization:

Japan, Toshiba Corporation

Poster Session 2 / 532

Methodical uncertainty of criticality precise calculations for fast lead reactor

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Criticality calculations for BFS-1 test facility with lead were performed using Monte-Carlo code MCU-BR to verify some evaluated neutron data files for fast spectra. These data files are RUSFOND, ENDF/B-VII.1, JEFF-3.2, JENDL-4.0, CENDL-3.1 and some combined data. The continuous energy treatment (ACE format) was used. Critical assemblies include the pellets consisted from fissionable materials, lead, stainless still and other. The average Keff evaluation for each critical assembly was obtained. Standard deviation for Keff at various data files is in interval 0,1% - 0,4% with probability of 0,55 - 0,82, for average Keff evaluation standard deviation is 0,14% with probability of 0,73.

Country/Int. Organization:

JOINT-STOCK COMPANY «N.A. DOLLEZHAL RESEARCH AND DEVELOPMENT INSTITUTE OF POWER ENGINEERING»

8.1 Professional Development and Knowledge Management - I / 533

Topical issues of training of specialists for fast nuclear power engineering and the closed nuclear fuel cycle

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Nuclear education and training for innovative projects are regarded. National Research Nuclear University (MEPI) history is shown as an example of the university, which has great experience in nuclear education and training, including innovative fast reactor projects. The activity classification and steps of technical specialist training for fast nuclear power engineering are presented. The efficiency of various educational technologies, implemented by MEPhI, is discussed. The report emphasizes the role of student teaching and research work and practice in the formation of specialists. The use of professional databases and international projects in the field of fast reactor technology is discussed separately. Stages of formation of the department of "Technology of closed nuclear fuel cycle", organized in MEPhI to carry out targeted training for the "Breakthrough" project, are described.

Country/Int. Organization:

Russian Federation

6.8 Experimental Facilities / 534

The DRESDYN project: A new facility for thermohydraulic studies with liquid sodium

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For the safe operation of SFRs there is a growing need for small and medium sized liquid metal experiments to study various thermohydraulic and safety aspects, comprising effects such as flow metering, local velocity measurements, gas bubble entrainment, and early gas bubble detection. We give a short description of the new large-scale infrastructure DRESDYN (DREsden Sodium facility for DYNamo and thermo-hydraulic studies) at HZDR. For the liquid sodium installations in the framework of DRESDYN a new experimental hall with an area of approximately 500 m2 became available in 2016. The total inventory of sodium will be 12 m3. The development of flow measurement techniques has a long tradition at HZDR that will be delineated in the talk. It covers contactless flow-rate sensors, local velocity measurements such as the Ultrasound Doppler Velocimetry (UDV) or the Contactless Inductive Flow Tomography (CIFT), as well as X-ray visualizations of liquid metal two-phase flows.

Country/Int. Organization:

Germany

6.7 Experimental Thermal Hydraulics / **535**

Eddy current flowrate and local ultrasonic velocity measurements in liquid sodium

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For the safe operation of sodium cooling systems a monitoring of the flow field is often desirable. We report first on the development of a new eddy current flowmeter (ECFM) and related tests in sodium. The objective of this sensor is its positioning above the fuel subassemblies and the detection of possible blockages of the sodium flow through the multitude of subassemblies. The sensor consists of a number of coils a part of which is fed by an excitation AC current. The assembly of coils is placed in a thimble and the measured flowrate is proportional to the integral flow around this thimble. In the second part we report on the preparation of local ultrasonic velocity measurements. Here the objective is to study the flow field resulting from a powerful electromagnetic pump installed at the PEMDYN facility of CEA in Cadarache (France). As for the ECFM, related test measurements were performed at the sodium facility NATAN of HZDR.

Country/Int. Organization:

Germany and France

7.2 Economics of Fast Reactors / **536**

Equipment cost estimation for pilot demonstration lead-cooled fast-neutron reactor BREST-OD-300

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One of the main problems in determining the investment in the construction of new nuclear facilities at the design stage is the cost estimation of non-standard equipment. The novelty and lack of experience are the features of early stage of the project. The uniqueness of the projected facilities makes it impossible to use catalogues and price-lists. It is required different approaches to estimate the cost of such equipment.
The methods of assessing the cost of projected non-standard reactor equipment can be divided into two main groups: analogy and resource or engineering build-up methods. Additional cost estimating methods include parametric methods and corrective amendments.
Methods and approaches of non-standard equipment cost estimation are differ from stage to stage, allowing to obtain a more accurate result. It is necessary to take into account features of the application of a method according to the stage of design work.
The essence of different approaches of equipment cost estimation is disclosed, advantages and disadvantages of different methods are analyzed, guidance on the applicability depending on the specific conditions of evaluation is provided.
The comparison of cost indexes is made.
Economies of scale and learning curve must be taken into account.
Rosatom Production System (RPS) is proposed as one of additional equipment cost estimation method.
Expected accuracy can be determined by cost estimate classification matrix according to Association for the Advancement of Cost Engineering International (AACE International) recommended practice.
All available methods and tools must be used to estimate cost of non-standard equipment at the design stage. Obtaining close results by means of different methods indicates reliable estimates of equipment cost.
Iterative approach to the assessment of the main BREST-OD-300 reactor equipment, based on decomposition and structural analysis, is presented taking into account the economic characteristics of potential manufacturing plants
It is important to develop and improve cost estimation methods taking into account best practices to achieve economic competitiveness.

Country/Int. Organization:

Russian Federation, ROSATOM, JSC "NIKIET"

Poster Session 1 / 537

The Computer model for the economic assessment of NPP pilot demonstration energy complex with BREST-OD-300 reactor (REM Proryv Project)

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The computer model for the economic assessment of Proryv Project (REM) was developed as a result of work in the field of simulation of technical and economic indicators.

REM is designed for supporting managerial decision-making during analyze of building NPP with thermal and fast reactors.
The targets of REM are noted, functional opportunities and technical demands are demonstrated.
Object-oriented programming and modeling was implemented to construct economic and material balances. It allows assembling required power utility system from various functional modules («Energy unit» «Fabrication/refabrication module», «Reprocessing module», «Radioactive waste management module»), depending on the unit type and the nuclear fuel cycle (NFC) technology.
Individual technical and economic indicators are set for each object, project economic efficiency (commercial, fiscal, social) is determined at different price level (forecast, current, deflated).
Indicators are integrated at the upper level for total energy complex economic efficiency. The upper level integral indicator is Levelized cost of energy (LCOE). It determines the total project competitiveness.
The examples of calculations of technical and economic indicators are illustrated, simulation data and optimization of Proryv project are reported.
REM functions allow to improve the quality of economic evaluations, reduce time and labor input of calculations, simplify work with large number of variable factors, increase visibility of the calculations and results.

Country/Int. Organization:

Russian Federation, ROSATOM, JSC "NIKIET"

8.2 Professional Development and Knowledge Management - II / 538

PERSONNAL TRAINING FOR THE "PRORYV" PROJECT AT THE SEVERSK TECHNOLOGICAL INSTITUTE OF NRNU MEPHI

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In Seversk on the base of Siberian Group of Chemical Enterprises (SGCE) the pilot project "PRORYV" on creation of nuclear power technologies of new generation based on the closed nuclear fuel cycle with the use of the fast neutron reactors is being implemented. It is planned to create a research and demonstration power complex (RDPC) including the BREST-OD-300 reactor, and modules of fabrication and refabrication of nuclear fuel. Implementation of such large-scale project at SGCE means the necessity of creating an effective system of the staff training and retraining.
At Seversk technological institute of National Research Nuclear University MEPhI (NRNU MEPhI) the complex personnel training system for the "PRORYV" project is going ahead of the construction and start-up of RDPC. The system includes various educational levels: middle school, higher education (bachelor and specialist programmes, magistracy, postgraduate course) and advanced training and retraining of the staff. In these educational activities highly-qualified scientific and pedagogical employees of NRNU MEPhI and employees of the Russian Federation national nuclear corporation ROSATOM with wide experience in the solution of innovative and technological tasks of nuclear industry are being involved. The educational process is organized on the basis of the modern interactive and multimedia training technologies using the modelling of key technological procedures of the closed fuel cycle. Joint research and educational center on the fast energetics is being created for effective interaction providing of NRNU MEPhI and the SGCE, integration of education, science and production.

Country/Int. Organization:

Russian Federation

1.5 LFR DESIGN & DEVELOPMENT / 539

BREST-OD-300 REACTOR FACILITY. DEVELOPMENT STAGES AND JUSTIFICATION

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BREST-OD-300, an innovative natural-safe fast reactor, is being developed as a pilot and demonstration prototype for the base commercial reactor facilities of future nuclear power with a closed nuclear cycle.
Coolant in the reactor facility is lead, the layout of the primary circuit is integral, the reactor vessel material is multilayer metal concrete. The reactor core design uses mixed uranium-plutonium nitride as the fuel, and the fuel elements are contained in shroudless fuel assemblies (FA). Small reactivity margin, excluding prompt-neutron runaway is provided in the core.
Decisions are based on a computational and experimental justification. To confirm the fuel serviceability, radiation tests of fuel elements are conducted in fast reactors. Full-scale fuel-free mockups of FA are tested. Tests have been conducted of the vessel elements. Experiments have confirmed the absence of a dependent break of steam generator tubes. Neutronic codes have been verified, including with the use of BFS critical assemblies. Loop facilities have been built on which studies are conducted to determine the radionuclide release from the coolant.
It has been shown based on the calculation results that the probability of the core damage (without core melting) for nuclear power plants with the BREST-OD-300 reactor facility does not exceed 8.65•10-9 1/year.

Country/Int. Organization:

JSC NIKIET, Moscow, Russia

Poster Session 1 / 541

Probabilistic Safety Analysis of NPP with BREST-OD-300 reactor

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As part of the PSA level 1 with BREST-OD-300 reactor the following main tasks have been solved:
- selection and grouping of initiating events. List of IE was formed as a result of the analysis (including specific local impact);
- success criteria (SC) for safety systems and functions. In the frame of this task several deterministic calculations were performed;
- analysis of accident sequences. For each groups of IE models of accident sequences were developed and event trees were constructed. Developed event trees for all groups of IE were included into the logical-probabilistic model of the unit to estimate the probability of the final states;
- analysis of systems. In the frame of this task logical-probabilistic models (fault trees) have been developed for the following systems:
a) the system of emergency cooling of reactor (SECR);
b) the steam generator leak localization system (SGLLS);
c) the integrated system of reactor control and shutdown (ISRCS);
d) the control system of technological process (CSTP);
e) the system of reactor normal cooling (SNC);

f) the gas system of reactor unit (GSRU);
g) the electrical power system;
h) the system of steam generator protection;
i) and others;
- data analysis. Evaluation of the frequency of IE groups was based on the generalized statistical data processing from different NPPs and research reactors;
- analysis of PSA results. The resulting integral risk value of severe core damage (category A) is 9,01•10-9 1 / year. It meets the target values of NP-001-15 (Regulatory requirements) for the total yearly probability of severe accidents 10-5, and also for the total probability of a large accidental release 10-7.

Country/Int. Organization:

JSC NIKIET, MOSCOW/RUSSIA

Poster Session 1 / 542

Corrosion behavior of tube steel for BREST-OD-300 steam generator

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Abstract

Once-through type steam generator (SG) is designed for BREST-OD-300 reactor unit. There is the steam overheating zone in the upper part of long-length heat exchange tubes. Consequently, under service conditions SG tubes to be exposed to three corrosion media –liquid lead (primary coolant), pressurized water, and superheated steel at temperatures in the range 350-505 ℃, pressure of 17 MPa.
To confirm appropriate corrosion resistance SG tubes material in NIKIET, CRIFM, CRISM Prometey research work has been conducting. Required corrosion properties of SG tube material are obtained due to usage of specially alloyed austenitic stainless steel (SS).
This paper presents the results of corrosion and corrosion-mechanical behavior studies, and material investigation results. Type 321 SS is used for comparison.
Use of modified SS as the material of SG tubes allows obtaining required corrosion properties for established SG service-life.

Keywords: austenitic stainless steel, corrosion, corrosion-mechanical properties, liquid lead, water coolant.

Country/Int. Organization:

N.A. Dollezhal Research and Development Institute of Power Engineering
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5.7 Chemistry Related Technology / 543

Development of steam-water cycle chemistry for steam generator of research reactor MBIR

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The compatibility of water chemistry with structural materials of steam-water cycle is most important condition for long-term operational reliability of steam generators. Limiting service life factor of the steam generators is usually corrosion damages of heat exchanger tubes from the steam-water side. In view of this the mostly proven in practice reference design and engineering solutions were used to ensure efficient, reliable and safe operation of the multipurpose fast research reactor MBIR under development in Russia by NIIAR (as an operator), NIKIET (as a chief designer), etc.
The reverse type steam generators for MBIR were developed by JSC "TASMO" and the Czech company ENERGOVÝZKUMO. The main design feature of reverse type steam generators is sodium coolant circulation within the tube bundle while the steam-water medium is going through the shell side. The design of these steam generators should ensure their reliable and safe operation during 200 thousand hours of service life. The above steam generators will provide both heat removal from the secondary sodium coolant loop to steam-water environment of the third coolant circuit and the generation of superheated steam for steam turbine.
The cutting research from reference steam generators has shown that corrosion damages of heat exchanger tubes were initiated and developed from the steam-water side under this environment. The main goal of water chemistry is the formation of the protective oxide films on heat exchanger tube surfaces which provides their low corrosion rate thanks to limitation of corrosion-active impurities in the steam-water environment (copper, chloride ion, sulfate ion, etc.). The main local corrosion mechanism of these tubes is crevice and pitting.
Neutral water chemistry was proposed for the steam generator of MBIR reactor due to a number of advantages over alternative options:
- Simple chemistry control and monitoring due to absence of any chemical reagent dosing into feed watert;
- Reduction of capital costs and the amount of waste due to absence of dosing system;
- The absence of hydrazine and ammonia dosing eliminates both toxicological hazards for personnel and ballast exchange capacity of ion exchangers in condensate polisher system
- Elimination of deposits from steam generator surfaces during operational transients.
Thus the proposed neutral chemistry mode for steam-water cycle of research reactor MBIR will reduce the capital and operating costs, improve environmental performance and ensure high reliability and design life of the steam generators.

Country/Int. Organization:

N.A. Dollezhal Research and Development Institute of Power Engineering,
Moscow, Russia

544

A new generation steel for heat exchangers tubes of reactors design with lead coolant

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Abstract

This paper deals with the concept of improvement of a new generation steel grade (ЭП302-Ш), which have better resistance to local corrosion in chloride-containing environments and maintaining high corrosion resistance in the flow of liquid lead. The main attention was paid to improve resistance to local damage like pitting corrosion, stress corrosion cracking and crevice corrosion. In order to improve the stability of austenite during thermal exposures and to maintain the high corrosion resistance both in lead flow and in chlorinated water, the main alloying elements including chromium and nickel were adjusted.
As a result of laboratory experiments that confirm high level of corrosion resistance, the newly developed steel ЭП302М-Ш grade were recommended to use it in heat exchange tubes of pilot project of nuclear reactor BREST-OD-300.

Country/Int. Organization:

N.A. Dollezhal Research and Development Institute of Power Engineering, Moscow, Russia

6.8 Experimental Facilities / 546

On the rational design of fuel assemblies for reactor facilities from the standpoint of providing vibration strength

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This report deals with a computational and experimental justification for the search of the optimal spacing for the axial fuel assembly (FA) spacer grid (SG) arrangement in reactor facilities subject to vibration resistance requirements. Practically, as is known, the spacing selection is postulated at the present time based on the earlier selected FA dimensions for both effective and decommissioned reactor facilities. However, in our opinion, this selection can be based on results of a computational and experimental justification. The major guidelines for the development of general rules for the rational FA design to ensure reliable operation and vibration resistance of FAs in conditions of impacts from induced hydroelastic vibrations in the axial coolant flow have been formulated with regard for the physical laws of interactions between the flow and elastic structures. A finite element FA model has been built, natural frequencies and modes of bending vibrations have been determined, mathematical relations have been defined to estimate variations in the amplitudes of bending vibrations, and major criteria have been formulated for ensuring the FA vibration resistance. The paper also presents experimentally obtained spectra of the standard multipin FA pressure fluctuations and vibrations in a hydraulic test bench in which water is used as the test environment. Specific recommendations have been developed for the axial FA spacer grid spacing selection.

Country/Int. Organization:

Joint Stock Company N.A. Dollezhal Research and Development Institute of Power Engineering, Moscow, Russia

Poster Session 2 / 547

THE STUDY OF THERMAL-HYDRAULIC PROCESSES IN THE STEAM GENERATOR OF THE BREST-OD-300 REACTOR FACILITY

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One of the most important elements of the BREST-OD-300 reactor facility is a steam generator (SG), which is a vertical heat exchanger with twisted pipes, immersed in liquid lead. To justify heat-hydraulic performance of SG and reliability of circulation was conducted complex of computational and experimental works.
Computational research were conducted with the help of numerical model of SG developed on the basis Relap5 heat-hydraulic code. Adequacy of modeling heat-hydraulic processes of numerical model SG has been confirmed by experiments conducted in JSC "IPPE» and JSC «NIKIET".
After verification of the numerical model was made computational analysis of heat-hydraulic stability SG, simulated work SG in stationary modes with different levels of power, in start-up conditions and emergency stop, including mods after breaking heat exchange tubes.

Country/Int. Organization:

JSC NIKIET, Moscow/Russia

6.1 CFD and 3D Modeling / 548

Numerical simulation of hydraulics and heat transfer in the BREST-OD-300 LFR fuel assembly

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This paper presents an analysis of hydraulic and heat transfer phenomena in a lead coolant flow in a fuel assembly (FA) of the BREST-OD-300 reactor core central subzone based on CFD simulations (RANS) using the STAR-CCM+ code. Based on the simulation results, coolant pressure, velocity and temperature fields, and the fuel cladding temperature distribution have been obtained. The FA design comprises a pin bundle (p/d = 1.38) with spacer grids, and the outer shroud has been eliminated. The influence of the FA design features on the hydraulic characteristics and the cladding temperature distribution has been shown.
The CFD model was validated on experimental data on the hydraulic characteristics of a full-scale FA model. A good agreement has been shown between calculated and experimental data on the pressure drop both for the FA parts (head, pin bundle and tail) and for the FA as the whole.

Country/Int. Organization:

Russia / JSC N.A. Dollezhal Research and Development Institute of Power Engineering (JSC "NIKIET")

3.5 General Safety Approach / 549

DETERMINISTIC SAFETY ANALYSIS OF REACTOR BREST-OD-300

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As part of the power unit with reactor BREST-OD-300 project justification, deterministic safety analysis with imposition of postulated single failure of systems and equipment, or human error on the initiating event (IE) was held in JSC "NIKIET". Work was done for the version of the core with jacketless TVS. The calculations were performed using the verified dynamically bound neutron-physical and thermal-hydraulic software package DINAR.
Results of the safety analysis of the power unit with reactor BREST-OD-300 are presented in the paper for up to four OE violations in normal operation (VNO), one from each group of the internal effects listed below. Selected VNO initiating events, accompanied by the greatest disturbance and deeper relative to the nominal power deviations of parameters important to safety. Initial events of violations in normal operation were considered including the following groups of effects:
- Initiating events that lead to the unauthorized introduction of positive reactivity;
- Initiating events that lead to the disruption of the heat sink from the core;
- The deterioration of heat removal by second circuit;
- Excess heat removal by second circuit.
Scenarios of IE with the imposition of a single failure of the safety systems, or human error take into account the requirements of the Russian nuclear power industry standards and regulations, according to which the power plant security must be provided in any of the initial event carried in project with imposition of one independent from the initiating event failure of any of the following safety systems: an active element or a passive element having mechanical moving parts, or a passive element with no moving parts, having a probability of failure of safety functions 10 -3 or more, or one independent from the initiating event of personnel mistakes in accordance with the principle of single failure. In addition to the one failure of the one of the elements listed above and independent from the initiating event, all failures resulting from this single failure or initial event, as well as non-detectable failures in the operation of the AC elements affecting the development of the accident were taken into account.
As the security criteria of the reactor facility in violation of the normal operation the exceedance of the established design limits of the power unit parameters were taken.

Country/Int. Organization:

JSC NIKIET, Moscow/Russia

Poster Session 2 / 551

Application of CFD simulation to validate the BREST-OD-300 primary circuit design

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Information has been obtained on the flow distribution along parallel paths, on the situation of the free levels, on the heat and mass transfer processes. Experimental determination of the hydraulic parameters is only possible for the components of the core by using full-scale mock-ups. The remaining elements of the circulation loop require calculation justification supported by experiments on fragmentary models. The three-dimensional flow in the first loop of the reactor requires using of computational fluid dynamics methods (CFD).
The computational model includes the porous domains (the core and the steam

generator modules) and the area for which the RANS simulation is performed. The hydraulic parameters of the porous domains are defined on the basis of experiments (for the components of the core) and the prior CFD calculations (for the SG modules). When creating the computational model the experience of the CFD code verification from the point of view of the modeling of the liquid metal coolants flow was taken into account.
The main design parameters of the reactor circulation loop are confirmed. The information on the spatial distribution of the thermal parameters of the coolant is useful for clarifying of the stress-strain state of the structural elements, and in the formulation of the requirements for the placement of the reactor control system sensors.

Country/Int. Organization:

JSC "NIKIET", Moscow, Russia

Poster Session 2 / 552

LES-SIMULATION OF HEAT TRANSFER IN A TURBULENT PIPE

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Study of turbulent pipe flows is extremely important because of its wide range of applications. In the past decades, many fundamental theoretical and experimental studies on wall-bounded flows have been performed: in the pipe, flat channel and boundary layer flow geometries. However, the internal fluid dynamics in these regions still far from being understood. Numerical simulation offers an opportunity to get detailed information on the flow structure, which is difficult to obtain experimentally.
In this paper, the numerical simulation of turbulent heat transfer in a circular pipe was performed in a wide range of Reynolds numbers using nonparametric MILES-method CABARET on grids with an incomplete resolution of the turbulence spectrum, as well as with the use of the STAR-CCM+ code in a LES-approximation. The calculation results was compared with the DNS calculations by other authors found in literature, as well as with the RANScalculations performed in the STAR-CCM+ code. The simulation showed a satisfactory accuracy in determining average, rms and integral characteristics of the flow, and revealed drawbacks in the existing model relations describing the local properties of turbulence. The authors have proposed a thermal wall function, which might be implement in the RANS-approximations.

Country/Int. Organization:

JSC «NIKIET», Moscow, Russia
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2.1 Commissioning and Operating Experience of Fast Reactors I / 553

USSR and Russian fast reactor operation through the example of the BN600 reactor operating experience and peculiarities of the new generation BN800 reactor power unit commissioning

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The fast-neutron nuclear power industry development began with the BR-5/10 experimental reactors (1959) followed by BOR-60 (1969).

The power reactor evolution started with the BN350 commissioning in 1973. In 1980 the BN600 operating up to now was put into operation. In 2015 the BN800 obtained the first criticality.

BN600 fast reactor power unit No. 3 of 600 MW power has been put into operation in April 1980 and is under day-to-day operating conditions. Over the operating period the advantages of sodium-cooled fast reactor facility were highly appreciated. The complex tasks were also solved to improve safety and cost-effectiveness of the BN600 reactor facility.

Since the commissioning the BN600 reactor facility core has been upgraded three times and the main equipment lifetime has been significantly increased. The work has been carried out to extend the lifetime until 2020, as part of which it has been shown that the strength conditions in all the critical reactor components are not infringed for 45 years of operation.

After the events at Japanese Fukushima NPP the action plan aimed at greater resistance of the BN600 reactor facility against external impacts was put in practice.

Over the operating period the following were carried out at the BN600 reactor:

1. About 500 experimental fuel sub-assemblies were tested to study structural materials and designs of different types, which in particular allowed the fuel burn-up to be dramatically increased.
2. The technologies of repairs and replacement of the large reactor and steam generator components (72 heat exchangers of the steam generators, 3 low pressure cylinders, 6 feedwater pumps, 3 emergency feedwater pumps) were mastered.
3. The experience of the production of the high-specific activity isotopes was accumulated.
4. The long life tests of the large components operating in sodium were carried out.

The most important outcome of the operation is a justification of the construction of the new fast-neutron reactor power units (BN800, BN1200).

For 36 years of the safe and reliable operation the main task was fulfilled, i.e. the operation of the powerful unit with the sodium cooled fast reactor and sodium steam generators was mastered.

The BN800 was designed using inherent safety principles and applying an additional reactor shutdown system based on the passive operating principle.

Country/Int. Organization:

Russia, Beloyarsk NPP

3.6 Safety Analysis / 557

Autonomous Reactivity Control

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Co-authors: Allen E Dubberley [2]; Carl Hellesen [3]; Christopher Keckler [4]; Ehud Greenspan [4]; Malwina Gradecka [4]; Roman Thiele [5]; Thomas Fanning [6]

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The Autonomous Reactivity Control (ARC) system was developed to ensure inherent safety of fast reactors while having a minimal impact on reactor performance and economic viability. The ARC system is a modification to a standard fast reactor fuel assembly, in which two liquid-filled reservoirs, one above and one below the core, are connected by a tube which replaces one of the fuel rods in the assembly. The system has a near-negligible impact on core operation and performance during standard conditions, but will act to passively introduce negative reactivity if temperatures rise above a pre-determined set point. Properly designed, the ARC-system can act as a thermostat in the core, autonomously controlling temperature without the need for any operator action, electrical systems or any moving mechanical parts. This actuation responds to temperature and relies solely on the laws of physics, and is therefore an inherent feedback mechanism. The ARC system is in active development at the University of California Berkeley & Argonne National Laboratory in the US and at Uppsala University in Sweden. This paper summarizes the state-of-the-art of these development efforts of the system itself as well as the results of full transient analysis of ARC-system equipped fast reactor cores.

Country/Int. Organization:

Uppsala University, Sweden;
University of California Berkeley, United States;
Argonne National Laboratory, United states

2.3 Decommissioning of Fast Reactors and Radioactive Waste Management / 560

Superphenix dismantling - Status and lessons learned

Authors: David Settimo[1]; Gérard PRELE[2]; Jean Claude Rauber[1]; Michel VELON[1]; Veronique BOUILLY[1]; joel guidez[3]

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Following Superphenix definitive shutdown announcement in 1997, enacted by a decree in late 1998, it was rapidly decided to start the reactor dismantling, for technical reasons (keeping in liquid form large amounts of sodium) and human resources availability.

Out of the 19 sodium fast reactors having been operated worldwide, 13 are dismantled or being dismantled. So there is already a great experience in this domain. Superphenix, taking over some of this methodology processes, got in 2006 the decree authorizing its definitive shutdown and dismantling, which allowed it to start sodium treatment and nuclear dismantling and to have completed by 2015 the entire sodium destruction.

Dismantling of a fast reactor presents specificities related to this presence of sodium and to the necessity to eliminate it, before being able to undertake the usual dismantling procedures. The procedures used to eliminate this sodium in the Superphenix primary vessel, are explained in this paper.

Explanations are given on the last events of these dismantling operations as the use of a dedicated robot to cut internal structures of the primary vessel where residual sodium was accumulated.

This Superphenix experience validates a methodology but shows that there are some remaining points needing further developments for complete elimination (oxidized NaK, oxidized sodium aerosols or cold traps).

Moreover, this experience enables to propose recommendations in terms of future reactor design, aiming at make their dismantling easier.

Country/Int. Organization:

FRANCE CEA EDF

6.6 Coupled Calculations / 561

Codes of New Generation Developed for Breakthrough Project

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In frame of Breakthrough project the system of codes of a new generation is being developing for justification of operational characteristics and safety of the NPP with sodium, lead and lead-bismuth coolants utilizing closed fuel cycle.
Use of the code system by design and industry organizations ensure reduction of uncertainties by the use of precise multi-scale models and more detailed geometry modeling and self consistent integrated analyses of nuclear reactors and fuel cycles.
Simulated objects include sodium (BN-600, BN-800, BN-1200, MBIR), lead (BREST-OD-300, BR-1200) and lead-bismuth (SVBR-100) reactors, NPP compartments and also objects of closed fuel cycle.
While building the system of new generation codes the attention is paid not only to the level of physical models but also to the realization of the modern technology of code development including effective use of high performance computation systems, contemporary code architectures and numeric schemes, etc.
Code system can be used during all stages of project development starting with the substantiation of design solutions up to probabilistic safety analyses of different levels, and building of training systems. To conduct PSA-1 the CRISS 5.3 code has been developed. For safety assessments including PSA-2 of NPP with fast reactors the EUCLID code has been developed as integrated performance and safety code for design justifications and safety analyses. It allows to model the behavior of different coolants and type of fuels.
RANS and LES CFD code LOGOS, and LES and DNS CONV-3D code, neutron-physical codes based on Monte-Carlo method MCU-FR and finite elements and discrete ordinates methods ODETTA have been developed for justification of designs. The BERKUT multi-scale code has been developed for mechanistic modelling of thermomechanical and physical-chemical behavior of oxide and nitride single fuel rod.
For PSA - 3 the system of codes for the spreading of radionuclides is realized consisting of several modules for the contamination areas in the regional scale (ROM) and site (ROUZ), in the water system (SIBILLA) and geological soils (GeRa).
The optimization of the technological processes of the closed fuel cycle (SNF reprocessing, nuclear fuel refabrication and radioactive waste management, including final disposal) the code VIZART is used.
In the presentation the short characteristics, status of development and verification and plans for future of the developing code system is presented.

Country/Int. Organization:

Russian Federation

5.7 Chemistry Related Technology / 562

Materials corrosion in Fast Reactor environment

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The design objectives of Fast Reactors (FR) are mostly related to safety, sustainability and economics; waste minimisation is also considered. To meet these objectives, the selection of structural and fuel clad materials is a crucial issue to be considered. Indeed, the materials in FR environment can be subject to intense mechanical stresses, a high energetic neutron field and circulating heat removal fluids [1]. The fluids considered to remove the heat from fast reactor cores should fulfil a number of properties. Among the possible coolants selected for FR, within the Generation IV initiative, there are either liquid metals (Sodium, Lead and Lead-Bismuth) or gas (Helium).
The focus of this overview is to address the importance that parameters, such as temperature, flow velocity, impurities etc., have on the materials corrosion in the liquid metals [2]. From these considerations, a summary is given on the available knowledge on corrosion mechanism and rates of austenitic and ferritic/martensitic steels (and their ODS variants).
The corrosion mechanism observed in liquid Na and in liquid Pb/Pb-Bi is driven by the oxygen concentration in the liquid metal. However, different corrosion products are observed in the two liquid metals.
The corrosion rate is also affected by the temperature gradient and the flow velocity of the liquid metal. Equations to calculate corrosion rates of steels in liquid metals have been defined. However, they all have an empirical character and need extensive experimental validation.
In addition, potential degradation of the mechanical properties of reference materials in contact with the liquid metal is also discussed. Due to the potential severity of these phenomena, independently from the intensity of their corrosion rates, they have an important impact on the materials selection.
The liquid metals corrosion data available for steels are then compared with typical design parameters and requirements of key components for both Sodium Fast Reactors (SFR) and Lead/Lead-Bismuth Fast Reactors (LFR) in order to discuss potentialities and challenges in FR environments.
Finally, this overview discusses corrosion protection methods investigated so far and on potential future needs to reach the design objectives stated.

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[2] C. Fazio, F. Balbaud. Chapter 2 in Structural Materials for Generation IV Nuclear Reactors. Edited by P. Yvon. Woodhead Publishing Series in Energy Nr. 106. Elsevier, 2016.

Country/Int. Organization:

European Commission Joint Research Centre

Poster Session 1 / 563

Key features of design, manufacturing and implementation of laboratory and industrial equipment for Mixed Uranium –Plu-

tonium Oxide (MOX) and Nitride fuel pellets fabrication in Russia

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The presentation describes the author's experience in design, manufacturing and installation of equipment used in Mixed Uranium –Plutonium fuel pellet fabrication in Russia. The key features of mixed uranium-plutonium oxide and nitride powders are described, as well as their influence on main process (furnaces, presses) and auxiliary (gloveboxes) equipment design. Technical solutions for working with low fluidity powders, automatic dimensional and weight control, automatic readjustment of the manufacturing parameters, automatic powder gathering are discussed. Conveyance of boats prone to deformation and gas separation systems, insulation material choice are described, as well as rules and regulations applicable for this kind of equipment.

Country/Int. Organization:

Russia, France

Poster Session 1 / 564

Feasibility of MA Transmutation by (MA, Zr)Hx in Radial Blanket Region of Fast Reactor and Plan of Technology Development

Author: Kazumi IKEDA[1]

Co-authors: Fumiki MIZUSAKO [2]; Hiroaki MUTA [3]; Kazuo IKEDA [4]; Ken KUROSAKI [3]; Kenji KONASHI [5]; Kunihiro Ito [4]; Mutsumi HIRAI [2]

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This paper shows a feasibility study of transmuting minor-actinide (MA) by MA-zirconium hydride, (MA, Zr)Hx in radial blanket region of a fast reactor and a plan of technology development for the MA target. The feature of this concept is that it has a great potential of transmutation and can be used in proven fast reactors, but naturally requires research and development.
The proposed (MA, Zr)Hx subassembly concept can be realized that the ratio of hydrogen to MA is enough for neutron energy spectrum shift and the loaded weight of MA is also enough for enhancing the transmutation because it densely contains both of MA and hydrogen. Preliminarily the MA transmutation rates were compared about four types of MA target: MA-Zr alloy pin; (MA, Zr)Hx one; lightly and heavily moderated combinations of (MA, Zr)O2 and ZrH1.6 ones. It was assumed that

they are loaded around an active core in a 280 MWe sodium-cooled reactor; 54 MA target assemblies are respectively arranged in the radial blanket zone. It was followed that the MA transmutation of (MA, Zr)Hx doubles or triples, compared with the other types.
One of the other issues is optimizing the irradiation condition and specification. Shorter terms from irradiation to acceptable decay heat for spent fuel storage, smaller power distortion of neighbor subassemblies, higher ratio of transmutation, and greater mass transmutation are preferable, but they are near incompatible. Therefore, the feasibility study is optimizing the irradiation condition and specification of the (MA, Zr)Hx target so as to harmonize the requirements.
We started to research and develop key technologies of this concept toward an innovative actinide fuel cycle, conducted as the nuclear system research and development program under the contract with MEXT and supported by Nuclear Safety Research Association in Japan. The items of R&D contains measurements of the physical properties of (MA, Zr)Hx, fabrication testing, and laboratory-scale reprocessing test of (MA, Zr)Hx samples. Pellets of MA hydride target will be fabricated by hydrogenating of MA-Zr alloys in a Sieverts system to produce homogenous mixed pellet without a crack. In the next phase, sample of (MA, Zr)H pellet will be irradiated in a fast reactor and their irradiation behavior are measured by post-irradiation tests. These R&Ds would create a practicable and effective strategy of MA transmutation.

Country/Int. Organization:

Japan

Panel 1: Development and Standardization of Safety Design Criteria (SDC) and Guidelines (SDG) for Sodium Cooled Fast Reactors / 565

Compliance of Korean SFR Safety Design Approaches with Generation-IV Safety Design Criteria (Korea, R. of)

Author: JAEWOON YOO[1]

Co-authors: JINWOOK CHANG [1]; TAE-HO LEE [1]

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Korea Atomic Energy Research Institute (KAERI) is developing Prototype Generation-IV SFR (PGSFR). The first design stage has been completed at the end of 2015 and the preliminary safety information document (PSID) has been issued as a main outcome of the phase. The safety design approach of PGSFR is compliance of defense-in-depth and to enhance the inherent and passive safety design features of a metal fuel and pool type sodium system. Additional design measures against the severe accident mitigation has been implemented into the PGSFR design. Inherent reactivity feedback resulting from the metal fuel properties plays a positive role during design basis accidents and design extension conditions, which is basic mechanism to prevent the severe accident as well as the fully passive safety grade decay heat removal systems. The ex-vessel cooling and self-actuated shutdown system provides additional design margin against severe accident propagation toward the goal of molten-fuel in-vessel retention. The compliance of PGSFR with Generation-IV SDC will be explained in more details during panel discussion.

Country/Int. Organization:

Korea Atomic Energy Research Institute, Republic of Korea

Panel 2: Small and Medium sized fast reactors / 566

Feasibility and Challenges for Self-sustainable Long-Life SMR without Refueling

Author: Yonghee Kim[1]

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This talk is concerned with feasibility and challenges for a self-sustainable fast-spectrum SMR (Small Modular Reactor), which can be operated without any refueling and reprocessing during the whole plant lifetime of ~50 years. For a competitive and proliferation-resistant fast-spectrum SMR, long-life core designs are reviewed and a compact SMR design is introduced and its design features and performances are discussed in terms of the core lifetime, fuel burnup, and inherent safety characteristics etc. Major technical challenges for the ultra-long-life SMR are discussed, which include lifetime of the fuel, coolant void reactivity, very high fuel burnup, passive decay heat removal, etc. In addition, self-sustainability of the long-life SMR is addressed in view of spent fuel recycle in an extremely simplified and clearly proliferation-resistant way. A super-simple melt-treatment of metallic spent fuels are introduced in this presentation for the self-sustainability of the fast-spectrum SMR.

Country/Int. Organization:

Republic of Korea

6.5 Uncertainty Analysis and Tools / 567

Recent and Potential Advances of the HGPT methodology

Author: Augusto Gandini[1]

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Recent and potential advances of the Heuristically based Generalized Perturbation Theory (HGPT) methodology are discussed:

- *The subcriticality monitoring method*

Basing on the HGPT methodology applied to subcritical systems, a procedure is described for the online monitoring of the subcriticality level of ADS reactors with minimal interaction with the plant normal operation.
The proposed method consists in compensating slow, small movements of a control rod with likewise slow, small alterations of the external source strength, so that the overall power is maintained constant.
The estimation of the subcriticality level requires the knowledge of a bias factor. This implies the standard precalibration of a control rod and the precalculation of the importance function associated with the reactor power control (in this case, the external neutron source strength).

- *The hot spot identification by sensitivity analysis and probabilistic inference*

A method is described by which the information obtained on-line through a system of neutron measuring devices such as self-powered neutron detectors (SPNDs, called also collectrons) inserted in the core of a nuclear power reactor allows the on-line detection of a possible hot spot during plant operation. The method is based on the HGPT techniques, for the calculation of the sensitivity coefficients of the integral quantities measured with the collectrons with respect to parameters representative of the hot spot, and on the use of statistical inference techniques, taking into account the errors associated with the measurements. The methodology allows to assess the effect on the quality of the hot point detection system following possible failures of the measuring devices during the core

life cycle. Such an assessment may be useful for defining an adequate protection strategy in terms of quality, number and distribution of the collectrons. This method has been initially aimed to be adopted in thermal power systems, but with advanced detection techniques underway it might be adopted also in fast ones.

- *Use of the GPT methodology for the analysis of reactivity worths with Monte Carlo*

Perturbation methods are part of the reactor physics foundation for the study of fundamental quantities considered in design and safety analysis of nuclear reactors. In deterministic codes standard perturbation theory (SPT) and generalized perturbation theory (GPT) methods have been historically developed and used. Monte Carlo codes, such as MCNP 6.1, can also perform, via adjoint weighted tally, SPT calculations of reactivity worths. A method is proposed to enable Monte Carlo codes to implement GPT.

Country/Int. Organization:

Sapienza University, Rome (Italy)

Plenary Session 28 June / 568

The European Commission contribution to the development of safe and sustainable fast reactor systems

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keynote speech will be provided later

Country/Int. Organization:

EC/JRC
JRC Directorate G –Nuclear Safety and Security

Poster Session 1 / 570

FACILITY FOR ADVANCED FUELS THROUGH THE SOL-GEL METHOD

Author: Joseph Mathew[1]

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Advanced fuels forms need to be developed for use with the evolving nuclear fuel cycles. The fabrication of sphere-pac fuel pins that employ microspheres of U,Pu mixed oxide (MOX) or microspheres containing the oxides of Minor Actinides (MA) prepared through the internal gelation process are particularly relevant in this context. Since the sol-gel process used in the fabrication of these microspheres offers significant advantages over the conventional powder metallurgical processes used in the fabrication of pellets, efforts are underway at IGCAR in collaboration with Bhabha Atomic Research Centre (BARC)), Mumbai in order to establish a remote handling facility suitable for fabricating a sphere-pac fuel pin containing MOX microspheres. This paper describes the details of a facility that is being created at IGCAR to accomplish the above and the recent experience on the test fabrication of sphere-pac fuel pins for test irradiation.
This facility comprises a jet-entrainment facility for the preparation of fine fraction UO2 microspheres (115±10 μm), a column gelation facility for the preparation of the coarse fraction U, Pu MOX (53% Pu) microspheres (775±75μm) through column gelation and equipment for characterization of the microspheres prepared in our laboratory. In addition a glove box train comprising facilites handling the microspheres, vibro-packing, fuel-pin welding and decontaminating the fuel pin was commissioned and qualified for handling MOX microspheres.
As many as 75 production runs were carried out for the production of UO2 microspheres with the jet entrainment set-up, earlier, in order to optimize the process parameters viz., size of the nozzle, temperature of gelation, composition of the broth, washing routine as well as the conditions for calcination and sintering. Both the MOX as well as the UO2 microspheres were checked for their physical integrity, dimensions and were found to conform to the desired chemical composition stipulated for the test irradiation in FBTR. Conditions for vibrocompaction were optimized. Quality control checks were performed on the fuel pins after the fabrication.
Trials were carried out for the preparation of Am containing UO2 fuel microspheres. The viscosity of a broth containing both U and Pu was measured in order to establish the optimum conditions for gelation of the sol. This is a "first of its kind"measurement, made in a custom-made facility created for this purpose.

Country/Int. Organization:

India, Indira Gandhi Centre for Atomic Research, Kalpakkam

Plenary Session 27 June / 571

Current status and future view of the fast reactor cycle technology development in Japan

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As stated in the "Fourth Strategic Energy Plan", which was approved by the Cabinet in April 2014, Japan continues to position nuclear energy as a major base-load power source even after the TEPCO's Fukushima Dai-ichi Nuclear Power Station accident. Its basic policy in the plan is to promote nuclear fuel cycle in terms of the efficient use of resources and reduction in volume and toxic level of high-level waste, and carry out fast reactor (FR) cycle R&D for the commercialization, taking advantage of international cooperation.

For the commercialization of FR cycle, Japan Atomic Energy Agency (JAEA) is conducting several R&D activities primarily focusing on 1) the reduction in volume and toxic level of radioactive waste and 2) the improvement of the safety of FRs and FBRs, as mentioned in the Fourth Strategic Energy Plan, in parallel with R&D utilizing international cooperation with bilateral frameworks such as ASTRID program with France and multilateral frameworks such as the Generation IV International Forum (GIF). In the nuclear fuel cycle R&D, SmART cycle project to conduct small-scale minor actinide (MA) recycling using existing facilities is in progress.

The prototype FBR Monju will be subject to a fundamental review, and the government's official policy on Monju together with FR development in concrete terms will be presented by the end of this year. As for the experimental FR Joyo, JAEA completed the upper core structure (UCS) replacement work and is preparing to make an application for earlier restart under the new regulatory requirements.

Country/Int. Organization:

Japan/ Japan Atomic Energy Agency

Poster Session 2 / 573

Participation of Mexico in the OECD/NEA SFR Benchmark using the Monte Carlo code Serpent

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In 2014, Mexico was honored with its acceptance as an observer member in the Technical Working Group on Fast Reactors of the IAEA. Afterwards, the Mexican participation in the fast reactors activities augmented when in 2016 the Mexican Team was accepted in the UAM SFR Benchmark of the OECD/NEA. The first technical specifications of the mentioned Benchmark consisted of four sodium-cooled fast reactors (3600 MWt metallic-fueled, 3600 MWt MOX-fueled, 1000 MWt carbide-fueled, and 1000 MWt MOX-fueled). The code selected for the full-core simulations was the Finnish Monte Carlo code Serpent version 2.1.26 and the calculations were performed using two different cross sections libraries, namely JEFF 3.1.1 and ENDFB 7.0. The geometry, material composition and Monte Carlo solution parameters used are briefly described in this paper together with the main results obtained by the Mexican team. Quite good agreement (in the order of tens of pcms) for results of keff, sodium void worth, and delayed neutron fraction was observed when comparing with the ones obtained by other participants that followed the same methodology (code and evaluated data libraries). Larger deviations were found when comparing with different methodologies, but in general the calculated solutions were reasonably close to the averaged results reported in the Benchmark. The use of the Monte Carlo code Serpent fulfills two objectives; firstly, to get confidence in the obtainment of reference solutions; and secondly, to generate homogenized cross sections to be used within the currently development of the Mexican neutron diffusion code for hexagonal-z geometry AZNHEX, which is part of the AZTLAN platform: Mexican platform for analysis and design of nuclear reactors. The participation of Mexico in this OECD/NEA Benchmark strengthens the Mexican fast reactors knowledge and allows the country to contribute more actively to the international efforts in the field.

Country/Int. Organization:

Instituto Nacional de Investigaciones Nucleares, Instituto Politécnico Nacional

1.7 ADS AND OTHER REACTOR DESIGNS / 574

The ALLEGRO experimental Gas Cooled Fast Reactor Project

Author: János Gadó[1]

Co-authors: Alfredo Vasile [2]; Branislav Hatala [3]; Grzegorz Wrochna [4]; Ladislav Bělovský [5]

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ALLEGRO is experimental fast reactor cooled by Helium being developed by the European V4G4 Consortium of the nuclear research organizations of the Czech Republic, Hungary, Poland and Slovakia associated with CEA, France. Development of ALLEGRO is an important step on the way to the Gas-cooled Fast Reactor, one of the six concepts selected by the Generation IV International Forum and one of the three fast reactors supported by the European Sustainable Nuclear Energy Technology Platform.
The main purpose of the facility is to develop:
- innovative refractory GFR fuels,
- components and systems related to the Helium technologies and
- a safety framework applicable to the specific characters of GFRs.
Starting from a reference design studied up to 2009, the project is exploring a new target of nominal power (in the range of 30 –75 MW thermal) and power density (in the range 50 –100 MW/m3) compatible with the safety limits and the design requirements. At the same time, the feasibility of a LEU UOX start-up core as alternative to a standard MOX core is being considered. This start-up core, to be used in the first period of the reactor operation, will include experimental positions dedicated to the refractory fuel development.
The paper describes the current status and the perspective steps of the design and safety studies and experimental work to demonstrate the feasibility of ALLEGRO.

Country/Int. Organization:

Hungary

1.7 ADS AND OTHER REACTOR DESIGNS / 575

Design Evolutions of the Molten Salt Fast Reactor

Author: Delphine GERARDIN[1]

Co-authors: Axel LAUREAU [1]; Carole SEUVRE [1]; Daniel HEUER [1]; Elsa MERLE-LUCOTTE [1]; Michel ALLIBERT [1]

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The CNRS has focused R&D efforts on the development of a new reactor concept called the Molten Salt Fast Reactor (MSFR). The MSFR, characterized by a circulating liquid fuel and a fast neutron spectrum, has been identified as a very interesting long term alternative to solid fuelled fast neutron systems in the Gen4 International Forum.

MSRs are liquid-fuelled reactors so that they are flexible in terms of operation or design choices, but they are very different in terms of design and safety approach compared to solid-fuelled reactors. The MSFR system includes three different circuits: the fuel circuit, the intermediate circuit and the power conversion circuit, together with normal and emergency draining tanks and on-site fuel processing units. This paper will focus on the new designs developed for the fuel circuit and the emergency draining system of the MSFR in the frame of the SAMOFAR European project. The fuel circuit, defined as the circuit containing the fuel salt during power generation, includes the core cavity and the recirculation/cooling sectors. These new designs result from physical and preliminary safety studies such as for the fuel circuit optimizing the use of the molten salt both as fuel and coolant, defining the operating procedures and minimizing the fuel leakage risks. Additional requirements

are considered for the emergency draining system to be able to confine the fuel and to evacuate the residual heat over very long time periods (months) with no human intervention and to guaranty that under no circumstance the salt may reach criticality in this area.

Country/Int. Organization:

FRANCE / LPSC-IN2P3-CNRS - Université Grenoble Alpes

6.6 Coupled Calculations / 576

Research and Development on Simulator of Fast Reactor in China

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With the closed fuel cycle strategy, China develops the fast reactor and advanced reprocessing technology, which supports the sustainable development of China nuclear energy. Technical solutions for the simulator of China Experimental Fast Reactor (CEFR), modeling and simulation are mainly introduced. CEFR adopted a pool-type FR technology with three-loops, which has 216 subsystems. The full scope real-time simulator was finished by Nuclear Power Simulation Research Center(NPSRC) at Harbin Engineering University（HEU）in collaboration with China Institute of Atomic Energy (CIAE) and validated by CEFR. According to the principle, characteristics of system, structure and operation of CEFR, the relevant research to determine the scope and degree, establish models and design systems for the simulation of CEFR has been accomplished. The model and software have been developed for 71 CEFR subsystems. The reactor physics, primary coolant system, secondary coolant system, third coolant system, auxiliary system and passive decay heat removal system and etc. are included in the simulator, which has been applied to instruct the debugging and experimental operation of CEFR and improve the control methods.

Country/Int. Organization:

Fundamental Science on Nuclear Safety and Simulation Technology Laboratory, Harbin Engineering University,China

3.5 General Safety Approach / 577

Safety assurance of the new generation of the Russian fast liquid metal reactors

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L.A.Bolshov, V.F.Strizhov, D.A. Klinov, A.A. Kamaev, V.V.Lemekhov, A.V.Moiseev, S.F.Shepelev, B.A.Vasiliev

The approaches to assurance and justification of safety of the new generation of fast liquid metal reactors with sodium and heavy coolants that are currently being developed in Russia are presented. The accepted safety concept deals with some aspects including:

• Assurance of nuclear safety by minimizing of potential hazard (reactivity excess, sodium void effect of reactivity), strengthening of the negative feedback (reactivity coefficient) and development of passive elements affecting reactivity based on hydraulic and temperature principle;
• Design solutions for safety improvement by integrated configuration of RI that excludes a loss-of-coolant accident;
• Using of the possibilities for long accumulation of residual energy without reaching a dangerous temperature level;
• Development of primary circuit systems for residual heat removal by natural circulation with passive principles of initialization of their operation;
• Improvement of reliability of main equipment and safety systems.
Design of the fast sodium reactor BN-800 that was put into operation in 2016 ensures safety at a level corresponding to GEN III+ nuclear installation due to inherent safety and improved and new safety system compared to the operating BN-600.
The BN-1200 design will meet the requirements for GEN IV nuclear installation taking into account the further optimization of safety systems and new technical solutions.
The purpose of developing the BREST-OD-300 reactor is to demonstrate the ability to achieve a high level of safety and competitiveness by using a new coolant, fuel and constructive solutions, as well as to gain experience of reactor closed fuel cycle operation within the frameworks of prototype power system.
The closed fuel cycle operation of reactor core in equilibrium mode using dense nitride fuel is possible for both projects.
The objective of new generation of nuclear installations is to exclude the severe accidents resulting in potential necessity of such protective measures as population evacuation and resettlement. The presentation shows that this objective is achieved by new Russian projects of power units equipped with BN-1200 and BREST-OD-300 reactor installations.
One of the basic elements of new project safety justification is a system of the Russian codes of new generation aimed at creation of new precision calculation systems describing a wide spectrum of physical processes and phenomena on the basis of multiphysics and multiscale modeling.
It is noticed that the development of installations with essentially new specifications has required an improvement of existing regulations, especially for lead coolant reactors. The main directions of legislation and regulation enhancement planned in the Russian Federation are presented; the high potential of international cooperation in this field is emphasized.

Country/Int. Organization:

Russian Federation

6.3 Neutronics - 1 / 578

International research center based on MBIR reactor –cornerstone for Generation 4 technologies development

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This report intended to provide an update on the International research center (IRC) based on the fast sodium research reactor MBIR development. The report will include the proposed IRC structure, key terms of participation, proposed areas for multilateral research, etc. It will also present the R&D possibilities that IRC members will have on the closed nuclear fuel cycle due to the Multi-functional radio-chemistry research facility, which is also being constructed at RIAR site.

MBIR reactor technical parameters (very high flux, up to 3 simultaneously working independent loops, horizontal and vertical channels, high experimental capacity and other features) ensure the needed experimental support for the R&D conducted to create the new generation innovative nuclear energy facilities. MBIR and Multi-functional radio-chemistry facility at one site will provide

an opportunity to execute and perfect the closed fuel cycle and radioactive waste utilization. In addition the combination of those facilities will allow to conduct the complex material testing research including creation of the new constructive materials, fuel and absorbing materials as well as to perform complex experimental tasks with the use of neutrons for fundamental studies.

Both MBIR reactor and the radio-chemistry facility were awarded the ICERR status as part of RIAR facilities in September 2016. The high flux neutron fast reactor facility is a powerful instrument and cannot be realized small scale or as a modular complex which leads to high capital costs and overcapacity for a single user. This is one of the reasons behind the idea of the international partnership where one reactor can be used by multiple international users and research can be conducted both on bilateral and multilateral basis.

Country/Int. Organization:

Russian Federation /
State Atomic Energy Corporation "Rosatom"

Poster Session 1 / 579

Development of Fast Reactors in the USSR and the Russian Federation; Malfunctions and Incidents in the Course of their Operation and Solution of Problems.

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The initial idea of potential nuclear fuel breeding originated in the USA and the first success in development of fast reactors designed for implementation of this idea was achieved there. With a very small delay, similar studies started in the USSR; however, that was the place where fast reactor development reached its peak. The chief scientific supervisor of these research studies in the USSR was A.I. Leypunsky. Great achievements in this area were made by scientists and engineers from France and the UK.
After completion of nuclear weapon tests, A.I. Leypunsky sent a Position Paper to the First Chief Directorate, where he stated the principal physical ideas and the high-priority tasks on fast reactors. These proposals were approved in the Government Decree of 1950. It was followed by development and construction of the critical facility BR- 1, research reactor BR-2 , research reactor BR-5 (BR-10), critical facility BFS-1, research reactor BOR-60, critical facility BFS-2, the pilot and demonstration power reactor BN-350, power reactor BN-600 and power reactor the BN-800.
Over the past 60 years operation of research and power reactors in our country has accumulated vast experience, including abnormal and emergency situations, their causes and ways of overcoming them, ensuring reliable and safe operation of fast reactors with sodium coolant.
In the future –assimilation of the closed fuel cycle, NPPs with the BN-1200 reactors, ensuring competitive economy.

Country/Int. Organization:

Russia. Institute for Physics and Power Engineering

Plenary Session 27 June / 580

Research and Pilot Fast Neutron Reactors in Russian Federation

as the Ground for Development of Worldwide Commercial Technologies

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As early as the exploration of nuclear energy based on the fission of heavy atoms, it became obvious that one of the key conditions for its future wide-scale application is nuclear breeding. For this purpose, in the middle of the last century, the Soviet Union started implementing an experimental infrastructure to develop and construct fast reactors. In a quick period of time the following test reactors were developed and commissioned: BR-1 (1955), BR-2 (1956) and BR-5 (1958). In 1961, a critical assembly BFS-1 was put into operation to simulate neutronic characteristics of fast reactors. In 1969, a fast test sodium-cooled reactor BOR-60 was commissioned having a steam turbine to produce electricity. The reactor is intended to test all the fast sodium reactor technologies. The same year, the world's largest critical assembly BFS-2 was constructed. For the next eleven years, commercial power reactors BN-350 (1973) and BN-600 (1980) were commissioned.
After the severe accident at the Chernobyl's 4th unit in 1986, the Soviet Union's intensive nuclear energy development program was suspended and the next following decades were devoted to the fundamental research in the reactor feasibility and safety as well as to the development of new reactor materials and design concepts.
Since the beginning of 1990s, the Russian Federation has conducted R&D and design activities to develop lead-bismuth- and lead-cooled fast reactors with inherent safety. The development activities related to the fast sodium reactors have been continued and, so far there was put into operation the BN-800 commercial power reactor with a hybrid core operating oxide and MOX fuel; the BN-1200 commercial fast sodium reactor project was developed as well.
Speaking about the test reactors, the BOR-60 lifetime has been extended till 2020 to continue the in-pile testing of the structural and fuel materials; a design of a new fast test reactor MBIR was developed and its construction has been started to further develop and experimentally support the wide-scale program for commercial power reactors of the next generation.

Country/Int. Organization:

Russia/JSC "SSC RIAR"

Poster Session 1 / 581

Complex discussion of inherent safety fast reactors start-up with enriched uranium concept (strategical, economical aspects, problems of neutron physics etc.). R&D program proposal

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Due to the growing population of Earth, the development of a full-scale nuclear power industry is becoming an increasingly challenging task in the 21st century and onwards. The Breakthrough («Proryv») Project is focused on inherently safe fast reactors which are expected to resolve, for a first time, the economic competitiveness problems of the nuclear power sector. In order to develop a full-fledged nuclear power industry based on such reactors within acceptable timeframe, these reactors must first be put into operation with enriched uranium.
The article provides the results of systemic calculations confirming this thesis. Moreover, it supports the economic benefits (in the nearest future) of the uranium-based start of fast reactors versus the uranium-plutonium start. For the first time, it demonstrates the possibility of a noticeably simpler transition from uranium fuel-based start to uranium-plutonium regime in the closed fuel cycle

compared to the previous alternatives (reduced number of structural changes in the core during the transient mode, less restrictive requirements to the start load, etc.). An R&D program is proposed in order to justify the start of inherently safe fast reactors on enriched uranium.

Country/Int. Organization:

RUSSIAN FEDERATION / Private institution «Innovation and technology center for the «PRORYV» project»

Poster Session 1 / **582**

(U,Pu)O2-x MOX pellet for Astrid reactor project

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Co-authors: Dominique FAVET [1]; Laurent PARET [2]

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Abstract

Since 2015, **AREVA and CEA** teams decided to launch yearly industrial tests of MOX pellets with an adapted GEN III design in the **MELOX** plant, to prepare the future manufacturing of MOX fuels bundle for Astrid reactor.
First campaign (2015) of tests was dedicated to demonstrate the feasibility of this manufacturing at half industrial scale; Main modifications involved the pelletizing station of LCT workshop (small scale line for MOX manufacturing) and one of the industrial furnaces, in order to define the range of main parameters (powder preparation, pelletizing and sintering steps and MOX pellet analyses procedures). Specified analyses results were performed in MELOX plant laboratory, completed with EPMA analyses on MOX pellet sent to CADARACHE CEA laboratory : First results show that required properties of these MOX pellet, meet the specified criteria defined by CEA teams, the most important one's are related to pellet design (dimensions and density), Pu distribution and stoichiometry.
Second campaign (2016) of tests, included a powder preparation step at industrial scale on one of the blender of the MELOX plant, in order to prepare the industrial manufacturing of MOX pellet for one fuel bundle, designed for a prototypical irradiation. Main results show again that the specified criteria are respected increasing the confidence in the process route.

Keywords: MOX, annular pellet, Astrid, EPMA.

Country/Int. Organization:

FRANCE / AREVA and CEA

Poster Session 1 / **583**

The Commercial Potential of the Dual-Component Nuclear Power System

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1. Introduction
2. NFC in Dual-Component Nuclear Power System
3. Advantages of Dual-Component Power System
4. Reducing RW amount going to disposal
5. Format of commercial proposal
6. Status of Russian nuclear infrastructure
7. Conclusions

Country/Int. Organization:

Russian Federation, TENEX

Plenary Session 26 June / 585

Research, development and deployment of fast reactors and related fuel cycle in China

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Country/Int. Organization:

China

Plenary Session 26 June / 586

Status of the French Fast Reactor Programme

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Country/Int. Organization:

France

Plenary Session 28 June / 587

Overview of NEA Activities Related to Fast Reactors

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Country/Int. Organization:

OECD/NEA

Plenary Session 28 June / 588

Overview of GEN-IV International Forum Activities. Status and Prospects of Fast Reactors

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Plenary Session 28 June / 589

INPRO: Fast Reactors and Enhanced Nuclear Energy Sustainability

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Country/Int. Organization:

IAEA

Panel 1: Development and Standardization of Safety Design Criteria (SDC) and Guidelines (SDG) for Sodium Cooled Fast Reactors / 590

Safety criteria for future Indian SFRs (India)

Author: Subhash Chander CHETAL[1]

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to be provided later by the author

Country/Int. Organization:

INDIA

Panel 2: Small and Medium sized fast reactors / 592

SVBR-100 as a possible option for developing countries, why?

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content to be provided by the author soon

Country/Int. Organization:

Russian Federation

Panel 2: Small and Medium sized fast reactors / 593

A safe and competitive lead-cooled small modular fast reactor concept for a short-term deployment

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content to be provided by the author soon

Country/Int. Organization:

ITALY

Panel 2: Small and Medium sized fast reactors / 594

Eligibility of Small Molten Salt Fast Reactor (S-MSFR)

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content to be provided by the author soon

Country/Int. Organization:

Switzerland

Panel 2: Small and Medium sized fast reactors / 595

Small fast reactors for arctic regions

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the title and content to be provided by author soon

Country/Int. Organization:

Sweden

Panel 1: Development and Standardization of Safety Design Criteria (SDC) and Guidelines (SDG) for Sodium Cooled Fast Reactors / 599

Russian SFR Safety Requirements and Approaches and Their Correspondence to Generation-IV SFR Safety Design Criteria (Russia)

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Panel 1: Development and Standardization of Safety Design Criteria (SDC) and Guidelines (SDG) for Sodium Cooled Fast Reactors / 600

Panel Discussion

Opening Session / 601

Opening Address by Director General, ROSATOM (by video message)

Opening Session / 602

Opening Address by Director General, IAEA (by video message)

Opening Session / 603

Welcome Note by Conference General Chair

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Poster Session 2 / 604

Characterization of LBE Non-isothermal Natural Circulation by Experiments with HELIOS Test Loop and Numerical Analyses

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Co-authors: H Ju [2]; IL SOON HWANG [1]; J Cho [3]; J Lee [2]; S Sohn [2]

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We present results of experiments with lead-bismuth eutectic (LBE) non-isothermal natural circulation in a full-height scale test loop, HELIOS, and numerical modeling results performed by a system thermal-hydraulics code. The experimental studies were conducted under steady state as a function of core power conditions from 9.8kW to 33.6kW. Local surface heaters on the main loop were activated and finely tuned by trial-and-error approach to make adiabatic wall boundary conditions. Activities on numerical modeling were carried out by a thermal-hydraulic system code MARS-LBE using the well-defined experimental data. It is found that the predictions were mostly in good agreement with the experimental data in terms of mass flow rate within 7% and temperature difference within 7%, respectively.

Country/Int. Organization:

Republic of Korea, Seoul National University

YGE Panel / 605

Introduction to the YGE Panel

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YGE Panel / 606

How the Next Generation of People will shape the Next Generation of Nuclear

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YGE Panel / 607

Innovative cold trap filtration technologies for reliable and economical exploitation of lead-bismuth eutectic cooled systems

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YGE Panel / 608

Development of Reverse Flow Blockage Device for Primary Sodium Pumps of Fast Breeder Reactor

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YGE Panel / 609

Developing an open-source multi-physics tool for simulating advanced nuclear reactors

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YGE Panel / 610

Development of Tri-iso-Amyl Phospahte (TiAP) based solvent extraction process as an alternate method for the processing of metallic alloy fuels (U-Pu-Zr and UZr)

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[1] *IGCAR*

YGE Panel / 611

Stability and bifurcation analysis of sodium boiling in a GEN IV SFR reactor core

Author: Bissen Edouard[1]

[1] *CEA - Cadarache*

Poster Session 1 / 612

Resting Bottom Fast Reactor

Author: Didier Costes[1]

[1] *CEA*

The resting bottom sodium vessel may receive several applications. For a fast reactor, the direct gas circuit without intermediary sodium may increase the thermal efficiency and, conveniently studied, presents no risk of gas arrival perturbing the core. A construction economy seems to appear and, at end of use, the under soil vessel needs no dismantling and could contain radioactive remains, again with economy.. This fast reactor should be studied.

Country/Int. Organization:

CEA, France

Opening Session / 613

Welcome Note by Conference General Co-Chair

Opening Session / 614

Welcome Note by Deputy Presidential Envoy in the Ural Federation District

Opening Session / 615

Fast Reactor Development and International Cooperation (by Honorary General Chair)

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Closing Session / 616

Concluding Report on the Technical Sessions

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Closing Session / 617

Report on Panel 1: Safety Design Guidelines

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Closing Session / 618

Report on Panel 2: Small and Medium sized fast reactors

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Closing Session / 619

Report on the Young Generation Event

Closing Session / 620

Closing Remarks by Conference General Chair

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Closing Session / 621

Closing Remarks by Conference General Co-Chair

YGE Workshop / 622

Introduction to the Workshop

YGE Workshop / 623

The Role of the IAEA in Fast Reactor Development and Knowledge Transfer

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YGE Workshop / 624

Knowledge Transfer and Management for an active fleet of fast reactors

YGE Workshop / 625

Knowledge Transfer and Management with interrupted development

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YGE Workshop / 626

Knowledge Transfer and Management during long outage periods

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YGE Workshop / 627

Knowledge Transfer to Young Generation and Technical Reconstruction of BFS Complex

YGE Workshop / 628

Group Discussion

YGE Workshop / 629

Group Presentation

YGE Workshop / 630

Workshop Conclusion