International Conference on Fast Reactors and Related Fuel Cycles FR22: Sustainable Clean Energy for the Future (CN-291)

Report of Contributions

Contribution ID: **6** Type: **ORAL**

Sketch Design of Fuel Sub-Assemblies for a SFR-150 MWe

Friday, April 22, 2022 10:30 AM (12 minutes)

During the years 2018 and 2019 of the ASTRID program, a simulation program on SFRs has been prepared by the CEA and its industrial partners –EDF and Framatome –featuring sketch studies of a smaller-size SFR with extended experimental purposes. The power of the core has been reduced to 150 MWe to minimize investment costs while keeping the capacity to demonstrate the feasibility of Pu multi-recycling and to qualify designs and technologies expected for the future industrial SFRs.
These requirements led to an evolutive design for the core and the fuel sub-assemblies (S/A) over the reactor lifetime. At the beginning, the fuel pins will be similar to the one in former Super-Phénix SFR with UPuO2 fuel containing Pu from reprocessed PWR-UOX fuels. In the next step, Pu coming from reprocessed PWR-MOX fuels will be introduced. Then the concepts for the future industrial SFRs will be qualified: low void worth core, ASTRID-like large pin-small wire bundle, ODS cladding⋯
This paper presents sketch design studies of fuel S/A for a 150 MWe SFR at the end of 2019.
The hexagonal wrapper tube can host either a 169-SPX-type-pins bundle or 127-ASTRID-type-pins bundle. The thermomechanical behavior of the fuel bundle has been calculated with DOMAJEUR code. The lower gas plenum of the fuel pins has been reduced thanks simulations with GERMINAL fuel performance code, developed within the PLEIADES software environment, considering a nominal operation up to 87.5 dpa followed by an unprotected loss-of-flow transient. The upper neutron shielding is made of steel and B4C rings housed in a leaktight compartment to stay compatible with the washing process, while limiting the secondary sodium activation and the irradiation level of diversified absorber rods electromagnet. The overall S/A length of 4.20 m has been reduced by 30 cm compared to ASTRID-600 in the perspective of costs reduction.

Country/Int. organization

France

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Session Classification: 1.3 System Innovations

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **7** Type: **ORAL**

Activities of the GIF Safety and Operation Project of Sodium-Cooled Fast Reactor Systems

Tuesday, April 19, 2022 4:22 PM (12 minutes)

Country/Int. organization

Japan

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Session Classification: 2.1 General Safety Approach

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **8** Type: **ORAL**

Recent studies on fuel properties and irradiation behaviors of Am/Np-bearing MOX

Thursday, April 21, 2022 10:40 AM (12 minutes)

There remain challenges in studies of properties and irradiation behaviors of mixed oxide (MOX) fuels, which aims at reduction in volume and toxicity of high-level radioactive wastes, because of the influential factors such that the fuel reaches very high temperature exceeding 2000 K and oxygen content in the fuel continuously varies depending on surrounding conditions. High temperature and steep temperature gradient of MOX in fast reactor bring about the unique phenomena of pore migration resulting in restructuring and redistribution of elements. In this study, we report our experimental results of the property studies on Am/Np-bearing MOX and discuss how these properties influences on the irradiation behaviors.
Oxygen potentials of Am/Np-bearing MOX have been collected by gas equilibrium technique and reported by the group of the authors. Both Am and Np inclusions in terms of substituting U increase the oxygen potential of MOX with the Am and the Np changing from quadrivalent to trivalent. The effects of Am/Np inclusion were analyzed via defect chemistry and quantitatively incorporated with the exiting models which relates oxygen-to-metal (O/M) ratio, contents of Pu, Am, and Np, temperature and oxygen partial pressure.
Inter-diffusion coefficients of U-Pu, U-Am and U-Np in MOX have been obtained by using diffusion couple technique. Although the measurement results could contain uncertainty, some important trends were obtained, i.e. the inter-diffusion coefficient of U-Am is the largest and that of U-Pu is the second. O/M is significantly influential such that the inter-diffusion coefficients were larger at the O/M=2 than those of O/M<2 by several orders.
The pore migration along temperature gradient during irradiation is considered to arise due to the vaporization and condensation of actinide species in pores. Especially, large vapor pressure of UO3 is the dominant property for the pore migration. The increase of the oxygen potential of MOX with Am/Np leads to more UO3 and the acceleration of the pore migration.
The redistributions of actinide elements were also considered with the relationship of the pore migration, i.e. diffusion in solid phase to relax the inhomogeneity caused by the vaporization and condensation of UO3. Thus, the inter-diffusion coefficients can directly influence on the magnitude of the redistribution.
The obtained properties were modelled with the parameters such as temperature and oxygen partial pressure. This enable the integrated time developing evaluation including the temperature profile of fuel irradiation by simulation code.

Country/Int. organization

Japan

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Session Classification: 3.2 Development of innovative fuels: design and properties irradiation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **9** Type: **POSTER**

Development of density control technologies for MOX pellet using dry recycled powders

Thursday, April 21, 2022 1:40 PM (2 hours)

Technology to utilize a dry recycled MOX powder has been developed as a part of MOX fabrication technology development for fast reactors. The purpose of this study is to develop a technology to control the density of MOX pellets with use of dry recycled MOX powder. A roll crusher and a jet mill were employed to prepare the recycled MOX powder which had three kinds of particle sizes (coarse, medium and fine). Sintering tests of MOX pellets were carried out as parameters of particle size and addition rate of dry recycled powder. The results are summarized as follows.

- For the coarse and medium dry recycled powders, a decrease in density due to addition was confirmed, but for the fine dry recycled powders, almost no decrease in density due to addition was confirmed. From this, it is considered that the fine dry recycled powder can be used in the same manner as the raw material powder such as the raw MOX powder as long as the addition rate is up to about 40 wt %.
- When dry recycled powder (coarse or medium) and pore former were added at the same time, a synergistic effect was produced in addition to the density reduction effect of both, and the density was lower than the expected density. In addition, this synergistic effect occurred within the range of this test at 10 wt% of coarse dry recycled powder + 2 wt% of pore former, or 15 wt% of medium dry recycled powder + 2 wt% of pore former. Further, it is considered that this synergistic effect can be alleviated by adding fine dry recycled powder.
- It is considered that the addition of coarse and medium dry recycled powder can delay the progress of sintering by adding it together with the pore former, and the influence can be suppressed by adding fine dry recycled powder.
- High dry recycled powder addition caused cracks in the pellets, but addition of 2% by weight of pore formers no longer observed cracks.

Country/Int. organization

Japan

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Presenter: NISHINA, Masahiro

Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **10** Type: **ORAL**

Development of simplified fuel fabrication technologies for fast reactors

Thursday, April 21, 2022 11:16 AM (12 minutes)

A high-density annular MOX fuel pellet fabrication technology has been developed for producing a low O/M ratio of less than 1.97 for fast reactors. The low O/M ratio sintered pellets aim to suppress the fuel-cladding chemical interaction (FCCI) at high burnup, and a simplified MOX pellet fabrication process (Short process) is a new production technology for this MOX fuel. The short process is a technology for producing pellets by tumbling granulation, die wall lubrication pressing, sintering, and O/M ratio adjustment using a raw MOX powder obtained by the microwave heating direct denitration method. Compared with the conventional process, the short process can reduce the number of processes from 23 to 8, which makes it possible to improve economic efficiency. In this report, the development situation of the short process was reviewed, and the test results of die wall lubrication pressing and O/M ratio adjustment technologies were extended for scale-up of the fabrication technology.
In the development of the die wall lubrication pressing technology, it is necessary to find the optimum operating conditions because a tumbling-granulated MOX powder is directly pressed without a mixing process with additional lubricant to fabricate annular pellets. The MOX granulated powder was fed by a feeder to a die, and was pressed with 8 cycles/min punch at about 510 MPa. The green pellets of about 55.3 %T.D. were sintered at 2023 K for 4 hours to obtain sintered pellets of 95 %T.D. or higher. The quality of the green annular pellets can be improved by optimizing the operating conditions of the die wall lubrication pressing.
Regarding the O/M ratio adjustment, as results of scaled-up tests by increasing the loading amount from 1.0 to 2.0 kgMOX/batch, and 5%H₂+95%Ar mixed gas flow rate from 5.0 to 10.0 l/min/kgMOX, the average O/M ratio increased from less than 1.97 to slightly higher than 1.97. As a result of the thermo-fluid dynamics simulation, it was revealed that a large part of gas did not pass through the mesh plate and leaked through the clearance between the mesh plate and the gas inlet. Further simulations indicate that the gas flow path can be improved by lengthening the lower end of the outer frame of the tiered mesh plates and installing a rod with an ejection hole in the center of each mesh plate. It is expected that these methods can reduce the O/M ratio to less than 1.97.

Country/Int. organization

Japan

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Session Classification: 3.2 Development of innovative fuels: design and properties irradiation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **11** Type: **ORAL**

Analysis of Fuel Burnup and Safety Parameters of Gas Cooled Fast Breeder Reactors

Friday, April 22, 2022 2:54 PM (12 minutes)

The Fast Reactor concept has been proposed by Generation-IV initiative as a potential candidate to develop safe, sustainable, reliable, proliferation-resistant and economic nuclear energy systems (GIF, 2002). Within fast reactor core, fission chain reaction is sustained by fast neutrons which result in a much higher and harder neutron flux than that of thermal reactors. This high neutron flux allow for the production of fissile materials from fertile nuclides through the so-called breeding process, whereas, part of fission neutrons is used to convert fertile nuclides (238U and 232Th) into fissile nuclides (239Pu and 233U, respectively).

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Two computational models, homogeneous and heterogeneous, of the large scale Gas cooled Fast cor
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Country/Int. organization

Egypt

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Session Classification: 3.4 Advanced Fuel Development

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **12** Type: **ORAL**

Coolant flow monitoring with an Eddy Current Flow Meter at a mock-up of a liquid metal cooled fast reactor

Thursday, April 21, 2022 11:28 AM (12 minutes)

As a possible part of the safety instrumentation in liquid metal cooled fast reactors, the Eddy Current Flow Meter (ECFM) is an important and robust tool for continuously monitoring the coolant flow and detecting coolant blockages in the reactor subassemblies for coolant temperatures of up to 700 ℃. This inductive sensor can be placed directly above the subassemblies, where changes of the coolant flow angle between 0° and 40° are expected in case of a coolant blockage at some subassembly. In this paper, we present the results of numerical simulations as well as experimental results for the influence of the flow angle with respect to the axis of the ECFM on the output signal of the sensor. In a second experimental setup, an array of ECFM is used to detect and localise coolant blockages by artificially blocking one or more flow channels. For both experiments, a simplified liquid metal flow model was constructed, using the eutectic alloy of Gallium, Indium and Tin to represent flow structures which may typically occur above the subassemblies. Due to the low melting point of this alloy, we were able to perform these experiments at room temperature and to validate the ECFM measurements by ultrasonic velocity measurements available at such lower temperatures. The results of our investigations give important insights into the performance of the ECFM under realistic conditions inside the reactor.

Country/Int. organization

Germany

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Session Classification: 5.2 Experimental Programs I

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **13** Type: **POSTER**

Eddy Current Flow Meter flow rate measurements in liquid Sodium at the SUPERFENNEC loop

Thursday, April 21, 2022 10:40 AM (2 hours)

The Eddy Current Flow Meter (ECFM) is a robust and reliable inductive sensor for measuring the flow rate of liquid metals. Since there is no direct contact between sensor and liquid metal, it can be used in chemically aggressive environments and at very high temperatures of up to 600 ℃. This allows the ECFM to be deployed, for example, as part of the safety instrumentation in liquid metal-cooled fast reactors directly above the subassemblies, in order to continuously monitor the flow rate of the coolant and to detect coolant blockages. In this paper we present the measurement results that were obtained with a high temperature prototype of the ECFM at the SUPERFENNEC Sodium loop at CEA Cadarache in France. There, we were able to evaluate the performance of the ECFM between temperatures of 200 ℃ to 400 ℃. In addition to the measurement results, we will discuss results of related numerical simulations and give a detailed description on the construction and choice of materials of the high temperature prototype.

Country/Int. organization

Germany

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Session Classification: Poster Session

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **14** Type: **ORAL**

Simulation of FFTF Individual Reactivity Feedback Tests with SAS4A/SASSYS-1 Code

Thursday, April 21, 2022 11:28 AM (12 minutes)

The Fast Flux Test Facility (FFTF) at the Hanford site in Washington was a 400 MW thermal, oxide-fueled, liquid sodium cooled test reactor, built to assist development and testing of advanced fuels and materials for fast breeder reactors. FFTF operated from 1980 until 1992, providing the U.S. Department of Energy (DOE) with the means to test fuels, materials, and other components in a fast neutron flux environment. One of the FFTF passive safety demonstration tests simulating loss-of-flow conditions without scram (LOFWOS) is currently being analyzed by the international community under an IAEA coordinated research project. In preparation for the passive safety demonstration tests in Cycle 8C, a series of individual reactivity feedback tests were carried out in FFTF. The primary goal of these tests was to check the core reactivity feedbacks in a systematic fashion by subjecting the core to various conditions of power, flow, and inlet temperature. These tests were carried out in Cycle 8A and consisted of quasi-static steps, where after each change the reactor was held at steady-state conditions for a period of about one hour to adjust to new steady-state conditions. The entire Cycle 8A test campaign consisted of about 200 steps. Each step was designed to simulate and validate specific features and reactivity feedbacks of the FFTF core. There were seven types of these individual reactivity feedback tests, targeting fuel reactivity Doppler and axial expansion feedbacks, coolant density feedback, structure reactivity feedbacks such as core radial expansion, as well as integral tests, such as the power reactivity coefficient. The data from the FFTF individual reactivity feedback tests provides a unique opportunity for validation of reactivity feedback modeling in fast reactor analysis codes. SAS4A/SASSYS-1 is one such safety analysis code that was developed at Argonne for transient simulation of liquid metal-cooled fast reactors. The structure of these tests provides data for code validation in a systematic fashion by separating reactivity feedbacks as much as was practically achievable. The quasi-static nature of these tests also simplifies code validation by eliminating the transient effects. This paper presents the results of the application of the SAS4A/SASSYS-1 code to a number of FFTF individual reactivity feedback steps, and compares the code predictions with the test data. The SAS4A/SASSYS-1 results show overall good agreement with the test, but at the same time several model improvement options were identified in this work.

Country/Int. organization

United States of America

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Session Classification: 6.3 Multiscale and Multiphysics Calculations

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **16** Type: **POSTER**

The neutronic study of the nitride fuel loaded CiADS core

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

China

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Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **18** Type: **ORAL**

Modelling and Simulation of Source Term for Sodium-Cooled Fast Reactor under Hypothetical Severe Accident: Primary System/Containment System Interface Source Term Estimation

Wednesday, April 20, 2022 2:04 PM (12 minutes)

Country/Int. organization

Japan

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Presenter: ONODA, Yuichi (Japan Atomic Energy Agency)

Session Classification: 2.3 Accident Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **19** Type: **ORAL**

Investigation on natural circulation for decay heat removal in reactor vessel of sodium-cooled fast reactor

Thursday, April 21, 2022 11:16 AM (12 minutes)

In Sodium-cooled Fast Reactors (SFRs), it is important to optimize the design and operate decay heat removal systems (DHRS) for safety enhancement against severe accidents which could lead to core melting. To clarify the natural circulation phenomena in a reactor vessel during operation of a decay heat removal system, water experiments have been conducted using a 1:10 scale experimental facility (PHEASANT) simulating the reactor vessel of loop-type SFRs. The dipped-type direct reactor heat exchanger (DHX), the penetrated-type DHX and reactor vessel auxiliary cooling system (RVACS) are mounted in PHEASANT. Moreover, the electric heaters are installed to simulate the core and fuel debris accumulated on the core catcher and upper plenum. Therefore, PHEASANT can simulate the natural circulation phenomena under the various conditions for decay heat sources and DHRS operation. In this paper, the natural circulation phenomena under the conditions of operating the dipped-type DHX and RVACS, respectively, were investigated by the results of PHEASANT experiments and experimental analyses. In the condition of operating the dipped-type DHX, the velocity field was quantitatively obtained by the particle image velocimetry and the characteristic of natural circulation phenomena were clarified by the velocity and temperature data. In the condition of RVACS operation, the temperature was measured using decay heat conditions as a parameter and the effect of decay heat condition on the natural circulation phenomena was investigated. In addition, from the comparison between the experimental results and simulation results, it was confirmed that the numerical simulation is applicable to the natural circulation flow field in the reactor vessel of loop-type SFRs.

Country/Int. organization

Japan

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Session Classification: 5.2 Experimental Programs I

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **20** Type: **ORAL**

SPECIFIC FEATURES OF THE EXPORT OF RUSSIAN TECHNOLOGIES OF FAST REACTORS AND A CLOSED NUCLEAR FUEL CYCLE

Tuesday, April 19, 2022 3:46 PM (12 minutes)

Country/Int. organization

Russian Federation

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Session Classification: 7.1 Sustainability: Economics, Environment, and Proliferation

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **21** Type: **ORAL**

Development of Integrated Severe Accident Analysis Code, SPECTRA for Sodium-cooled Fast Reactor

Thursday, April 21, 2022 1:52 PM (12 minutes)

Analytical evaluation of severe accidents (SAs) in sodium-cooled fast reactors (SFRs) becomes increasingly important. The progress of the SAs has been previously evaluated by transferring the analytical results between the multiple analysis codes with different roles. In this study, a new code named SPECTRA (Severe-accident PhEnomenological computational Code for TRansient Assessment) was developed for integrated analysis of the in- and ex-vessel phenomena. This paper provides the newly developed analytical models and the analysis of a loss of reactor level (LORL) event as one example of the SAs.
The SPECTRA code consists of the in- and ex-vessel modules which have a thermal hydraulics module as a base part. The in-vessel thermal hydraulics module computes complicated multi-dimensional behavior of liquid sodium and gas by using the multi-fluid model considering compressibility. Relocation of a molten core is computed by the dissipative particle dynamics method which has an advantage from the viewpoint of its wide applicability. A lumped mass model is employed for computation of the ex-vessel multi-component gas including aerosols. The fully implicit scheme is applied to the both thermal hydraulics modules in order to enable computation with a large time step width. The analytical models for sodium fire, sodium-concrete interaction, and debris-concrete interaction are integrated into the ex-vessel thermal hydraulics module. The in- and ex-vessel modules are coupled by exchanging the amount of leaked sodium and debris at every time step.
The LORL event is considered as one example of the SA scenarios. A sodium coolant leaks from a damaged pipe in a primary cooling loop and causes sodium fire. In case a molten core and sodium leak from a damaged lower head of a reactor vessel (RV), sodium-concrete interaction and debris-concrete interaction occur in the compartment under the RV. This event progress was computed in a simplified domain including the RV, the primary cooling loop, and the ex-vessel multi cells. The analytical result showed lowering of the liquid level due to sodium leak, boiling of the coolant around the core region, and molten core relocation in the in-vessel region. As for the ex-vessel region, the atmosphere temperature and pressure increased due to sodium fire, sodium-concrete interaction, and debris-concrete interaction. The basic capability to reproduce SA scenarios was demonstrated through this analysis.

Country/Int. organization

Japan

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Session Classification: 6.4 Simulation Tools for Safety Analysis

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **22** Type: **ORAL**

A Status of Experimental Program to Achieve In-Vessel Retention during Core Disruptive Accidents of Sodium-Cooled Fast Reactors

Friday, April 22, 2022 11:06 AM (12 minutes)

To achieve in-vessel retention for mitigating the consequences of core disruptive accidents (CDAs) of sodium-cooled fast reactors, controlled material relocation (CMR) has been proposed as an effective safety concept. CMR is not only aiming at eliminating the potential for exceeding prompt criticality events that affect the integrity of the reactor vessel, but also enhancing the potential for the in-vessel cooling of degraded core materials during CDAs. Based on this concept several design measures have been studied, and, to evaluate their effectiveness, experimental evidences to show relocation of molten-core material were required. With this background, a series of experimental program called EAGLE (Experimental Acquisition of Generalized Logic to Eliminate re-criticalities) has been carried out collaboratively over 20 years between Japan Atomic Energy Agency and National Nuclear Center of the Republic of Kazakhstan (NNC/RK) using an out-of-pile and in-pile test facilities of NNC/RK. The EAGLE program is divided into three phases, they are called EAGLE-1, EAGLE-2 and EAGLE-3, to cover whole phase after core-melting begins. The subject for EAGLE-1 and the first half of EAGLE-2 is CMR in the early phase of CDA in which the core melting progresses rapidly driven by positive reactivity insertions. The subject for the later half of EAGLE-2 and whole EAGLE-3 is CMR in the later phase of CDA in which the gradual core melting by decay heat and relocation and cooling of degraded core materials occur. In this paper, the major achievement of the EAGLE program and future plans are presented.

Country/Int. organization

Japan

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Presenter: Dr KAMIYAMA, KENJI (Japan Atomic Energy Agency)

Session Classification: 2.4 Severe Accidents

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **24** Type: **ORAL**

Development of Multi-level Simulation System for Core Thermal-hydraulics Coupled with Plant Dynamics Analysis - Prediction of Transient Temperature Distribution in a Subassembly under Inter-subassembly Heat Transfer Effect -

Thursday, April 21, 2022 11:52 AM (12 minutes)

In the design study of sodium-cooled fast reactor, various activities from sensitivity analysis on whole plant dynamics using simple model to detailed analysis on local phenomena of interest are being performed. In conventional way, the analyses on whole plant dynamics and local phenomena are performed individually and the mutual interaction between them are considered through the settings of boundary conditions for each individual analysis. The final result through the individual analyses may contain excessive conservativeness. Therefore, JAEA has developed the multi-level simulation system in which detailed analysis codes for local phenomena of interest are coupled with a plant dynamics analysis code in order to obtain evaluation results considering the mutual interaction with reasonable conservativeness by updating the boundary conditions successively in coupling process.
In this study, focusing on core thermal-hydraulics, the coupling analysis method using a plant dynamics analysis code named Super-COPD and a subchannel analysis code named ASFRE to evaluate temperature distribution in a subassembly during the transient from forced circulation to natural circulation has been developed as a part of multi-level simulation system. During the transient, the consideration of thermal interaction between whole core and in-subassembly is important because flow re-distribution is caused and temperature distribution in a subassembly is affected by that in adjacent subassembly due to inter-subassembly heat transfer effect in the core. In the coupled analysis using the sequential two-way method, Super-COPD runs first to update flow rate, inlet temperature and heat flux on the boundary wall of the subassembly for ASFRE calculation. Subsequently, ASFRE calculates thermal-hydraulics and pressure drop in the subassembly and turns to Super-COPD calculation in the next time step.
After confirmation of basic functions of the coupling method in a simple geometry, the numerical analyses on the test in EBR-II (SHRT-45R) were performed with two models of the specific subassembly of XX09 with thermo-couples; one was the subchannel model of ASFRE in the coupling method and another one was included in the core model of Super-COPD. Through the comparison of the temperature distributions among the results using only Super-COPD and the coupling method, and the measurement in XX09, it was shown that the coupled analysis could predict transient temperature distribution in a subassembly under inter-subassembly heat transfer effect and it was indicated that the multi-level simulation by changing the level of detail of the analysis model between the method with the plant dynamics code and the coupling method could be performed.

Country/Int. organization

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Session Classification: 6.3 Multiscale and Multiphysics Calculations

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **25** Type: **ORAL**

EXPERIENCE OF OPERATIONAL CHEMICAL CLEANING OF BN-600 STEAM GENERATOR EVAPORATORS FROM CORROSION PRODUCT DEPOSITS

Tuesday, April 19, 2022 4:34 PM (12 minutes)

Country/Int. organization

Russian Federation

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Presenter: Ms LEGKIKH, Kristina

Session Classification: 8.1 SFR Commissioning, Operation, and Decommissioning

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **26** Type: **ORAL**

Comparative multi-criteria analysis of scenarios of the Russian nuclear energy development in the context of uncertainty knowledge about the future

Tuesday, April 19, 2022 4:22 PM (12 minutes)

Country/Int. organization

Russian Federation

Primary authors: Mr KOROBEYNIKOV, Valeryi (IPPE JSC); MOSEEV, Andrei (moseev andrei); YEGOROV, Alexander (IPPE); Dr DEKUSAR, Viktor (IPPE)

Presenter: YEGOROV, Alexander (IPPE)

Session Classification: 7.1 Sustainability: Economics, Environment, and Proliferation

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **27** Type: **ORAL**

Modeling the optimal economic structure of a global deploying nuclear power system with fast and thermal reactors in a partially closed nuclear fuel cycle

Tuesday, April 19, 2022 3:10 PM (12 minutes)

Country/Int. organization

Russian Federation

Primary authors: Mr USANOV, Vladimir (IPPE JSC); YEGOROV, Alexander (IPPE)

Presenter: YEGOROV, Alexander (IPPE)

Session Classification: 7.1 Sustainability: Economics, Environment, and Proliferation

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **28** Type: **ORAL**

The initial stage of closing the NFC of two-component nuclear power. Challenges and solutions

Wednesday, April 20, 2022 10:40 AM (12 minutes)

Country/Int. organization

Russian Federation

Primary authors: Mr GULEVICH , Andrey (IPPE JSC); Dr DEKUSAR, Viktor (IPPE); KOROBEINIKOV, Valerii; MOSEEV, Andrei (moseev andrei)

Presenter: Mr GULEVICH , Andrey (IPPE JSC)

Session Classification: 3.1 Fuel Cycle Scenarios

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **29** Type: **POSTER**

CORROSION HYDROGEN MASS TRANSFER IN FAST REACTOR STEAM GENERATORS OF THE SODIUM-WATER TYPE

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

Russian Federation

Primary authors: SMYKOV, Vladimir (SSC IPPE, JSC); Ms LEGKIKH, Kristina (SSC IPPE,JSC); Ms KANUKHINA, Svetlana (SSC IPPE,JSC)

Presenter: SMYKOV, Vladimir (SSC IPPE, JSC)

Session Classification: Poster Session

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **30** Type: **ORAL**

PROBLEMS OF DECOMISSIONING FAST REACTORS AND WAYS OF THEIR SOLUTION ON THE BASIS OF THE BR-10 RESEARCH REACTOR

Tuesday, April 19, 2022 5:10 PM (12 minutes)

Country/Int. organization

Russian Federation

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Presenter: Dr SMYKOV, Vladimir (SSC IPPE,JSC)

Session Classification: 8.1 SFR Commissioning, Operation, and Decommissioning

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **33** Type: **POSTER**

Characterization of the Molten Chloride Fast Reactor fuel cycle options

Friday, April 22, 2022 10:30 AM (2 hours)

Molten Salt Reactors, as a whole reactor category, belong to the GenIV reactors. They can be designed as thermal, epithermal of fast systems for variety of applications. Especially the Molten Chloride Fast Reactors (MCFRs) provide very hard neutron spectra and very high neutron economy. Hence, MCFRs can be operated as breeders in the closed U-Pu and Th-U cycles or as breed-and-burn reactors in open U-Pu fuel cycle. This high fuel cycle performance is, nonetheless, accompanied by unfavorable fuel salt transparency for neutrons and results in bulky cores.
In this paper, several operating modes of MCFR are simulated, analyzed and characterized. The EQL0D routine, developed for this purpose at Paul Scherrer Institut, is applied on several fuel cycle scenarios. The major evaluated parameters are the actinide mass balance, core size, neutron spectrum, achievable burnup and radiotoxicity generated per unit of power.
The results show that breeding in MCFRs is possible in both Th-U and U-Pu closed cycles. However, the Th-U cycle provides much lower kinf and results in much bigger core. The breeding in closed U-Pu cycle is possible, and the core size is comparable to the fast fluoride salt reactor operated in the Th-U cycle. The breed-and-burn cycle in MCFRs is possible. Nevertheless, the tight neutron economy require minimization of neutron leakage and the core is thus extremely large.
The dependency of core size on fuel cycle parameters like: refueling rate, fissile material share in the feed, or presence of blankets is also analyzed.

Country/Int. organization

Switzerland

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Presenter: KREPEL, Jiri (Paul Scherrer Institut)

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **34** Type: **ORAL**

MSR Fuel Cycle and Thermo-Dynamics Simulations

Friday, April 22, 2022 1:54 PM (12 minutes)

The Molten Salt Reactor (MSR) is unique, with respect to other Gen IV concepts as well as current LWRs, in the fact that the liquid fuel comes with a slew of safety-relevant features, which are chemically distinct from those otherwise encountered. These features are often neglected in the scope of neutronics-based investigations into the topic, where more heed is paid to the isotopic composition rather than the elemental or chemical one.

The overall aim of the present work is to investigate the chemical behaviour of an MSR depending on the setup, fuel cycle as well as initial fuel composition. In order to achieve this analysis, EQL0D, a MATLAB based fuel evolution routine, as well as a Gibbs'energy minimization program (GEMS) are employed. While EQL0D uses inputs such as fuel composition, geometry and reactor power in order to produce an isotopic composition, GEMS can be used on this obtained composition in order to make a prediction on the chemical speciation of the salts present in the system. When the speciation is known, more qualified statements on the volatility, miscibility as well as influence of the redox conditions on the system can be made.

In this paper, simplified MSFR-like systems are used for the neutronics simulation in order to generate representative fuel compositions which can be passed on to GEMS. There are a total of four cases, with all combinations of the fertile isotopes Uranium-238 / Thorium-232 as well as chloride-/ fluoride-based carrier salts. For each case, a separate equilibrium composition is generated by EQL0D, which is used as an input for GEMS to perform a speciation on the most important elements of the system. From there, it is possible to add less prevalent elements to the chemical-thermodynamic simulation to make a prediction as to their chemical state, and therefore volatility, or sweep through various conditions such as temperature or redox environment to investigate which elements are primarily at risk of being transformed into a different state that affects their safety-relevant parameters.

Country/Int. organization

Switzerland

Primary authors: Mr DIETZ, Jonathan (Paul Scherrer Institut); Mr NICHENKO, Sergii (Paul Scherrer Institut); KREPEL, Jiri (Paul Scherrer Institut)

Presenter: Mr DIETZ, Jonathan (Paul Scherrer Institut)

Session Classification: 6.6 Fuel Performance and Material Modelling

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **36** Type: **ORAL**

Spatial interdependence of safety related effects in ESFR-SMART core

Wednesday, April 20, 2022 11:40 AM (12 minutes)

Country/Int. organization

Switzerland

Primary authors: KREPEL, Jiri (Paul Scherrer Institut); Mr PONOMAREV, Alexander (Paul Scherrer Institut); MIKITYUK, Konstantin (Paul Scherrer Institut)

Presenter: KREPEL, Jiri (Paul Scherrer Institut)

Session Classification: 6.1 Neutronics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **37** Type: **POSTER**

NEXT GENERATION NUCLEAR POWER: RADIOLOGICAL SUSTAINABILITY AND ECOLOGICAL ADVANTAGES

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

Russian Federation

Primary author: IVANOV, Viktor

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Presenter: IVANOV, Viktor

Session Classification: Poster Session

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **38** Type: **POSTER**

Design of experimental scheme for activation method of China demonstration fast reactor

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

China

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Session Classification: Poster Session

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **39** Type: **ORAL**

Creep and Creep-Fatigue Behavior of an Advanced Stainless Steel (Alloy 709) - Application to Sodium-Cooled Fast Reactors

Wednesday, April 20, 2022 10:40 AM (12 minutes)

Country/Int. organization

Saudi Arabia

Primary author: Dr ALOMARI, Abdullah (King Abdulaziz City for Science and Technology)

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Presenter: Dr ALOMARI, Abdullah (King Abdulaziz City for Science and Technology)

Session Classification: 4.2 Structural, Novel, and Large Components Materials

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **41** Type: **ORAL**

Investigation of sodium purification

Tuesday, April 19, 2022 2:12 PM (12 minutes)

Country/Int. organization

Russian Federation

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Presenter: Dr ALEKSEEV, Viktor (SSC IPPE)

Session Classification: 4.1 Advanced Reactor Cladding and Core Material, Coolants, and Related Chemistry

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **45** Type: **ORAL**

Fuel cycle closure for high power fast neutron reactor

Wednesday, April 20, 2022 11:04 AM (12 minutes)

Country/Int. organization

Russian Federation

Primary authors: RODINA, Elena (JSC "Proryv"); RACHKOV, Valery (Track leader); KHOMIAKOV, Iurii (JSC "Proryv"); EGOROV, Alexander (JSC "Proryv")

Presenter: RODINA, Elena (JSC "Proryv")

Session Classification: 3.1 Fuel Cycle Scenarios

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **46** Type: **ORAL**

Development of an Artificial Neural Network for predicting spatial interdependencies of reactivity effects in Sodium Fast Reactors

Friday, April 22, 2022 10:42 AM (12 minutes)

Artificial Neural Networks (ANN) are presented as a very powerful tool for modelling complex systems. This approach is becoming increasingly widespread and it has a great potential for nuclear reactor applications. In this work, an ANN is developed for predicting sodium void effects in a large Sodium Fast Reactor core and their spatial interrelations.
The ultimate goal is to provide more realistic inputs to the thermal-hydraulics code TRACE for point-kinetics-based transient analysis of the most recent ESFR core conception. With that goal, an ANN is developed and trained to provide the global sodium density effect and Doppler effect, receiving as input the normalized sodium density and temperatures at the different regions of the core. Local reactivity effects are computed using ERANOS deterministic code for an extensive set of combined scenarios in order to train the ANN.
In this work, the main aspects regarding the optimization of the ANN are presented. A neuron trimming exercise is carried out for getting the most consistent architecture. The developed model can predict the reactivity evolution taking into account the mutual interdependencies of sodium void and Doppler effect. ANN's performance is analyzed by comparing its output in a real transient simulated by TRACE with traditional approaches.

Country/Int. organization

Switzerland

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Presenter: Mr JIMÉNEZ-CARRASCOSA1, Antonio (Politécnica de Madrid (UPM))

Session Classification: 6.5 Integrated Analysis and Digitalization

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **47** Type: **ORAL**

Selection, testing and development of qualification procedure for ALLEGRO gas-cooled fast reactor fuel

Thursday, April 21, 2022 10:52 AM (12 minutes)

On the basis of detailed review, the fuel types were proposed for the new design of the ALLEGRO gas-cooled fast reactor. The first core will be built with MOX or UOX fuel in 15-15Ti stainless steel cladding. These fuel types have been widely used in different sodium-cooled fast reactors. The second core of ALLEGRO will use refractory fuel. The primary candidate is carbide fuel –(UPu)C or UC –in SiC cladding.
15-15Ti and SiCf/SiC type claddings were tested in high temperature helium atmosphere with different impurities in order to investigate the effect of high temperature treatment and impurities on the mechanical load bearing capabilities of these cladding materials. Ballooning tests were performed with 15-15Ti cladding tubes and it was shown that they can keep their integrity at high temperature. The failure pressure of samples tested at 960-1000 ℃ was above 18 MPa.
Qualification procedures have been proposed for the start-up and refractory ALLEGRO fuel. The technology readiness level approach was applied and the basic step of qualification procedure were identified. Using the currently available information the further needs were specified, which include experimental activities, design work, development of numerical models, technology developments, establishment of fuel fabrication capabilities, irradiation in research reactors and post-irradiation examination of fuel.

Country/Int. organization

Hungary

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Presenter: Mr HÓZER, Zoltán (Centre for Energy Research)

Session Classification: 3.2 Development of innovative fuels: design and properties irradiation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **50** Type: **ORAL**

A statistical design method for steady state creep applied to Grade 91 components

Friday, April 22, 2022 2:18 PM (12 minutes)

Current methods for the design of high temperature fast reactor components are deterministic, often based on deterministic structural analysis compared to factored design material data. These methods often produce very conservative designs and the exact design margin – for example, the probability of premature component failure – often cannot be easily quantified. A statistical design method for high temperature components could better quantify the design margin in current deterministic design methods, provide a basis for tuning the design margin in structural design methods to produce more efficient, but still adequately safe, components, and provide regulators and designers increased confidence in the predicted life of fast reactor structural components. This contribution describes a complete statistical design and analysis method for primary load, steady state creep in high temperature reactor components. The method has two parts: a steady state creep analysis method based on a Stokes flow solution and statistical methods for quantifying the distribution of rupture life and creep rate in Grade 91. The Stokes flow analysis greatly reduces the time required to simulate the steady-state stress distribution corresponding to a given set of loading conditions. The statistical creep rate and rupture distributions provide the underlying data for the design analysis. The complete method applies the Monte Carlo approach to sample the creep rate and rupture stress distributions, providing a probabilistic assessment of the life of the component. Here the method is applied to a Grade 91 structural component in the context of a sodium fast reactor. The statistical analysis can be compared to a deterministic design analysis to quantify the design margin in terms of the component reliability.

Country/Int. organization

United States of America

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Presenter: MESSNER, Mark (Argonne National Laboratory)

Session Classification: 6.6 Fuel Performance and Material Modelling

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **51** Type: **ORAL**

Simulation of fission gas release in the 3-D fuel performance code OFFBEAT

Friday, April 22, 2022 1:30 PM (12 minutes)

The OpenFOAM Fuel BEhavior Analysis Tool, i.e. OFFBEAT, is a research-oriented multi-dimensional fuel behavior solver under development at the Laboratory for Reactor Physics and Systems Behaviour at the EPFL and at the Paul Scherrer Institut, in Switzerland. OFFBEAT relies upon the C++ library OpenFOAM and it aims to be a complement to traditional fuel performance codes for the study of 2D and 3D complex phenomena occurring in fuel rods. Although initially developed for light water reactors, the tool displays a wide flexibility and work is ongoing on its extension to Fast Reactor applications. Among various ongoing developments, this paper presents the integration in OFFBEAT of an accurate fission gas release model. The implementation presented in this report is based on the 0-D inert gas behavior code SCIANTIX, developed at the Politecnico di Milano (Italy) and designed to be coupled with fuel performance codes. The correct implementation of this fission gas release model in OFFBEAT is assessed with validation tests, using experimental data from the International Fuel Performance Experiments (IFPE) database.

Country/Int. organization

Switzerland

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Presenter: FIORINA, Carlo (EPFL, Switzerland)

Session Classification: 6.6 Fuel Performance and Material Modelling

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **52** Type: **ORAL**

Verification of the SPL module of the neutron diffusion code AZNHEX through Neutronics Benchmark of CEFR Start-Up Tests

Wednesday, April 20, 2022 11:04 AM (12 minutes)

Country/Int. organization

Mexico

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Presenter: Dr LOPEZ-SOLIS, Roberto (National Institute for Nuclear Research)

Session Classification: 6.1 Neutronics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **53** Type: **POSTER**

Verification and validation of the CEFR Serpent model for the generation of reference solutions and Cross Sections database for the deterministic code AZNHEX

Friday, April 22, 2022 1:30 PM (2 hours)

The National Institute for Nuclear Research (ININ) of Mexico, participates in the IAEA-CRP on Neutronics Benchmark of the Chinese Experimental Fast Reactor (CEFR) Start-Up Tests, which was proposed by China Institute of Atomic Energy (CIAE). The Mexican participation in this Benchmark is focused in two main goals: the first one, the use of SERPENT code for the generation of reference solutions and a Macroscopic Cross Sections (XS) database and,the second one, the use of the previously generated XS to verify and validate AZNHEX, a deterministic domestic code, nowunder-development, devoted to Fast Reactors calculations, that is part of the Mexican platform for nuclear reactor analysis: AZTLAN platform.
The Benchmark exercises include the following tests: fuel loading and approach to criticality, control rod worth measurements, sodium void reactivity, temperature reactivity, reactivity due to fuel subassembly position swap, foil activation and reactivity due to thermal expansion. In this paper, a complete Verification and Validation (V&V) process is described for each one of the exercises mentioned above by comparing the numerical results using SERPENT with the experimental data available. Temperature and thermal expansion effects in all the materials are considered for an accurate representation of the model. Several scripts were created, and are briefly described, to simplify and automate input generation considering the control rods position and for the generation of the XS database. The V&V of AZNHEX code with the XS generated with SERPENT is presented in another dedicated paper.

Country/Int. organization

Mexico

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Presenter: Dr LOPEZ-SOLIS, Roberto (National Institute for Nuclear Research)

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **54** Type: **POSTER**

Model validation of the ASTERIA-SFR code related to freezing phenomena of liquid and liquid/particle mixtures based on THEFIS experimental results

Wednesday, April 20, 2022 1:40 PM (2 hours)

Country/Int. organization

Japan

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Presenters: Dr ISHIZU, Tomoko (Regulatory Standard and Research Department, Secretariat of Nuclear Regulation Authority (S/NRA/R)); Mr FUJITA, Satoshi (Regulatory Standard and Research Department, Secretariat of Nuclear Regulation Authority (S/NRA/R))

Session Classification: Poster Session

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **55** Type: **POSTER**

DEVELOPMENT OF SUBMERGED ELECTROMAGNETIC PUMP FOR LIQUID LEAD

Wednesday, April 20, 2022 10:40 AM (2 hours)

Country/Int. organization

Russian Federation

Primary authors: Mr OBUKHOV, Denis (Joint Stock Company "D.V. Efremov Institute of Electrophysical Apparatus"(JSC «NIIEFA»)); Mr CHAIKA, Pavel (Joint Stock Company "D.V. Efremov Institute of Electrophysical Apparatus"(JSC «NIIEFA»)); Mr GOLOVANOV, Mikhail (Joint Stock Company "D.V. Efremov Institute of Electrophysical Apparatus"(JSC «NIIEFA»)); KIRILLOV, Igor (Joint Stock Company "D.V. Efremov Institute of Electrophysical Apparatus"(JSC «NIIEFA»)); Mr KRIZHANOVSKY, Sergey (Joint Stock Company "D.V. Efremov Institute of Electrophysical Apparatus"(JSC «NIIEFA»)); Mr KOMOV, Kirill (Joint Stock Company "D.V. Efremov Institute of Electrophysical Apparatus"(JSC «NIIEFA»)); Mr LABUSOV, Alexey (Joint Stock Company "D.V. Efremov Institute of Electrophysical Apparatus"(JSC «NIIEFA»)); Mr PRESLITSKY, Gennadiy (Joint Stock Company "D.V. Efremov Institute of Electrophysical Apparatus"(JSC «NIIEFA»)); Mr TUKEEV, Pavel (Joint Stock Company "D.V. Efremov Institute of Electrophysical Apparatus"(JSC «NIIEFA»)); Mrs FEDERIAEVA, Valeriia (Joint Stock Company "D.V. Efremov Institute of Electrophysical Apparatus"(JSC «NIIEFA»))

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Session Classification: Poster Session

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **56** Type: **ORAL**

STATE OF DEVELOPMENT OF LEAD COOLANT TECHNOLOGY COMPONENTS FOR BREST-OD-300 REACTOR

Wednesday, April 20, 2022 11:40 AM (12 minutes)

Country/Int. organization

Russian Federation

Primary authors: ORLOV, Alexander; PLISEINA, Tatiana (JSC NIKIET)

Presenter: ORLOV, Alexander

Session Classification: 4.2 Structural, Novel, and Large Components Materials

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **57** Type: **POSTER**

CURRENT STATE AND ISSUES OF THE HEAVY LIQUID METAL COOLANT TECHNOLOGY DEVELOPMENT (PB, PB-BI)

Wednesday, April 20, 2022 10:40 AM (2 hours)

Country/Int. organization

Russian Federation

Primary authors: Mr ASKHADULLIN, Radomir (Institute of Physics and Power Engineering, Join--Stock Company (IPPE JSC)); Mr LEGKIKH, Alexander (Institute of Physics and Power Engineering, Join-Stock Company (IPPE JSC)); Mr UL'YANOV, Vladimir (Institute of Physics and Power Engineering, Join-Stock Company (IPPE JSC)); Mr VORONIN, Igor (Institute of Physics and Power Engineering, Join-Stock Company (IPPE JSC))

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Session Classification: Poster Session

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **58** Type: **POSTER**

Modeling of water leak into sodium in the BN-600 steam generator

Friday, April 22, 2022 1:30 PM (2 hours)

The report presents the results of comparing the calculated data and readings of devices for monitoring water leakage into sodium, observed during a real leak in the BN-600 steam generator.
BN-600 implemented a section-modular scheme of a sodium-water steam generator. The damage of the heat exchange surface of the BN-600 steam generator occurred mainly in the initial period of plant operation (1980-1985).
The calculations were performed using two codes designed to analyze the efficiency of the steam generator protection system in case of "small" leaks and the secondary circuit protection system against overpressure in "intermediate" and "large" leaks. The use of two calculation codes made it possible to simulate the operation of the BN-600 steam generator protection system in case of a water leak in the steam generator, taking into account its leak evolution from "small" to "large".
The SLEAK code was used to calculate the readings of the control devices for "small"leaks in the BN-600 steam generator: IVA-1 –control of hydrogen in sodium, mounted at the outlet of each steam generator section; KAV-7 –control of hydrogen in gas, mounted on the expansion tank; ITI and ISHIT are systems detecting gas phase appearance in the sodium flowing through the SG relief pipelines and the sodium sampler line to IVA-1.
Readings of the BN-600 "large"leak monitoring devices were obtained by the LLEAK-3C code such as: the pressure sensor in the expansion tank and magnetic flow meters mounted at the outlet of the steam generator sections.

Country/Int. organization

Russian Federation

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Presenter: MIAZDRIKOVA, Olga (Scientific Centre of the Russian Federation –Leypunsky Institute for Physics and Power Engineering, Joint-Stock Company" (IPPE JSC))

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **59** Type: **ORAL**

The Severe Accident Management of the high-power SFR with loss of the heat removal from the core

Wednesday, April 20, 2022 2:28 PM (12 minutes)

Country/Int. organization

Russian Federation

Primary authors: PAKHOMOV, Ilia; Mr KAMAEV, Aleksei

Presenter: PAKHOMOV, Ilia

Session Classification: 2.3 Accident Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **61** Type: **ORAL**

FEASIBILITY STUDY OF HETEROGENEOUS TRANSMUTATION OF AMERICIUM IN FAST REACTORS

Thursday, April 21, 2022 1:40 PM (12 minutes)

The most dangerous of the minor actinides is americium. Transmutation of external americium in the fuel of a fast reactor is possible when its content is over than 1% heavy atoms, however the lower content of an americium, on the contrary, it will accumulate. But curium isotopes with a high heat release are formed from it, complicating the unloading of spent assemblies. Therefore, the content of americium in the fuel should not exceed 1% (which corresponds to the equilibrium state and actually closes the possibility of transmutation of external americium), and the retaining time such fuel in the in-reactor storage should be at least 2 years.
Many researchers believe that heterogeneous transmutation in separate assemblies or blankets is preferable. However, the concentration of americium transmutation products in a small number of burnout assemblies will lead to a manifold increase in the residual heat release in them, and the discharge of such assemblies from the reactor will become very problematic.
Heterogeneous transmutation in the blankets of devices with a strong moderator (zirconium or yttrium hydride) seems to be more rational. Theoretically, this method makes it possible to convert all loaded americium into fission products in one campaign, eliminate the need for multiple handling of it and its transmutation product - curium, and also eliminate the problem of high residual heat release. In this way, all "own" americium, which is formed in the fast reactor, can be converted into fission products.
At the same time, it seems economically feasible to burn out in fast reactors not americium itself, but its predecessor, 241Pu. This is possible due to the use of "fresh" plutonium from VVER spent fuel, which will allow reducing the annual production of americium by almost 2 times without developing expensive technologies.
An unpleasant feature of neptunium transmutation is the formation of the plutonium-236 isotope, which decays into uranium-232 and then into a whole series of high-energy gamma emitters. Therefore, burning out neptunium in fuel should be recognized as inexpedient, and burning it out in the same irradiation devices with moderator as americium seems to be the most preferable.

Country/Int. organization

Russian Federation

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Session Classification: 3.3 Reprocessing, Partitioning, and Transmutation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **62** Type: **ORAL**

Dutch Thermal Hydraulic Design and Safety Support for LMFRs

Thursday, April 21, 2022 10:52 AM (12 minutes)

Liquid metal fast reactor have a prominent role in the roadmap of the Dutch nuclear stakeholders. As nuclear service provider in the Netherlands, the Nuclear Research and consultancy Group (NRG) has established an elaborate program on liquid metal thermal hydraulics. This paper describes the thermal hydraulic design and safety support activities of NRG. The paper will start with the development of tools to allow thermal hydraulic system analyses. For this, the SPECTRA code, under development at NRG, has been adapted to facilitate to application of various liquid metals. The paper will provide a short description of the tool and show some examples of liquid metal applications. Since liquid metal reactors typically employ large liquid metal pools in which 3-D effects are unavoidable, a generic multi-scale modelling approach is being developed coupling the SPECTRA code to CFD codes. The paper will update the reader on the progress. Gradually, the paper will zoom in on more detailed analyses employing CFD codes for liquid metal pools, including important components like pumps, heat exchangers, and the core. For the core, past and ongoing activities will be shown related to validation of CFD approaches based on assemblies as they are designed on the drawing board, but also application and where possible validation of deformed and blocked fuel assemblies. Finally, the fundamental activities on understanding and pragmatic engineering model development for turbulent heat flux will be presented.

Country/Int. organization

Netherlands

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Presenter: ROELOFS, Ferry (NRG)

Session Classification: 6.3 Multiscale and Multiphysics Calculations

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **63** Type: **ORAL**

Fabrication and reprocessing of mixed uranium-plutonium nitride fuel for reactor BREST

Thursday, April 21, 2022 2:04 PM (12 minutes)

Dense nuclear fuel for fast reactors (FR) is the preferred option. In the Russia, as part of the "PRORYV" project, the development of key technologies of closed nuclear fuel cycle (CNFC) for FR with dense mixed nitride uranium-plutonium fuel (MNUP) is underway. MNUP is a new complex product in the field of nuclear power technologies. CNFC with FR ensures:
– no spent nuclear fuel (SNF) accumulation;
– radiwaste management based on the principles of radiation-equivalence;
– technological support for the non-proliferation regime;
– competitiveness with other large-scale energy technologies.
For industrial implementation FR CNFC on the basis of MNUP fuel an experimental demonstration energy complex (EDEC) is being created at the Siberian Chemical Combine site. It consists of BREST-300 reactor with lead coolant and CNFC facilities. The latter includes a MNUP fuel fabrication/refabrication module (FRM) and SNF reprocessing module (RM). In 2022 it is planned to put into operation the FRM and in 2024 to begin construction of RM of EEDC. At the FRM the technology of carbothermal synthesis of MNUP fuel will be implemented. At the RM the technology of combined (pyro+hydro) and hydrometallurgical reprocessing of FR SNF are under development.
To date, R&D have been conducted to justify the use of MNUP fuel in FR. Over 1000 fuel pins with MNUP fuel have been successfully irradiated in BN-600. A complex of post-reactor studies, including destructive radiochemical studies, has been conducted.
R&D on the reprocessing of MNUP SNF includes the fundamental possibility of using pyroelectrochemical technological operations have been shown. The technical feasibility of the following hydrometallurgy operation has been demonstrated experimentally:
• Voloxidation of the MNUP SNF (recovery > 99.9 % tritium and 98 % 14C);
• Extraction and crystallization refining of U+Pu+Np mixture (purification factor of 5*106);
• Recovery >99.9 % of actinides including Am and Cm;
• Microwave denitration for mixed U-Pu-Np, U-Am, U-Cm oxides preparation;
• Separation 1 g of Am and 0,1 g of Cm;
• Waste vitrification in borosilicate glass in a remotely removable cold crucible.
An integrated system of models and codes for all technological modifications of the non-reactor part of CNFC and for coordinated modelling of heterogeneous processes and phenomena are under development. The system under development uses both existing and newly developed models and codes designed to describe technological processes and apparatuses, nuclear and radiation safety, criteria of ignition, combustion, behaviour of structures and engineering systems under critical loads, etc.

Country/Int. organization

Russian Federation

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Session Classification: 3.3 Reprocessing, Partitioning, and Transmutation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **64** Type: **POSTER**

R&D on recovery and separation of americium and curium under ”Proryv” project

Thursday, April 21, 2022 1:40 PM (2 hours)

New nuclear fuel cycles include reducing the long-term radiotoxicity of nuclear waste by separation and transmutation of long-lived transplutonium elements. Therefore, selective recovery of transuranic elements, especially actinides (III) –americium and curium –from high-level waste generated during spent nuclear fuel reprocessing is an important issue. Processes for extracting americium (III) from PUREX-process raffinates are under development in Europe, USA, Russia, Japan and other countries.
Russia is currently actively working on the separation of americium and curium under the "Proryv" project. A two-stage flowsheet is envisaged, including extraction group recovery of americium and curium, followed by sorption separation of americium and curium. Dynamic tests of different extraction systems (CMPO, TOGDA, Dyp7 in polar fluorinated diluent (F3), UNEX-T) were carried out after intensive laboratory studies. Based on the results of the tests, TODGA - F3 system was chosen to test on the real high-level radioactive waste. "Hot" dynamic test of actinides (III) recovery from PUREX-process raffinates using extraction system TODGA - F-3 was carried out at Production Association "Mayak". No less than 99.9% of americium was extracted from during processing of BN 600 and VVER-440 spent nuclear fuel. The total working time of the test was 70 hours.
Extractants based on asymmetric diglycolamides (DGA) are currently being studied for transplutonium elements extraction, instead of TODGA, will make it possible to use saturated hydrocarbons as diluents and to abandon fluorine containing F-3.
The chromatography separation of Cm and Am from rare earth and transplutonium elements concentrate was tested at the Production Association "Mayak". The used concentrate was obtained during processing of VVER-440 SNF. Around 14 g Cm was allocated. The Cm-Am fraction contained about 4.6 g Cm and about 40 g Am. 65 g of pure Am fraction were obtained.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **66** Type: **POSTER**

The working capacity analysis of boron carbide after two-year operation as an emergency protection material of the fast reactor

Wednesday, April 20, 2022 10:40 AM (2 hours)

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **67** Type: **ORAL**

Verification of SARAX Code for the Transient Analysis of Sodium-cooled Fast Reactor

Thursday, April 21, 2022 11:04 AM (12 minutes)

This paper describes the verification work of SARAX code for the transient analysis of a sodium-cooled fast reactor (SFR). The Advanced Burner Test Reactor (ABTR) benchmark created by Argonne National Laboratory (ANL) was modeled and calculated. The reference core is the 250 MWt sodium-cooled fast reactor, which includes neutronics calculation of the core at the beginning of equilibrium cycle, and also several transient analysis sequence such as ULOF (Unprotected Loss-of-Flow) accident. The SARAX code is a neutronics analysis package developed by the NECP team at Xi'an Jiaotong University and aiming for the advanced reactor R&D. It consists in a cross-section generation code named TULIP, a steady state neutronics calculation code named LAVENDER and a transient analysis code named DAISY. In this paper, the 33-group homogenized cross sections of all materials were generated using TULIP. LAVENDER gave the results of steady state parameters like power distribution, critical control rod position, reactivity coefficients and kinetics parameters. Then, DAISY simulated the transient progress with a space-dependent point-kinetics model and a parallel multi-channel thermal-hydraulics model and gave the results of peaking fuel temperature, cladding temperature and coolant temperature. The simulation of ULOF transients showed that SARAX gave comparable results with the design code of ANL and the SAS4A code, which verified the complete code system for transient calculations of SFR.

Country/Int. organization

China

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Session Classification: 6.3 Multiscale and Multiphysics Calculations

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **68** Type: **ORAL**

MECHANISTIC MODELLING OF AEROSOL EVOLUTION IN AN SFR CONTAINMENT FOLLOWING A HYPOTHETICAL SEVERE ACCIDENT

Wednesday, April 20, 2022 1:52 PM (12 minutes)

Country/Int. organization

India

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Session Classification: 2.3 Accident Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **69** Type: **POSTER**

Modelling of radionuclide release from primary system during a hypothetical severe accident in an SFR

Wednesday, April 20, 2022 1:40 PM (2 hours)

Country/Int. organization

India

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Session Classification: Poster Session

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **70** Type: **ORAL**

Safety Analysis of Small Modular Sodium Fast Reactors in Anticipated Transients Without Scram Scenarios

Wednesday, April 20, 2022 2:52 PM (12 minutes)

Country/Int. organization

China

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Presenter: Mr JIN, Xin

Session Classification: 2.3 Accident Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **71** Type: **ORAL**

Export Potential and Commercialization Conditions of Fast Reactors Considering Non-Proliferation Items

Tuesday, April 19, 2022 4:46 PM (12 minutes)

Country/Int. organization

Russian Federation

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Session Classification: 7.1 Sustainability: Economics, Environment, and Proliferation

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **79** Type: **ORAL**

Codes of new generation –sustainable platform for numerical modeling of installations in the Proryv project

Thursday, April 21, 2022 10:40 AM (12 minutes)

Since 2010, domestically produced software required for design decision making and safety assessment of nuclear power plants with fast reactors has been developed in Russian Federation under "Proryv"project, namely, under one of its subprojects –Codes of New Generation.
As a priority, a task was set and successfully accomplished covering the development of 24 software products addressing various areas: neutronics (Monte-Carlo method –MCU-FR, kinetic approximation –CORNER, diffusion approximation –DOLCE VITA); thermal hydraulics (DNS and LES approach –CONV-3D, RANS and LES approach –LOGOS, channel approximation –HYDRA-IBRAE/LM); thermomechanics and fission products release from fuel rods (BERKUT-U); probabilistic safety analysis (CRISS 5.3); radiation effects of releases (outside the industrial site boundaries –ROM, within the industrial site boundaries –ROUZ, in freshwater bodies –SIBILLA); dynamics of soil and groundwater contamination due to radioactive and chemical substances migration (GeRa); modeling the balance of material and nuclide flows within closed nuclear fuel cycle (VIZART) and others. Special attention was given to the development of integral multiphysics codes for safety assessment of power units –EUCLID and radiation safety justification –COMPLEX. The state-of-the-art mathematical models and effective numerical algorithms are used in the codes. They are developed by cooperation of leading Russian research centers and can be effectively used both on personal computers and high-performance computing systems. The modern approaches to collective software development resulting in a significant improvement of software quality are applied in the process of code development.
By the end of 2020, TRL level of the developed software averaged to 8.5 (TRL level 9 suggests that Rostechnadzor certifies the software, as well as the code is put into production). Large-scale validation on experimental data obtained on operating reactor units (BOR-60, BN-600, BN-800), as well as on unique small-scale experiments focused on investigation of separate phenomena conducted recently in the Russian Federation can be considered as an undoubted advantage of the codes developed.
The codes of new generation are actively used for the safety assessment of nuclear facilities, to teach student in universities, to model benchmarks under IAEA CRP and are considered by other industries as a promising software allowing to address production tasks.
The contribution presents the basics for new generation code development and briefly overviews the state-of-the-art in their development, verification and validation, as well as the plans for their further evolution.

Country/Int. organization

Russian Federation

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Session Classification: 6.3 Multiscale and Multiphysics Calculations

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **80** Type: **ORAL**

A novel method of manufacturing a heavy integrated support ring in fast reactor

Wednesday, April 20, 2022 11:04 AM (12 minutes)

Country/Int. organization

China

Primary authors: Prof. MINGYUE, Sun (Institute of Metal Research, Chinese Academy of Sciences); Dr BIN, Xu (Institute of Metal Research, Chinese Academy of Sciences); Dr BIJUN, Xie (Institute of Metal Research, Chinese Academy of Sciences); Prof. DIANZHONG, Li (Institute of Metal Research, Chinese Academy of Sciences)

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Session Classification: 4.2 Structural, Novel, and Large Components Materials

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **82** Type: **POSTER**

ON MEASUREMENT OF OXYGEN CONCENTRATION IN SODIUM BY MEANS OF PLUG INDICATOR

Wednesday, April 20, 2022 10:40 AM (2 hours)

Country/Int. organization

Russian Federation

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Presenter: LOGINOV, Nikolai

Session Classification: Poster Session

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **83** Type: **ORAL**

First fully adjusted set of parameters for the corrosion product contamination code OSCAR-Na

Friday, April 22, 2022 1:42 PM (12 minutes)

The OSCAR-Na code has been developed during the last decade 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 transfer of metallic elements between steel and sodium is due to dissolution and precipitation at the interface, as well as to diffusion in the steel and through the sodium boundary layer. The key parameters of the transfer model are 1) element diffusion in steel, considered to be enhanced under irradiation 2) element diffusion through the sodium boundary layer 3) element equilibrium concentration in the sodium at the interface and 4) oxygen enhanced iron dissolution rate.

For the first time, a full set of parameters has been evaluated for each element (Fe, Ni, Cr, Mn, Co) as a function of temperature through comparison of simulations with measurements in sodium loops and in sodium fast reactors. Thus, concentration profiles in steel at the interface (local depletion due to preferential release of the mostly soluble elements) and mass losses after 6000 hours of sodium exposure at 538 °C and 604 °C have been correctly simulated for the different elements in the STCL sodium loop, as well as the contamination profile along a PHENIX intermediate heat exchanger for Mn-54, Co-58 and Co-60 after about two years of operation.

These results provide a satisfying calibration of the OSCAR-Na code, which validation domain is for the time being restricted to sodium temperature between 400 °C and 650 °C, sodium velocity higher than 4 m/s and oxygen content in the sodium lower than 5 ppm. The considered steel is supposed to be 316 SS.

This paper presents the values of the different parameters retained in OSCAR-Na modeling. They are compared to published values. The discrepancy between adjusted and published values for element diffusivity in steel (higher in the code) and element solubility in sodium (lower in the code) is discussed. The validation process of the OSCAR-Na code will be pursued to extend the validation domain.

Country/Int. organization

France

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Session Classification: 6.6 Fuel Performance and Material Modelling

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **84** Type: **ORAL**

Current status of development of 3D DNS CONV-3D code: one- and two-phase flow models

Thursday, April 21, 2022 11:40 AM (12 minutes)

In IBRAE RAN in "Codes of New Generation"subproject of "Proryv"project one- and two-phase models are being developed to simulate heat and mass transfer processes in the separate elements of nuclear reactor. Those models are realized in the LES and DNS CONV-3D code.
The one-phase models are based on the algorithms with small scheme diffusion, for which the discrete approximations are constructed with use of finite-volume methods and fully staggered grids. For solving convection problem the regularized nonlinear monotonic operator-splitting scheme has been developed. The Richardson iterative method with FFT solver for Laplace's operator as preconditioner is applied for solving pressure equation. Such approach to the elliptical equations with variable coefficients gives multiple acceleration in comparison with the usual conjugate gradients method. For modeling of 3D turbulent flows both DNS and LES approaches are used.
The one-phase module of CONV-3D code is fully parallelized and has perfect scalability, thus it is effective on high-performance computers such as "Lomonosov"(MSU, Russia). The one-phase module has been validated against data of well-known and just got experimental data for various liquids, including lead and sodium used as coolants, in a wide range of Rayleigh numbers between 10^6 and 10^16, and Reynolds numbers in the range of 10^3 –10^5.
The two-phase models take into account interphase heat and mass transfer, stratification of the two-phase flow and separation of the gas component through the interface using equations of state such as condensed gas and the Noble-Able. The two-phase module in CONV-3D code is fully parallelized and has perfect scalability on a CPU and GPU systems. The algorithm of two-phase module is based on the use of HLL (Harten-Lax-van Leer) and HLLC (Harten-Lax-van Leer-Contac) solvers and two-step MUSCL (Monotonic Upstream-centered Scheme for Conservation Laws) predictor-corrector. The validation base includes experiments in which the heat and mass transfer and sodium boiling in the pipes were investigated.
This contribution presents several examples of code application for solving such problems as flow in fuel assemblies, tubes and ring channels, as well as natural convective flows in the elements of reactor. The results of two-phase flows modeling on the series of tests, including the problem of sodium boiling in a round pipe, are also shown.
In all cases the good agreement of numerical predictions with experimental data has been found, that specifies the applicability of the developed CONV-3D code to solve CFD problems for designing and operating NPPs.

Country/Int. organization

Russian Federation

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Session Classification: 6.3 Multiscale and Multiphysics Calculations

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **85** Type: **ORAL**

Models of the integral EUCLID/V2 code for numerical simulation of severe accidents in a sodium-cooled fast reactor with MOX and MNUP fuels

Thursday, April 21, 2022 1:40 PM (12 minutes)

For the modeling of severe accidents in a sodium-cooled fast reactor coupled multiphysics EUCLID/V2 code is being developed in Russian Federation in Codes of New Generation subproject of "Proryv"project. Multiphysics code allow calculating all relevant processes occurring during severe accident: reactor power change including due to boiling and melting, coolant boiling and dryout, cladding and fuel melting (for MOX fuel), as well as fuel dissociation (for MNUP fuel), movement and solidification of the resulting melt, the formation of a pool of melt, the release and transport of fission products in the reactor and beyond and others.
To simulate thermohydraulic processes HYDRA-IBRAE/LM module is used in the EUCLID/V2 code. This module simulates processes in one- and two-phase coolant flow. For fuel rod behavior modeling the BERKUT module is used. The processes of core damage are represented by the severe accident module SAFR. Also EUCLID/V2 code contains the DN3D neutronics module and the AEROSOL-LM module for calculation of the FP transport in the reactor facility and in the containment compartments. MCU-FR module based on Monte-Carlo method is used to estimate the secondary criticality of the core configuration during severe accident.
In the contribution the brief description of each module is given as well as the algorithms used to make the computational grids of all modules to be consistent and for modules coupling. Some results of the EUCLID/V2 V&V calculations are also presented.

Country/Int. organization

Russian Federation

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Presenter: Dr USOV, Eduard (IBRAE RAN)

Session Classification: 6.4 Simulation Tools for Safety Analysis

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **86** Type: **ORAL**

France-Japan Collaboration on Thermodynamic and Kinetic Studies of Core Material Mixture in Severe Accidents of Sodium-Cooled Fast Reactors

Friday, April 22, 2022 2:30 PM (12 minutes)

In the framework of the current implementing arrangement on France-Japan collaboration on Sodium-cooled Fast Reactors (SFRs) from 2020 to 2024, the R&D tasks called "Thermodynamic and Kinetic Studies of Core Material Mixture"is intended to improve models on material interactions at thermodynamic equilibrium and kinetics of reactions for use in severe accident simulation codes with experimental data production. This task includes experimental study on chemical interaction between (U,Pu)O2, B4C and stainless steel (SS), high temperature thermodynamic and thermophysical properties of complex mixtures studies, and modelling of mixtures thermodynamics and liquefaction kinetics.
The previous implementing arrangement (2014-2019) was organized in two phases:
• The first phase (2014-2015) focused on the comparison between French and Japanese thermodynamic databases using the Calphad method. The French and Japanese teams developed a modelling approach for the severe accident simulation code SIMMER based on possible accidental scenarios during the reactor degradation.
• In the second phase (2016-2019), CEA and JAEA has conducted thermodynamics studies, experimental programs to support core material mixture modelling, and the development of models for severe accident simulation code.
These collaborative tasks were successfully accomplished and continuous R&D items were identified.
Based on the previous collaboration, CEA and JAEA defined the following sub-tasks under the current implementing arrangement:
 Kinetics of interaction in core material mixtures,
 Physical properties of core material mixtures,
 High temperature thermodynamic data for the UO2-Fe-B4C system,
 Experimental studies on B4C-SS kinetics and B4C-SS eutectic material relocation (freezing) with/without sodium,
 B4C/SS eutectic and kinetics models for SIMMER code systems,
 Methodology for the modelling of mixtures liquefaction kinetics.
This paper describes major R&D results obtained in the France-Japan collaboration under the previous implementing arrangement as well as experimental and analytical roadmaps under the current arrangement.

Country/Int. organization

France

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Session Classification: 5.3 Experimental Programs II

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **87** Type: **ORAL**

The Status of the ALFRED Project

Tuesday, April 19, 2022 2:00 PM (12 minutes)

Country/Int. organization

Italy

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Session Classification: 1.1 Overviews and Fundamentals of Fast Reactors

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **88** Type: **ORAL**

DIGITAL TECHNOLOGIES FOR PROJECT DEVELOPMENT ODEC AND PEC AND DIGITAL TWINS

Friday, April 22, 2022 10:30 AM (12 minutes)

In the conditions of the modern international market, not only the safety of nuclear facilities, but also their economic performance is critically important. When developing its facilities, the project direction "Proryv" tries to solve these problems comprehensively on the basis of new reactor technologies and closing the nuclear fuel cycle. Given the high degree of novelty of projects, the following key difficulties arise:
• the need for a large number of participating organizations to participate in the project, which use in their work heterogeneous information, calculation and modeling tools that require integration;
• significant uncertainty with the way of achieving the final results, a large amount of R&D performed, the results of which constantly cause changes in the projects of objects.
In addition to the standard set of modern CAD and engineering software used in the industry for the development of NPP projects, a comprehensive digital solution has been developed and applied to ensure that the specified economic indicators are achieved in compliance with all safety requirements for the development of ODEC and PEC projects. This solution includes:
• unified information space - a set of databases, data transmission channels, hardware and software and methodologies that ensure the joint work of project participants, common information services for private projects and integration of IT systems of participating organizations;
• information models of objects-a continuously updated structured set of electronic data and documents about objects and technologies of project direction, necessary and sufficient at each stage of the life cycle;
• integrating projects and consolidated 3D models of objects that provide visual navigation of objects, link documentation with the requirements management system of projects, as well as 4D models for handling equipment of ODEC and PEC objects;
• calculation complexes based on integrated mathematical models that allow for advanced construction and simulation modeling of objects in various modes of operation –normal operation, violations of normal operation and emergency, which is necessary for the development and testing of automated control systems, search and elimination of collisions on technological parameters.
In fact, integrating projects with calculation complexes based on integral computational mathematical models are digital twins of ODEK and PEC objects, accompanying real objects at all stages of the life cycle.

Country/Int. organization

Russian Federation

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Presenter: FEDOROVSKII, Andrei

Session Classification: 6.5 Integrated Analysis and Digitalization

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **89** Type: **ORAL**

EXPERIMENTAL CAPABILITIES OF THE RESEARCH REACTOR FACILITY MBIR. MAIN AREAS OF THE RESEARCH PROGRAMME IN THE INTERESTS OF THE GENERATION 4 REACTORS

Wednesday, April 20, 2022 2:40 PM (12 minutes)

Country/Int. organization

Russian Federation

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Session Classification: 5.1 Experimental Reactors and Facilities

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **91** Type: **ORAL**

Progress in system thermohydraulic code HYDRA-IBRAE/LM models development for fast reactor simulation

Wednesday, April 20, 2022 2:52 PM (12 minutes)

The system thermohydraulic code HYDRA-IBRAE/LM is designed for the simulation of non-stationary thermohydraulic processes in liquid metal and water circuits of fast reactors under normal operating conditions, anticipated operational occurrences and accidents. The code uses a two-fluid model in all flow regimes except for dispersed annular flow, where a three-fluid model is applied. Besides advanced mathematical models, the code has advanced pre- and postprocessor and utility for performing multivariative calculations, uses MPI and OpenMP parallelization. The code is being developed in "Codes of New Generation"subproject of "Proryv"project.
New models described in the present paper expand the code applicability and reduce the degree of conservatism in the nuclear power plants safety assessment.
One direction of the code development is the improvement of dispersed phase transport model. The new model accounts for the flow parameters time dependence and allows performing correct calculation of the bubble diameter necessary for the description of heat transfer and interfacial friction. The interfacial area density equation describes its dynamics and determines bubble diameter in one-group approximation. A more sophisticated heterogeneous multi-group model allows describing bubble dynamics and processes of separation and stratification.
The improved post dryout model was implemented for water coolant. Post dryout flow is treated as steam-water annular two-phase flow. In considered conditions deposition of droplets from the flow core on the hot tube wall takes place. The new model describes post dryout heat transfer in the pressure range 3-16 MPa with high accuracy.
The turbine model was developed and implemented in the code. The model describes the thermal-hydraulic processes taking place as the energy of superheated steam transforming into mechanical work. The model is based on the universal thermodynamical relations and self-consistently calculates enthalpy and pressure drops at the turbine stages. This approach allows determination of the thermal-hydraulic parameters of the coolant (temperature, density, pressure, enthalpy, entropy, mass fraction of water) in all elements of the turbine for a given load.
The work was performed on enabling the description of water behavior at supercritical parameters which is extremely important for correct modeling of the experiments where supercritical water was used.
Performed validation calculations made it possible to refine correlations used in the closure relations.

Country/Int. organization

Russian Federation

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Session Classification: 6.2 Thermal Hydraulics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **92** Type: **ORAL**

Status of Generation-IV Lead Fast Reactor Activities

Tuesday, April 19, 2022 1:48 PM (12 minutes)

Country/Int. organization

Italy

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Session Classification: 1.1 Overviews and Fundamentals of Fast Reactors

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **93** Type: **ORAL**

SOCRAT-BN INTEGRAL CODE: DEVELOPMENT, VALIDATION AND CURRENT STATUS

Thursday, April 21, 2022 3:04 PM (12 minutes)

Integral computer codes SOCRAT-BN have been developed at the Nuclear Safety Institute of the Russian Academy of Sciences (IBRAE RAN) in the frame of the Federal Target Program «New-Generation Nuclear Power Technologies for the Period 2010–2015 and up to 2020». The first version SOCRAT-BN/V1 was developed for the period 2010-2014 to simulate design basis (DBA) and beyond design basis accident (DBDA) at a nuclear power plant with sodium fast reactor (SFR). In 2016, the first version of the code was certified by "Scientific and Engineering Centre for Nuclear and Radiation Safety"(SEC NRS). For 2014-2017, the second version of the code, SOCRAT-BN/V2, was developed. It had extended the first version to severe accident with core melting. The second version of the code was certified in 2019.
Currently, SOCRAT-BN is used for safety assessment of operating and projected SFR for Russian power plants. Also, it is planned to use SOCRAT-BN for supporting projects to be constructed abroad.
The physical models of the SOCRAT-BN are divided into two blocks: a steady state and a transient one. The steady state block is applied to simulate the accumulation of fission products and the state of the fuel elements for the period of time that the reactor operates before an emergency event. The transient block is applied to simulate the temperature state of the reactor, the transfer of fission products before release to the environment, deformation of fuel and cladding, neutronics processes during the destruction of the core, melting and movement of fuel.
The report represents description of the basic code physical models, its validation and the current state of the code.

Country/Int. organization

Russian Federation

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Session Classification: 6.4 Simulation Tools for Safety Analysis

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **94** Type: **ORAL**

Mechanistic code BERKUT-U: self-consistent modeling of fuel rods thermomechanical behavior and processes in the fuel of fast breeder reactors

Friday, April 22, 2022 2:30 PM (12 minutes)

The BERKUT-U mechanistic fuel performance code has been designed at Nuclear Safety Institute of the Russian Academy of Sciences (IBRAE RAN) since 2012 in frame of "Codes of new Generation" subproject of "Proryv"project. The code is intended for self-consistent computational simulation of the stress-strain state and temperature distribution in fuel rods with nitride or oxide fuel, with a gas or liquid metal sublayer under irradiation in fast reactors with liquid metal coolant under normal and transient operational conditions.

The BERKUT-U code has a modular structure, the main modules of the code are:

• Thermophysical module –simulates the distribution of heat fluxes and temperatures inside a fuel rod with known sources of heat release and heat transfer conditions at the border «external surface of the fuel rod cladding-to-coolant».

• Thermomechanical module –simulates the evolution of the stress-strain state of fuel pellets and fuel rod cladding, predicts the mechanical state of the cladding and fuel rod performance.

• Fuel module MFPR –self-consistently simulates a set of processes occurring in fuel: microstructural changes and swelling, production and radioactive transformations of fission products, their intragranular and intergranular migration, accumulation and release of gaseous fission products under the fuel rod cladding, distribution of fission products and fuel components over molecular and phase states, takes into account the influence of fission products on the thermophysical properties of the fuel.

• Module describing the fuel-to-cladding gap –simulates the redistribution of radioactive fission products over the phase (condensed and gaseous) states and their transfer along the fuel-to-cladding gap, for the gas of fuel-to-cladding gap calculates the thermal conductivity of the gas mixture depending on the gas composition, for the liquid metal sublayer simulates the dissolution of the cladding and the transfer of corrosion products in liquid lead or sodium.

• Database module of thermophysical and mechanical properties of fuel rod materials –calculates the mechanical and thermophysical properties of materials of fuel rod of fast reactors and issues them upon request of all code modules.

The validation of the BERKUT-U code carried out over the past three years on the data of post-irradiation studies of about fifty fuel rods indicates that the developed fuel performance code BERKUT-U makes it possible to reliably predict the behavior of fuel rods under irradiation in fast reactors. The some of the calculation results obtained in comparison with experimental data are represented in the contribution.

Country/Int. organization

Russian Federation

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Session Classification: 6.6 Fuel Performance and Material Modelling

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **95** Type: **ORAL**

Aerosol module for modeling of the fission product behavior in FR cooling circuits and NPP compartments

Thursday, April 21, 2022 2:04 PM (12 minutes)

This contribution presents an overview of models of an aerosol module designed to simulate the behavior of fission products in the circuits and compartments of nuclear power units with fast reactors with sodium or lead coolants. Aerosol module AEROSOL-LM is included in the thermal-hydraulic HYDRA-IBRAE/LM code. Together they represent a unified code with a common interface for calculating the processes of thermal-hydraulics and the fission products transport both in gaseous and aerosol forms. The AEROSOL-LM module allows calculating the relevant processes of aerosol dynamics: nucleation, coagulation, condensation and sedimentation. A specific feature of the module is the simulation of multicomponent and polydisperse aerosols.
In particular, for sodium reactors the behavior of sodium combustion aerosols in NPP compartments is simulated. For lead cooled fast reactors the oxygen transport, the formation and behavior of corrosion particles are considered. The aerosols formation and transport between rooms of the NPP including those resulting from melt-concrete interaction are also modeled by the aerosol module.
The results of module validation are also briefly presented in the contribution.

Country/Int. organization

Russian Federation

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Session Classification: 6.4 Simulation Tools for Safety Analysis

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **96** Type: **POSTER**

Calculation of the materials activation with BPSD code

Friday, April 22, 2022 1:30 PM (2 hours)

The isotopic kinetics code BPSD is developed by IBRAE RAN in "Codes of New Generation"sub-project of "Proryv"project. BPSD solves fuel, absorber (boron carbide, dysprosium hafnate) transmutation, coolant (lead, sodium) and steel activation problems. Moreover, it carries out activation and residual heat calculations of materials. BPSD is intended to model materials, applied in fast reactors with sodium and lead coolants and closed nuclear fuel cycle facilities.
BPSD is one of the modules of integral multiphysics EUCLID code used to simulate the liquid metal cooled fast neutron reactor systems under normal operating conditions, anticipated operational occurrences, design basis accidents. Also BPSD is included in the integral code COMPLEX for radiation safety assessment of reactor and nuclear fuel cycle facilities.
The isotopic kinetics problem is solved for cases with the fixed transmutation chains. Each chain accords to its material. Steel chain contains 501 nuclides, lead chain –201 nuclides, boron carbide chain –115 nuclides, dysprosium hafnate –99 nuclides. Chain takes into account the impurities of the materials considered. The two linked transmutation chains (actinide chain and fission products chain) are used in the isotopic kinetic problem solution.
The problem is solved by an iterative method. It enables to calculate any type of transmutation chains and to exclude negative solution appearance. In addition to calculation of the nuclide concentrations, the problem of their uncertainty (caused by input data uncertainty –initial material composition, decay constant, reaction rate) estimation –is solved.
Transmutation chains realized in BPSD code are built on the base of the ROSFOND database. The CONSYST-RF/BNAB-RF system intended to calculate nuclear constants is also used in the code.
The calculation data of the materials irradiation obtained by BPSD code in comparison with the experimental data are presented in the contribution.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **97** Type: **ORAL**

Models of the integral EUCLID/V2 code for numerical modeling of different regimes of lead-cooled fast reactor

Thursday, April 21, 2022 2:40 PM (12 minutes)

The EUCLID/V2 integral multiphysics 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, anticipated operational occurrences, design basis accidents and severe accidents. The EUCLID/V2 code includes the system thermohydraulics module (HYDRA-IBRAE/LM), spatial time-dependent neutronics module (DN3D/CORNER), quasi two-dimensional fuel rod module (BERKUT), the module of burnup and decay heat calculations (BPSD), the module of fission, activation and corrosion products transport in primary loop and gas system of a reactor facility (AEROSOL-LM), the tritium migration module (TRITIUM), the module of fission product source calculation, the fuel rod and core disruption module (SAFR), the modules of mass transfer and fission product transport calculation in the reactor containment compartments (HYDRA-IBRAE/LM or KUPOL-BR), the module of simulation of radiation situation beyond industrial site of a NPP (ROM).
The mentioned modules are multi-purpose, their models do not depend on a coolant or fuel type and may be used to simulate reactor facilities with sodium, lead or lead-bismuth coolant and nitride or oxide fuel. However, some special models needed for behavior simulation of reactor facilities with lead coolant have been implemented into the EUCLID/V2 integral computer code. They are the model of solid phase impurities transport in a primary loop of a reactor facility with heavy liquid metal coolant, the model for a steam generator tube rupture simulation of a reactor facility with the lead coolant, the model of fission product source calculation taking into account physicochemical interaction between the nitride fuel and the lead coolant, the nitride fuel dissociation model, the lead melt and concrete interaction model, the lead freezing model.
At present, the V&V of the listed above modules and models is being carried out on analytical and numerical tests and experimental results.
In the contribution the brief description of the above mentioned models and some V&V results are presented.

Country/Int. organization

Russian Federation

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Session Classification: 6.4 Simulation Tools for Safety Analysis

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **98** Type: **POSTER**

Integral code COMPLEX for radiation safety assessment of reactor and nuclear fuel cycle facilities

Friday, April 22, 2022 1:30 PM (2 hours)

The computational code COMPLEX for radiation safety assessment of reactor and nuclear fuel cycle facilities is a set of programs (modules) combined by exchange data files and a pre- and post-processing system. The code is being developed in the "Codes of new generation" subproject of the "Proryv" project. The code includes the following modules:

- reactor core calculation modules based on Monte Carlo method (MCU-FR), diffusion approximation (DOLCE VITA) and discrete ordinates method (CORNER);
- nuclide kinetics module (BPSD);
- radiation sources calculation module (RASTAS_M);
- radiation shielding calculation modules based on Monte Carlo method (MCU-FR) and finite element method (ODETTA);
- group constants system (CONSYST/ABBN-RF).

Almost all separate modules (except for RASTAS_M) are certified or are completing certification in Rostechnadzor.
The application area of the COMPLEX includes storage and transportation units for fresh and spent fuel assemblies, reactor core, reactor plant, NPP equipment and premises, closed nuclear fuel cycle facilities.
The calculating scenario of the COMPLEX code, in which all modules are used, consists of the following stages:

1. At the first stage, the assembly campaign is simulated (fuel assembly, control rod or other core elements), the neutron flux is calculated;
2. At the second stage, burnup calculations of the fuel, absorber and activation of structural materials are carried out; the call of the modules can be iterative (by burnup steps);
3. At the third stage, the calculation of ionizing radiation sources is carried out for specific materials, assemblies or sets of assemblies;
4. At the fourth stage, the neutron, gamma or coupled neutron-gamma transport problem is solved and the dose equivalent rate is calculated at the detection points.

Thus, due to a wide application area the code COMPLEX is an important and relevant tool for radiation safety assessment of liquid metal fast reactors and closed nuclear fuel cycle facilities

Country/Int. organization

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Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **102** Type: **POSTER**

Irradiation-Thermo-Mechanical Coupling Analysis and Calculation of Fast Neutron Oxide Fuel Element

Wednesday, April 20, 2022 10:40 AM (2 hours)

Country/Int. organization

China

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Session Classification: Poster Session

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **104** Type: **ORAL**

CEFR Physical Start-Up Tests: the Core Specifications and Experiments

Tuesday, April 19, 2022 1:12 PM (12 minutes)

Country/Int. organization

China

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Session Classification: Special Session: IAEA Coordinated Research Projects

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **106** Type: **ORAL**

Material Data Acquisition Activities to Develop the Material Strength Standard for Sodium-cooled Fast Reactors

Wednesday, April 20, 2022 11:16 AM (12 minutes)

Country/Int. organization

Japan

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Presenter: TOYOTA, Kodai (Japan Atomic Energy Agency)

Session Classification: 4.2 Structural, Novel, and Large Components Materials

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **107** Type: **ORAL**

Experimental modeling of a fuel element simulator vibration in a coolant flow

Wednesday, April 20, 2022 2:04 PM (12 minutes)

The elements of the fuel assembly including individual fuel pins are affected by the coolant flow. This can lead to mechanical vibrations. The cyclic loading of the fuel element cladding material accompanying these vibrations causes an additional effect on the fuel element material. This can cause the damage of the fuel cladding material, especially in its contact with the spacer elements. The natural frequency of vibrations of a fuel element in a liquid flow depends on various factors. One of the main ones is the method of fastening the fuel element. Usually, the lower end of the fuel element is rigidly sealed, while the upper one either rests or moves freely in a small gap. The added mass of the liquid has a noticeable effect on the frequency of rods oscillations.
In the contribution the results of experimental study of the oscillations of fuel rod simulators in fuel assembly models with a lead-bismuth eutectic (LBE) coolant flow are represented. Measurements were carried out on a 7 pins model of the fuel assembly. An annular channel with an equivalent diameter equal to the hydraulic diameter of the fuel assembly model was also used for measurements. The Reynolds number in the experiments was varied in the range of 5·103 –4·104. In addition to the lead-bismuth coolant, a water coolant was also used, which made it possible to compare the results and determine the features of the vibrations of the fuel element simulator in liquids with different densities. In addition, for the water coolant, the distribution of the averaged and pulsating component of the liquid velocity was measured both along the axis of a single fuel element simulator and behind its head.
During the experiments, the following parameters were controlled and measured:
- temperatures of various elements;
- temperature of the coolant at the inlet and outlet of the test section;
- coolant flow rate at the test section inlet;
- profiles of displacements (pulsations) of the fuel element simulator.

As a result the data on the dynamic stability of the fuel element simulator in lead-bismuth and water coolants, data on the regularities of vibrations of the fuel element simulator, depending on the method of its fastening, the length and flow rate of the coolant, were obtained.

Country/Int. organization

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Session Classification: 6.2 Thermal Hydraulics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **108** Type: **POSTER**

Simple Design Comparison of uranium nitride pin cell assembly and matrix fuel assembly for a Lithium Cooled Fast Reactor

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

China

Primary author: PINEDA, Raul (IAEA FELLOW)

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Presenter: PINEDA, Raul (IAEA FELLOW)

Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **110** Type: **POSTER**

Experience of Using CFD Models for Development of High-Temperature Furnace Equipment for Fabrication of Mixed Nitride Uranium-Plutonium Fuel Pellets

CFD modeling was extensively used for the development of high-temperature furnaces for the carbothermal synthesis of uranium and plutonium nitrides and a furnace for the sintering of mixed nitride uranium-plutonium fuel pellets. This equipment is intended for use at the Pilot Demonstration Energy Complex (PDEC) being constructed in Seversk, Russia. The CFD-model of the carbothermal synthesis furnace was developed with the SolidWorks Flow Simulation software to obtain a three-dimensional temperature distribution both inside the furnace and in the bulk material charged, as well as typical gas flow patterns and a gas velocity distribution throughout the furnace. The CFD model was verified using experimental data on the temperature profile at three points inside the furnace measured during heating, isothermal exposure, and cooling within a temperature range from 20 to 1650 °C during acceptance tests of the manufactured equipment. The CFD model was used to verify the engineering solutions selected and formulate recommendations on operation modes of the furnace. In particular, the modeling results demonstrated a wide range of process parameters, such as the heater temperature, the gas temperature and flow rate, that ensure a temperature of 1650±50 °C throughout the bulk material required for the carbothermal synthesis of uranium and plutonium nitrides. It is shown that the maximum difference in temperature throughout the bulk material does not exceed 62 °C.

A horizontal pusher-type sintering furnace was developed, wherein mixed nitride uranium-plutonium fuel pellets successively move through heating, sintering, and cooling zones with different gas media. The CFD model of the furnace channel was developed with the Ansys Fluent software and underwent benchmark testing on a specifically built bench for gas-dynamic investigations using a full-size channel model. The engineering solutions were proved to ensure the sustainable operation of three gas zones (argon-nitrogen-argon) in the furnace channel at a sintering temperature of ~1950 ℃.

Application of the CFD models reduced the time of developing the high-temperature furnace equipment and facilitated the justification of the engineering solutions. The developed models allow simulations of various operation modes including possible emergencies and will be used to support the operation of the high-temperature furnaces.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **112** Type: **ORAL**

Implementation of LFR Experimental Infrastructures in Romania

Friday, April 22, 2022 2:06 PM (12 minutes)

Romania through RATEN ICN is deeply involved in the development and implementation of ALFRED Demonstrator being the reference site for ALFRED construction. One of its roles is to demonstrate the effectiveness of the nuclear option as a reliable component of any sustainable energy scenario of the future.
The ALFRED Project aims to the development, up to the full demonstration, of the LFR technology, one of the most promising Gen-IV concepts being significantly safe, sustainable, economically competitive, and not-proliferant.
The project will gather the existing centre of excellence on lead technology located in Italy with a new one, to be realized in Romania, where the infrastructures presently missing in the European landscape, but required for supporting the design of an LFR, are planned to be constructed.
Six new research infrastructures will be built or are under construction on RATEN ICN site: ATHENA (full-scale testing of the components, assessment of systems behavior in a pool configuration, etc.), ChemLab (coolant and cover gas chemistry, auxiliary systems development), HELENA2 (multi-purpose - pump, valves, sub/assemblies and erosion/corrosion investigations in lead), ELF (long-running system tests (endurance)), MELTIN'Pot (fuel-(clad)-coolant interaction) and HANDS-ON (core simulator for S/As manipulation and handling tests).
The purpose of the LFR experimental infrastructures is:
o to support ALFRED licensing process (demonstration of the complete control of the phenomena, qualification of the materials, component, equipment, validation and verification, etc.);
o to use the infrastructure to find the solution for the open issues;
o to create the skills and competences for lead technology;
o to explore beyond the frontiers of the field, synergies with other fields.
The ALFRED infrastructure is suitable to investigate the key points related to the heavy liquid metals and to support the technological development of the LFRs, while
the demonstrator itself will support the qualification of materials subject to fast spectrum neutron irradiation in a representative environment.
FALCON Consortium (created in 2011 by RATEN-ICN (Romania) together with ENEA and ANSALDO NUCLEARE (Italy) holds the skills to carry out safety studies, licensing as well as planning and execution of experimental campaigns in support of LFR technology development.
ALFRED infrastructure is fully integrated in an experimental roadmap expected to integrate the ALFRED reactor and to support the development of the LFR technology, beyond the construction of the demonstrator itself, towards the deployment of a commercial fleet.

Country/Int. organization

Romania

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Presenter: GUGIU, Daniela (RATEN ICN)

Session Classification: 5.3 Experimental Programs II

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **116** Type: **ORAL**

Phénix Control Rod Withdrawal test analysis using a multiphysics methodology

Thursday, April 21, 2022 11:16 AM (12 minutes)

Before the definitive shutdown of the Phénix reactor, a series of end of life tests were performed in 2009 and 2010, by CEA (Commissariat à l'Energie Atomique et aux Energies Alternatives), EDF (Électricité de France) and AREVA. The main objectives were to enlarge experimental database for the research and design of Sodium cooled Fast Reactors (SFR). Due to this important opportunity, the IAEA (International Atomic Energy Agency) decided to establish a Coordinate Research Project from 2007 to 2011 to stimulate computational codes validation among different countries involved in fast reactors development. In this context, a benchmark was established on the "static Control Rod Withdrawal Test"(CRW) with the objective of the investigation on the flux and power local deformations related to different control rod insertions in the core. Such valuable experimental data are useful to improve calculation schemes used to analyze control rod withdrawal transient, which could potentially trigger a core melting accident in a SFR. The objective of the current study was to perform multi-physics simulation based on a loosely coupled approach to take into account local Doppler feedback effect on power deformations. The probabilistic particle transport code Serpent 2 (VTT Technical Research Centre of Finland, Ltd), associated with the JEFF-3.1 nuclear library, was chosen as reference neutron calculation code and was coupled to an in-house static thermal-hydraulic solver. Firstly, purely neutron transport calculations were done in order to build-up and check the overall core model. The uncoupled results were compared to benchmark results already published and show a correct accordance with experimental and calculated results providing that a heterogeneous description of control rods was used. A convergence study to estimate the required precision level of neutron calculations with respect to multiplication factors and power estimations was also performed. Secondly, coupling between neutron and thermal-hydraulics solvers were done through the Serpent 2 multiphysics interface with a regular exchange of the main coupled parameters such as fuel temperatures and neutron deposited powers for each axial node of each subassemblies of the fissile core. The coupled results on the power deviation are globally slightly nearer to the experimental ones than uncoupled results but are affected by probabilistic uncertainties and batch-to-batch inter correlation problems responsible for light power oscillations with respect to the number of simulated neutrons instead of a straight convergence.

Country/Int. organization

France

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Presenter: Ms FORNO, Barbara (CEA Cadarache)

Session Classification: 6.3 Multiscale and Multiphysics Calculations

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **119** Type: **ORAL**

HETEROGENEOUS BURNING OF MINOR ACTINIDES IN A FAST REACTOR

Thursday, April 21, 2022 2:16 PM (12 minutes)

Transmutation of minor actinides (MA) into stable or short-lived ones by their irradiation in reactors will alleviate the problem of long-term activity of spent nuclear fuel (SNF), increase the efficiency of nuclear fuel due to energy produced by MA fission, and also accumulate and produce useful radionuclides. The economic efficiency of closed-cycle nuclear power cannot be achieved without MA disposal and safe final isolation of radioactive waste.
Fast reactors are the most suitable for homogeneous MA transmutation and heterogeneous MA burning.
With homogeneous transmutation, MA in a small amount (less than 5%) is introduced into the standard nuclear fuel. With this approach, MA will be both burnt and accumulated from MA introduced into nuclear fuel, as well as from uranium and plutonium of standard nuclear fuel. During repeated recycling of such nuclear fuel, its nuclide composition stabilizes and MA accumulation rate is compared with their decrease rate, and the equilibrium SNF nuclide composition is reached.
The concept of heterogeneous MA burning involves their inclusion into inert matrices (no uranium and plutonium) and placement in separate fuel assemblies (fuel rods) either in the fast reactor core or blanket.
Heterogeneous MA burning in the fast reactor blanket has a more flexible strategy for MA handling than homogeneous MA transmutation and can be used to achieve high MA burning with minimal effect on reactor characteristics. The use of inert matrices will avoid the formation of secondary MA.
In Russia, there is a unique opportunity for MA transmutation in existing fast reactors (BN-600, BN-800). Therefore, a technology for MA separation from SNF and production of fuel with MA should be developed, and scientific research and reactor experiments should be performed.

Country/Int. organization

Russian Federation

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Presenter: TUZOV, Alexander (JSC "SSC RIAR")

Session Classification: 3.3 Reprocessing, Partitioning, and Transmutation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **122** Type: **POSTER**

Hybrid high power fast breeder reactor with metallic fuel and additives consisting with lightweight atoms

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

Russian Federation

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Presenter: DROBYSHEV, IURII (Russian Federation)

Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **123** Type: **ORAL**

PRE-DESIGN OF A PASSIVE DECAY HEAT REMOVAL SYSTEM WITH A PHASE CHANGE MATERIAL FOR SMR-SFR

Wednesday, April 20, 2022 11:16 AM (12 minutes)

Country/Int. organization

France

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Presenter: Mr PANTANO, Alessandro (CEA)

Session Classification: 2.2 Safety Design and Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **124** Type: **ORAL**

Uranium and mixed uranium-plutonium nitrides thermal stability

Thursday, April 21, 2022 12:04 PM (12 minutes)

The thermogravimetric method was used to study the behavior of uranium nitride and mixed uranium-plutonium nitride (MNIT) in a helium flow and a helium with nitrogen gas mixture at temperatures up to 2173 K. When heated in helium in the low-temperature range (˂1773 K), a mass loss was found, which amounts to hundredths of a percent. In this case, mass loss occurs in 2 stages, accompanied by the release of nitrogen and it is not associated with the decomposition of uranium or plutonium mononitrides. It has been shown that sintered nitride fuel pellets may contain several percent of uranium sesquinitride U2N3, which decomposes in this range. Nitride fuel pellets were heated in a gas mixture of helium with nitrogen to study the formation of higher nitrides. In the case of uranium mononitride this led to the formation of uranium sesquinitride U2N3 in the temperature range of 673-723 K. However, upon further heating (>1173 K), U2N3 decomposes again to uranium mononitride in 2 stages. The sequential formation and decomposition of uranium sesquinitride led to the destruction of the sample. At the same time multiple heating of the MNIT fuel (U0.79Pu0.21)N in the helium-nitrogen gas mixture does not lead to the formation of U2N3. It is also shown that the partial pressure of nitrogen at its content of 5 vol.% in the helium flow significantly exceeds the equilibrium partial pressure of nitrogen over the samples of uranium nitride and MNIT fuel in the entire test temperature range, which inhibits the decomposition of uranium mononitride up to 2173 K. However, in the case of MNIT fuel at a temperature >1773 K a clearly observed mass loss on the thermogravimetric curve occurs. Therefore, even in an atmosphere containing nitrogen, it was not possible to suppress the decomposition process of the MNIT fuel.

Country/Int. organization

Russian Federation

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Session Classification: 3.2 Development of innovative fuels: design and properties irradiation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **126** Type: **ORAL**

Development of methodology to evaluate mechanical consequences of vapor expansion in SFR severe accident transients: lessons learned from previous France-Japan collaboration and future objectives and milestones

Friday, April 22, 2022 10:30 AM (12 minutes)

In the frame of France-Japan collaboration, one of the objectives is to define and assess the calculation methodologies, and to investigate the phenomenology and the consequences of severe accident scenarios in sodium fast reactors (SFRs). A methodology whose purpose is to assess the loadings of the structures induced by a Fuel Coolant Interaction (FCI) taking place in the sodium plenum of SFR has been defined in the frame of the collaboration between France and Japan during 2014-2019. The work progress will be spread over the period 2020-2024 and the main objectives and milestones will be introduced in this paper. The objective of studies is to comprehensively address the margin between the limit of integrity of the main vessel structures and the loadings resulting from severe accidents.
For this purpose, the SIMMER mechanistic calculation code simulates core disruptive accident sequences in SFRs. However, SIMMER cannot be used for main vessel loading assessment while it does not take into account fluid structure interactions. That is the reason why, associated with SIMMER code, a fluidstructure dynamics tool evaluates this interaction i.e. EUROPLEXUS is used in CEA studies and AUTODYN tool is used in JAEA studies. In this paper, a benchmark study is described in order to illustrate the evaluation of vapour expansion phase in the hot plenum. To do that, joint input data are used on the basis of an ASTRID 1500 MWth core degraded state after the power excursion which leads to vapour expansion. The most penalizing case was evidenced in this study by suppressing the action of transfer tube in-core mitigation devices in SIMMER input deck and thus privileging the upward molten core ejection. Since the risk of main vessel failure by cumulative stresses is an issue often discussed in the SFR concepts, the calculation methodology presented in this paper based on chaining of SIMMER code with another Fluid/structure evaluation tool is very promising. The future perspectives are highlighted.

Country/Int. organization

France

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Presenter: BACHRATA, Andrea (CEA)

Session Classification: 2.4 Severe Accidents

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **127** Type: **ORAL**

iMAGINE - a Breakthrough Technology for Closing the Fuel Cycle without Reprocessing

Friday, April 22, 2022 12:06 PM (12 minutes)

The energy trilemma and UN-SDG 7 are drivers for energy research to support the UK governments net-zero emissions law. Nuclear reactors are a highly attractive candidate for reliable, 24/7 available, low-carbon electricity generation. However, current technology reactors and their related fuel cycles suffer from unreasonably high cost, a lack of sustainability, and a waste problem due to the absence of recycling. Molten salt technologies will be a key step into the future of nuclear supporting a disruptive way of optimizing the whole nuclear system to enhance sustainability and affordability. Main advantage, compared to existing technologies, is elimination of complex solid fuel production and fuel cycle technologies. Molten salt systems will be a breakthrough for most efficient fuel use, by operating on existing spent fuel while drastically reducing the cost of nuclear and solving the long-term waste problem. However, developing a disruptive, highly sustainable and affordable fuel cycle –instead of just a reactor –requires a strong inter-disciplinary approach, linking physics, engineering, and chemistry.
Primary key is to deliver the essential step into any new reactor technology: a zero-power facility to research the game-changing technology in safe settings, to advance knowledge and capabilities in the technology to grow the skills base in the UK. Core activity is improving simulation and demonstration of innovative control and safety features to allow a qualified response to regulatory requests and to support the formation of the required skilled workforce to support BEIS, aiming to achieve an industrially demonstrated, market ready product in 2050.
The proposal pushes the breakthrough technology delivering significantly improved sustainability indices, characterized by:
• avoiding mining (major source of eco-toxicity, carbon emissions, and cost) & avoiding enrichment (major energy consumption, proliferation-risk, and cost)
• reducing waste production and storage demand by the reuse of existing spent fuel & eliminating highly-radiotoxic transuranium isotopes (reducing the final disposal challenge)
• eliminating reprocessing (proliferation risk, prohibitively expensive prior step for closed fuel cycle) & solid fuel production (major cost driver and radiation source in closed fuel cycle)
• replacing reprocessing with demand driven salt clean-up & applying low pressure technology
The ultimate aim is to prepare the UK for a net-zero future using highly-innovative technologies. The impact of the proposed technology and the attractiveness of the vision is evidenced by the rapid take-up through the major industrial technology developers, including Terrestrial Energy, Terrapower, Elysium Industries⋯.
An overview on the research plan will be given.

Country/Int. organization

United Kingdom of Great Britain and Northern Ireland

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Presenter: Prof. MERK, Bruno (University of Liverpool)

Session Classification: 1.3 System Innovations

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **128** Type: **ORAL**

TECHNICAL AND ECONOMICAL FEATURES OF COMMERCIAL SODIUM FAST REACTOR IN FRANCE

Tuesday, April 19, 2022 3:34 PM (12 minutes)

Country/Int. organization

France

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Presenter: Mr SETTIMO, DAVID (EDF)

Session Classification: 7.1 Sustainability: Economics, Environment, and Proliferation

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **129** Type: **ORAL**

APPLICATION OF DIGITAL TWIN OF FAST REACTOR PLANT FOR CONTROL SYSTEM ALGORITHM TESTING

Friday, April 22, 2022 10:54 AM (12 minutes)

Enhancement of efficiency of state-of-the-art computation systems and software modernization make it possible to elaborate computer model development technology to the technology of supercomputer twin development. The reactor plant supercomputer twin implies three levels of detailing:
- The upper level model permitting consideration of effects of non-isothermal coolant flows, distribution of power density fields in the core, neutron flux expansion to ionization chambers. The model is implemented based on integration of 3D codes by means of special software and permits to perform multi-physical computations of different operation modes of reactor plants.
- The medium level model intended to perform dynamic analyses and considering effects obtained at multi-physical computations. It is implemented on the basis of associated neutron and thermal-hydraulic analysis controlled by the control system model.
- The real-time model intended to adjust and optimize control system algorithms of the reactor plant. It is implemented based on data obtained from the developed models of upper levels.
Super-computer multi-processor computations and application of state-of-the-art 3D computation codes, thermal-hydraulic and neutronic models permit to use this tool at all life cycle stages:
- at the design stage to justify normal operation modes, modes with equipment failures and emergency modes; adjustment of control algorithms of equipment and systems;
- at the stage of Automated Process Control System development to master at the test facility and to test equipment;
- at the stage of operation, as a tool for training of operators and to trace plant-prototype condition.
This paper presents description of digital twin development technology of the fast reactor plant and its application for control system algorithm testing.

Country/Int. organization

Russian Federation

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Presenter: USHATIKOV, Anton

Session Classification: 6.5 Integrated Analysis and Digitalization

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **131** Type: **ORAL**

BOR-60 REACTOR OPERATING EXPERIENCE, WORK ON IMPROVING SAFETY AND EXTENDING LIFETIME

Tuesday, April 19, 2022 4:58 PM (12 minutes)

Country/Int. organization

Russian Federation

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Session Classification: 8.1 SFR Commissioning, Operation, and Decommissioning

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **132** Type: **ORAL**

Comparison of calculation methods for lead cooled fast reactor reactivity effects

Wednesday, April 20, 2022 10:40 AM (12 minutes)

Country/Int. organization

Hungary

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Session Classification: 6.1 Neutronics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **133** Type: **POSTER**

Two-Component Energy Industry under Conditions of Closed Nuclear Fuel Cycle: Economic Benefits

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

Russian Federation

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Presenter: ROSLAYA, Mariia

Session Classification: Poster Session

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **134** Type: **ORAL**

Multi-criteria comparison of the efficiency of minor actinides burning in different nuclear reactors based on the INPRO/IAEA KIND approach

Thursday, April 21, 2022 2:40 PM (12 minutes)

This paper presents a comparison of the efficiency of minor actinides (MA) burning in various type of nuclear reactors with a fast neutron spectrum. A set of criteria for comprehensive comparison of reactor technologies, based on the INPRO/IAEA KIND approach to multi-criteria assessment, has been prepared. This set of criteria includes indicators in such areas as the efficiency of MA burning, economics, safety, environment, readiness of reactor technology and infrastructure for its implementation. The evaluation and comparison procedure was carried out using the KIND-ET tool. It is shown that a comprehensive multicriteria analysis of various aspects of the technologies, as expected, led to estimates that differ from the approach in which technologies are compared only single criterion and without taking into account the influence of other equally important factors. And the cumulative assessment of technologies largely depends on the set development objectives. This means that each of the listed options can take the first place in the rating when certain priorities are selected.
Keywords: multi-criteria evaluation, sodium fast reactors, lead fast reactors, MSR, technology comparison, project KIND, IAEA, INPRO.

Country/Int. organization

Russian Federation

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Presenter: KVIATKOVSKII, Stepan (SC Rosatom)

Session Classification: 3.3 Reprocessing, Partitioning, and Transmutation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **135** Type: **POSTER**

Neutronic Calculation of CEFR Core using Different Nuclear Data Libraries

Friday, April 22, 2022 1:30 PM (2 hours)

The uncertainities of evaluated nuclear data represent one of the most important sources of uncertainity in the reactor physics simulation. The improvement of these data used is requiered for the development, safety assesment and licensing process of a reactor. Is generally recognised the need for further investigation (experimental included) regarding the uncertainities on some main cross-section (e. g. 238U, 242Pu, minor actinides etc.).
The paper deals with the investigation of keff discrepancies induced by the differences among the cross-sections from ENDF/B-VIII.0, JEFF-3.3 and JENDL-4.0 libraries. For this study, a benchmark neutronic calculations for the first criticality of China Experimental Fast Reactor core configuration have been performed using the Continuous-energy Monte Carlo Reactor Physics Burn-up Calculation Code - SERPENT 2, version 2.1.31. The reactor reached the first criticality for a load of 72 fuel subassemblies at cold state (250℃±5℃) with only one regulating rod inserted at a certain position; all other control rods have been withdrawn out-of-core position.
The results of SERPENT 2 code show a relatively large variation in the keff values obtained with different libraries, as following: ENDF/B-VIII.0 library yields excess reactivity of 98 pcm while JEFF3.3 and JENDL-4.0 yield excess reactivity of 243 pcm and 627 pcm, respectively.

Country/Int. organization

Romania

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Presenter: MOISE, Andreea (Institute for Nuclear Research Pitesti)

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **136** Type: **ORAL**

DEVELOPMENT OF BN REACTOR TECHNOLOGY IN RUSSIA

Tuesday, April 19, 2022 1:00 PM (12 minutes)

Country/Int. organization

Russian Federation

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Presenter: SHEPELEV, Sergey (JSC «Afrikantov OKBM»)

Session Classification: 1.1 Overviews and Fundamentals of Fast Reactors

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **138** Type: **ORAL**

The role of pyrochemical processing in a NetZero economy in the UK

Thursday, April 21, 2022 1:52 PM (12 minutes)

The programme described here is part of a UK government investment in Nuclear to be delivered by a collaboration of UK Government, The UK National Nuclear Laboratory (NNL), Industry and Academia. The programme will contribute to international understanding of development and demonstration requirements for pyro-processing as well as emerging applications for salt separations enabling a broad base of future fuel recycling. This paper will outline the programme and present early findings.

Electrorefining as an actinide separation technique offers potential to provide useful fissionable material as part of a closed fuel cycle. It is specifically designed for irradiated metallic fuel, but other fuel types can be processed with pre-treatment; it uses a molten chloride salt electrolyte such as a eutectic mixture of lithium chloride and potassium chloride. The production of a baseline flowsheet model and demonstration of key engineering aspects such as scalability and in-line process monitoring will raise the technology readiness level of pyro-processing technology in the UK.

Basic molten salts data capture is limited in the UK, data will be generated by NNL and university partners through post-doctoral research programmes and physical properties and materials data captured in a centralised database. This will not only be used longer term to develop potential future facilities but is being used directly to inform other areas of the wider pyro-processing project such as the flowsheet development and pilot-scale demonstrator work packages.

A lab-scale Pu-active pyro-processing rig being developed providing a key UK capability. This rig will provide new data on the behaviour of Pu and other components in the molten salt. It will complement the concept design of a pilot-scale rig is being developed to investigate the engineering practicalities of operating a full-scale pyro-processing plant such as scalable pumping systems, instrumentation and control (process and accountancy), materials of construction etc. this will also allow us to assess the impact of introducing a fluoride-based salt matrix.

The activities and outputs from the Salt Engineering and Salt Science activities will enable NNL to evaluate, on behalf of the UK, the case for recycle in advanced fuel cycles at a broad level and set out the potential drivers or switching points between advanced aqueous and pyro-processing technologies. The results will inform the direction of any follow-on project through the development of a future UK roadmap in pyro-processing and identify opportunities for international collaboration and leveraged investment.

Country/Int. organization

United Kingdom of Great Britain and Northern Ireland

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Presenter: Dr EDMONDSON, Michael (The National Nuclear Laboratory)

Session Classification: 3.3 Reprocessing, Partitioning, and Transmutation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **140** Type: **POSTER**

Sustainability of nuclear and non-nuclear power generation options under Russian conditions: a comparative evaluation study

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

Russian Federation

Primary authors: ANDRIANOV, Andrei (INPE NNRU MEPhI); KVIATKOVSKII, Stepan (SC Rosatom); Mr PTITSYN, Pavel (SC Rosatom, Private Enterprise "Science and Innovations", Center of Analytical R&D); Mr KUPTSOV, Ilya (SC Rosatom, Private Enterprise "Science and Innovations", Center of Analytical R&D)

Presenter: KVIATKOVSKII, Stepan (SC Rosatom)

Session Classification: Poster Session

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **141** Type: **ORAL**

Treatment of sodium of Superphenix Fast Breeder Reactor

Tuesday, April 19, 2022 3:58 PM (12 minutes)

Country/Int. organization

France

Primary authors: VILLANI, Dominique (Framatome); Mr ROSSAT-MIGNOD, Pascal (EDF); Mrs PIGNON, Isabelle (Cyclife Engineering)

Presenter: VILLANI, Dominique (Framatome)

Session Classification: 8.1 SFR Commissioning, Operation, and Decommissioning

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **142** Type: **ORAL**

Effect of Reactor Technology on Economics of SMR Projects

Tuesday, April 19, 2022 4:10 PM (12 minutes)

Country/Int. organization

Russian Federation

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Presenter: ZHURAVLEV, Ilya

Session Classification: 7.1 Sustainability: Economics, Environment, and Proliferation

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **143** Type: **ORAL**

ON THE POSSIBILITY TO CHANGE THE ISOTOPIC COMPOSITION OF PLUTONIUM FROM THE SPENT MOX FUEL OF PWRs IN FAST REACTORS

Friday, April 22, 2022 1:30 PM (12 minutes)

The purpose of this article is to investigate the use of fast reactors for changing the isotopic composition of Pu for a better reuse in thermal reactors. The possibility to change or adjust (the term «improve» can also be used) the isotopic composition of Pu from MOX SNF of thermal reactors is determined by the fundamental characteristic of fast reactors –their capability for nuclear breeding, and is directly related to the conversion ratio (CV). It is obvious that in order to be effective for the whole two-component nuclear power, the change in Pu isotopic composition should not cause shortage of plutonium for fast reactors themselves. Therefore, fast reactors should have a high enough CV as well as fertile blankets. The only current technology to meet these requirements is that of fast sodium cooled reactors. Single or multiple recycle of plutonium from thermal reactors in the fast reactor fuel, use of uranium-235 or special (target) assemblies with reduced plutonium content are the possible ways to achieve the purpose in BN-800. This article investigate the possibility of large-scale Pu improvement in commercial fast BN-type reactors with increased conversion ratio. The perspective of an experiment on the BN-800 reactor that would let one demonstrate this possibility in the experimental way is also discussed. Measurable parameters of the change in the isotopic composition of plutonium in BN-800 are discussed in the present report.

Country/Int. organization

Russian Federation

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Presenter: Prof. GULEVICH, Andrey (IPPE)

Session Classification: 3.4 Advanced Fuel Development

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **146** Type: **ORAL**

EFFECTIVE FUEL SUPPLY OF TWO-COMPONENT NUCLEAR ENERGY SYSTEM WITH VVER-BN REACTORS

Tuesday, April 19, 2022 4:34 PM (12 minutes)

Country/Int. organization

Russian Federation

Primary authors: MAROVA, Elena (JSC "Afrikantov OKBM"); SHEPELEV, Sergey (JSC «Afrikantov OKBM»); BAKANOV, Mikhail (Rosenergoatom Concern JSC); Mr TEPLOV , Pavel (JSC "Rosenergoatom Concern"); GULEVICH, Andrey (SSC RF-IPPE); TROYANOV, Vladimir (JSC "Rosenergoatom Concern"); Mr DEKUSAR , Viktor (JSC "SSC RF IPPE"); KLINOV , Dmitrii; Mr KOROBEYNIKOV , Valerii (JSC "SSC RF IPPE"); ALEKSEEV, Pavel (NRC "Kurchatov Institute"); NEVINITSA, Vladimir (National Reseach Center "Kurchatov Institute"); Mr FOMICHENKO, Peter (NRC Kurchatov Institute)

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Session Classification: 7.1 Sustainability: Economics, Environment, and Proliferation

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **147** Type: **ORAL**

Study on actinide conversion capabilities of Molten Salt Reactors (MSR)

Wednesday, April 20, 2022 12:04 PM (12 minutes)

Country/Int. organization

France

Primary authors: MESTHIVIERS, Laura (LPSC/IN2P3/CNRS); HEUER, Daniel (LPSC/IN2P3/CNRS, Grenoble INP, UGA); MERLE, Elsa (LPSC-IN2P3-CNRS / Universite Grenoble Alpes)

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Session Classification: 6.1 Neutronics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **148** Type: **ORAL**

THERMOHYDRAULIC TESTS IN JUSTIFICATION OF DESIGN CHARACTERISTICS OF THE BREST-OD-300 RP STEAM GENERATOR

Thursday, April 21, 2022 10:52 AM (12 minutes)

In order to substantiate the design characteristics of the steam generator of the BREST-OD-300 reactor plant (RP) developed at NIKIET JSC, IPPE JSC carried out thermohydraulic tests of various models of the lead-heated steam generator. Initially, to confirm the design characteristics and thermal-hydraulic stability at the parameters of the nominal, partial and start-up modes, a model of a twisted steam generator was tested, consisting of two three-tube modules with a longitudinal lead flow around a bundle of heat transfer tubes. The influence of operating parameters on thermohydraulic characteristics and hydrodynamic stability is shown in the case of operation of one module, as well as in the joint operation of two models in the investigated range of operating parameters.
At the second stage, tests of the standard model of the BREST-OD-300 RP steam generator were carried out with lead flowing around 18 heat exchange tubes. The model consisted of two collectors, each of which included a bundle of nine heat transfer tubes. Data were obtained on the hydrodynamic stability of steam generating tubes and the entire model as a whole when operating in the entire range of change in operating parameters, necessary for creating a databank and further verification of calculation codes describing the ongoing thermohydraulic processes. The boundary of thermohydraulic stability has been experimentally confirmed.
The tests of the standard model, as well as the model of a steam generator with a longitudinal flow of a lead coolant, were carried out in a wide range of changes in operating parameters. The feed water pressure varied from 16.5 to 18.7 MPa (on the model with longitudinal media flow, tests were carried out at a supercritical pressure of 24.3 to 25.7 MPa), the water flow through the heat transfer tube varied from 10 to 120% of the nominal value. The feed water temperature varied from 340 to 350 ℃, the lead temperature at the model inlet varied from 390 to 536 ℃, and the lead consumption varied from 10 to 100% of the nominal value.
A series of works devoted to the study of heat transfer from the lead coolant with a transverse flow around a package of heat transfer tubes has been completed. A model with a transverse lead flow around the steam-generating tubes has been created, on which studies were carried out on the effect of the oxygen concentration in lead on heat transfer in normal heat transfer modes.

Country/Int. organization

Russian Federation

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Session Classification: 5.2 Experimental Programs I

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **150** Type: **ORAL**

Advanced Reactor Experiments for Sodium Fast Reactor Fuels (ARES) Project: Transient Irradiation Experiments for Metallic and MOX Fuels

Thursday, April 21, 2022 11:28 AM (12 minutes)

Advanced Reactor Experiments for Sodium Fast Reactor Fuels (ARES) is a joint project between the U.S. Idaho National Laboratory (INL) and the Japanese Atomic Energy Agency (JAEA) to investigate the transient fuel performance of metallic and MOX fuels. The project has the specific goals of experimentally evaluating the transient failure modes of high burnup metallic and MOX fuels, guided by advanced modeling and simulation (M&S) tools, but also to support development and validation of M&S tools. The recent availability of the Transient Reactor Test (TREAT) facility provides the opportunity to renew in-pile evaluation of advanced fuel designs. Sodium Fast Reactor (SFR) experiments leverage a rich inventory of fuels irradiated in EBR-II and FFTF, currently residing at INL, and still supporting research programs at the INL. As part of the ARES project, a heat-sink capsule with liquid metal specimen bond has been designed and M&S is being used to develop the detailed experimental conditions for the planned experiments. A series of fresh fuel commissioning tests is planned in TREAT for 2021. These tests will evaluate the performance of the hardware and instrumentation to measure temperature response, fuel elongation, and fuel failure. The four irradiated fuel tests are planned for fully intact, high-burnup pins (2x metallic pins and 2x MOX pins), which achieved nearly ~13 at% burnup in EBR-II, to be irradiated in TREAT in 2022. Test instrumentation will include optical-fiber-based distributed temperature sensors, thermocouples, acoustic emission detectors, a capsule pressure sensor, self-powered neutron detectors (SPND) as well as the refurbished TREAT fuel motion monitoring system (hodoscope). Fresh fuel tests will incorporate additional advanced diagnostics including optical-fiber-coupled pyrometry and fuel elongation sensors. The experiments will benefit from advanced pre- and post-transient examination capabilities available at INL and actively used for similar examinations. Long-term test development underway includes a full circulating sodium loop and the ability to refabricate fuel specimens. This paper will present more detailed description of the planned test design, matrix, and conditions with an emphasis on modeling guiding experiment design.

Country/Int. organization

United States of America

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Session Classification: 3.2 Development of innovative fuels: design and properties irradiation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **151** Type: **ORAL**

The fluid structure interaction of narrow gaps between thin-wall coaxial structures in fast reactors

Friday, April 22, 2022 2:06 PM (12 minutes)

In order to protect the key equipment from high temperature in fast reactor, main pumps and main vessel is shielded by single or multiple hot screens, forming narrow fluid gaps. However, these fluid gaps bring some difficulties in seismic analysis by introducing the fluid structure interaction effect. Added mass, a simplified but important parameter of fluid structure interaction effect, is much larger than the structure's mass itself especially when the gap between two coaxial cylinders is narrow. Moreover, the 2D beam-model based added mass formula generally used in engineering design is over conservative, making the structure burden large extra mass, however, there is no specific added mass guideline for such thin-wall and narrow-gap structure available. To study the fluid structure interaction of main pumps and main vessel with its hot screens, a series of dynamics/seismic experiments of coaxial cylinders with different gap sizes and height-radius ratios are carries out. The fluid pressure and acceleration distribution of such structures under different modal shapes are measured. A data processing method is established to transfer experimental results to added mass. Finally, the correlation between added mass and parameters like gap sizes and height-radius ratios are obtained, which can be useful in structural assessment of key equipment with fluid structure interaction effect.

Country/Int. organization

China

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Session Classification: 6.6 Fuel Performance and Material Modelling

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **152** Type: **ORAL**

Modelling of postulated reactivity insertion in a Generation IV Molten Salt Reactor

Wednesday, April 20, 2022 12:04 PM (12 minutes)

Country/Int. organization

France

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Session Classification: 2.2 Safety Design and Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **155** Type: **ORAL**

The SAIGA in-pile experimental program to qualify the SIMMER calculation tool in SFR Severe Accident Conditions

Friday, April 22, 2022 10:54 AM (12 minutes)

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 a SFR fuel bundle with molten fuel discharge device which are planned to be housed in the IGR reactor (Impulse Graphite Reactor operated by NNC-RK). The purpose of the SAIGA program is to qualify the SIMMER computer code on the SEASON platform based on one in-pile test conducted with two fuel sub-assemblies and one discharge tube representative for an in-core mitigation device dedicated to severe accident situations. This test should be representative, as much as possible, for the phenomena encountered during Unprotected Loss-Of-Flow Total (ULOF) severe accident sequences.
A sodium loop will be built and connected to the in-pile experimental device to drive the sodium flow inside the fuel pin bundle with experimental conditions close to the SFR nominal conditions before triggering of the ULOF sequence. Over accidental transient period with sodium flow reduction, the constant neutron heating from IGR reactor will lead to degrade a first sub-assembly (16 Kazakh fuel pins by sub-assembly) to produce some molten fuel material and the propagation of this degraded fuel will be followed by fine instrumentation towards both a second sub-assembly and a discharge tube.
For this scenario, the feasibility study defined the main characteristics of the experimental device and the operating conditions for the test to be conducted in the IGR reactor. The purpose of this study was to assess the capacity of the IGR reactor to provide the necessary neutron flux during the transient, to demonstrate the capacity to carry out on-line or post-test measurements of the variables of interest, and to assess the schedule of one test incorporating the safety file. Also, the sodium loop feeding the test device and its instrumentation were studied and their feasibility demonstrated.

Country/Int. organization

France

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Session Classification: 2.4 Severe Accidents

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **156** Type: **ORAL**

Potential Role of Fast Reactors with Heterogeneous Fuel Assembly in Development Nuclear Power Structure

Wednesday, April 20, 2022 11:28 AM (12 minutes)

Country/Int. organization

Russian Federation

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Session Classification: 3.1 Fuel Cycle Scenarios

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **157** Type: **POSTER**

Controlled thermonuclear fusion: potential role of a joint (Th-U-Pu) nuclear fuel cycle

Thursday, April 21, 2022 1:40 PM (2 hours)

This paper aims at finding solutions of so important problems of nuclear power as decreasing the scope and the number of technological operations, as well as enhancing the proliferation resistance of fissile materials in nuclear fuel cycle by means of minimal changes in the cycle. The method is including fusion neutron sources with thorium blanket into future nuclear power system. In addition to production of light uranium fraction consisting of 233U and 234U, high-energy 14-MeV neutrons emitted in the process of fusion (D,T)-reaction can generate 231Pa and 232U through (n,2n)- and (n,3n)-reactions.

It has been demonstrated that admixture of 231Pa into fresh fuel composition can stabilize its neutron-multiplying properties thanks to two well-fissile consecutive isotopes 232U and 233U, products of radiative neutron capture by 231Pa. Coupled system of two well-fissile isotopes can allow us to reach the following goals: the higher fuel burn-up and, as a consequence, the longer fuel lifetime; the shorter scope and the lower number of technological operations in nuclear fuel cycle; the better economic potential of nuclear power technologies. Such a fuel cycle presumes shifting from 235U to 233U as more attractive fuel material for thermal nuclear reactors. Uranium component will be protected from unauthorized proliferation by the presence of light uranium isotope 232U. The use of well-mastered traditional uranium-based fuels in power LWR will be preserved. The idea suggests fresh fuel fabrication for power LWR without applications of isotope separation technologies.

Country/Int. organization

Russian Federation

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Presenter: Dr KULIKOV, Evgeny (National Research Nuclear University MEPhI)

Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **158** Type: **POSTER**

On substantial slowing down of the kinetics of fast transient processes in fast reactor

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

Russian Federation

Primary authors: Dr KULIKOV, Gennady (National Research Nuclear University MEPhI); Prof. SHMELEV, Anatoly (National Research Nuclear University MEPhI); Dr APSE, Vladimir (National Research Nuclear University MEPhI); Dr KULIKOV, Evgeny (National Research Nuclear University MEPhI)

Presenter: Dr KULIKOV, Evgeny (National Research Nuclear University MEPhI)

Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **159** Type: **POSTER**

Investigation of characteristics of fast power reactor with an additional function of large-scale production of plutonium-238

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

Russian Federation

Primary authors: Prof. SHMELEV, Anatoly (National Research Nuclear University MEPhI); Dr GERASKIN, Nikolay (National Research Nuclear University MEPhI); Dr KULIKOV, Gennady (National Research Nuclear University MEPhI); Dr KULIKOV, Evgeny (National Research Nuclear University MEPhI); Dr APSE, Vladimir (National Research Nuclear University MEPhI); Dr GLEBOV, Vasily (National Research Nuclear University MEPhI)

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Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **161** Type: **POSTER**

Transmutation efficiency of minor actinides in fast-and thermal-spectrum molten salt reactors

Thursday, April 21, 2022 1:40 PM (2 hours)

Long-lived minor actinides (MA) like, Neptunium, Americium, and Curium are the major burden of nuclear power. Long-lived MAs have not yet been used as nuclear fuel. Therefore, the transmutation of long-lived MAs is introduced as an alternative to direct final disposal. In current work, we compare the performance of MA transmutation in a critical Single-fluid Double-zone Thorium-based Molten Salt Reactor (SD-TMSR) and a Small Molten Salt Fast Reactor (SMSFR). We study the dynamic of Keff and core reactivity with different MA loads, shift of the neutron spectrum, time evolution of MA and basic nuclides inventory that affect the core stability, as well as the transmutation coefficient (TC). The TC of long-lived MA is calculated using the Monte Carlo code SERPENT-2. The total neutron flux in SD-TMSR and SMSFR can reach 4.1x1014 and 1.8x1015 n/cm2s, respectively. The results show that SD-TMSR consumes about 50% of the generated Pu isotopes in the fuel salt; however, SMSFR consumes about 86.5% of the generated Pu isotopes. During burnup, we apply online reprocessing and refueling, so the core remains critical, and the total mass of fuel in the core and blanket is practically constant. The results show that both reactors efficiently transmute 237Np, 241Am, 243Am and 243Cm, while SMSFR has a higher TC than SD-TMSR. TC of total MA reaches 54.84% and 87.97% in SD-TMSR and SMSFR, respectively.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **162** Type: **POSTER**

Revealing the dependencies of partitioning americium-241 and uranium using sorption technology based on solid-phase extractant TODGA

Thursday, April 21, 2022 10:40 AM (2 hours)

The objective of this work is to reveal the dependences of the separation of americium-241 and uranium using sorption technology based on the solid-phase extractant TODGA. In technologies for the purification of radioactive waste of low and medium activity levels with low contents of actinides, sorption and ion exchange methods are widely used due to their high selective. The required selectivity level, under extreme external conditions of the environment, is inherent in solid-phase extractants obtained using ligands. To extract americium-241 efficiently and separate from uranium, it is necessary to use solid-phase extractants based on N, N, N ', N' - tetraoctyldiglycolamide (TODGA), which will allow the return of fissile materials to the nuclear fuel cycle and reduce the hazard class of the removed radioactive waste. For the effective use of experimental modified TODGA sample, it is necessary to identify the dependences of the extraction of americium-241 and its separation with uranium using model solutions that simulate real radioactive waste. Therefore, the determination of the kinetic parameters of the extraction of americium-241 and its separation with uranium in the process of sorption processing using experimental modified TODGA samples is necessary for the most efficient separation process. In this work, the kinetic parameters and diffusion coefficients of americium-241 and uranium in the process of their sorption on experimental TODGA samples were determined. The kinetics of the process was investigated with the determination of the reaction rate, which was analyzed based on the dependence of the change in the concentration of the extracted element from the volume upon contact with the extractant on time, which depends on the value of the diffusion coefficient. The estimation of the reaction rate dependence on the value of the diffusion coefficients is carried out. Based on the dependence of the diffusion coefficients on the characteristics of the TODGA prototypes, the increased rate of sorption for americium-241 in comparison with uranium was determined by calculation. According to calculations of the diffusion coefficients of americium-241 and uranium in the experiments carried out with the samples under study, a prototype TODGA with the highest kinetic characteristics was determined.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **163** Type: **ORAL**

Verification and validation of neutronic codes using the start-up fuel load and criticality tests performed in the China Experimental Fast Reactor

Tuesday, April 19, 2022 1:24 PM (12 minutes)

Country/Int. organization

Mexico

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Session Classification: Special Session: IAEA Coordinated Research Projects

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **164** Type: **ORAL**

Postirradiation characterization of AFC metallic fuel alloys concepts.

Friday, April 22, 2022 1:42 PM (12 minutes)

A long-term objective of the Advanced Fuels Campaign (AFC) is the investigation into enabling technologies that allow for improving nuclear fuel performance and for the transmutation of minor actinides in sodium fast reactor. As part of this development, candidate fuel compositions and forms are irradiated in a cadmium-shrouded positions at the INL's Advanced Test Reactor (ATR), and they are subsequently examined at the Material Fuel and Complex (MFC) facilities. In addition to and complementary to ATR experiments, systematic characterization of experiments irradiated in true fast reactors (e.g. Phenix, EBR-II, FFTF) are also performed in order to assess ATR simulated fast reactor testing effectiveness.
Recent irradiation experiments have explored new alloys and geometric forms beyond what has historically been irradiated (U-10Zr / U-20Pu-10Zr, 75% smeared density, sodium bonded fuel) to overcome primary limiting performance factors, such as: high swelling rate at higher burnup (design tested: annular fuel, low smear density, alternative alloying metals) and fuel cladding chemical interaction (FCCI) from lanthanides fission products (design tested: additives, liners / coating) and to assess the performance of adding minor actinides (Am, Np) to the metallic fuel systems.
The minor actinides behavior has been assessed through a suite of characterization techniques on three transmutation metallic fuel experiments and focus was concentrated on microstructural evolution under irradiation. Metallic fuel samples were taken from sibling experiments AFC-1H (experiment in ATR) and DOE1 FUTURIX-FTA (experiment in Phenix) both comprising metallic fuel alloys of 35U-29Pu-4Am-2Np-30Zr composition. In addition, samples were analyzed from a unique transmutation experiment in EBR-II, X501, with a fuel composition of U-20.2Pu-9.1Zr-1.2Am-1.3Np.
Regarding the innovative fuel design, engineering scale postirradiation examination have been performed to understand the overall behavior of this new fuel form at relatively low burnup (between 2-4 %FIMA) from various irradiation experiments in ATR (AFC-3 and -4 series). The main alloys studied were U-10Zr annular fuel, U-10Mo, U-10Zr with Pd addition.
Postirradiation examinations at engineering and microstructural scale for transmutation metallic fuel alloys and from AFC-3 series experiments will be presented and discussed in this work.

Country/Int. organization

United States of America

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Presenter: CAPRIOTTI, Luca (Idaho National Laboratory)

Session Classification: 3.4 Advanced Fuel Development

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **165** Type: **POSTER**

ADS for Energy Production and 233U breeding in HEU-Thorium Oxide system

Friday, April 22, 2022 1:30 PM (2 hours)

For power production and 233U breeding from thorium, a preliminary neutronic design of an Accelerator-Driven Sub-critical System (ADS) is presented. The ADS reactor core design with "HEU–Thorium Oxide fuel"was coupled with proton accelerator and spallation target. The neutron source (ADS system) feasibility of HEU burning and isotopes production was evaluated. The multiplication factor Keff, the production of 233U and depletion of 235U were computed using the MCNPX 2.7.0 code. The results indicated that the introduction of thorium fuel with HEU into the ADS core gives an efficient method to produce 233U isotopes and to burn 235U isotopes more efficiently. Additionally, less minor actinides (MA) production and generation of energy can be achieved.

Country/Int. organization

Egypt

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Presenter: Mr ALI, Ahmed (EGYPTIAN ATOMIC ENERGY AUTHORITY)

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **166** Type: **ORAL**

The solution of nuclide kinetic equation for fast reactor in the OpenBPS code with options of choosing calculation method and uncertainties analysis.

Wednesday, April 20, 2022 11:52 AM (12 minutes)

Country/Int. organization

Russian Federation

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Presenter: BUKHTIIAROV, Ivan

Session Classification: 6.1 Neutronics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **167** Type: **ORAL**

TRAINING OF NEW GENERATION SPECIALISTS IN THE FIELD OF FAST NEUTRON REACTORS AND NUCLEAR FUEL CYCLE CLOSURE

Thursday, April 21, 2022 1:40 PM (12 minutes)

The Strategy is based on the formation of a two-component nuclear energy based on a closed nuclear fuel cycle with fast neutron reactors.
The solution of long-term tasks of creating a two-component nuclear power with a closed NFC based on fast reactors is associated with the need to create a system for training, attracting and developing young professionals based on new principles.
Replenishment of the industry with young qualified personnel is a fundamental task that allows us to maintain the strategic direction of development.
Russia has a strong scientific and educational engineering and physical school. Russian nuclear education is one of the most advanced in the world. Currently, the formation of a two-component nuclear energy based on a closed nuclear fuel cycle with fast reactors poses for the profile educational system the task of training a new generation of researchers based on interdisciplinary knowledge, with fundamental training and practical skills. In addition, a new generation of specialist researcher should have a broad scientific outlook and modern management skills in science (principal investigator, etc.).
In order to solve the large-scale scientific and technological problems associated with fast reactors and closed NFC, a system of long-term planning for training qualified personnel, attracting and continuous development is also necessary, based on close cooperation of the industry with specialized universities.
The report discusses a comprehensive approach to the preparation and implementation of a program for the development of scientific and technical competencies for two-component nuclear power with a closed NFC based on fast neutron reactors. The key components of the program are considered, including the formation of a target plan for the training and retraining of personnel, improving the quality of educational programs, developing network interaction with specialized universities, and international cooperation in the field of education. The report describes the experience of NRNU MEPhI in the creation and implementation of interdisciplinary educational programs aimed at training a new generation researchers, as well as creating conditions for the development of university internationalization and the export of nuclear education.

Country/Int. organization

Russian Federation

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Presenter: Mr TIKHOMIROV, Georgy (NRNU MEPhI)

Session Classification: 9.1 Education, Profesional Development, and Knowledge Management

Track Classification: Track 9. Education, Profesional Development, and Knowledge Management

Contribution ID: **168** Type: **POSTER**

Transmutation of minor actinides in a fast reactor with uranium-curium fuel

Thursday, April 21, 2022 1:40 PM (2 hours)

As a result of the operation of nuclear reactors, a certain amount of Cm is produced, which is included in the minor actinides series (MA). Among the long-lived Cm isotopes, Cm243 and Cm245 should be noted. Their fission cross section is over 2.5 barn. In this regard, Cm can be used as a fuel in a fast neutron nuclear reactor.
For the scientific research, was used a model of the RBEC reactor (a fast natural circulation reactor with a lead-bismuth coolant), developed at the Kurchatov Institute (Moscow, Russia).
(U + Cm)N was used as fuel. Uranium - waste uranium with an enrichment of 0.1% in the isotope U235. The efficiency of different approaches to the placement of MA in fuel (homogeneous and heterogeneous) was considered. This was for the transmutation of Cm and other elements from the minor actinides series.

Country/Int. organization

Russian Federation

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Presenter: Ms TEREKHOVA, Anna

Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **174** Type: **POSTER**

Study on Sodium Fire PSA Methodology for Pool-Type Sodium cooled Fast Reactor

Wednesday, April 20, 2022 1:40 PM (2 hours)

Country/Int. organization

China

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Presenter: WANG, Jing (China Institute of Atomic Energy)

Session Classification: Poster Session

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **175** Type: **ORAL**

FABRICATION AND PERFORMANCE ASSESMENT OF ODS FECRAL CLADDING TUBE

Tuesday, April 19, 2022 1:24 PM (12 minutes)

Country/Int. organization

China

Primary author: Prof. LIU, Shi (Institute of Metal Research, China Academy of Sciences)

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Presenter: Prof. LIU, Shi (Institute of Metal Research, China Academy of Sciences)

Session Classification: 4.1 Advanced Reactor Cladding and Core Material, Coolants, and Related Chemistry

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **178** Type: **ORAL**

The δ-ferrite transformation behavior and mechanical properties of 316H weld metal during high temperature service

Wednesday, April 20, 2022 11:28 AM (12 minutes)

Country/Int. organization

China

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Presenter: WEI, Shitong

Session Classification: 4.2 Structural, Novel, and Large Components Materials

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **179** Type: **POSTER**

ELABORATION OF THE THERMAL-HYDRAULIC CHARACTERISTICS OF THE REACTOR PLANT BASED ON THE OPERATION EXPERIENCE OF THE POWER UNIT WITH BN-800 REACTOR

Friday, April 22, 2022 10:30 AM (2 hours)

The analysis and elaboration of the thermal-hydraulic characteristics based on the results the reactor plant (RP) commissioning allow validating the algorithms of passing the modes and sufficiency of the margins applied in the project related to the thermal-hydraulic characteristics of the main equipment.
The paper presents the comparative analysis of the start-up algorithms and operation modes at various power levels of the BN-800 RP of the power unit 4 of the Beloyarsk NPP applied in the project and realized during the commissioning of the unit. There are presented the comparison results of the operation mode parameters and thermal-hydraulic characteristics of the main equipment obtained during operation of the BN-800 RP with the calculated ones using verified and validated software ТР-БН (TR-BN). Based on the summation of the operation data of the BN-800 RP the conclusion regarding sufficiency of the margins applied in the project for the thermal hydraulic characteristics of the intermediate heat exchanger and main circulation pumps of the primary and secondary circuits.
The results of the completed research are used for validation of algorithms of passing the modes during normal operation of the BN RP and operation conditions of the main equipment.

Country/Int. organization

Russian Federation

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Presenter: FADEEV, Ilia

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **180** Type: **POSTER**

Development of in-vessel source term evaluation method for ULOF events in sodium-cooled fast reactor

Wednesday, April 20, 2022 1:40 PM (2 hours)

Country/Int. organization

Japan

Primary author: Mr SONODA, Hiroki (Regulatory Standard and Research Department, Secretariat of Nuclear Regulation Authority (S/NRA/R))

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Presenter: Mr SONODA, Hiroki (Regulatory Standard and Research Department, Secretariat of Nuclear Regulation Authority (S/NRA/R))

Session Classification: Poster Session

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **187** Type: **ORAL**

Experimental and computational studies of heat exchange for liquid metals boiling in fuel assembly models at accidental conditions

Thursday, April 21, 2022 11:04 AM (12 minutes)

The higher level of modeling for dynamic liquid metal boiling is important for comprehensive analysis of neutron-physical and thermohydraulic characteristics of fast reactors cores for safety justification at accidental conditions (UTOP, ULOF).
Experimental data obtained at IPPE JSC have showed that boiling of liquid metals in fuel assemblies of fast reactors has a complex structure which is characterized by both stable and pulsation regimes with significant fluctuations of parameters that can cause a boiling crisis.
Stable nucleate boiling in fuel assembly models is observed only at limited heat fluxes; and transition to unstable bubble-slug boiling regimes is determined by various factors. In an assembly with a low surface roughness of pin simulators, progressing of unstable (slug) boiling with sharp fluctuations of the coolant flow rate and overheating of the pin simulator wall causes the boiling crisis; in fact, there is no any margin before the crisis. For the pin simulators with higher roughness a transition from unstable slug boiling regimes to stable annular-dispersed boiling regimes is observed due to a liquid film on the surface of the pin simulators.
The hydrodynamic interaction of fuel assemblies can result in to considerable increase in the amplitude of fluctuations in the flow rate of the coolant (”resonance” of the flow rate pulsations) and to the “locking” or inversion of the flow rate of the coolant in the loops, increase in the temperature of the coolant and the pin claddings (the effect of interchannel instability) and, finally, to the boiling crisis.
During sodium boiling in a model fuel assembly with a “sodium cavity” located at the top of the reactor core (the cavity is designed to compensate the positive sodium void reactivity effect when boiling occurs) the possibility of prolonged sodium cooling of the pins in the fuel assemblies for these conditions is shown.
The generalization of data on heat transfer at liquid metals boiling in fuel assemblies is carried out; a cartogram of the flow regimes for two-phase flow of liquid metals in fuel assemblies is developed.
The model of the two-phase flow of liquid metal used in the calculations of accidental conditions has been improved; it has a significant impact on the calculation results. The results of comparing the data of calculated and experimental studies are presented.

Country/Int. organization

Russian Federation

Primary authors: Mrs KUZINA, Julia (SSC RF - IPPE); Mr SOROKIN, A.P. (IPPE JSC); Mrs DENISOVA, N.A. (IPPE JSC); Mr IVANOV, E.F.

Presenter: Mrs KUZINA, Julia (SSC RF - IPPE)

Session Classification: 5.2 Experimental Programs I

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **190** Type: **ORAL**

OPTIMIZATION OF BUILT-IN PRIMARY SODIUM PURIFICATION SYSTEM FOR ADVANCED BN REACTOR PLANT

Friday, April 22, 2022 10:42 AM (12 minutes)

While developing an advanced BN reactor plant the tasks were put to reduce a reactor plant cost with obligate meeting of safety requirements and reliability increase of reactor plant equipment and systems.
A purification system with cold traps (CT) placed in the reactor vessel (built-in purification system) was applied in the large BN reactor plant for primary sodium purification.
The developed CTs were equipped with unchangeable electromagnetic devices (a pump and a pump-throttle) to provide flowrate through the CT.
While developing an advanced BN, the primary sodium purification system was optimized - the cold trap design was optimized.
The electromagnetic pump and the electromagnetic pump-throttle were excluded from the cold trap design.
Sodium supply was arranged from the pressure chamber of the reactor.
To control sodium flowrate, the cold trap was equipped with changeable mechanical regulating devices with built-in flowmeters, and a changeable throttling device.
In addition, a CT cooling circuit was optimized within the framework of optimization. Eccentric collectors were used for cooling agent inlet and outlet; it permitted to increase CT impurity capacity by ~ 1.75 times and to increase purification system efficiency by ~ 1.5 times.
The performed optimization permitted to increase CT reliability and repairability, reduce quantity of CTs for replacement over the reactor plant service life by two times approximately; this made a positive effect on decrease of reactor plant construction cost and costs of CT replacement and disposal of spent CT.

Country/Int. organization

Russian Federation

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Presenter: Mr RUKHLIN, Sergey (JSC "Afrikantov OKBM")

Session Classification: 1.3 System Innovations

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **194** Type: **ORAL**

Coupled neutronic/thermal-hydraulic simulation of Unprotected Loss of Flow Test at Fast Flux Test Facility

Wednesday, April 20, 2022 1:40 PM (12 minutes)

Country/Int. organization

Switzerland

Primary authors: Mr PETROVIĆ, Đorđe (SCK•CEN); Dr KONSTANTIN, Mikityuk (Paul Scherrer Institute)

Presenter: Dr KONSTANTIN, Mikityuk (Paul Scherrer Institute)

Session Classification: 2.3 Accident Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **196** Type: **POSTER**

ULTRASONOSCOPY SYSTEM “VIZUS”FOR SODIUM-COOLED BN-TYPE REACTORS

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

Russian Federation

Primary authors: Mr LESIUKOV , Dmitrii (JSC “Afrikantov OKBM”); Mr VORONTSOV , Vitalii (JSC “Afrikantov OKBM”); Mr GUBYRIN , Michail (JSC “Afrikantov OKBM”)

Presenter: Mr LESIUKOV , Dmitrii (JSC “Afrikantov OKBM”)

Session Classification: Poster Session

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **197** Type: **ORAL**

EXPERIMENTAL TEST FACILITY TO TEST A PROTOTYPE OF THE AIR HEAT EXCHANGER GATE FOR THE ADVANCED BN REACTOR PLANT. DESIGN AND CONSTRUCTION ITEMS

Friday, April 22, 2022 2:18 PM (12 minutes)

To enhance safety, an emergency heat removal system (EHRS) is provided as a part of the reactor plant. One of the main elements of this system is an air heat exchanger (AHX), equipped with a device for air flowrate control with passive opening principle, which is a gate.
To ensure operability of this gate, high-temperature experimental test facility was developed and manufactured. This test facility permits to test the gate under operation conditions similar to the standard ones (temperature higher than 500℃).
A gate prototype is installed on a special box (3000 × 3000 mm) located on the site.
For uniform heating of air in the gate box equipped with an electric heating system of 170 kW electric power; and a special slot gap with optimum space provided by a spacing system is made.
To arrange flowrate, cold air supply and hot air discharge, air duct paths with required valves and a blower (fan) were designed. Because of the high temperature, the air circuit was unclosed; air duct outlet with special flare emission beyond the bounds of the building was arranged.
High-temperature thermal insulation is used at the test facility to ensure safe operation.
An information-and-measuring system was designed for on-line reporting on test facility parameters (temperature, pressure, flowrate, etc.).
Considering large overall dimensions and a high-altitude location of the test facility, special process tooling was developed in the course of test facility construction.
A process was developed and process tooling was manufactured to provide the slot gap in the course of manufacture.
A process was developed to simplify thermal insulation installation.
Now, the gate is tested at the made experimental test facility.

Country/Int. organization

Russian Federation

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Presenter: Mr GRUSHKO, Maksim (JSC "Afrikantov OKBM")

Session Classification: 5.3 Experimental Programs II

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **199** Type: **ORAL**

Recent thermal hydraulic studies of Gas Fast Reactor demonstrator ALLEGRO

Wednesday, April 20, 2022 1:52 PM (12 minutes)

The helium cooled high-temperature fast-spectrum reactor (GFR) with closed fuel cycle is one of the six GEN IV reactors selected by the Generation IV International Forum (GIF) to be developed for the foreseeable future. The European reference concept of the GFR technology is a unit with an envisaged power of 2400 MWth, which is currently in the pre-conceptual design phase. Prior to the building of the full scope facility the viability of the GFR technology will be proven by means of the ALLEGRO demonstrator with an envisaged thermal power of 75 MWth. The ALLEGRO development is led by the V4G4 Centre of Excellence consortium associating research organizations, companies and laboratories from Czech Republic (UJV Rez), France (CEA) Hungary (MTA-EK), Poland (NCBJ) and Slovakia (VUJE). One of the key tasks of ALLEGRO is to test the new ceramic refractory fuel for the industrial version of GFR2400. In this paper the latest outcomes of thermal hydraulic calculations of ALLEGRO are summarized. First, the work that has been done under the EU VINCO project is reviewed. It was carried out by V4G4 consortium aiming to transfer the GFR technology know-how from the CEA to the V4G4 and to establish the platform for continuation of the ALLEGRO demonstrator development. It comprises the methods, specific calculations and outcomes of the ALLEGRO thermal hydraulic benchmark which were carried out by the V4G4 partners using the CATHARE, RELAP and MELCOR codes. Based on the benchmark future experimental program is proposed using helium-cooled experimental facilities the S-ALLEGRO build in Czech Republic and the STU helium loop operating in Slovakia. Subsequently, a short summary of a recent work is presented, in which the hot duct break scenario is studied for the two and the three-loop ALLEGRO versions. The preliminary results of this analysis showed that the three-loop ALLEGRO has better cooling performance in case of hot duct break. Finally, the gas mass flow distribution in two parallel geometrically identical pipes is investigated, when they are heated with different power and when they have the same pressure loss. The results show that the pipe (or a closed subassembly in the reactor core) heated with higher power usually has lower coolant mass flow rate, which deteriorates the cooling capabilities of the subassemblies in a real reactor.

Country/Int. organization

Slovakia

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Presenter: Mr KVIZDA, Boris (VUJE, a.s.)

Session Classification: 6.2 Thermal Hydraulics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **200** Type: **ORAL**

Possibility of Simulating Natural Circulation in Fast Neutron Reactors Using a Light Water Test Facility

Wednesday, April 20, 2022 1:40 PM (12 minutes)

The paper evaluates the possibility of modeling the heat transfer phenomena in a liquid-metal coolant using a light water test facility. It considers the natural circulation of the coolant in the upper plenum of the fast-neutron reactor. A large nuclear power reactor (like the BN-1200 project) was selected as a reactor installation to be modeled. As the referent one was accepted the IPPE B-200 facility.
To validate the model, the similarity theory and the "black box"method were used. The paper uses the experience of a number of researchers in this field, in particular, the accepted assumptions which do not result in serious loss in modeling accuracy. The governing criteria of similarity were estimated based on the fundamental differential equations of convective heat transfer, so were the conditions under which it is possible to model sodium coolant by using light water with adequate accuracy. The paper presents the scales of the parameters used for the model-reactor comparison.
The introduction presents the paper purpose, considers the relevance of this topic, the utilized approaches –the similarity theory and the "black box"method, their limits to applicability. The general restrictions of the water test facility structural features are provided.
The first section provides the governing criteria derivation from the fundamental equations.
The second section includes obtaining the scales of the parameters.
The third section presents estimating the water test facility characteristics depending on its geometric scale. The conclusion about the possibility of the water-based modeling the liquid-metal coolant behavior is presented.
The paper includes 2 pictures, 2 tables, 23 references.

Country/Int. organization

Russian Federation

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Presenter: Mr URALOV, Dmitrii

Session Classification: 6.2 Thermal Hydraulics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **201** Type: **ORAL**

Preserving and transferring knowledge in the field of fast reactor technologies. Experience of the Obninsk Institute of Nuclear Power Engineering MEPhI

Thursday, April 21, 2022 2:16 PM (12 minutes)

Obninsk Institute for Nuclear Power Engineering has been realizing education and training of specialists for nuclear industry since establishing in 1950s. Educational programs are developed for nuclear power plants as well as for research and development institutions of Russian Federation and abroad. Due to close connections with scientific Institute of Physics and Power Engineering named A.I. Leipunsky and other organizations of the Rosatom State Corporation a lot of professors and researchers in Obninsk branch of MEPhI university have experience in operation and research in fast nuclear reactors such as BN-350, BN-600, BN-800 and others.
The paper describes experience in teaching, preserving and transferring knowledge in fast reactor technologies for more than 60 years. Nowadays the specific of fast reactors can be taught in separate sections of courses (for example, fast reactor physics as a part of general Nuclear Reactor Physics course), separate courses (like Liquid metal coolants) and finally in specially tailored master's program like Physics and technologies of fast reactors. The different ways of preserving and transferring knowledge are considered such as inviting famous specialists as professors, joint publications, providing laboratory practice and practical training (including preparing graduation thesis) for students on the base of Rosatom's companies. The modern distant learning technologies (video courses, online lectures, using VR and AR technologies, etc.) are discussed as well.

Country/Int. organization

Russian Federation

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Presenter: NAKHABOV, Aleksandr (Obninsk Institute for Nuclear Power Engineering NRNU MEPhI)

Session Classification: 9.1 Education, Profesional Development, and Knowledge Management

Track Classification: Track 9. Education, Profesional Development, and Knowledge Management

Contribution ID: **203** Type: **POSTER**

Experimental and Numerical Study on Temperature Fluctuation in The Upper Plenum of Fast Reactor

Wednesday, April 20, 2022 1:40 PM (2 hours)

Country/Int. organization

China

Primary authors: Mr DU, Yongqi (North China Electric Power University); Prof. LU, Daogang (North China Electric Power University); Dr CAO, Qiong (North China Electric Power University); Mr SI, Yu (North China Electric Power University); Mr MA, Benhao (North China Electric Power University)

Presenter: Mr DU, Yongqi (North China Electric Power University)

Session Classification: Poster Session

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **204** Type: **ORAL**

Integrating safety at the first design stages: a new methodology for safety-oriented SFR core design

Wednesday, April 20, 2022 10:52 AM (12 minutes)

Country/Int. organization

France

Primary author: Dr DROIN, Jean-Baptiste (CEA Cadarache)

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Presenter: Dr DROIN, Jean-Baptiste (CEA Cadarache)

Session Classification: 2.2 Safety Design and Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **205** Type: **ORAL**

The influence of isotopic composition of plutonium in fast reactor fuel on the reactivity margin

Wednesday, April 20, 2022 11:52 AM (12 minutes)

Country/Int. organization

Russian Federation

Primary authors: TEREKHOVA, Anna (no); KARAZHELEVSKAIA, Yulia

Presenter: KARAZHELEVSKAIA, Yulia

Session Classification: 3.1 Fuel Cycle Scenarios

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **206** Type: **ORAL**

Physical modeling of hydrodynamics and heat exchange in fast reactors with liquid metal coolants

Wednesday, April 20, 2022 1:52 PM (12 minutes)

Country/Int. organization

Russian Federation

Primary authors: Mrs KUZINA, Julia (SSC RF - IPPE); Mr SOROKIN, A.P. (IPPE JSC)

Presenter: Mrs KUZINA, Julia (SSC RF - IPPE)

Session Classification: 5.1 Experimental Reactors and Facilities

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **207** Type: **ORAL**

Optimization of the secondary loops on the ESFR SMART project

Friday, April 22, 2022 11:06 AM (12 minutes)

The ESFR SMART project is a European sodium fast reactor project, which follows on from the EFR project, then the CP ESFR project, and whose goal is to present an improved project in terms of safety, taking into account the new rules to be applied in particular following the post Fukushima provisions. For the primary circuit a number of safety options have already been presented (ref 1, 2, 3), based on simplifications as well as passive and forgiving systems. In the same spirit, this paper proposes an optimization of the secondary circuits to improve their intrinsic safety and their compactness.
First, these circuits have the role of evacuating the power of the reactor, and in case of shutdown will actively participate in the evacuation of the residual power. The use of these circuits has been favored because it is the loop commonly used by the operator for this purpose, and in all operating circumstances. They were therefore pre-dimensioned (ref 4 , 5) to be able to evacuate this power alone even after the loss of the water circuits, only by natural convection of the air around the modules of steam generators. Moreover, in each secondary loop a dedicated system connected to the exchanger is capable of performing this function on its own, in passive natural convection and even in the event of draining of the secondary circuit.
Second, the REX of the SFRs shows that sodium leaks mainly take place at the level of the secondary circuits .Proposals are made for the secondary piping to increase the possibilities of rapid detection and mitigation . For this, we propose to use straight pipes where the expansions are taken up by bellows. This helps to minimize lengths and welds. This also allows the use of an offset thermal insulation allowing a quick and more reliable sodium leak detection. This point greatly improves the compactness of the circuit and therefore makes it possible to significantly reduce secondary buildings.
Then, for the sodium / water reactions, the choice of modular steam generator has been made to increase quickness of detection at the outlet of each module. It allows also minimizing consequences on the operation of the reactor able to operate with one unavailable module
Finally, some propositions are made on the use of passive thermal pump to assure passively a minimal sodium flow rate.
The conclusion summarizes the necessary R&D to allow this optimization.

Country/Int. organization

France

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Presenter: GUIDEZ, joel (CEA)

Session Classification: 1.3 System Innovations

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **208** Type: **ORAL**

APPROACHES TO FORM THE BN 1200 CORE START LOADING USING MOX-FUEL AND MNUP-FUEL

Friday, April 22, 2022 11:30 AM (12 minutes)

Two types of mixed uranium-plutonium fuel are considered for BN-1200 reactor within the framework of "Proryv"project: conventional, well-mastered MOX-fuel and advanced mixed uranium-plutonium nitride (MNUP) fuel having higher density.
The main design mode of reactor operation is operation in the equilibrium mode with scattered batch refuelings. As compared with equilibrium state, the start-up core is formed of fresh FSAs completely. Taking this fact into account, in the course of forming of the start-up core consisting of FSAs with fuel enrichment by plutonium corresponding to the equilibrium mode with retention of total quantity of FSAs in the core, it will have an excess reactivity margin. The reactivity margin of the start-up core shall be predicted considering its possible deviation caused by fuel manufacturing tolerance and by uncertainties of core critical parameters estimation. For this reason, in the start-up core design, even increased fuel enrichment may be considered, and, in any case, measures to compensate possible excess reactivity margin due to development of an appropriate core layout shall be provided.
Basic criteria to select the start-up core layout are to meet regulatory requirements on reactivity balance and to provide the possibility of reactor operation at nominal power without exceeding design parameters of FSA operation.
The paper considers possible methods to compensate excess reactivity margin of the start-up core, presents basic approaches to form the start-up core loading, and describes its layout in the case of MOX-fuel and MNUP fuel application.

Country/Int. organization

Russian Federation

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Presenter: BELOV, Sergey

Session Classification: 6.5 Integrated Analysis and Digitalization

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **209** Type: **ORAL**

DISTINCTIVE FEATURES OF THE BN-800 CORE IN THE COURSE OF TRANSITION TO COMPLETE MOX-FUEL LOADING

Friday, April 22, 2022 11:42 AM (12 minutes)

To solve problems of BN-800 transition from the hybrid core consisting of FSAs with pellet-type uranium oxide fuel and FSAs with pellet-type and vibropacked MOX-fuel to the core loaded completely with pellet-type MOX-fuel, it is necessary to develop and implement a proper core design.
In the developed design it is provided that the transition is performed by replacement of hybrid core spent FSAs with fresh FSAs with pellet-type MOX-fuel. The FSAs shall be replaced during three sequential refuelings (eighth, ninth, and tenth refuelings of the core).
The paper compares layouts of the hybrid core and the full MOX-fuel core and shows change of the core composition at transition to complete loading with MOX-fuel.
The transition period is characterized by step-wise increase of neutron flux density in the core caused by nuclear distinctive features of plutonium relative to uranium-235. Nevertheless, corresponding increase of linear power of fuel pins of hybrid core FSAs operated in the transition period is compensated by fissile isotope content decrease in the course of fuel burnup.
The transition period has also some distinctive features concerning hydraulic characteristics. Change of ratio of quantities of FSAs of different types with different hydraulic resistance leads to correspondent re-distribution of sodium flowrates through them and to change of the total hydraulic resistance of the core.
The paper presents data on sodium pressure drop over the core and on the temperature state of FSAs during transition period. The paper shows that FSA operation parameters do not exceed the justified values. No reactor power limitation is required during reactor operation in the transition period.

Country/Int. organization

Russian Federation

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Presenter: BELOV, Sergey

Session Classification: 6.5 Integrated Analysis and Digitalization

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **210** Type: **ORAL**

Application of the practical elimination concept within the framework of the ESFR-SMART project to improve the intrinsic safety of the sodium-cooled fast reactor

Tuesday, April 19, 2022 4:34 PM (12 minutes)

Country/Int. organization

France

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Session Classification: 2.1 General Safety Approach

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **213** Type: **ORAL**

RESULTS OF POST-IRRADIATIONS EXAMINATIONS OF MIXED NITRIDE PINS WITH GAS AND LIQUID METAL SUB-LAYERS

Friday, April 22, 2022 1:54 PM (12 minutes)

Today the investigations have been completed on helium-bonded fuel pins with mixed uranium-plutonium nitride of BN-600/BN-800, BN-1200 and BREST reactors types after irradiation as a part of ten EFAs of BN-600 reactor up to maximum burn-up of 7.5;6.0 and 4.5at%, respectively. Also the PIE of mixed nitride pins of BREST type with helium and lead sub-layers after intermediate tests to maximum burn-up of 4,8 and 3,9at% as part of dismountable EFAs of BOR-60 reactor are over. As a result of post-irradiation examinations, the main intra-fuel pin processes that affect on materials properties change and fuel pins state, their relationship with the initial nitride state were identified. The regularities and quantitative characteristics of nitride swelling, the behavior of fission products, cladding corrosion and mechanical properties change are revealed.

The average for the irradiation time rate of fuel swelling in cross sections near the central core plane in gas-bonded pins of different EFAs irradiated in BN-600 reactor is equal to (1.6-2.0)%/at%. With an increase of fuel burn-up there is a tendency to slow down of swelling rate. According to the results of irradiation in BOR-60 reactor, the fuel swelling rate in lead-bonded pins is lower than in helium-bonded pins: 1.4±0.2 and 1.7±0.2%/at%, respectively.

The presence of carbon and oxygen impurities in the fuel can lead to local areas of claddings carburization and oxidation, randomly distributed along the height and perimeter of its inner surface. The key characteristics of the fuel that determine the cladding carburization and oxidation, and ways to prevent them, are determined. The maximum depth of the corrosion zone is located in the upper part of the fuel column and is equal to 40μm.

The results of cladding mechanical tests showed the significant margin of strength and ductility of cladding materials along the entire height of fuel pins. High values of strength characteristics indicate a weak influence of corrosion on the strength of studied claddings materials. The mechanical cladding properties of a lead–bonded pins have the same pattern of change along the pin height depending on irradiation and testing temperatures, as of fuel pins with helium sublayer, keeping enough ductility in the area of low-temperature irradiation embrittlement. At irradiation and test temperatures of 350-380℃, the values of the strength limit from 1070 to 1200MPa were obtained with the values of the total elongation of not less than 2%.

Country/Int. organization

Russian Federation

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Session Classification: 3.4 Advanced Fuel Development

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **216** Type: **POSTER**

DEVELOPMENT OF THE TECHNICAL APPROACH FOR RESEARCH OF THE SODIUM COOLANT CURRENT IN THE INTEGRAL TYPE REACTOR

Friday, April 22, 2022 10:30 AM (2 hours)

The paper presents the results of the end-to-end mathematical modeling of the BN reactor with integral equipment layout. The developed approach permits to validate RP characteristics and to study the process of the transfer of the predecessor of the delayed neutrons with the primary circuit coolant in the conditions of stratified current.
The approach includes a complex of specially developed models:
- turbulent heat transfer model, permitting to consider the specific character of the liquid metallic sodium coolant in the codes of the computational fluid dynamics;
- model of the transfer of the predecessor of the delayed neutrons, going out from FAs with leaking fuel rods, and considering their radioactive decay.
The results of verification and validation of the technical approach are provided including validation the possibility of modeling of the heat exchanging equipment and the core with simplified structures of these elements.
The developed approach allows solving problems, which are significantly important for validation of the service life and increase of safety of the sodium cooled fast neutron reactors.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **217** Type: **ORAL**

Operating Experience of FBTR

Tuesday, April 19, 2022 4:10 PM (12 minutes)

Country/Int. organization

India

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Session Classification: 8.1 SFR Commissioning, Operation, and Decommissioning

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **219** Type: **ORAL**

Project of a multipurpose lead reactor with a hard neutron spectrum

Wednesday, April 20, 2022 2:40 PM (12 minutes)

Country/Int. organization

Russian Federation

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Session Classification: 1.2 Innovative Design Advances

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **220** Type: **ORAL**

Development status of commercial SMRs and its experience to China

Tuesday, April 19, 2022 3:58 PM (12 minutes)

Country/Int. organization

China

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Session Classification: 7.1 Sustainability: Economics, Environment, and Proliferation

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **223** Type: **POSTER**

INTEGRATED RADIATION AND HYGIENIC APPROACH TO PRODUCTION SAFETY. ASSESSMENT OF THE IMPACT ON PUBLIC HEALTH

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **224** Type: **ORAL**

Complex of experimental facilities for design and safety justification of fast reactors with liquid metal coolants

Wednesday, April 20, 2022 1:40 PM (12 minutes)

Country/Int. organization

Russian Federation

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Presenter: Mrs KUZINA, Iuliia (IPPE JSC)

Session Classification: 5.1 Experimental Reactors and Facilities

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **226** Type: **POSTER**

RISK FACTORS OF COMPLEX RADIATION AND NON-RADIATION EFFECTS ON THE HEALTH OF PERSONNEL IN ASSESSING THE IMPACT OF THE PRODUCTION OF INNOVATIVE FUEL FOR FAST REACTORS

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **227** Type: **POSTER**

COMPARATIVE ANALYSIS OF CALCULATIONAL AND EXPERIMENTAL DIFFERENCES OF THE NEUTRON-PHYSICAL CHARACTERISTICS OF THE BN-800 REACTOR

Friday, April 22, 2022 10:30 AM (2 hours)

During the operation of the BN-800 reactor, a large amount of experimental data has been accumulated on critical states, the effectiveness of the control system, etc. It should be noted that the loading of the hybrid core in the initial period was constantly changing: different ratios of fuel assemblies with uranium fuel and MOX fuel, as well as the number of fuel assemblies in the intermediate and highly enrichment zone. On the eve of the transfer of the core to the full load of MOX fuel, it is necessary to make sure that the neutronic characteristics are adequately reproduced by all calculation codes.
The aim of this work is to calculate the neutronic characteristics of the BN-800 core with a hybrid fuel load and compare them with the available experimental data.
The work on the analysis of the methods of computational support of the operation of the BN-800 reactor has been performed. The calculations were performed using the MMKK and MMKC codes, in which the Monte Carlo method is implemented, as well as with JARFR and GEFEST800 in the diffusion approximation. Comparison of the calculation results for all involved codes with the measurement results made it possible to estimate the methodological error in calculations of the main neutron-physical characteristics.
The calculations were performed for the first seven micro-campaigns of the BN-800 reactor.
The analysis of such neutronic characteristics as:
- the value of the reactivity margin in the conditions at the beginning and end of the reactor micro-campaigns;
- the values of the subcriticality levels of the reactor during refueling and after the withdrawal of the safety rods;
- efficiency of single control rods and control rod groups;
- the temperature-power effect of reactivity;
- the reactivity loss rate during micro-campaign.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **231** Type: **POSTER**

COMPLEX RADIATION AND HYGIENE STUDIES OF RADIATION IMPACT FACTORS ON PRODUCTION PERSONNEL, MIXED NITRIDE URANIUM-PLUTONIUM FUEL FOR FAST NEUTRON REACTORS

Thursday, April 21, 2022 10:40 AM (2 hours)

This work is carried out in order to assess the compliance of the radiation protection of personnel working at the complex experimental installations of JSC «SChE» with the requirements of the national radiation safety standards to limit the generalized risk of potential exposure and the IAEA recommendations for not exceeding the control level of the minimum significant radiation risk.
On the basis of monitoring the dynamics of the ambient dose rate equivalent (ADER) of photon and neutron radiation ADER at the workstations of the complex experimental installations 1 and 2, the regularities of the dose formation have been studied. Doses of external exposure of personnel were estimated. In accordance with the recommendations of the ICRP and the IAEA, the radiation risks of personnel were assessed.
The report will present the estimates obtained for the personnel working at the complex experimental facilities # 1 and # 2 on the external exposure doses for gamma radiation and neutron radiation. The annual expected effective dose of internal exposure of personnel will be estimated, and the level of the minimum significant radiation risk will be calculated.
The results obtained and the developed methods will be used to ensure the radiation safety of personnel during the transition from experimental installations to pilot industrial implementation of the technology for the production of mixed uranium-plutonium nitride (MNUP) fuel.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **232** Type: **ORAL**

RADIATION AND HYGIENE ASSESSMENT OF EXTERNAL EXPOSURE FACTORS OF PERSONNEL WORKING AT EXPERIMENTAL FACILITIES IN THE PRODUCTION OF MIXED NITRIDE URANIUM-PLUTONIUM FUEL

Wednesday, April 20, 2022 12:04 PM (12 minutes)

Country/Int. organization

Russian Federation

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Session Classification: 3.1 Fuel Cycle Scenarios

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **233** Type: **ORAL**

Neutronics Benchmark of CEFR Start-Up Tests: Reaction Rates and Reactivity Coefficients

Tuesday, April 19, 2022 1:48 PM (12 minutes)

Country/Int. organization

United States of America

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Presenter: KIM, Taek Kyum (Argonne National Laboratory)

Session Classification: Special Session: IAEA Coordinated Research Projects

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **234** Type: **POSTER**

Comparative analysis of minor actinides transmutation in a molten-salt burner reactor based on LiF-NaF-KF and LiF-BeF2 salts

Thursday, April 21, 2022 1:40 PM (2 hours)

In Russia, research is actively underway to develop a specialized molten-salt burner reactor (MSR-burner) of minor actinides (MA) from spent nuclear fuel of power reactors. Two candidate fluoride salts, LiF-BeF2 [1] and LiF-NaF-KF, are considered as the solvent of the reactor fuel components. The purpose of the present paper is to study MA transmutation in the MSR-burner based on selected salts in the equilibrium mode of reactor operation at different volumes of the core. The calculations were performed using PRIZMA+RISK software package developed at the "RFNC-VNIITF named after Academ. E.I. Zababakhin" [3,4].
The LiF-BeF2 salt has a low solubility limit of actinide fluorides, which leads to the need to feed the reactor with a significant amount of Pu and, consequently, to a low efficiency of MA transmutation. By reducing of volume of the active zone increases the consumption of Pu and reduces the efficiency of transmutation. In contrast to LiF-BeF2, LiF-NaF-KF eutectic is characterized by a relative high solubility of actinide fluorides. For a MSR-burner based on this salt, Pu is needed mainly for starting; in the equilibrium mode reactor consumes only MA. In this case, the maximum efficiency of MA transmutation can be achieved in a wide range of core volume: from 2 m3 to 30 m3 with a concentration of actinide fluorides from 17 to 10%, mol., respectively.

[1] Ignatiev V., Feynberg O., I. Gnidoi, et al. Molten salt actinide recycler and transforming system without and with Th-U support: Fuel cycle flexibility and key material properties. Ann. Nucl. Energy, 2014, v.64, p.408-420.
[2] Lizin A.A., Tomilin S.V., Gnevashov O.E., et al. PuF3, AmF3, CeF3, and NdF3 solubility in LiF-NaF-KF melt. Atomic energy, 2013, v.115, No.1, p.11-16.
[3] Zatsepin O.V., Kandiev Ya.Z., Kashaeva E.A., et al. Calculation for the VVER-1000 core by the Monte-Carlo method implemented in the PRIZMA code. Voprosy atomnoy nauki i techniki. Serija: Jadernye konstanty, 2011, No.4, p.64-73.
[4] Modestov D.G. The RISK-2014 code to solve nuclear kinetics problems, RFNC-VNIITF preprint No.243, Snazhinsk, 2014.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **235** Type: **POSTER**

RADIATION-HYGIENIC ASSESSMENT OF INTERNAL EXPOSURE FACTORS OF PERSONNEL WORKING AT EXPERIMENTAL FACILITIES IN THE PRODUCTION OF MIXED NITRIDE URANIUM-PLUTONIUM FUEL

Thursday, April 21, 2022 10:40 AM (2 hours)

The report presents results of studies of the physicochemical characteristics of radioactive workplace aerosols formed during the production of mixed nitride uranium-plutonium fuel (activity particle-size distribution, nuclide composition, lung absorption type, elemental composition, reactive properties in the air). Taking into account these characteristics, the dose coefficients were calculated and the annual committed effective doses of internal exposure of personnel were calculated.

Next impactors were used for activity particle-size distribution analysis: AIP-2, IPHRT (developed by SRC FMBC named by A.I. Burnazyan), an electric low-pressure impactor HR-ELPI, Andersen cascade impactor and others. At the study of morphological characteristics, we used a scanning electron microscope (SEM) LYRA-3 equipped with an X-ray microanalyzer (RMA) X-max 80. X-ray structural analysis was performed on an XRD-7000 X-ray diffractometer. Analysis to determine the mass fraction of nitrogen and oxygen was carried out on the LECO analyzer; measurement of the content of uranium and plutonium was conducted on the mass spectrometer "TRITON+". Lung absorption type assessment was performed by dialysis through membrane filters in a pulmonary fluid simulator.

The high reactivity of mixed nitride uranium-plutonium (MNUP) compounds causes instant oxidation of the thoracic fraction of MNUP fuel aerosols upon contact with air, however, the intake of MNUP into the body is possible orally as part of the extrathoracic fraction (particles of 100 - 500 μm in size which have oxide film emerging upon interaction with air and inhibiting further oxidation of nitride). Dissolution of these particles in gastric juice can lead to the release of the nitride core, followed by a rapid entry of radionuclides into the organs and tissues of the body through the gastrointestinal tract.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **236** Type: **POSTER**

Influence of preheating temperature on delta-ferrite formation and mechanical properties of 12%Cr steel weld metals

Wednesday, April 20, 2022 10:40 AM (2 hours)

Country/Int. organization

China

Primary author: Dr WU, Dong

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Presenter: Dr WU, Dong

Session Classification: Poster Session

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **238** Type: **ORAL**

Overview of a Sodium Fast Reactor Thermal Hydraulic Test Facility

Wednesday, April 20, 2022 2:04 PM (12 minutes)

Country/Int. organization

United States of America

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Presenter: WEATHERED, Matthew

Session Classification: 5.1 Experimental Reactors and Facilities

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **239** Type: **ORAL**

TYPES OF CHEMICAL COMPOUNDS IN THE ASSESSMENT OF RADIATION AND HYGIENIC HAZARDS WHEN WORKING WITH IRRADIATED NITRIDE FUEL

Friday, April 22, 2022 2:30 PM (12 minutes)

At the moment, in the world nuclear power industry there are proposals for the transition from oxide to other types of fuel, which can be much more cost-effective and used in the technology of closing the nuclear fuel cycle based on fast reactors. One of such fuels is uranium-plutonium nitrides. During the fission of uranium and plutonium nuclei in fast neutron reactors, a large number of isotopes of chemical elements and various chemical compounds with these isotopes are formed. These elements and compounds have different chemical properties.
The purpose of this report is to consider various types of chemical compounds of fission products formed in irradiated nitride fuel in fast reactors and to assess the radiation-hygienic hazard of work with irradiated nitride fuel.
The report examined the types of chemical compounds formed both directly in the fuel matrix during the operation of reactors, and their possible transitions into other chemical compounds and forms during the processing of irradiated nuclear fuel.
As a result, the report contains the main types and forms of chemical compounds of fission products present in irradiated nitride fuel, such as lanthanide nitrides and intermetallic compounds, the physicochemical and radiotoxicological properties of which have not been studied in detail, which can lead to an incorrect assessment of the contribution to the internal dose of workers during ensuring radiation monitoring or emergency situations. This stands means the loss or complete absence of control over radiation safety during the reprocessing of nitride fuel is possible. It is necessary to holding a computational and analytical assessment of the real radio-hygienic hazard and the significance of these compounds, in internal exposure and experimental confirmation of the presence of these compounds in the air of the working area of enterprises processing irradiated nitride fuel.

Country/Int. organization

Russian Federation

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Session Classification: 3.4 Advanced Fuel Development

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **240** Type: **ORAL**

Overview of the Versatile Test Reactor Safety Analysis

Wednesday, April 20, 2022 10:40 AM (12 minutes)

Country/Int. organization

United States of America

Primary author: SUMNER, Tyler (Argonne National Laboratory)

Co-authors: FANNING, Thomas (Argonne National Laboratory); THOMAS, Justin (Argonne National Laboratory)

Presenter: SUMNER, Tyler (Argonne National Laboratory)

Session Classification: 2.2 Safety Design and Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **241** Type: **ORAL**

Safety Analysis of the ARC-100 Sodium-Cooled Fast Reactor

Tuesday, April 19, 2022 3:58 PM (12 minutes)

Country/Int. organization

United States of America

Primary author: SUMNER, Tyler (Argonne National Laboratory)

Co-author: MOISSEYTSEV, Anton (Argonne National Laboratory)

Presenter: SUMNER, Tyler (Argonne National Laboratory)

Session Classification: 2.1 General Safety Approach

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **244** Type: **POSTER**

Experimental investigation of the fluid-structure interaction effect between adjacent equipment supports in a fast reactor

Thursday, April 21, 2022 10:40 AM (2 hours)

The equipment supports are in constricted arrangement in the main vessel of the fast reactor. Under the condition of earthquake, equipment supports may sustain damage caused by the interaction between equipment supports and fluid, therefore, the evaluation of the fluid-structure interaction effect is an important aspect of the structural safety assessment of fast reactors. Using the added mass to assess the fluid-structure interaction effect is a more common method. ASME standard gives the added mass's formula for a single cylinder immersed in an infinite fluid domain. However, this formula is not suitable for the complex arrangement of equipment supports in the annulus region of main vessel. To investigate the added mass of equipment supports in fast reactor, in this paper, a number of experiments are carried out. A simplified and scaled model of fast reactor was designed, containing a main pump support cylinder, two intermediate heat exchanger (IHX) support, and two independent heat exchanger (DHX) support. Through the modal experiment and the sine wave experiment of the shaking table, the circumferential and axial pressure distribution data under different arrangements are recorded. For the purpose of assessing the fluid-structure interaction effect between equipment supports and fluid, the additional mass matrix of the support cylinders is derived according to the experimental results. The experiments provide a reference and basis for the arrangement of the equipment supports in the annular region of the fast reactor and the correction of the additional mass formula.

Country/Int. organization

China

Primary authors: Mr DUAN, Dexuan (North China Electric Power University); Prof. LU, Daogang (North China Electric Power University); Dr LIU, Yu (North China Electric Power University); Mr HUANG, Yijun (North China Electric Power University)

Presenter: Mr DUAN, Dexuan (North China Electric Power University)

Session Classification: Poster Session

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **245** Type: **POSTER**

Some results of using partial equations for calculations of transient processes in fast breeder reactors

Friday, April 22, 2022 1:30 PM (2 hours)

Calculations of non-stationary processes of fast neutron reactors taking into account the spatiotemporal dependence of the neutron field is a rather complex process due to the significant influence on the calculation results of delayed neutrons, which make up a very small, less than a percent, part of all the neutrons of the reactor in its critical state. This circumstance is due to the fact that the spectra of all neutrons, both prompt and delayed, are in the working range of the reactor's neutron spectrum, in contrast to thermal neutron reactors.
The author's use of solutions to a system of partial equations for estimating transient processes shows the influence of differences in time changes in the form functions of prompt and delayed neutrons on the estimation of transient processes (physical characteristics) of a fast-breeder reactor, for example, the reactivity of the reactor. By partial equations, the author understands a system of separate equations for prompt and each of the groups of delayed neutrons, as well as, if necessary, neutrons of an external source.
The results of using partial neutron transfer equations for estimating the time behavior of a fast reactor are presented on the example of computational studies of test models of the MET1000 and MOX1000 fast breeder reactors developed within the framework of the Generation-IV project.
The paper shows that in transient processes, for example, during the discharge of control rods, due to the difference in the form functions of prompt and delayed neutrons, the real reactivity introduced by the rods exceeds the reactivity obtained from calculations of stationary problems by 5-10%, depending on the type of fuel in the reactor (MOX and uranium fuel).

Country/Int. organization

Russian Federation

Primary author: Prof. SELEZNEV, Evgeny (All-Russian Research Institute for Nuclear Power Plants Operation)

Presenter: Prof. SELEZNEV, Evgeny (All-Russian Research Institute for Nuclear Power Plants Operation)

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **248** Type: **POSTER**

Analysis of sodium fire accident after upgrade of ventilation system of primary loop's corridor

Wednesday, April 20, 2022 1:40 PM (2 hours)

Country/Int. organization

China

Primary author: Mrs YANG, jiayin

Co-authors: Mr HU, wenjun; Mr LI, shirui; Mr ZHAO, lei

Presenter: Mrs YANG, jiayin

Session Classification: Poster Session

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **252** Type: **ORAL**

Gear Test Assembly: First Liquid Metal Component Testing in METL

Tuesday, April 19, 2022 1:36 PM (12 minutes)

Country/Int. organization

United States of America

Primary authors: KENT, Edward (Argonne National Laboratory); Mr BELCH, Henry (Argonne National Laboratory); GRANDY, Christopher (Argonne National Laboratory); Dr BAKHTIARI, Sasan (Argonne National Laboratory); CHIEN, Hualte (Argonne National Laboratory); Mr KULTGEN, Derek (Argonne National Laboratory); WEATHERED, Matthew

Presenter: KENT, Edward (Argonne National Laboratory)

Session Classification: 4.1 Advanced Reactor Cladding and Core Material, Coolants, and Related Chemistry

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **254** Type: **ORAL**

Thermally conductive liquid-metal sublayer in fuel element

Tuesday, April 19, 2022 1:12 PM (12 minutes)

Country/Int. organization

Russian Federation

Primary authors: Ms ORLOVA, Ekaterina (INPE, NRNU MEPhI); SAMOKHIN, Dmitrii (National Research Nuclear University MEPhI); ORLOV, Michael (Private institution «Innovation and technology center for the «PRORYV» project»)

Presenter: ORLOV, Michael (Private institution «Innovation and technology center for the «PRORYV» project»)

Session Classification: 4.1 Advanced Reactor Cladding and Core Material, Coolants, and Related Chemistry

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **255** Type: **POSTER**

JUSTIFICATION OF CRITICAL EXPERIMENTS ON STAND FKBN-2 TO VERIFY NEUTRON-PHYSICAL SOFTWARE FOR CALCULATIONS OF THE MOLTEN-SALT REACTOR

Thursday, April 21, 2022 10:40 AM (2 hours)

In order to reduce the long-term potential hazard of waste from the reprocessing of spent nuclear fuel from thermal reactors and to increase the enviroment attractiveness of nuclear power in our country, work is underway to create a molten-salt reactor-burner of minor actinides (MSR-B). The first stage on this path is the creation of an investigative molten-salt reactor (IMSR) for testing key technological solutions for the full-scale MSR-B. A carrier fuel based on the two-component LiF-BeF2 system was chosen as the base for the IMSR.
At present, there are no computer software certified for neutron-physical calculation of a reactor with fuel based on molten salts of the LiF-BeF2 type, adopted for design work on the IMSR. There are also large uncertainties in the choice of neutron constants that exceed the permissible limits for the accuracy of calculations when justifying of the choice of specific core parameters when performing design work.
Experiments with critical multiplying systems that simulate the IMSR core in terms of the composition of materials and spectral characteristics are required to justify and subsequently certify neutron-physical software as applied to the IMSR project. One of the possible instalations for carrying out such experiments is the stand for critical assemblies FKBN-2. An important task is to justify the possibility of setting up critical experiments on this stand.

Country/Int. organization

Russian Federation

Primary authors: YUDOV, Aleksey; KHMELNITSKI, Dmitry; MODESTOV, Dmitry; VOLVOV, Igor; BELONOGOV, Mikhail; Mr TRAPEZNIKOV, Mikhail; ANDREEV, Sergey; SIMONENKO VADIM; LITVIN, V; SOKOLOV, Yuri

Presenter: Mr TRAPEZNIKOV, Mikhail

Session Classification: Poster Session

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **258** Type: **ORAL**

Development of the Versatile Test Reactor (VTR) Probabilistic Risk Assessment

Wednesday, April 20, 2022 11:28 AM (12 minutes)

Country/Int. organization

United States of America

Primary authors: GRABASKAS, David (Argonne National Laboratory); Mr ANDRUS, Jason (Idaho National Laboratory); LI, Yunlong (Jonathan) (GE Hitachi Nuclear Energy); Mr HENNEKE, Dennis (GE-Hitachi Nuclear Energy); BUCKNOR, Matthew (Argonne National Laboratory)

Presenter: Mr ANDRUS, Jason (Idaho National Laboratory)

Session Classification: 2.2 Safety Design and Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **259** Type: **POSTER**

Development of the Simplified Radionuclide Transport (SRT) Code Version 2.0 for Versatile Test Reactor (VTR) Mechanistic Source Term Calculations

Wednesday, April 20, 2022 1:40 PM (2 hours)

Country/Int. organization

United States of America

Primary authors: GRABASKAS, David (Argonne National Laboratory); BUCKNOR, Matthew (Argonne National Laboratory); Dr JERDEN, James (Argonne National Laboratory)

Presenter: GRABASKAS, David (Argonne National Laboratory)

Session Classification: Poster Session

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **260** Type: **POSTER**

Increase of nuclear power plant hydrogen safety using zirconium accumulator

Wednesday, April 20, 2022 1:40 PM (2 hours)

Country/Int. organization

Russian Federation

Primary authors: ORLOVA, Ekaterina; SAMOKHIN, Dmitrii (National Research Nuclear University MEPhI); Mr ORLOV, Andrey (National Research Nuclear University MEPhI (Obninsk Institute for Nuclear Power Engineering)); Mr BELOZEROV, Vladimir (National Research Nuclear University MEPhI (Obninsk Institute for Nuclear Power Engineering))

Presenter: ORLOVA, Ekaterina

Session Classification: Poster Session

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **262** Type: **ORAL**

Physical feasibility of MA transmutation in a two-component nuclear energy system in Russia

Thursday, April 21, 2022 2:52 PM (12 minutes)

The transition to a two-component nuclear power structure using thermal (TR) and fast reactors (FR), as asserted by the «Russian nuclear power development strategy to 2050 and outlook to 2100» (Strategy-2018), is directed at finding optimal solutions and resolving relevant issues pertaining to the currently established nuclear energy system in Russia. A core issue in this regard is managing the long-lived MA inventory, which have a substantial impact on overall nuclear power radiological safety for time-frames that could be considered historically significant. The study presents findings related to analyzing the capability of a commercial fleet of FRs to successfully resolve these issues by including MA in the closed nuclear fuel cycle without any fundamental changes to their expected characteristics regarding safety and competitiveness parameters.
Three different Russian scenarios were considered in the study: 1) full transition to a large-scale FR dominated nuclear fleet reaching 92 GWe capacity by the end of the 21st century, 2) mixed composition of VVER (43% capacity) and FR (57% capacity) with comparable installed capacity 3) a moderate nuclear power development scenario reaching 72 GWe by the end of the century with a 57% VVER and 43% FR mix. At this rate it is calculated that 36-67 tonnes ofAm and 67-120 tonnes of MA (Am+Np) would be accumulated from the VVER fleet.
Two algorithms are proposed for Am and Np utilization. In the first approach, MA and Pu obtained from reprocessing VVER spent fuel are simultaneously used for FR start-up, after which the fuel will reach an equilibrium state following multiple recycling in the fast reactor. The maximum concentration of MA content in the fuel was calculated to be at 2% (1,1% Am and 0,9% Np), and 0,5% when the fuel reaches equilibrium state.
If radiation-related limiting factors for handing nuclear fuel with high MA content are taken into account, comparable MA utilization efficiency could be achieved with lower MA fuel concentration if they are introduced evenly throughout the FR operation lifecycle. Continuous addition of MA from VVER reactors to FR fuel with 2% MA concentration will increase MA utilization threefold compared to the first approach.
The results of the study conclude that the MA arising from VVER spent fuel accumulation in all scenarios considered could be successfully utilized without dedicated MA-burners, although the complexity of the issue intensifies as fewer FRs are introduced into the power mix with increased MA content in their fuel.

Country/Int. organization

Russian Federation

Primary authors: KHOMYAKOV, Yury (Private institution «Innovation and technology center for the «PRORYV» project»); RODINA, Elena (Innovative &Technology Center by "PRORYV" Project); RACHKOV, Valery (Track leader); KASHIRSKII, Andrei (JSC "Proryv")

Presenter: KHOMYAKOV, Yury (Private institution «Innovation and technology center for the «PRORYV» project»)

Session Classification: 3.3 Reprocessing, Partitioning, and Transmutation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **264** Type: **POSTER**

NON-DESTRUCTIVE METHOD FOR DETERMINING STEEL CORROSION COEFFICIENTS IN LEAD

Wednesday, April 20, 2022 10:40 AM (2 hours)

Country/Int. organization

Russian Federation

Primary author: Mr GOLOSOV, Oleg (Research institute of nuclear materials, (INM JSC))

Co-authors: Ms KOZLOVA, Anastasiya (Research institute of nuclear materials, (INM JSC)); Ms GLUSHKOVA, Natalya (Research institute of nuclear materials, (INM JSC)); Ms KUZINA, Tatyana (Research institute of nuclear materials, (INM JSC)); Mr TSYGVINTSEV, Vladimir (Research institute of nuclear materials, (INM JSC))

Presenter: Mr GOLOSOV, Oleg (Research institute of nuclear materials, (INM JSC))

Session Classification: Poster Session

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **265** Type: **POSTER**

Preliminary Shielding Analysis for the Versatile Test Reactor

Wednesday, April 20, 2022 1:40 PM (2 hours)

Country/Int. organization

United States of America

Primary author: FEI, Tingzhou (Argonne National Laboratory)

Co-authors: Dr BAYS, Samuel (Idaho National Laboratory); HEIDET, Florent (Argonne National Laboratory)

Presenter: FEI, Tingzhou (Argonne National Laboratory)

Session Classification: Poster Session

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **268** Type: **ORAL**

Application of a Risk-Informed Performance-Based Approach for the Authorization of the Versatile Test Reactor

Tuesday, April 19, 2022 4:46 PM (12 minutes)

Country/Int. organization

United States of America

Primary authors: ANDRUS, Jason (Idaho National Laboratory); GRABASKAS, David (Argonne National Laboratory); BUCKNOR, Matthew (Argonne National Laboratory); Mr HENNEKE, Dennis (GE-Hitachi Nuclear Energy); LI, Yunlong (Jonathan) (GE Hitachi Nuclear Energy); Mr REISS, Troy (INL); Mr GERSTNER, Doug (INL)

Presenter: ANDRUS, Jason (Idaho National Laboratory)

Session Classification: 2.1 General Safety Approach

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **269** Type: **ORAL**

Progress in conceptual design of a pool-type sodium-cooled fast reactor in Japan

Tuesday, April 19, 2022 1:36 PM (12 minutes)

Country/Int. organization

Japan

Primary authors: Mr KATO, Atsushi (JAEA); Mr KUBO, Shigenobu (JAEA); Dr CHIKAZAWA, Yoshitaka (JAEA); Mr MITAGAWA, Takayuki (JAPC); Mr UCHIDA, Masato (JAPC); Mr SUZUNO, Tetsuji (MFBR); Mr ENDO, Junji (MFBR); Mr KUBO, Koji (MFBR); Mr MURAKAMI, Hisatomo (MFBR); Ms TOKIZAKI, Minako (MFBR); Mr OKAZAKI, Hitoshi (MFBR); Mr UZAWA, Masayuki (MFBR); Mr SHIMIZU, Ryo (MFBR)

Presenter: Mr KATO, Atsushi (JAEA)

Session Classification: 1.1 Overviews and Fundamentals of Fast Reactors

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **270** Type: **ORAL**

France-Japan Collaboration on the SFR Severe Accident Studies: Outcomes and future work program

Tuesday, April 19, 2022 3:46 PM (12 minutes)

Country/Int. organization

Japan

Primary authors: Mr KUBO, SHIGENOBU (JAEA); Dr YAMANO, HIDEMASA (JAEA); Mr SHIBATA, A; Mr IITSUKA, T (MHI); Dr PAYOT, F (CEA); Dr BERTRAND, F (CEA); Dr BACHRATA, A (CEA); Dr GOSSE, S (CEA); Dr JOURNEAU, C (CEA); Dr SAAS, L (CEA); Dr CARLUEC, B (FRAMATOME); Dr CZARNY, O (FRAMATOME)

Presenter: Mr KUBO, SHIGENOBU (JAEA)

Session Classification: 2.1 General Safety Approach

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **271** Type: **ORAL**

Conceptual design of ultra-long life hybrid micro modular reactor cooled by potassium heat pipe

Wednesday, April 20, 2022 3:04 PM (12 minutes)

Country/Int. organization

Korea, Republic of

Primary author: Mr JANG, Seongdong (Korea Advanced Institute of Science and Technology (KAIST))

Co-author: KIM, Yonghee (Korea Advanced Institute of Science and Technology)

Presenter: Mr JANG, Seongdong (Korea Advanced Institute of Science and Technology (KAIST))

Session Classification: 1.2 Innovative Design Advances

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **272** Type: **ORAL**

Objectives and Status of Neutronics Sub-exercises of the OECD/NEA Benchmark for Uncertainty Analysis in Modelling for Design, Operation and Safety Analysis of SFRs

Wednesday, April 20, 2022 11:16 AM (12 minutes)

Country/Int. organization

United States of America

Primary author: BOSTELMANN, Friederike (Oak Ridge National Laboratory)

Co-authors: AURES, Alexander (GRS); Prof. PAUTZ, Andreas (Paul Scherrer Institut); WAN, Chenghui (NECP); LEE, Deokjung (UNIST); FRIDMAN, Emil (Helmholtz-Zentrum Dresden-Rossendorf); FRIDMAN, Emil (HZDR); HILL, Ian (OECD/NEA); TRIVEDI, Ishita (NCSU); HOU, Jason (NCSU); ZENG, Kaiyue (NCSU); VELKOV, Kiril (GRS); IVANOV, Kostadin (NCSU); BUIRON, Lauren (CEA); BERNER, Nadine (GRS); STAUFF, Nicolas (Argonne National Laboratory); LIANG, Qiao (NECP); ZWERMANN, Winfried (GRS); DU, Xianan (UNIST); ZHENG, Youqi (NECP); JO, Yunki (UNIST)

Presenter: BOSTELMANN, Friederike (Oak Ridge National Laboratory)

Session Classification: 6.1 Neutronics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **275** Type: **ORAL**

Core Design of 100MWe Advanced Nitride-fueled Simplified Liquid Metal Cooled Fast Reactor

Wednesday, April 20, 2022 2:16 PM (12 minutes)

Country/Int. organization

Korea, Republic of

Primary author: Mr NGUYEN, Tung Dong Cao (Ulsan National Institute of Science and Technology)

Co-authors: Mr KIM, Ji Yong (Ulsan National Institution of Science and Technology); Ms CHOE, Jiwon (Ulsan National Institute of Science and Technology); Prof. BANG, In Cheol (Ulsan National Institute of Science and Technology); Prof. LEE, Deokjung (Ulsan National Institute of Science and Technology)

Presenter: Mr NGUYEN, Tung Dong Cao (Ulsan National Institute of Science and Technology)

Session Classification: 1.2 Innovative Design Advances

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **277** Type: **ORAL**

Modeling and Simulation of Source Term for Sodium-Cooled Fast Reactor under Hypothetical Severe Accident: Sodium Fire and Radionuclide Transport in Containment

Wednesday, April 20, 2022 2:16 PM (12 minutes)

Country/Int. organization

United States of America

Primary authors: Dr CHANG, Jong E. (TerraPower, LLC); Mr LI, Shirui (CIAE); Ms REN, Lixia (CIAE); Mr SUN, Hongping (XJTU); Mr ZHANG, Yapei (XJTU); Mr LIEGEARD, Clement (CEA); Mr A., John Arul (Indira Gandhi Centre for Atomic Research, India); S., Raghupathy (Indira Gandhi Centre for Atomic Research, Kalpakkam); Ms MOSUNOVA, Nastasya (Nuclear Safety Institute of the Russian Academy of Sciences (IBRAE RAN)); Mr TARASOV, O.V. (IBRAE RAN); Mr HERRANZ, L.E. (CIEMAT); Ms GARCIA, Monica (CIEMAT)

Presenter: Dr CHANG, Jong E. (TerraPower, LLC)

Session Classification: 2.3 Accident Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **281** Type: **ORAL**

Neutronics Benchmark of CEFR Start-Up Tests: Temperature Coefficient, Sodium Void Worth, and Swap Reactivity

Tuesday, April 19, 2022 1:36 PM (12 minutes)

Country/Int. organization

Korea, Republic of

Primary authors: CHOE, Jiwon (Ulsan National Institute of Science and Technology); BATKI, Bálint (HUNGARIAN ACADEMY OF SCIENCES CENTRE FOR ENERGY RESEARCH); DAVIES, Una (University of Cambridge); WON, Jong Hyuck (Korea Atomic Energy Research Institute (KAERI)); LEE, Min Jae (Korea Atomic Energy Research Institute (KAERI)); BATRA, Chirayu (IAEA); KRIVENTSEV, Vladimir (IAEA); BODI, Janos (Paul Scherrer Institute); MIKITYUK, Konstantin (Paul Scherrer Institut); ZHENG, youqi; Dr DU, Xianan (Xi'an Jiaotong University); Prof. LEE, Deokjung (Ulsan National Institute of Science and Technology); QUOC TRAN, Tuan (Ulsan National Institute of Science and Technology); PATAKI, István (Centre for Energy Research (CER)); TÓTH, Martos (Centre for Energy Research (CER)); FRIDMAN, Emil (Helmholtz-Zentrum Dresden-Rossendorf); KIM, Taek Kyum (Argonne National Laboratory); Dr JARRETT, Mike (Argonne National Laboratory); GOMEZ TORRES, Armando Miguel (Instituto Nacional de Investigaciones Nucleares); LOPEZ, Roberto (National Institute for Nuclear Research); TANINAKA, Hiroshi (Japan Atomic Energy Agency); SZOGRADI, Marton (VTT Technical Research Centre of Finland); Dr GIUSTI, Valerio (Università di Pisa); DI PASQUALE, Simone (Nuclear and Industrial Engineering (N.I.N.E.)); PETRUZZI, Alessandro (Nuclear and Industrial Engineering (N.I.N.E.)); SCIORA, Pierre (CEA)

Presenter: CHOE, Jiwon (Ulsan National Institute of Science and Technology)

Session Classification: Special Session: IAEA Coordinated Research Projects

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **283** Type: **POSTER**

A Study on the Development of a Procedure Complexity Evaluation and Optimization for Operating Procedures of China Experimental Fast Reactor

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

China

Primary author: Mrs ZHOU, Qi

Presenter: Mrs ZHOU, Qi

Session Classification: Poster Session

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **285** Type: **ORAL**

Neutronics Benchmark of CEFR Start-Up Tests: An IAEA coordinated research project

Tuesday, April 19, 2022 1:00 PM (12 minutes)

Country/Int. organization

IAEA

Primary authors: BATRA, Chirayu (IAEA); KRIVENTSEV, Vladimir (IAEA); GOMEZ TORRES, Armando Miguel (Instituto Nacional de Investigaciones Nucleares); KIM, Taek Kyum (Argonne National Laboratory); Dr JARRETT, Mike (Argonne National Laboratory); DAVIES, Una (University of Cambridge); FRIDMAN, Emil (Helmholtz-Zentrum Dresden-Rossendorf); ZHENG, youqi; Dr DU, Xianan (Xi'an Jiaotong University); Prof. LEE, Deokjung (Ulsan National Institute of Science and Technology); QUOC TRAN, Tuan (Ulsan National Institute of Science and Technology); CHOE, Jiwon (Ulsan National Institute of Science and Technology); BATKI, Bálint (HUNGARIAN ACADEMY OF SCIENCES CENTRE FOR ENERGY RESEARCH); PATAKI, István (Centre for Energy Research (CER)); TÓTH, Martos (Centre for Energy Research (CER)); BODI, Janos (Paul Scherrer Institute); MIKITYUK, Konstantin (Paul Scherrer Institut); TANINAKA, Hiroshi (Japan Atomic Energy Agency); SZOGRADI, Marton (VTT Technical Research Centre of Finland); DAŘÍLEK, Petr (VUJE Inc., Okružná 5, SK91864 Trnava, Slovakia); Dr GIUSTI, Valerio (Università di Pisa); DI PASQUALE, Simone (Nuclear and Industrial Engineering (N.I.N.E.)); PETRUZZI, Alessandro (Nuclear and Industrial Engineering (N.I.N.E.)); SCIORA, Pierre (CEA)

Presenter: MORELOVA, Nikoleta (IAEA)

Session Classification: Special Session: IAEA Coordinated Research Projects

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **286** Type: **POSTER**

Leak-Before-Break Design of Double-Walled Once-Through Steam Generators for Lead Cooled Fast Reactor

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

Korea, Republic of

Primary authors: KIM, Taeyong (UNIST); LEE, Jeonghyeon (Ulsan National Institute of Science and Technology); Mr HWANG, Il Soon (UNIST); Prof. KIM, Ji Hyun (UNIST)

Presenter: KIM, Taeyong (UNIST)

Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **293** Type: **POSTER**

Cognitive Information Retrieval Based on Ontological Model of Knowledge Representation

Thursday, April 21, 2022 10:40 AM (2 hours)

The technologies of information retrieval in a database with full-text semantic indexing are considered. The information retrieval process is considered as a cognitive-oriented process. The semantic image of the document context is presented as an ontology. An ontology is defined as a set of three interconnected systems (functional, conceptual and terminological), on which the operation of comparing elements of different systems at the level of signs is defined. A functional system (a system of tasks, objects, processes, properties of the subject area) represents objects and situational relationships between them in the context of the target activity. The objects of the conceptual system are stable concepts, and the set of relationships is limited to generic and associative relationships. The terminological system reflects the properties of a natural language at the level of signs –terms for which relationships of equivalence and inclusion, as well as linguistic relationships, are specified. The semantics of the document are represented as a multi-meta-hyper-graph of the ontology. Such graph in the nodes contains named entities (concepts, names, values, etc.), and in the edges –typed relationships, extracted from the text also taking into account the location of them. This made it possible to build up mechanisms (algorithms) for a new type of information retrieval –searching semantic dependencies and neighborhoods within the text of a document or a selection of documents. Such cognitive information retrieval is considered as a search for a path on a multi-meta-hyper-graph of an ontology dynamically formed on the basis of ontological images of found documents or their fragments. A Cognitive subject tree is used to fix and manage search directions. A Cognitive subject tree is a hierarchically ordered structure of a personified representation of a subject area (project, task). Such a structure integrally identifies tasks / knowledge, since each structural and semantic component (node) of the tree includes not only keywords, but also relevant documents texts, query expressions, indexes of classifiers, etc. The prototype of the information retrieval system has been developed and experimental databases have been created.

Country/Int. organization

Russian Federation

Primary authors: MAKSIMOV, Nikolay (National research nuclear university (Moscow Engineering Physics Institute)); Mrs GOLITSINA, Olga (National Research Nuclear University MEPhI); MONANKOV, Kirill (National Research Nuclear University MEPhI)

Presenter: MAKSIMOV, Nikolay (National research nuclear university (Moscow Engineering Physics Institute))

Session Classification: Poster Session

Track Classification: Track 9. Education, Profesional Development, and Knowledge Management

Contribution ID: **298** Type: **POSTER**

HEAT TRANSFER CALCULATION AND SERVICE LIFE TIME ESTIMATION OF SUBMERGED ELECTROMAGNETIC PUMP FOR LIQUID LEAD

Wednesday, April 20, 2022 10:40 AM (2 hours)

Country/Int. organization

Russian Federation

Primary authors: FEDERIAEVA, Valeriia (JSC "NIIEFA"); CHAIKA, Pavel (JSC "NIIEFA"); KOMOV, Kirill (JSC "NIIEFA"); OBUKHOV, Denis (JSC "NIIEFA"); PRESLITSKY, Gennadii (JSC "NIIEFA"); TUKEEV, Pavel (JSC "NIIEFA")

Presenter: FEDERIAEVA, Valeriia (JSC "NIIEFA")

Session Classification: Poster Session

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **300** Type: **ORAL**

Research and development of nuclear fuel for fast neutron reactor

Friday, April 22, 2022 2:42 PM (12 minutes)

The following topics are considered in this presentation:
The main issues of nuclear power are connected with spent nuclear fuel management, radioactive waste management and limited stocks of uranium.
Rosatom's strategy of two-component nuclear energy system with closed nuclear fuel cycle based on fast reactors is presented.
The map of Rosatom's fuel company TVEL activity shows the key points of research and development of traditional and innovative types of nuclear fuel.
The main directions of fast reactor fuel activity under TVEL coordination are considered in this report:
Construction and commissioning of industrial BN-800 MOX fuel plant in Krasnoyarsk. The main result achieved during last years is the loading into the BN-800 core the first batch of MOX-fuel assemblies manufactured on fully automatized production lines.
Main elements of the Rosatom strategic project «Proryv» aimed on creating a new technology platform for the nuclear industry with a closed NFC. A Pilot Power Supply Complex (ОДЭК/ODEK) including the U-Pu nitride fuel fabrication-refabrication module (MFR) and a lead-cooled fast reactor BREST-OD-300 are under construction on the TVEL's enterprise at Seversk site.
The following key ways of research and development of fast reactor nuclear fuel are also presented in the report:
- the BN-600 reactor: adoption of a new steel type as a fuel rod cladding material to increase the duration of the reactor fuel campaign;
- the BN-800 reactor: transition to the core fully loaded by MOX fuel and subsequent adoption of a new material of fuel rod cladding to ensure longer fuel campaign;
- the Chinese CFR600 fuel. Development of control and protection system assemblies, appropriate FA and CPSA mock-ups testing, adjustment to FA and CPSA fabrication;
- experimental justification of U-Pu nitride fuel for BREST-OD-300 and BN-1200 reactors. Development and manufacturing of experimental fuel assemblies with U-Pu nitride fuel at Seversk pilot line to irradiate in the BN-600 reactor. The results of reactor tests and PIE will be used to justify the performance of BREST-OD-300 and BN-1200 fuel assemblies with nitride fuel;
- development of sodium-cooled BN-1200 reactor core and assemblies with U-Pu nitride and oxide nuclear fuel;
- development of lead-cooled BREST reactor's core and assemblies with U-Pu nitride fuel;
- development and implementation of new structural materials for fast neutron reactors.

Country/Int. organization

Russian Federation

Primary author: KHRAMTSOV, Maksim

Presenter: KHRAMTSOV, Maksim

Session Classification: 3.4 Advanced Fuel Development

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **302** Type: **ORAL**

French-Japanese experimental collaboration on fuel-coolant interactions in sodium-cooled fast reactors

Friday, April 22, 2022 11:30 AM (12 minutes)

Fuel-coolant interactions (FCI) are important phenomena occurring in the event of core disruptive accidents in sodium-cooled fast reactors (SFR). A new phase of experimental research into FCIs has been launched through a collaboration between French and Japanese organisations. Bespoke high-speed, high-resolution X-ray imaging at the JAEA MELT facility has enabled successful visualisation, in the highest resolution achieved to date, of the quenching of a jet of molten steel in a pool of molten sodium. The FCI experimental program is accompanied by calibration tests undertaken using "phantom"models, which are necessary for detecting imaging artefacts, such as vignetting and optical distortion, and assisting in the development of algorithms to reconstruct melt fragments in 3D from X-ray attenuation data. The CEA have developed SPECTRA (Software for Phase Extraction and Corium TRacking) for processing the images acquired at the MELT facility, enabling the segmentation of melt and vapour phases from the raw images and the tracking of discrete melt fragments traversing the imaging window. The first experiment at the MELT facility under this collaboration revealed evidence of extensive crust formation at the interface between the melt and sodium. Rapid vaporization of entrained sodium appeared to lead to fracturing of the crust during a thermal fragmentation event, resulting in a debris population containing both fine fragments and large frozen jet shells. JAEA have now commissioned a new MELT test section, roughly 10-times greater in sodium capacity, to observe the interaction at increased scale. In parallel, the SERUA (Sodium boiling Experimental Ring for Understanding of fuel-coolant interAction) facility is currently being developed by the CEA to investigate film-boiling heat transfer at the interface between corium droplets and molten sodium, in support of computational research into FCIs in sodium.

Country/Int. organization

France

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Presenter: JOURNEAU, Christophe (CEA)

Session Classification: 2.4 Severe Accidents

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **304** Type: **ORAL**

GFR Research and Development Programme in V4 countries

Tuesday, April 19, 2022 2:24 PM (12 minutes)

Country/Int. organization

Czech Republic

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Presenter: Mr VÁCHA, Petr (UJV Řež, a.s.)

Session Classification: 1.1 Overviews and Fundamentals of Fast Reactors

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **306** Type: **ORAL**

Basis for the Safety Approach (BSA) for Design & Assessment of Generation IV Nuclear Systems

Tuesday, April 19, 2022 3:10 PM (12 minutes)

Country/Int. organization

France

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Presenter: GAUTHÉ, Paul (CEA)

Session Classification: 2.1 General Safety Approach

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **308** Type: **POSTER**

Application of Model Based System Engineering in Design of Digital Fast Reactor Nuclear Power Plant

Friday, April 22, 2022 10:30 AM (2 hours)

Due to the complexity of the fast reactor project and its technical uncertainty, the design needs a long period. In order to improve design and research ability of fast reactor and develop the technology of digital reactor, bring in the model-based systems engineering method for requirement analysis, function decomposition and architecture design and weigh the overall design, to the benefit of discovering design defects early, guaranteeing the traceability and consistency of technical state, reducing duplication of effort for the basis of subsequent systems and equipment design. In this paper, the Harmony-SE process of IBM standard is studied. Combined with the actual situation, the standard process is tailored to form the system demand analysis and dynamic and static architecture design process based on Harmony-SE. Taking a certain system as an example, the system demand analysis and dynamic and static architecture design are carried out.

Country/Int. organization

China

Primary author: YUJIE, Guo

Presenter: YUJIE, Guo

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **309** Type: **ORAL**

Novel neutronics design of the MYRRHA core

Wednesday, April 20, 2022 2:28 PM (12 minutes)

Country/Int. organization

Belgium

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Presenter: FIORITO, Luca

Session Classification: 1.2 Innovative Design Advances

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **311** Type: **POSTER**

New Finite Element Neutron Kinetics Code System FENNECS/ATHLET for Coupled Safety Assessment of (Very) Small and Micro Reactors

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

Germany

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Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **312** Type: **POSTER**

Conceptual Core configuration for increasing Power of Fast Breeder Reactor to 40 MWt

Thursday, April 21, 2022 10:40 AM (2 hours)

Fast Breeder Test Reactor (FBTR) in India is designed for 40 MWt Thermal 13.2 MWe. At present FBTR is operating at 32 MWt with 56 fuel sub assemblies (FSA) of 48 Mark I and 8 MOX type fuel sub assemblies. Mark I FSA are of Pu-U Carbide fuel with 70% Pu and MOX FSA are of PuO_2 (44%) and UO_2 (56%). Due to constraint on minimum shut down margin of 4200 pcm, the core could not be expanded and hence the power could not be increased to design power. A conceptual core configuration has been suggested and safety analysis was being carried out, by introducing four poison sub assemblies, Boron SA, with 50% B10 concentration, in the second ring, which would enable to expand the core and increase the power to the design power of 40 MWt and at the same time minimum criterion on shut down margin also will be met. The envisaged core for 40 MWt comprises 70 numbers of Mark I FSA.

Country/Int. organization

India

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Session Classification: Poster Session

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **313** Type: **POSTER**

Research on the Impact of Advanced Rule Design System on the Digitization of Reactor Building Model

Friday, April 22, 2022 1:30 PM (2 hours)

This paper proposes a more stringent method for customizing project rules. This method customizes the comprehensive rules of the project and component reference database on the digital plant design platform based on some general design codes, standards and item classification principles in nuclear engineering, digitalization requirements in reactor design, plant layout, project management, material procurement and construction, etc. In order to improve the correlation between design specifications and digital power plants, enhance the data consistency among different design disciplines, standardize the three-dimensional layout design of nuclear power plants, ensure the consistency between the digital power plant model and the real power plant, the rules are sorted out, analyzed and transformed systematically in this paper. These rules include the naming and classification principles of items in nuclear power projects, model data composition structure, essential attribute content, component selection filters, material performance, model parameters, output content format, basic requirements for plant layout for reactor design, etc. Through refinement and improvement, this paper finally forms a systematic rule customization scheme, which includes parameters such as process, operating conditions, materials, fluids, specifications, safety, quality assurance, seismic and radioactivity levels, as well as items naming rules, project database, component reference database, three-dimensional modeling, information integration, attribute inheritance, data extraction and other rules. This scheme can make the three-dimensional arrangement more standard, the operation steps more concise, and greatly reduce the attribute range of manual input by the designer. It can effectively promote the accurate and appropriate expression of process and instrumentation process design scheme in the reactor building model. Significantly shorten the project design cycle. Data integration and transmission between rules enable system attributes to be deeply inherited and automatic checking and judging of operating conditions parameters and pressure and temperature limits in physical properties of component materials. This scheme can make the three-dimensional layout more standard, the operation steps more concise, greatly reduce the attributes range of manual input by the designer, and obtain better application feedback in automatic drawing and material report. These rules provide more comprehensive data support for coupling experiments, data integration, process simulation and digital handover of different disciplines and depths.

Country/Int. organization

China

Primary authors: WANG, Jin; Prof. YANG, Hongyi

Presenters: WANG, Jin; Prof. YANG, Hongyi

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **316** Type: **ORAL**

Neutronics analysis of CEFR Start-up tests at IGCAR using FARCOB and ERANOS 2.1 Code Systems

Wednesday, April 20, 2022 10:52 AM (12 minutes)

Country/Int. organization

India

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Session Classification: 6.1 Neutronics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **317** Type: **ORAL**

Creep and Tensile Properties of Indian Advanced Fast Reactor Clad tubes (IFAC-1) for Future FBRs

Tuesday, April 19, 2022 1:00 PM (12 minutes)

Country/Int. organization

India

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Presenter: Dr G. V., Prasad Reddy (Head, Creep Studies Section & Assistant Prof., HBNI)

Session Classification: 4.1 Advanced Reactor Cladding and Core Material, Coolants, and Related Chemistry

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **318** Type: **ORAL**

KEY ASPECTS OF COMPETITIVENESS FOR INDUSTRIAL ENERGY COMPLEX WITH FR AND CLOSED NFC

Tuesday, April 19, 2022 3:22 PM (12 minutes)

Country/Int. organization

Russian Federation

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Presenter: Mr TOLSTOUKHOV, Dmitriy

Session Classification: 7.1 Sustainability: Economics, Environment, and Proliferation

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **319** Type: **ORAL**

LOW ENRICHMENT NUCLEAR FUEL BASED ON URANIUM-ZIRCONIUM CARBONITRIDE: REACTOR TESTS AND PREPARATION FOR STUDIES AT CRITICAL ASSEMBLIES

Friday, April 22, 2022 2:18 PM (12 minutes)

Uranium-zirconium carbonitride has been developed at the LUCH FSUE and is a high-density high-temperature fuel with high heat conductivity capable of being used in various types of reactors, including fast reactors. The main problem hindering wide application of this fuel is insufficient knowledge of its behavior under irradiation, especially at high burnup. In the USSR, HEU UZrCN (96% by U-235) fuel underwent reactor testing to a low burnup of approximately 1%. However, to confirm practicality of application of this fuel, it needs to be reactor tested at a high burnup. In the framework of a joint Belarussian-American-Russian effort, LEU (19.75% by U-235) UZrCN fuel will undergo reactor testing to a burnup of approximately 40% in the SM-3 reactor at the JSC "SSC RIAR".

To conduct the prolonged reactor experiment, an irradiating device with an experimental capsule containing UZrCN pellets has been made, neutronic analysis and thermophysical analysis have been carried out and a programme of pre-irradiation experiments has been implemented.

In June 2019 a methodical experiment was carried out in the SM-3 reactor at the JSC "SSC RIAR"in order to confirm operability of the irradiating device developed. Testing of the irradiating device and refinement of the prolonged irradiation experiment procedure were carried out at cell 11 of the reflector of the SM-3 reactor at the JSC "SSC RIAR". The period of irradiation equaled 23.3 effective days. The mean power density in the tested pellets throughout the methodical reactor testing was 516 W/cm3. Burnup achieved for the pellets studied was 0.63% fifa.

The "Giacint"and "Kristal"critical facilities of the Scientific Institution "JIPNR - Sosny"will be used for studying neutronic characteristics of critical and subcritical fast assemblies simulating the physical particulars of the cores of advanced gas- or liquid metal cooled fast reactor systems and accelerator driven systems.

This paper provides a detailed description of the results of preparatory works for conducting the reactor testing and the results of the methodical reactor experiment. Also presented are the experimental programme and the descriptions of the design and composition of the fast critical assemblies with UZrCN fuel.

Country/Int. organization

Belarus

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Session Classification: 3.4 Advanced Fuel Development

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **322** Type: **ORAL**

Influence of Low Dose Irradiation on Permanent Core Structural Materials of PFBR

Tuesday, April 19, 2022 1:48 PM (12 minutes)

Country/Int. organization

India

Primary authors: Mr KUMAR , Ran Vijay (Indira Gandhi Centre for Atomic Research); KOLHATKAR, Ashish (IGCAR); V V, JAYARAJ (Indira Gandhi Centre for Atomic Research); Dr C, Padmaprabu (Indira Gandhi Centre for Atomic Research); Ms M, Shreevalli (Homi Bhabha National Institute (HBNI)); VENKATASUBRAMANIAN, Karthik (Indira Gandhi Centre for Atomic Research); C.N., VENKITESWARAN (IGCAR); PADMANABHAN, Parameswaran (Indira Gandhi Centre for Atomic Research); RAMACHANDRAN, Divakar (Indira Gandhi Centre for Atomic Research); Dr K, Shaju Albert (Group Director, Indira Gandhi Centre for Atomic Research)

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Session Classification: 4.1 Advanced Reactor Cladding and Core Material, Coolants, and Related Chemistry

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **323** Type: **ORAL**

MYRRHA, the Belgian prototype that fascinates the world

Tuesday, April 19, 2022 2:12 PM (12 minutes)

Country/Int. organization

Belgium

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Presenter: Dr DE BRUYN, Didier (SCK CEN)

Session Classification: 1.1 Overviews and Fundamentals of Fast Reactors

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **324** Type: **ORAL**

Preliminary testing of ALFRED DHR System

Friday, April 22, 2022 1:42 PM (12 minutes)

Lead fast reactors are of particular interest thanks to the characteristics of the coolant which simplifies numerous plant choices while maintaining high levels of safety and reliability. In Europe, ALFRED is the main technology demonstrator and leveraging on the knowledge developed in the evolutionary generation III+ plants it makes use of passive safety systems, for example for the decay heat removal. One of the main limitations of passive safety systems is the lack of an intrinsic control power removal. This is particularly restraining when they are applied to liquid metal reactors because the coolant solidifies at a higher temperature than the external environment, which commonly constitutes the final heat sink. Coolant solidification poses the risk of inhibiting the natural circulation of the primary coolant and with it the decay heat removal, as well as possible mechanical stresses induced by the volume change from liquid to solid states. Ansaldo Nucleare patented a passive safety system for decay heat removal able to passively control the power removed to the final heat sink, exploiting the presence of non-condensable gases in strategic positions of the circuit. The system consists of 3 loops, each one is connected to one steam generator and is equipped with an Isolation Condenser immersed in a pool and connected to a gas storage tank. In 2016, thanks to a partial funding from the Italian Ministry for Economic Development, Ansaldo Nucleare together with ENEA, SIET and SRS started a project called SIRIO to design, build and test an experimental facility scaled with respect to ALFRED's DHR system to qualify the operating principle. Scaling is carried out by keeping constant power density and adopting the same operating conditions of pressure and temperature, together with the height of the real system. This paper aims to describe the facility by showing the scientific scaling aspects and discuss the first experimental campaign carried out on the facility. The behaviour of the main parameters of interest for the safety system are reported, such as the total pressure and temperature in the various areas of the system, which allow us to infer the concentration and transport of non-condensable gases. The results also guarantee to quantify the regenerative heat transfer of the bayonet steam generator that contribute to the operation of the passive power control system. The experimental data will be used for the qualification of the operating principle and the detailed design of the full scale system.

Country/Int. organization

Italy

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Presenter: Mr CARAMELLO, Marco (Ansaldo Nucleare S.p.A.)

Session Classification: 5.3 Experimental Programs II

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **325** Type: **ORAL**

Development of SFR core degradation simulation code SIMMER-V and its validation & verification studies

Thursday, April 21, 2022 12:04 PM (12 minutes)

In the framework of reactor safety analysis of Sodium cooled Fast Reactors (SFR) using applicable code systems, the CEA and JAEA are involved in the achievement of SIMMER-V (owned by JAEA and co-developed by JAEA and CEA) developments dedicated to the simulation of the Severe Accident (SA) events of SFR.

The demand for new physical models into SIMMER-V rises in SIMMER previous versions related works, since new features are necessary in order to give a mechanistic evaluation of SA events. In order to enhance functions of SIMMER-V code system, a ranking of additional models has been proposed i.e. the detailed pin model or more adapted heat exchange correlations. Regarding the importance of predicting severe accident progression and consequences, verification and validation (V&V) work has been planned and conducted to get high value feedback on the applicability of the SIMMER-V to SFR severe accident simulation.

This paper outlines the CEA-JAEA collaboration on SIMMER-V development including V&V work, its achievement and perspective. The following two exercises of v&v work are highlighted.

First exercise presented in this article will consist on the validation of a natural convection correlation in fuel molten pools through the SCARABEE experimental program dedicated to the study of the Total Instantaneous Blockage (TIB) severe accident scenario in a SFR. Among the different tests performed, the program BF ("Bain Fondu"which means molten pool in French) aimed to study the behavior of molten and boiling heat-generating fuel pools using real reactor materials. Then, this study focuses on the evaluation of BF tests input data set using the latest SIMMER-V version and on the implementation and validation of the Chawla-Chan natural convection correlation in SIMMER-V sources.

The second exercise will focus on the verification of fluid dynamics scheme on SIMMER-V considering the updated test (2019) of classical benchmark problem of the Ideal Shock Tube, which was already part of SIMMER-III validation test basis. The main objectives are 1) to confirm the capability of SIMMER- V to simulate, in 1-D geometry, a single compressible fluid flow with strong pressure gradients, and 2) to verify the stability of results when varying mesh discretization, pressure solver options, time step control, domain decomposition and check basic conservation laws.

Country/Int. organization

France

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Presenter: Dr MARTIN LOPEZ, Elena (CEA)

Session Classification: 6.3 Multiscale and Multiphysics Calculations

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **332** Type: **ORAL**

Development of Plasma Nitriding as alternate hardfacing technique for Large components of FBR and Assessment of static In-Sodium Stability of Plasma Nitrided Layer

Wednesday, April 20, 2022 10:52 AM (12 minutes)

Country/Int. organization

India

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Presenter: Dr MYTHILI, R. (IGCAR)

Session Classification: 4.2 Structural, Novel, and Large Components Materials

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **336** Type: **ORAL**

Experiment and Numerical Simulations on SFR Core-catcher Safety Analysis after Relocation of Corium

Friday, April 22, 2022 11:54 AM (12 minutes)

An In-vessel core catcher located in European type SFR is a safety design feature to guarantee the integrity of the SFR during a core-melting accident. The core catcher collects and distributes the relocated melt from the core region via discharging tubes to avoid firstly a significant molten pool in the core, and secondly the local thermal attack on bottom part of the vessel. The heat transfer and ablation behavior of core-catcher material under the thermal load of the core melt at high-temperature and with high decay heat will be studied experimentally with a new facility in Karlsruhe Institute of Technology and numerically in CEA within the ESFR-SMART European project.

The new facility, named ESFR-LIVE, is a 3-dimensional model of Core-catcher in a length scale of 1:6. The lower part of the tray-type core-catcher is a truncated cone and the upper part is a cylinder with 1 m diameter. The cooling of liquid Na at all boundaries is simulated by a water cooling channel enclosing the test vessel, and a cooling lid at the upper surface of the simulate. Four planes of individually controllable heater in the vessel enable the variation of the height of the core melt, and the shapes of the melt pool. The distribution of bulk temperature, boundary temperature, wall temperatures and heat flux can be measured or determined. The simulant of core-melt is the eutectic NaNO3-KNO3 mixture, which is representative for the character of the general liquid oxide melt. Similarity comparisons on geometry, material properties and heat transfer features are presented in this paper. With the similar pool height, the EFSR-LIVE facility can well capture the dimensionless heat transfer features, e.g. Ra and Nu.

Computational Fluid Dynamics (CFD) pre-calculations of the LIVE-ESFR experiments will be performed on the basis of the ESFR-LIVE test conditions. These simulations will be performed with the TrioCFD code using High Performance Computing on ten million of mesh nodes. A parametric study according to the heating power distributions will be presented.

Regarding the turbulent flows with large Rayleigh numbers, these simulations require to use Large Eddy Simulation (LES) models. This study is supplemented by a mesh resolution analysis and a validation of TrioCFD on natural convection turbulent flows using past molten-pool experiments (BALI) and academic results.

Country/Int. organization

Germany

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Session Classification: 2.4 Severe Accidents

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **338** Type: **ORAL**

System Safety Assessment of the Generation IV Lead Fast Reactor

Tuesday, April 19, 2022 3:34 PM (12 minutes)

Country/Int. organization

European Commision

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Session Classification: 2.1 General Safety Approach

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **340** Type: **ORAL**

Mechanisms Engineering Test Loop (METL) Facility

Wednesday, April 20, 2022 2:28 PM (12 minutes)

Country/Int. organization

United States of America

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Session Classification: 5.1 Experimental Reactors and Facilities

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **341** Type: **ORAL**

Evaluation of an increase of the power density for the French commercial Sodium Fast Reactor and optimization study at 1100 MWe with the SDDS tool

Friday, April 22, 2022 11:54 AM (12 minutes)

In order to enhance the competitiveness and to reduce the construction cost of the future industrial Sodium Fast Reactors (SFR), several options are explored which need further R&D studies or design assessment. Among them, the possibility to reduce the size of the reactor vessel has been investigated through the reduction of the core diameter and the increase of the power density thanks to several optimisation studies conducted by the R&D department of EDF.

To this end, an in-house multi-physics optimization tool called SDDS has been used. The basis of the method is to predict the performance of a large number of core designs using surrogate models. The surrogate models are themselves created using the results of a parametric calculation scheme based on the following codes: ERANOS for the neutronic, MAT5DYN for the thermal-hydraulic transients and GERMINAL for the thermomechanical fuel performance.

A first optimization study has been performed in 2018 to define a compact core design for the 1000 MWe French commercial SFR. As a result, two designs were selected as they offered a good compromise between the safety criteria and a reduced core diameter: a twelve Sub-Assembly (SA) rings core with a smaller core diameter and a thirteen SA rings core with better safety margins.

This paper focuses on a second optimisation study which has been performed more recently on a 1100 MWe power reactor in order to evaluate the impact of an elevation of 10% of the nominal power on the results. The analysis of the results shows that the main trends (e.g. large pellet, large fertile plate height, etc.) are the same as the ones observed for the optimum designs in the 2018 study. However, the designs selected in 2018 do not meet some of the safety criteria anymore after the increase of their power density. Thus, a new core design with 13 SA rings has been proposed with a better compromise between safety performances and core diameter to operate at 1100 MWe.

Country/Int. organization

France

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Presenter: Dr GÉRARDIN, Delphine (EDF)

Session Classification: 1.3 System Innovations

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **343** Type: **ORAL**

Integration of Small Modular Lead Fast Reactor with Energy Storage for load-following operation in high V-RES penetration electricity markets

Friday, April 22, 2022 11:30 AM (12 minutes)

Energy decarbonisation, through the transition from fossil fuels to V-RES electricity production and the electrification of transport & heating sectors, may jeopardise the electricity supply security on the long term, because of the growing power demand and the increased production volatility. While advanced and modular reactor designs can make nuclear an attractive low-carbon solution to diversify the energy mix and address the power demand increase, a paradigmatic change is required in both NPP design and operation to increase load-following mode attractiveness. Indeed, although the current nuclear Gen-III/III+ fleet provide good load following capabilities and some operators (especially in high nuclear share markets as France) find profitable operating NPP in load-following mode, most of nuclear generating units are operated in baseload mode. This paper investigates the feasibility and the potential of integrating a cost-effective Energy Storage system into a Small Modular Lead Fast Reactor, to achieve load-following performances while maintaining the reactor at high power levels minimizing power excursions. Indeed, the integration of Energy Storages in Gen-IV reactors may significantly boost nuclear competitiveness in high V-RES penetration electricity markets, by combining the economic benefits of running nuclear reactors at high power (i.e., efficient use of capital invested in plants, simplicity and reliability of the operations) with the plant load-following capacity, compensating V-RES volatility. The paper investigates the Energy Storage option under a wide and comprehensive perspective, from the description of the reference electricity market with the identification of specific national grid requirements down to the Energy Storage technology selection, integration with the balance-of-plant and preliminary sizing, to best-fit the load-following demand and LFR specificities. Romania has been selected as reference scenario for the investigation, due to the representative energy-mix with high RES penetration (42%), a large use of hydropower (27%) to compensate for wind and solar volatility as well as the consolidated use of nuclear power (18%) as baseload.

Country/Int. organization

Italy

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Session Classification: 1.3 System Innovations

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **345** Type: **ORAL**

ALFRED High priority R&D Needs

Friday, April 22, 2022 1:30 PM (12 minutes)

The identification of the R&D priorities and needs is a necessary step to complete the development, up to the qualification and demonstration, of the solutions envisaged for LFR technology. In particular, this is of paramount importance to allow the design, licensing and construction of industrial systems.
In the present work, starting from the key scientific aspects (including lead chemistry monitoring and control, thermal-hydraulics in large pools, components qualification and integral system operation), the necessary experimental activities are identified in support of the ALFRED safety demonstration program.
According to the identified R&D needs and considering the current status of the ALFRED conceptual design as well as of the research infrastructures'implementation in Romania, a prioritization of activities is also proposed, which takes into account also the actual worldwide level of knowledge on the LFR technology.
Finally, by taking advantage of the experimental activities, the needs for Verification and Validation (V&V) of the computational tools to be used for safety demonstration are also aligned.

Country/Int. organization

Italy

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Session Classification: 5.3 Experimental Programs II

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **348** Type: **POSTER**

ALFRED DHR system scaling verification and numerical pre-test analysis

Thursday, April 21, 2022 10:40 AM (2 hours)

Passive safety systems are used in generation III+ evolutionary reactors and in generation IV advanced reactor designs, especially for the decay heat removal following an accidental event. These systems allow with one or more loops the heat transfer from the primary system to the external environment through the natural circulation of fluids or through boiling and condensation phenomena. A limitation of these systems is the lack of intrinsic mechanisms that allow the passive control of the power removed, and this is particularly important for reactor designs that involve the use of liquid metals as a coolant, as the solidification temperature is always higher than the temperature of the ultimate heat sink. Under these circumstances, there is the possibility that the primary system coolant may freeze in the colder regions of the system, opposing to the natural circulation of the fluid and in eventually inducing mechanical stresses due to the volume variation between liquid and solid state. Ansaldo Nucleare has patented a passive safety system for the ALFRED reactor that allows to control the power removed to the environment by making use of non-condensable gases placed in strategic positions of the safety system. The DHR system consists of 3 loops, each connected to one steam generator through the feedwater and the steamline. These circuits have an isolation condenser immersed in a pool and connected to an expansion tank. During safety system operation, the non-condensable gases are passively transported in the system between the tank and the isolation condenser proportionally to the decay heat, degrading the heat transfer in the condenser and strongly delaying the onset of solidification in the primary coolant. This paper is based on the scientific effort dedicated through the European H2020 PIACE project and reports the scaling verification of the decay heat removal system to be carried out at the SIRIO experimental facility. Pre-test analyses performed by means of the RELAP5-3D system code are presented, assessing the applicability of an existing facility configuration to the revised design of the DHR system of ALFRED. The results show how the experimental facility is able to represent the most important phenomena underlying the operating principle of the system such as pressure behaviour, noncondensables gas transport and coolant temperature control above solidification point.

Country/Int. organization

Italy

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Presenter: Mr CARAMELLO, Marco (Ansaldo Nucleare)

Session Classification: Poster Session

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **350** Type: **ORAL**

Regulatory Perspectives on Analytical Codes and Methods for Advanced Reactors

Thursday, April 21, 2022 2:28 PM (12 minutes)

Analytical codes and methods are used extensively in the design and safety analysis of nuclear reactors. These are commonly used to analyse the response of a complex engineering system to postulated events with potentially severe health, financial, and environmental implications. Regulatory agencies establish requirements and/or expectations on the nuclear power plant designer or licensee for the development and use of analytical codes and methods in order to ensure the quality, credibility, and confidence in the analyses produced by the analytical codes and methods. In addition, regulatory agencies have used analytical codes and methods to perform confirmatory analyses as part of due diligence during a regulatory review. The Task Group on Analytical Codes and Methods (TGACM) of the OECD-NEA Working Group on the Safety of Advanced Reactors (WGSAR) has performed a review to (1) identify and clarify the requirements and best practices applicable to nuclear power plant designers for the development and use of analytical codes and methods used in the design and safety analysis of nuclear power plants, and (2) identify best practices for the use of confirmatory analyses by regulatory agencies.
In this paper first results of this on-going work will be presented, based on the responses to a survey from Canada, France, Germany, Italy, Russia, UK and USA. The partly different procedures and expectations on regulatory approval of codes and methods, quality assurance program and handling of possible bugs and errors will be discussed. For example, some countries require the code developer to undergo a certification process. In addition, in some countries qualification of code users and / or organisations is required.
The second part of the survey is related to confirmatory analysis. Because the objective of these confirmatory analysis is mainly linked to support the regulator, the required capabilities and expectations are partly different to codes used for the design and optimization of advanced reactors. Independent from the claimed inherent safety capabilities of reactor concept, a simulation of severe accident phenomena is expected by regulatory authority.
In the review an overview on existing codes used for advanced reactors will be provided. The capabilities of these will be discussed and compared with safety relevant phenomena of these advanced reactor concepts.
In conclusion, the regulatory expectations related to codes used for advanced reactors should be considered in the development of these codes. Comparing code capabilities with safety relevant phenomena, the review provides information on further code development needs.

Country/Int. organization

Germany

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Presenter: KLEIN-HESSLING, Walter

Session Classification: 6.4 Simulation Tools for Safety Analysis

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **351** Type: **ORAL**

The Versatile Test Reactor (VTR) Approach to Sodium Fire Hazards Analysis and Protection System Methodology

Wednesday, April 20, 2022 3:04 PM (12 minutes)

Country/Int. organization

United States of America

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Presenter: BUCKNOR, Matthew (Argonne National Laboratory)

Session Classification: 2.3 Accident Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **353** Type: **POSTER**

Software for Simulation of Fast Reactor Operation in a Closed Nuclear Fuel Cycle (SC RTM-2)

Friday, April 22, 2022 10:30 AM (2 hours)

Fast reactor (FR) operation in closed nuclear fuel cycle (CNFC) is accompanied by the change in isotopic composition of a recycled fuel during a prolonged period of time (10-30 years). Series of similar calculations are required to determine optimal parameters of core charge and operational conditions during this transient phase of FR operation. To solve this problem, a RTM software complex has been developed. The software complex incorporates simulation of the reactor core and process stages of a fuel cycle. The RTM is intended for simulation of normal FR operating modes in the CNFC. Aside of simulation of the transient mode, the software complex is used to evaluate different scenarios of afterburning of long-lived actinides.
The software complex unites several computational modules: a code for neutron physical computation in diffusion approximation, a computational code for nuclear kinetics, and a code for simulation of process stages of a fuel cycle. The nuclear kinetics code allows one to calculate detailed isotopic composition of fission fragments in order to evaluate the activity and heat generation of spent nuclear fuel and radioactive waste. The process stage model is a balance model. The computational codes are united and controlled by a system shell. The system shell is a graphical interface designed to input initial data, and to view, compare and process the results of calculations. The system shell also includes a subsystem that checks the consistency of the initial data, and the scenarios and algorithms of solving benchmark problems.
The computational codes used in the RTM are not newly-developed products. However, integrating these codes into one software tool with a common system of initial data input and analysis of results makes it possible to obtain a new product for solving numerous problems of the same type. Such an integration of codes expands the capabilities of simulation of FR fuel cycles, reduces the scope of work required from a user to set the initial data, decreases the number of introduced errors, and enhances the efficiency of analysis of the obtained results.

Country/Int. organization

Russian Federation

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Presenter: POPOV, Ivan

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **356** Type: **POSTER**

The Code Complex for Computational Evaluation of Technical Solutions and Optimization of Processes Parameters of CNFC

Friday, April 22, 2022 1:30 PM (2 hours)

The advanced development of NPE assumes gradual embedding fast neutron reactors, which enshures the most complete use of the uranium and thorium resources. Even on the theoretical level there are some alternative solutions for organization of operation of reactor facility and the fuel cycles. The experimental verification requires the considerable amount of time and serious material resources. The mathematical simulation is an effective method for evaluation of acceptability of technology development and organization of fuel cycle. Also these decisions must be coordinated at every stage of NFC: strategic, system and technologycal.
RFNC-VNIITF has been developed the mathematical model and software platform ATEK since 2008. The platform makes it possible to simulate the objects of NFC with different level of detailing on the base of unified principles. The code ATEK-NFC is designed for computational assesment of techno-economic characteristics of scenarios of NFC development. The code LOGOYAT is created for assessing the development fo infrastructure of handling spent fuel scheme. The code ODC is oriented to asses the economical characteristics of NPE components. The speciality of simulation on the platform ATEK is consideration of nuclide composition, which allows evaluating activity and heat generation at every stage of NFC and is the base of for computational evaluation of nuclear and radiation safety. The code VIZART allows calculating the characteristics of technologies and manufactures of NFC, assessing the pricipal performability of technologies, optimizing technology lines layout. The results of calculation for material balances and sequence diagrams of equipment operation are used while preparing initial data for development of the equipment and designing technology lines of reprocessing, fabrication and radioactive waste handling modules whithin "Proryv" project.
Safety is an obligatory term for every object fo nuclear energy use. When enclosing NFC part of technologies are developed for the first timeand does not have counterparts while highly active materials are involved into the cycle and their use experimentally is ellaborated or impossible. Since 2017 the work on developing the system of calcution models has been carried out for substiantiation of safety of tchnology stages of nuclear fuel cycle. The system of straight-through calculations is developed consisting of code VIZART, CFD-model of apparatus? codes for calculation of critical charaacteristics, radiation exposure, assesment of fire and explosion safety. Obtained calculational information on concentrations, pressures and temperatures was transformed to codes for calculating neutron-physical charcteristics and radiation exposures using the automatic software tools to estimate the critical technology characteristics.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **357** Type: **ORAL**

MULTIPURPOSE RESEARCH FACILITY MBIR AND POLY FUNCTIONAL RADIOCHEMICAL COMPLEX (R&D COMPLEX) AS A UNIQUE RESEARCH PLATFORM

Wednesday, April 20, 2022 3:04 PM (12 minutes)

Country/Int. organization

Russian Federation

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Presenter: ZAGORNOV, Alexander (Rosatom)

Session Classification: 5.1 Experimental Reactors and Facilities

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **360** Type: **ORAL**

Proposal of a compact core design for the 1000 MWe French commercial Sodium Fast Reactor by means of the SDDS multi-objective optimization tool

Wednesday, April 20, 2022 2:52 PM (12 minutes)

Country/Int. organization

France

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Presenter: Dr POUMEROULY, Sandra (EDF)

Session Classification: 1.2 Innovative Design Advances

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **361** Type: **ORAL**

Presentation of the new European project PUMMA devoted to Plutonium management in the whole fuel cycle

Wednesday, April 20, 2022 10:52 AM (12 minutes)

Country/Int. organization

France

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Session Classification: 3.1 Fuel Cycle Scenarios

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **363** Type: **ORAL**

Pilot Demonstrational Fast Reactor with Lead Coolant BREST-OD-300

Wednesday, April 20, 2022 1:52 PM (12 minutes)

Country/Int. organization

Russian Federation

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Presenter: Mr LEMEKHOV, V.V. (JSC NIKIET)

Session Classification: 1.2 Innovative Design Advances

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **364** Type: **ORAL**

Computational Studies of Advantages of Lead-Cooled Fast Reactor Core

Friday, April 22, 2022 11:06 AM (12 minutes)

Concept of the BREST reactor with lead coolant and dense heat-conductive nitride fuel envisages the development of an equilibrium core with complete breeding of fissionable nuclides in the core (core breeding ratio of ~ 1) without a blanket compensating for reactivity reduction due to fuel burn-up and fission product build-up. This makes it possible to operate the reactor in the period between two regular refuelings with a low reactivity margin (< βeff). At the same time, efficient utilization of uranium is ensured by means of conversion of U-238 to Pu-239 in the fast reactor spectrum and a possibility of transmutation of the produced minor actinides during the reactor operation in the closed nuclear fuel cycle.

To carry out computational studies of the BREST reactor core, a design code system is used, which includes the FACT-BR diffusion software system, the MCU-BR software tool based on the Monte-Carlo method, and the IVIS-BR thermophysical module. Analytical models of the MCU-BR code provide the most precise description of geometry and composition of the core, reflectors and other components affecting the neutronic characteristics of the reactor. In calculations, a neutronic cross-section library based on the ENDF/B-VII.1, JENDL-4.0, ROSFOND actual authenticated data files was used. FACT-BR and MCU-BR have been certified for calculations of the BREST-OD-300 reactor with lead coolant and nitride fuel. Experimental results obtained with the BN-350, BN-600 reactors and BFS test facility with lead were used for validation.

During the computational studies, impact of manufacturing tolerances on Keff and reactivity margin was evaluated. Maximum deviation of each manufacturing parameter was conservatively considered, including the most significant ones: plutonium mass fraction, mass of fuel, nitrogen content. The estimated overall technological error was 1.17, and the total error for Keff and reactivity margin was 1.36 % δk/k. Feasibility of low reactivity margin during rated reactor operation (~ 0.54 βeff) was demonstrated even for the initial period of operation given the compensation of the methodical, constant and manufacturing uncertainties by technical measures taken at the first criticality stage. Calculations of fuel and absorber burn-up, power distribution in the core, worth of control and safety rods, reactivity effects and other core characteristics were carried out.

The developed and validated design code system, the results obtained in computational studies can be used to support design concepts for the BREST-OD-300 reactor and commercial lead-cooled fast reactors.

Country/Int. organization

Russian Federation

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Presenter: Mr MOISEEV, A.V. (JSC NIKIET)

Session Classification: 6.5 Integrated Analysis and Digitalization

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **365** Type: **POSTER**

POWER CONTROL OF THE FAST NUCLEAR-BURNING-WAVE REACTOR

One of the most important problems in the further development of nuclear energy, from the point of view of its public acceptance, is the problem of safety. Thus, the development of new concepts for nuclear fission reactors with so-called "intrinsic safety" is a very urgent task. An equally important problem for the sustainable development of nuclear power is the need to expand the fuel base by involving uranium-238 and thorium-232.
The concept of a fast reactor (FR) operating in a self-sustaining nuclear burning wave (NBW) mode, proposed in [1], also known as the Traveling Wave Reactor and the CANDLE, if implemented, is capable to solve both of these problems, and in a very effective way. The "intrinsic safety" of the reactor is based on the specific mechanism of negative reactivity feedback inherent in the NBW mode, which ensures automatic maintenance of the critical state of the reactor even under external influences [2]. The use of depleted uranium and thorium as the main fuel with high burnup [3] makes it possible to use a "one-through"scheme instead of an expensive closed fuel cycle.
In our work, the possibility of controlling the NBW reactor power by changing the efficiency of a neutron reflector is investigated. Such a possibility is an important in the context of widespread use of weather-dependent wind and solar energy. The consideration was carried out on the basis of the approach developed in [4, 5], using the numerical solution of the multigroup nonstationary neutron diffusion equation together with the system of equations for fuel burnup and nuclear kinetics of delayed neutron precursors. A cylindrical multi-zone FR with U-Pu cycle fuel, in which NBW propagates in the axial direction, is considered. The calculations took into account the presence of a structural material Fe and a Pb-Bi coolant in the reactor core. The reflector consisted of 90% Pb-Bi and 10% Fe. Optimal algorithms are proposed for bringing the NBW reactor to a given power (both with decreasing and increasing) using a proportional-differential method for controlling the tantalum-181 content in a radial reflector.

1. Feoktistov L.P, Sov. Phys. Doklady, 34 (1989) 1071.
2. Fomin S.P., et al., Int. conf. "FR-13", Paris, 2013, paper CN-199-457.
3. Fomin S.P., et al., Prog. Nucl. Energy, 50 (2008) 163; 53 (2011) 800.
4. Fomin S.P., et al., Int. conf. "Global 2009", Paris, 2009, paper 9456.
5. Malovytsia M.S., et al., Ann. Nucl. Energy 148 (2020) 107699.

Country/Int. organization

Ukraine

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Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **366** Type: **POSTER**

VERSATILE TEST REACTOR: CORE SYSTEM DESIGN REQUIREMENTS TO SUPPORT ADVANCED REACTOR DEVELOPMENT

Thursday, April 21, 2022 10:40 AM (2 hours)

The Versatile Test Reactor (VTR) is a reactor under development in the United States of America to provide a very high-flux fast neutron source. This reactor will accelerate the testing of advanced nuclear fuels, materials, and other potentially irradiated components. As this reactor design effort is underway to support eventual construction and operation, a necessary step is the development of design requirements and objectives for all components and systems of the VTR. Such requirements are necessary in any engineering project to ensure the delivered product can perform its'mission safely, while providing a means for integration of the various design teams working on interfacing systems, and providing a basis for successful project execution.

Many of the VTR nuclear core requirements are consistent with those found for typical reactor designs. For example, inherently safe feedback behavior is required as a part of the design, various fuel material performance limits shall be met during certain scenarios, and the occupational and public dose limits must be below site and regulatory limits. However, some requirements are unique to VTR due to the reactor being a test reactor that can support the needs of experimenters. For example, to support the needs of experimenters the reactor shall be designed to allow the use of any non-control/safety assembly position in the core as an un-instrumented experiment position. The reactor also must be able to accommodate multiple different materials and/or fuels under irradiations experiment campaigns at a time. This paper will present and discuss these reactor core system design requirements with the goal of disseminating these requirements to potential experimenters as early as possible and providing an example of design requirements application to a modern large engineering project.

Country/Int. organization

United States of America

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Presenter: NELSON, Adam (Argonne National Laboratory)

Session Classification: Poster Session

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **367** Type: **ORAL**

Development and Demonstration of Diffusion-type Hydrogen Meters for Sodium-cooled Fast Reactors

Tuesday, April 19, 2022 2:24 PM (12 minutes)

Country/Int. organization

United States of America

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Session Classification: 4.1 Advanced Reactor Cladding and Core Material, Coolants, and Related Chemistry

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **368** Type: **POSTER**

SIMULATION OF THE FAST FLUX TEST FACILITY LOSS-OF-FLOW WITHOUT SCRAM ACCIDENT SCENARIO USING THE SAM COMPUTER CODE

Wednesday, April 20, 2022 1:40 PM (2 hours)

Country/Int. organization

United States of America

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Presenter: HOLLRAH, Brent (Texas A&M University)

Session Classification: Poster Session

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **369** Type: **ORAL**

ALFRED FLOW BOCKAGE ANALYSIS

Wednesday, April 20, 2022 2:28 PM (12 minutes)

In the context of Lead-cooled Fast Reactor development and safety assessment, the flow blockage in a fuel sub-assembly is considered among the most relevant issues to be addressed. Hence, the event shall be postulated assessing its consequences, also considering that grid-spaced fuel assemblies could partially mitigate the occurrence of sudden blockages with respect to wire-spaced fuel assemblies.
The Advanced Lead-cooled Fast Reactor European Demonstrator (ALFRED) is a 300 MWth pool-type reactor aimed at demonstrating the safe and economic competitiveness of the Generation IV LFR technology. The ALFRED design, currently being developed by ANSALDO NUCLEARE and ENEA in the frame of the FALCON Consortium, is based on prototypical solutions intended to be used to boost the DEMO-LFR development.
Within the scope of FALCON consortium and in the frame of investigating the thermal-hydraulics of the average ALFRED FA, a CFD computational model is built looking for the assessment of its thermal field in nominal flow conditions and when affected by a blockage. Starting from the experience in this kind of simulations and in experimental work, the whole model of the ALFRED Fuel Assembly is first presented and calculation of flow and temperature field in nominal conditions is carried out. RANS simulations of idealized blockage scenarios adopting three different spacer grid locations (under the active length, at half active length, above the active length). Results showed that the most likely blockage in the lower grid positioned before the active region do not perturb the temperature distribution in the fuel assembly, while the ones at the central grid may have strong consequences and lead to a clad temperature peak behind the blockage with possible clad failure. In particular, the CFD analysis on the ALFRED FA suggested to install the spacer grids –or at least the first one –in the not-active region to avoid any clad failure due to an internal blockage.

Country/Int. organization

Italy

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Presenter: Dr DI PIAZZA, Ivan (ENEA Brasimone R. C.)

Session Classification: 6.2 Thermal Hydraulics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **370** Type: **ORAL**

CFD Simulations on a hexagonal 61-pin wire-wrapped fuel bundle with STARCCM+ and comparison with experimental data.

Wednesday, April 20, 2022 2:40 PM (12 minutes)

In this work, several CFD simulations of the hexagonal 61-pin fuel bundle replica of the Thermo-hydraulic Research Lab at Texas A&M University were performed. This fuel assembly geometry is a helically wire wrapped bundle of rod pitch-to-diameter ratio of 1.89 and helix pitch-to-diameter ratio of 29.93, as in Sodium-cooled fast breeder reactors (FBR). The experimental activity has produced a large set of data that is compared against simulations. The purpose of the present study is to optimize computational fluid dynamics and compare the code predictions of pressure drops (friction factors) against experimental data and available correlations. The friction factor is of paramount importance in determining reactor features like pump specification and safety limits.

The commercial software STARCCM+ Version 14.04.013 was chosen to perform the simulations. Four different turbulence models were utilized to compare with the experimental data: k-epsilon (standard and realizable) and k-omega (standard and SST). The code predictions were also compared with the upgraded Chen and Todreas detailed correlation (UCDT).

For the geometry under study, the laminar to transition (RebL) and transition to turbulent (RebT) Reynolds numbers are RebL=494 and RebT=13,554. The experimental data was generated within Reynolds ranging from 439 to 13,766. Therefore, the simulations were focused in the transition regime, although there were simulations run at laminar Reynolds numbers.

The Incompressible Reynolds-Averaged Navier-Stokes Equation (RANS) with an eddy viscosity turbulence model was used in the calculations. Two meshes were utilized. The first mesh was used to perform simulations with these turbulence models; however, a second mesh was created with the intention of enhancing the results of k-ω SST.

Within the laminar regime, no turbulence model was needed. The simulation results exhibited a good comparison with the experimental data with an error of 2.81 % and the UCTD, with and error of 7.60%.

At the transition regime, the k-omega standard model with the first mesh produced the best prediction of the experiment (all simulation values were within the uncertainty interval) and the UCTD, while the two k-epsilon models overpredicted the friction factor. Regarding the k-omega SST model, results obtained with the first mesh were more distant to the experiment than the standard model. A better prediction was obtained using the second mesh, because this mesh had an average Y+ less than 1, which was not the case of the first mesh. Using Y+<1 is a requirement for the SST to accurately predict pressure and velocity mean field.

Country/Int. organization

United States of America

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Presenter: BOVATI, Octavio (Texas A&M)

Session Classification: 6.2 Thermal Hydraulics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **374** Type: **POSTER**

Irradiation Effects of T91 Ferritic/Martensitic Steel

Wednesday, April 20, 2022 10:40 AM (2 hours)

Country/Int. organization

China

Primary authors: Mr LI, Junhong (China Institute of Atomic Energy); Mr HAO, Xianchao (Institute of Metal Research, Chinese Academy of Sciencies); Mr SU, Xiping (China Institute of Atomic Energy); Mr YIN, Tong (China Institute of Atomic Energy)

Presenter: Mr LI, Junhong (China Institute of Atomic Energy)

Session Classification: Poster Session

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **375** Type: **POSTER**

Numerical Investigation of Cellular Convection in the Cover Gas space of Fast Breeder Test Reactor

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

India

Primary authors: Mr CHATURVEDI , Kanha (Dept. of Atomic Energy, IGCAR); Mr CHAUHAN, Amit Kumar (Dept. of Atomic Energy, IGCAR); Mr KUMARESAN, Natesan (Scientific Officer)

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Presenter: Mr CHATURVEDI , Kanha (Dept. of Atomic Energy, IGCAR)

Session Classification: Poster Session

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **376** Type: **ORAL**

ANALYSIS OF THE SGTR ACCIDENT FOR SAFETY JUSTIFICATION OF TWO-CIRCUIT LEAD COOLED REACTOR.

Wednesday, April 20, 2022 11:40 AM (12 minutes)

Country/Int. organization

Russian Federation

Primary authors: Mr SHVETSOV, IURII (JSC "PRORYV"); Mr KHOMYAKOV, Iurii (JSC "PRORYV"); RACHKOV, Valery (JSC "PRORYV"); Mr SUSLOV, Igor (JSC "PRORYV")

Presenter: Mr SHVETSOV, IURII (JSC "PRORYV")

Session Classification: 2.2 Safety Design and Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **378** Type: **POSTER**

INDUSTRIAL ENERGY COMPLEX WITH FAST NEUTRON REACTOR

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

Russian Federation

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Presenter: PETRENKO, Andrey

Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **379** Type: **ORAL**

Overview of critical experiments with fast metal cores held on assembly machine FKBN-2

Thursday, April 21, 2022 11:52 AM (12 minutes)

A brief description given for critical experiments held at RFNC-VNIITF on assembly machine FKBN-2 and ROMB critical assembly specially built for verification of neutron transfer simulation codes. The design of ROMB allows to use it for critical experiments with nuclear materials having fast neutron spectrum (highly enriched uranium, plutonium and its'mix). ROMB assembly contains of wide set of construction materials (such as depleted uranium, beryllium, beryllium oxide, steel, titanium, lead, tungsten, vanadium, molybdenum etc.) shape fitted to parts made of nuclear materials. This permits to hold critical experiments with heterogeneous structures which differ in content, including combinations with neutron moderators. Similar critical assemblies may appear in emergency cases concerned with nuclear fuel cell damage. Benchmark critical experiment set up and its'features are discussed. Brief description of the experiments held earlier and planned now for verification of neutron transfer simulation codes is given as an example of possibilities of ROMB assembly and experiment using it.

Country/Int. organization

Russian Federation

Primary authors: YUDOV, Aleksey; ANDREEV, Sergey; SOKOLOV, Yuri

Presenter: SOKOLOV, Yuri

Session Classification: 5.2 Experimental Programs I

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **380** Type: **POSTER**

Experiment-calculated method for determination of prompt neutron lifetime in fast metal cores intended for verification of neutron transfer simulation codes

Thursday, April 21, 2022 10:40 AM (2 hours)

Here described a method for determination of mean prompt neutron lifetime in fast metal cores during critical experiments held in RFNC –VNIITF using assembly machine FKBN-2. The evaluation of derivative using experimental dependence between asymptotic decrease coefficient and core parts gap α(H) was proposed further to determination of the delayed critical state of the core. The value characterizes the transient prompt neutron process in the core and accurate within coefficient determines the mean prompt neutron lifetime in the system. By-turn the coefficient may be calculated using contemporary neutron transfer simulation.
The experimental results of and routine criticality experiments data may be used for verification of computer codes and cross section databases.
Approbation of the method was held using fast metal cores of U and Pu in different mass ratio. The experiments were performed on assembly machine FKBN-2 leaded by time-correlated measurements. Benchmark experiment modeling was carried out and Monte-Carlo simulation was done for critical and time-correlated experiments using different cross section databases.

Country/Int. organization

Russian Federation

Primary authors: ANDREEV, Sergey; SOKOLOV, Yuri; Mr KHMELNITSKY, Dmitry (RFNC-VNIITF named after Academ. E.I. Zababakhin)

Presenter: ANDREEV, Sergey

Session Classification: Poster Session

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **381** Type: **POSTER**

Nuclear Hydrogen and Fast Reactors

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

Romania

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Presenter: Dr IORDACHE, Ioan (ICSI Rm. Valcea)

Session Classification: Poster Session

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **382** Type: **ORAL**

Perspectives and discussions on the modes and development path of China's commercial closed nuclear fuel cycle

Wednesday, April 20, 2022 11:16 AM (12 minutes)

Country/Int. organization

China

Primary author: Mr XIAO, MIN (China Nuclear Power Technology Research Institute)

Presenter: Mr XIAO, MIN (China Nuclear Power Technology Research Institute)

Session Classification: 3.1 Fuel Cycle Scenarios

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **384** Type: **POSTER**

INVESTIGATION OF THE SOLUBILITY OF ACTINIDE FLUORIDES FOR THE CHOICE OF A SALT SOLVENT FOR A MOLTEN-SALT REACTOR-BURNER OF MINOR ACTINIDES

Thursday, April 21, 2022 10:40 AM (2 hours)

Characteristics of the molten-salt reactor-burner (MSR-burner) of minor actinides (MA), which are concentrated in spent nuclear fuel of power reactors, depend significantly on the physical-chemical properties of the fuel composition. In particular, the MA transmutation efficiency is mainly determined by the concentration of actinide fluorides in the molten-salt fuel composition [1]. In this regard, the theoretical and experimental research of the actinide fluorides solubility in the molten-salt solvent to justify the choice of the molten-salt fuel composition is a relevant task. In Russia, fluoride salt systems based on LiF-BeF2 [2] and LiF-NaF-KF [3] are considered as basis of molten-salt fuel composition. The purpose of this work is an experimental determination of the solubility of actinide fluorides and their simulators in these fluoride salt systems.
This report presents the procedures for the experimental research of the actinide fluorides solubility in fluoride salt systems, such as thermal analysis method by cooling curves, differential scanning calorimetry, elementary analysis. The working out of these techniques was carried out by the determining individual solubility of NdF3 and CeF3 in salt solvents under investigation. Satisfactory agreement of experimental data with literature ones was obtained.
The results of calculating of different MSR-burner versions and the analysis of literary data were used for determination of characteristic compositions of molten-salt fuel mixture. Temperature dependencies of individual solubility of actinide fluorides and simultaneous solubility of actinide fluorides and their simulators in investigated molten salt solvents were obtained, and the elemental composition of molten salt samples was determined.

Country/Int. organization

Russian Federation

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Presenter: САННИКОВА, Полина

Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **385** Type: **ORAL**

TECHNOLOGICAL SUPPORT OF THE NON-PROLIFERATION FOR SVBR-100 FUEL CYCLES

Tuesday, April 19, 2022 4:58 PM (12 minutes)

Country/Int. organization

Russian Federation

Primary author: Dr KOMLEV, Oleg (JSC “AKME-engineering”)

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Presenter: Dr KOMLEV, Oleg (JSC “AKME-engineering”)

Session Classification: 7.1 Sustainability: Economics, Environment, and Proliferation

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **387** Type: **ORAL**

CHOICE OF A COOLANT FOR A MODULAR SMALL POWER REACTOR SVBR-100

Wednesday, April 20, 2022 2:04 PM (12 minutes)

Country/Int. organization

Russian Federation

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Presenters: Prof. TOSHINSKII, Georgii (JSC "AKME-engineering"); TOSHINSKII, Georgii

Session Classification: 1.2 Innovative Design Advances

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **391** Type: **POSTER**

Fast Reactor Source Term Modeling and Simulation Functional Requirements and Gap Assessment

Wednesday, April 20, 2022 1:40 PM (2 hours)

Country/Int. organization

United States of America

Primary authors: SHAHBAZI, Shayan (Argonne National Laboratory); GRABASKAS, David (Argonne National Laboratory)

Co-authors: THOMAS, Sara (Argonne National Laboratory); ANDREWS, Nathan (Sandia National Laboratories); CLARK, Andrew (Sandia National Laboratories); LUXAT, David (Sandia National Laboratories)

Presenter: SHAHBAZI, Shayan (Argonne National Laboratory)

Session Classification: Poster Session

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **393** Type: **POSTER**

Impact of Core Materials on The Fuel cladding Irradiation Damage in Breed-and-Burn Fast Reactors

Wednesday, April 20, 2022 10:40 AM (2 hours)

Country/Int. organization

Mongolia

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Presenter: Dr SAMBUU, Odmaa (School of Engineering and Applied Sciences, National University of Mongolia)

Session Classification: Poster Session

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **395** Type: **ORAL**

Simulation of ULOF initiation phase in ESFR-SMART with SIMMER-III

Friday, April 22, 2022 10:42 AM (12 minutes)

A large 3600 MWth European Sodium Fast Reactor (ESFR) design was proposed in 2000s. It is studied now in the EURATOM ESFR-SMART project. A new core configuration with several new safety measures, including a reduced to a near-zero value coolant void reactivity effect, mainly due to introduction of a sodium plenum above the core and core flattening, has been established recently. We investigate the efficiency of these measures by simulating transients such as unprotected loss of coolant flow (ULOF) with the SIMMER-III code, starting from nominal conditions till molten fuel discharge from the core. In the initiation transient phase, before structure and fuel melting, sodium boiling happens in the described simulations. The reactivity oscillates after the boiling onset due to subsequent boiling and flooding in the upper fissile core part and the sodium plenum above, where the void effect is negative. These oscillations are associated with interaction of different flow channels. In the paper we investigate these phenomena by considering different modelling options. New SIMMER capabilities for taking in account core thermal and control rode driveline expansion reactivity effects are also presented and their influence on the transient behavior is discussed. We also compare to results of ULOF simulations performed with other codes for ESFR-SMART.

Country/Int. organization

Germany

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Presenter: CHEN, Xue-Nong (Karlsruhe Institute of Technology (KIT), Institute for Nuclear and Energy Technologies (IKET))

Session Classification: 2.4 Severe Accidents

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **397** Type: **POSTER**

Impact of Cladding Material on Neutronic Balance in Breed-and-Burn fast reactors

Wednesday, April 20, 2022 10:40 AM (2 hours)

Country/Int. organization

Japan

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Presenter: OBARA, Toru (Tokyo Institute of Technology)

Session Classification: Poster Session

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **398** Type: **POSTER**

Development of Burnup Analysis System for rotational and Spiral Fuel Shuffling scheme in Breed-and-Burn Fast Rectors

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

Viet Nam

Primary author: Dr HOANG VAN, Khanh (aInstitute for Nuclear Science and Technology, Vietnam Atomic Energy Institute)

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Presenter: Dr HOANG VAN, Khanh (aInstitute for Nuclear Science and Technology, Vietnam Atomic Energy Institute)

Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **399** Type: **ORAL**

Towards design guidelines for fast reactor oxide fuel pins with high Pu content: driving post irradiation examination by benchmarking European fuel performance codes

Thursday, April 21, 2022 11:40 AM (12 minutes)

In the framework of the European Commission call for proposal "Horizon 2020", the project Plutonium Management for More Agility, called PuMMA, is granted. This project starts in October 2020 and will last four years. A work package is dedicated to the behaviour and safety of mixed oxide fuels with high plutonium content, which is essential for plutonium multi-recycling or plutonium burning in fast reactors. This paper describes main goals and status of this work package. Major task is the comparison of a large set of european fuel performance codes (FPC) on the basis of three passed experimental irradiations of oxide fuel pins containing around 45 % of plutonium: CAPRIX, irradiated Phenix French Reactor, TRABANT 1 and TRABANT2, irradiated in High Flux Reactor, HFR, in the Netherland.
The first phase of the work consists in the definition of irradiation conditions for fuel pins simulation, involving CEA and NRG. In a second phase, 10 various FPC will be used by 13 European nuclear research organizations in order to simulate these three irradiations: SIMMER-V, TRANSURANUS, OFFBEAT, FEMAXI, FUROM, FRED, MACROS, FINIX, TRAFIC and GERMINAL. Results will be compared in terms of global and local quantities: fission gas release, fuel pin elongation, profilometry, central hole radius, Pu redistribution, internal corrosion, etc. Moreover, this first set of simulations will be used to define the post-irradiation examinations programme, which will be carried out in the framework of the PuMMA project in JRC-ITU and CEA facilities. In a third phase, simulation tools will integrate new thermal properties measurements to be realized in PuMMA (other workpackage), the back-up of first comparisons, and these irradiations simulations will be re-launched and compared to experimental measurements. This mixed approach simulation/examination will allow to improve fuel codes reliability and to reduce uncertainties in the design process of this kind of fuel, which is outside of the validation area of all the existing codes. The experimental programme will be devoted to FPC validation as well as knowledge improvement. Last part of the work package will also tend to propose specific safety recommendations for the design of this kind of fuel.

Country/Int. organization

France

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Presenter: BLANC, Victor (French Atomic Energy Commission)

Session Classification: 3.2 Development of innovative fuels: design and properties irradiation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **406** Type: **ORAL**

Reference Fuel Options for Generation-IV Sodium-cooled Fast Reactors

Wednesday, April 20, 2022 11:40 AM (12 minutes)

Country/Int. organization

France

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Presenter: SERRE, Frédéric (Commissariat à l'Energie Atomique et aux Energies Alternatives (CEA))

Session Classification: 3.1 Fuel Cycle Scenarios

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **411** Type: **POSTER**

New Concepts and Methodologies for the Effective Deployment of Gen IV reactors

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

Nigeria

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Presenter: Mr WILCOX, BOMA (Nigeria Atomic Energy Commission)

Session Classification: Poster Session

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **412** Type: **ORAL**

Nuclear Fuels for Fast Reactors-A Review

Friday, April 22, 2022 2:06 PM (12 minutes)

This paper reviews the status of fast reactor fuels. The main focus of the development of fast reactor fuel is their potential for actinide transmutation and high burn up. Metallic, oxide, carbide, nitride and dispersion type fuels are being used as fast reactor fuels. Metallic fuels, because they produce an extremely hard neutron spectrum, are neurotically ideal for fast reactors. As compared to other fuels,the thermal and mechanical performance of metallic fuels provide high breeding ratio and large safety margins in normal and accident conditions. U-Pu-Zr are considered best metallic fuels and their properties are well documented.U-Pu-Zr fuel system contains up to 8% minor actinides (MA) and mixture of rare earth element and has excellent compatibility with coolant. U-Mo based alloy fuel system is anticipated to have benefits avoiding phase separation, and increased fuel thermal conductivity. In addition, the fabrication of these metallic fuels is easy and economical.

However other fuels such as oxide fuels have been studied for its potential to be used in fast reactor for normal and transient conditions. Low density, poor thermal conductivity and chemical reaction with coolant are main drawbacks of oxide fuels.Despite these, mix oxide fuel (U,Pu)O2 has been widely used for fast reactor because of its high melting point and excellent irradiation behavior. Considered as advanced fuel concepts, carbide and nitride fuels have been investigated as an alternative of metallic and oxide fuel for fast reactor. The nitrides fuels have better thermal conductivity and safety margins as compared to oxide fuels. Dispersion type fuel concept were developed to increase the transmutation rate of minor actinides .Use of minor actinides in special mechanical fuel forms is reported to be useful for reduction of high level radioactive waste inventory.These fuel options have been considered w.r.t. fabrication, characterization, irradiation performance, design, high burn up and safety criteria. The technical issues related with the innovative fuels for advanced fast reactors and several irradiation test performed have been discussed and experience research programs of different countries summarized. The recently completed, currently ongoing and planned projects on fast reactors and related fuel cycles of different countries are also part of this paper. By comparing the data of fuel types and reactor designs,the future plan for deployment of a fast research reactor and commercial FBRs in combination with PWRs as sustainable energy system scenarios in Pakistan are also discussed in this paper.

Country/Int. organization

Pakistan

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Presenter: Mr MAQBOOL, Abu Baker (PAEC)

Session Classification: 3.4 Advanced Fuel Development

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **414** Type: **POSTER**

CFD ANALYSES OF THE ALFRED HOT PLENUM

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

Netherlands

Primary author: VISSER, Dirk (NRG)

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Presenter: VISSER, Dirk (NRG)

Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **415** Type: **ORAL**

Analysis of the natural circulation capacity of decay heat removal system in pool-type sodium-cooled fast reactor

Thursday, April 21, 2022 2:16 PM (12 minutes)

The structure of pool type sodium-cooled fast reactor (SFR) is complex, which leads to the complicated thermal-hydraulic phenomena in the process of natural circulation for decay heat removal. The determination of natural circulation flow path and the decay heat removal capacity of natural circulation of each flow path are issues to be considered in the design of SFR. Core flow distribution, flow and heat transfer of inter-wrapper flow, thermal stratification of the sodium pool, thermal hydraulic interaction between the core and the sodium pool, and arrangement of the decay heat removal system are factors that affect the decay heat removal capacity in the reactor. Therefore, this paper analyzes the influence of the arrangement scheme of decay heat removal system on the removal of decay heat.

Firstly, the system program THACS is used to establish the decay heat removal system of coupling primary circuit and external circuit of SFR, and the analysis is carried out for the condition of station black out (SBO). Secondly, sensitivity analysis is conducted for the arrangement scheme of the decay heat removal system, so as to evaluate the decay heat removal capacity of the reactor. Two decay heat removal systems selected for comparative analysis are non-penetrating direct reactor auxiliary cooling system (NPDRACS) and penetrating direct reactor auxiliary cooling system (PDRACS). The results indicate that both decay heat removal system arrangements can effectively remove decay heat from the core. And for large SFR, the decay heat removal capability of the PDRACS is better, because the cold sodium from direct heat exchanger (DHX) can cool the core assemblies directly.

Country/Int. organization

China

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Session Classification: 6.4 Simulation Tools for Safety Analysis

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **416** Type: **POSTER**

Modeling of the coolant region in the ALFRED core in case of thermal expansion

Friday, April 22, 2022 10:30 AM (2 hours)

From a neutronic point of view, the effects of thermal expansion on the reactivity of a reactor core are an important feedback mechanism, both in steady-state and during many postulated accidents sequences. It is therefore necessary to model the expanded configuration in terms of shapes, densities and volumes as accurately as possible. Unfortunately, this is not easy for those regions that expand differently due to a temperature gradient. This is the case of the coolant region along the active height, where the change in temperatures determines a change in the flow area and consequently a change in both the mass and volume of the coolant itself. In this study, three alternative approaches to modelling the coolant region are theoretically discussed. In the first model, masses and volumes of the expanded configuration are calculated for the entire subchannel using a single averaged pitch, and uniform density along the active height as evaluated by the average expansion of the coolant. In the second model, the first approach is enhanced by taking into account the axial change in mass, i.e., a discretization of the subchannel in equivalent regions is introduced so that in each axial region a more precise value of the coolant density can be used. Finally, in the third model, starting from the idea of explicitly preserving the reaction rates by preserving the coolant inventory compared to the real case, the mass is again calculated for each axial region in which the subchannel has been discretized but using an effective density which is derived from the physical density applied to the real-case volume of that region. These three approaches applied to the elementary cell of the ALFRED core are then compared by detailed analysis with MCNP6.1. This is done to assess, on one hand, their discrepancy on the reactivity of the system and, on the other hand, their demanding in terms of model setup and computational costs. An insight on the cost-benefit ratio of each model is in fact necessary to obtain quantitative information for establishing a reference calculation route catching the aimed details while balancing the required efforts.

Country/Int. organization

Italy

Primary authors: Mr PERGREFFI, Roberto (ENEA); Mr GRASSO, Giacomo (Italian National Agency for New Technology, Energy and Sustainable Economic Development (ENEA)); Mr LODI, Francesco (ENEA)

Presenter: Mr PERGREFFI, Roberto (ENEA)

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **417** Type: **ORAL**

New ASTRID SFR - Intermediate Heat Exchanger (IHX) and internal vessel interface system: qualification tests onto a scale 1 representative mock-up

Wednesday, April 20, 2022 12:04 PM (12 minutes)

Country/Int. organization

France

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Session Classification: 4.2 Structural, Novel, and Large Components Materials

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **421** Type: **POSTER**

ENDF/B-VIII.0 NUCLEAR DATA SENSITIVITY AND UNCERTAINTY (S/U) ANALYSIS OF KEY SAFETY-RELEVANT REACTIVITY COEFFICIENTS FOR THE ALFRED CORE

Friday, April 22, 2022 10:30 AM (2 hours)

ENEA has a long-lasting expertise in the design of Gen IV nuclear reactors, in particular the ones cooled by liquid Lead (LFRs). In the EU context, through the participation to the FALCON Consortium, ENEA is pursuing all the activities required to support the construction of ALFRED –the European demonstrator of the LFRs –in Romania.

S/U analyses are a paramount step for the licensing of such an innovative reactor. In fact, no previous LFR experience can be used for validating neutronic calculations justifying the design of the core, so that a thorough assessment of the calculation uncertainties has to be used in front of the safety authorities asked to license ALFRED construction.

In fact, S/U analyses are used for establishing the adequateness of the assumed safety margins, as one of the key goals in designing the demonstrator, by verifying that such safety margins cope –with the aimed confidence –with the relative uncertainties.

The objective of this paper is to present the S/U analysis of the ALFRED reactor in order to assess the impact of the nuclear data uncertainties on the core reactivity and on the most important safety- relevant reactivity effects: e.g. coolant density effect, temperature-related geometric effects, control rod worth, delayed neutron fraction, etc.

Both the sensitivity and uncertainty analyses are here presented so to give the full picture of the parameters investigated, outlining what are the most important isotope-reaction couples both from the purely physical and nuclear data quality standpoints.

S/U analyses are performed using one of the most up-to-date nuclear data evaluations, ENDF/B-VIII.0, prepared in a special format readable by the selected neutronic code, ERANOS, which does not accept libraries in the standard ENDF-6 format.

Moreover, regarding the needed covariances, in order to avoid inconsistencies and with the aim of enhancing the confidence on the obtained results, a new homemade one also based on the state-of-the-art evaluation ENDF/B-VIII.0, was generated and used.

Country/Int. organization

Italy

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Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **422** Type: **ORAL**

REALISATION OF AN ADJUSTED NUCLEAR DATA LIBRARY BASED ON ENDF/B-VIII.0 NUCLEAR DATA EVALUATIONS FOR THE ALFRED CORE

Wednesday, April 20, 2022 11:28 AM (12 minutes)

Country/Int. organization

Italy

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Session Classification: 6.1 Neutronics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **423** Type: **ORAL**

Versatile Test Reactor (VTR) Experimental Capabilities

Friday, April 22, 2022 2:42 PM (12 minutes)

The Versatile Test Reactor (VTR) is a proposed fast neutron spectrum test facility that will provide irradiation capabilities not currently available within the U.S. The Idaho National Laboratory (INL), in conjunction with five other U.S. national laboratories, 19 universities and 10 industry partners, is working to develop the VTR to provide an irradiation-testing facility capable of performing a wide range of tests to meet current and future experimental needs. The VTR will allow many research institutions to have access to fast neutrons that will support the development of advanced nuclear technologies. The VTR is envisioned to be a 300 MWth sodium-cooled, pool-type fast neutron spectrum (fission spectrum) reactor for neutron irradiation testing and research, and is anticipated to perform multiple irradiation test campaigns during its operational lifespan of up to 60 years. The proposed configuration of the VTR core comprises 66 driver-fuel assemblies, up to six dedicated/fixed test locations for instrumented vehicles, cartridge loop tests, and rapid transfer (rabbit) test vehicles, and the ability to replace a specific number of the 66 driver-fuel assemblies with open test assemblies. The ultimate goal of the experimental capabilities of the VTR is to provide a platform and test capability that will help increase the technology readiness level of advanced reactor fuels, materials, instrumentation, and sensors. In addition, it will provide the ongoing test capabilities that will support and sustain current and future nuclear reactor continuous technology improvements. This paper describes the different experiment vehicles being developed, and the different lab, university, and industry teams that are leading the design and development of each of the experiment vehicles.

Country/Int. organization

United States of America

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Presenter: WEAVER, Kevan (Idaho National Laboratory)

Session Classification: 5.3 Experimental Programs II

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **424** Type: **ORAL**

Investigation on Human Resources Needs and Competences Building for ALFRED Implementation in Romania

Thursday, April 21, 2022 2:28 PM (12 minutes)

ALFRED is the demonstrator of Lead Fast Reactor (LFR) technology. According to strategic documents (at national level and of FALCON international consortium), it is planned to be built on Mioveni nuclear platform. An experimental infrastructure consisting of six experimental facilities (ATHENA, HELENA2, ELF, ChemLab, HandsON, Meltin'Pot) and a coordination Hub is planned to be built on the same site in support of the licensing process and technology development.
The development of the LFR technology faces various challenges including: (1) Research and Development (R&D) open issues (such as development and behaviour of the structural and cladding materials, the control of lead and cover gas chemistry, development of the instrumentation and control, fuel and fuel cycle, deterministic and probabilistic analyses, thermal-hydraulics for large pool configuration of molten lead, etc.) and (2) the novelty of the qualification, demonstration, validation and verification process for the ALFRED demonstrator. An appreciable number of high qualified personnel is estimated to perform the envisaged activities. In this context, the human resources including the competences building process are considered as crucial factors for the success of the implementation.
Considering the complexity of the scientific activities, the high degree of specialization and the existing offer of the workforce market, an education and training program is essential. Update/adaptation of the existing curricula in the education programme of the Romanian universities is needed, and can be achieved for example by new specializations/courses devoted to Generation IV systems (with a focus on LFR technology), a dedicated internship programme in European or worldwide experimental facilities, specialised/dedicated/thematic workshops, summer schools on simulation and experimental activities, etc.
This paper presents the outcomes of the investigation on human resources needs for ALFRED implementation in Romania, developed in the framework of Romanian PRO ALFRED project. The jobs estimation has been accounted for the ALFRED infrastructure R&D activities, operation and realization of the specific activities for each experimental facility and for the Hub, as well as for the safe operation of the ALFRED demonstrator.
Around 600 jobs have been identified for the operation of ALFRED demonstrator and its support infrastructure, as well as for the R&D related activities. For each identified job, the specializations and the minimal competencies have been established.
To see how the existing Romanian educational programs cover the minimal competences required by ALFRED infrastructure, an expert judgement evaluation has been performed in University of Pitesti and University Politehnica of Bucharest.

Country/Int. organization

Romania

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Session Classification: 9.1 Education, Profesional Development, and Knowledge Management

Track Classification: Track 9. Education, Profesional Development, and Knowledge Management

Contribution ID: **428** Type: **POSTER**

Removal of Radiocesium from High-Level Liquid Waste using Inorganic Ion-exchangers

Thursday, April 21, 2022 1:40 PM (2 hours)

The present study demonstrates the use of inorganic ion-exchanger (IX) to condition the high-level liquid waste (HLW) by selective separation of one of the major radionuclide, cesium-137 (137Cs) from it. 137Cs possesses a broad range of potential applications in societal and agricultural area. In addition to this, the selective separation of 137Cs from HLW would drastically bring down secondary waste generation and reduce burden on the off-gas treatment in the vitrification process. Here, we have successfully demonstrated the conversion of Cs loaded IX to compact waste form.
Among the various adsorbents, Ammonium Molybdo-Phosphate (AMP) was preferred and used as IX in the present study because it shows high selectivity towards Cs+ in the presence of various metal ions (alkali, transition, lanthanides and actinides) and is stable under highly acidic & irradiation condition. Despite these advantages, the powder form of IX is not readily adaptable and does not provide ideal flow dynamics for continuous column operations. With a view to bring it to a usable form, synthesis of composite forms of IX (20-30%) in Poly-Ether-Sulfone (PES) was carried out. By adjusting the flow rate of the polymer liquid, particles with an average size ranging from 150 to 710 μm were obtained using a dual nozzle device that allows the break–up of polymer solution by air blowing. The polymer particles of 355-600 μm in diameter were mainly used for Cs extraction studies.
The obtained beads were characterized for thermal stability using thermogravimetry (TG), phase purity by X-ray diffraction (XRD) and functional group identification by Fourier transform (FT)-infra-red (IR) spectroscopy. The thermograms of IX and IX-PES beads showed few steps of decomposition reactions, it may be due to the loss of moisture, ammonia, PES and MoO3 from AMP-PES. To assess the efficiency of the IX-PES beads, its cesium extraction capacity and distribution coefficient were determined using actual HLW. The extraction capacity and distribution coefficient of cesium in actual HLW (3M acidic) was 4369.3 cm3/g and 0.4 meq/g, respectively. Column studies (L/D=3) were also carried out with an HLW flow rate of 0.5mL/min. The performance of the column was evaluated by plotting a breakthrough curve. Pellet formation of the inactive cesium loaded IX beads was successfully demonstrated using a manual pelletiser with a pressure of 150 kg/cm2.

Country/Int. organization

India

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Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **432** Type: **ORAL**

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

Tuesday, April 19, 2022 1:24 PM (12 minutes)

Country/Int. organization

United States of America

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Presenter: HILL, Robert (Argonne National Laboratory)

Session Classification: 1.1 Overviews and Fundamentals of Fast Reactors

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **433** Type: **ORAL**

Approach for ALFRED licensing in Romania

Tuesday, April 19, 2022 4:10 PM (12 minutes)

Country/Int. organization

Italy

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Presenter: GRASSO, Giacomo (Italian National Agency for New Technology, Energy and Sustainable Economic Development (ENEA))

Session Classification: 2.1 General Safety Approach

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **434** Type: **POSTER**

The ”ALFRED White Book”: a business card of the project

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

Italy

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Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **435** Type: **ORAL**

Overview of the R&D programs led by the past at IRSN on sodium fire

Wednesday, April 20, 2022 2:16 PM (12 minutes)

Country/Int. organization

France

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Session Classification: 5.1 Experimental Reactors and Facilities

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **436** Type: **ORAL**

Versatile Test Reactor (VTR) Project Mission and Status

Wednesday, April 20, 2022 2:52 PM (12 minutes)

Country/Int. organization

United States of America

Primary authors: PASAMEHMETOGLU, Kemal (Idaho National Laboratory); ROGLANS-RIBAS, Jordi (Argonne National Laboratory); O'CONNOR, Thomas (U.S. Department of Energy Office of Nuclear Energy)

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Session Classification: 5.1 Experimental Reactors and Facilities

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **437** Type: **ORAL**

GEN IV INTERNATIONAL FORUM WEBINARS INITITATIVE

Thursday, April 21, 2022 2:04 PM (12 minutes)

Collaboration and support among national laboratories, industry, universities, and research and development organizations are vital to not only maintain a skilled and competent nuclear workforce but also to avert the risk of human resource shortages. However, despite numerous efforts in coordinating and promoting nuclear education, there is still a lot to be done for developed and developing countries to either maintain and/or build a skilled nuclear workforce to address the increasing demand for technical skills. The Gen IV International Forum (GIF) Education and Training Working Group (ETWG) was created in November 2015 to respond to this demand, by proposing webinars that focus on advanced reactors systems and cross-cutting subjects. As of today (28 September 2020), the GIF ETWG has produced, podcasted and posted 46 Webinars covering the six Gen IV systems and various subjects addressing e.g. the economics of the nuclear fuel cycle, sustainability aspects of Gen IV systems, nuclear fuels and materials challenges, the thorium fuel cycle, energy conversion systems, and lessons learned for knowledge management and preservation. The GIF webinars are presented live by internationally recognized subject matter experts. They are recorded and archived at www.gen-4.org, and have been recently converted to the YouTube platform as video. The development of GIF webinars, with their expansion of topics, is intended to inform and stimulate not only junior scientists' interest, but also managers, key decision makers and the general public about advanced reactors introducing foreseen advantages but also key R&D to be developed. Details and examples of the GIF webinar modules from the initial concept to the full realization will be presented. Future topics for webinars that are planned beyond May 2021, will be announced.

Country/Int. organization

United States of America

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Presenter: Dr PAVIET, Patricia (Pacific Northwest National Laboratory)

Session Classification: 9.1 Education, Profesional Development, and Knowledge Management

Track Classification: Track 9. Education, Profesional Development, and Knowledge Management

Contribution ID: **438** Type: **POSTER**

CRITICALITY SENSITIVITY ANALYSIS IN RELATION TO EMPTIES OF A FAST REGENERATOR NUCLEAR REACTOR

Friday, April 22, 2022 10:30 AM (2 hours)

The current energy production, resulting from the concepts related to the fission of fissile nuclides, nuclear energy, is of the order of 397,650 MWe produced by the 449 nuclear plants in operation and another 54,364 MWe to be supplied by another 54 under construction on the planet [2], data that demonstrate the growth in installed capacity and the installation of electrical energy from nuclear fission. Thus, nowadays, the projection of an increase in the share of nuclear energy in energy production and supply is notorious, with the need for annual availability of approximately 62.825 thousand tons of mineral resources of its fundamental item, natural uranium, for the which is estimated at approximately 7.988 million tons of world reserves [3]. This work, applying spherical coordinates, presents the modeling of the FBR core considering the approximation of diffusion equation to one and two energy groups for conditions without void and also with the insertion of 5.87% of void in the coolant. Taking as reference the analytical approach developed, programs were elaborated in FORTRAN language that allowed the calculation of the flow distribution, the absorption, the leakage, the kef f, and the reactivity coefficient. The detailed results allowed showing the behavior of FBR and the sensitivity of the kef f and the reactivity coefficient to the presence of void, which presented the same trend of the results obtained through the SCALE software. Therefore, the exposed modeling proved to be a powerful tool in the initial phases of the nuclear reactor core design.

Country/Int. organization

Brazil

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Presenter: FIEL, Joao Claudio (IME)

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **440** Type: **ORAL**

Target Accuracy Requirements and an evidence-based background for MSFR safety assessment

Thursday, April 21, 2022 2:52 PM (12 minutes)

From the very beginning of the nuclear era, a mission of fast neutron reactors (FRs) has been foreseen [1], [2] in its double functionality, combining a generation of power and a conversion of fertile materials (U-238 and Th-232) into new nuclear fuels to make energy resources practically inexhaustible.
Since then, due to conjuncture changing, one used to consider fast reactors as elements of the global fuel cycle intended to play as traditional roles –to be a robust source of energy and artificial fissile materials, –as complementary ones –to burn plutonium and minor actinides accumulated in LWRs spent fuel [3].
At the same time, the nuclear engineering community considers, inter alia, some innovative FR concepts with fuel reprocessing fully immersed in a nuclear reactor like Molten Salt Fast Reactors (MSFRs) [4].
Over there, of course, enhanced flexibility of the fuel cycle in MSFRs should be supported by their extraordinary safety potential [3]. Unfortunately, in the case of MSFRs due to a fundamentally limited operational background, assessors have to rely, largely, on comprehensive simulations than on pure expert judgment.
Of course, these simulations, models, and relevant tools should be somehow validated against objective observations to ensure a sufficiency of their Predictive Capability Maturity (PCM) [5]. Yet any integral experiment (IE) taken alone cannot reproduce phenomena or processes of interest while an extrapolation basing on indirect data and Data Assimilation (DA) can [6].
These indirect IEs might be selected 1) focusing on phenomena essential as for reactor control as for accidental process management, 2) combining nuclear-driven and non-nuclear experimental data in the validation of relevant codes and models, and 3) establishing realistic Target Accuracy Requirements (TARs) for the best estimate and penalizing models.
We are discussing some factors of informativeness of IEs in terms of pre-defined "accidental states" (ASs)–subsequent phases of the core degradation.

[1] E.Fermi, The Future of Atomic Energy, United States, (1946),
http://www.osti.gov/accomplishments/documents/fullText/ACC0043pdf
[2] Nuclear Fuel Cycle Science and Engineering, 1st Edition, Ian Crossland, Woodhead Publishing, 2012
[3] J.Serp et al., The molten salt reactor (MSR) in Generation IV: Overview and perspectives, Prog. Nucl. Energy, 77, 2014
[4] I.Babuska, J.T.Oden, Verification and validation in computational engineering and science: basic concepts, Computer Methods in Applied Mechanics and Engineering, 193, 36–38, 2004
[5] G.Palmiotti et al, A global approach to the physics validation of simulation codes for future nuclear systems, Annals of Nuclear Energy, 36(3), 2009

Country/Int. organization

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Session Classification: 6.4 Simulation Tools for Safety Analysis

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **445** Type: **POSTER**

Experimental study on sodium insulation interaction and its effect on structural material

Wednesday, April 20, 2022 10:40 AM (2 hours)

Country/Int. organization

India

Primary authors: Mr CH SSS, Avinash (Indira Gandhi Center for Atomic Research); Mrs V, Snehalatha (IGCAR); Mr E, Hemanth Rao (IGCAR); Mr M, Nanibabu (IGCAR); Mr POLAKI, S R (IGCAR); Mr DAS, Sanjay Kumar (IGCAR); Dr D, Ponraju; Mr S, Athmalingam; BALASUBRAMANIAM, Venkatraman (Indira Gandhi Centre for Atomic Research, Kalpakkam DAE, India)

Presenter: Mr CH SSS, Avinash (Indira Gandhi Center for Atomic Research)

Session Classification: Poster Session

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **449** Type: **ORAL**

Transient 3D simulations for the ASTRID reactor: preliminary results for the ULOF initiation phase

Friday, April 22, 2022 11:42 AM (12 minutes)

An Unprotected Loss Of Flow transient (ULOF) in the 1500MWth Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) reactor is investigated with SIMMER-IV, a 3D multi-phase, multi-velocity and multi-component fluid-dynamics and neutronics code. The 2D RZ code version, SIMMER-III, is a working horse for fast reactor severe accident studies at KIT and other institutions, but the 2D approach affects simulation of reactivity feedbacks and of behavior of reactor materials under accident conditions. On the other hand, 3D SIMMER ULOF calculations take a lot of time and computer memory and were not tried for a full-vessel pool-type fast reactor model at KIT and EdF before recently.

Recent developments for SIMMER-IV, including introductions of a new neutron transport solver based on the PARTISN code and of a new procedure for generation of few-group cross-sections during the transient, offer new simulation capabilities. Also more computer power is available now. Therefore, an effort was done at KIT, in collaboration with EdF, to perform calculations for a full-vessel 3D model of the primary ASTRID circuit. After a first attempt, modifications were introduced in the code and the employed reactor model for improving their performance.

In the paper we inform on our experience with 3D SIMMER calculations, present preliminary results of 3D steady state and transient simulations for the ULOF initiation phase, do preliminary comparisons with results of 2D analyses.

Country/Int. organization

Italy

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Session Classification: 2.4 Severe Accidents

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **451** Type: **POSTER**

Study on the Method of Correction of Fast Reactor Power Distribution by MCNP

Friday, April 22, 2022 10:30 AM (2 hours)

The tally cards F6 and F7 in MCNP program allow users to calculate reactor power. After a time of operation, the fission products increased, which caused the delayed energy in the reactor. Thus, the power directly calculated by F6 and F7 would not correspond with the real value, and for the fast reactor, the energy distribution of fuel and other structural materials will also deviate from the actual value. In this paper, to obtain more accuracy of core power distribution by the MCNP power tally cards, the first core of China Experimental Fast Reactor (CEFR) is taken as an example, the distribution of the neutron energy, prompt γ energy, delayed γ energy and delayed β energy are calculated. The delayed γ energy and delayed β energy which cannot be calculated directly have been corrected. The delayed β is regarded as deposited in the fuel area, while delayed γ would transport in the whole reactor range. A source of body type of delayed γ is described, the heat release distribution is evaluated with the effect of delayed γ energy.

Country/Int. organization

China

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Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **459** Type: **ORAL**

Computational fluid dynamics study for estimation of dilution for failed fuel location system

Friday, April 22, 2022 2:42 PM (12 minutes)

In a pool type sodium cooled fast reactor, in case of detection of failure of a fuel subassembly (FSA) by global delayed neutron detection system, localization of failed subassembly would be done using the Failed Fuel Location Module (FFLM). This is achieved by sampling sodium at exit of each subassembly and looking for presence of delayed neutrons. For a 500 MWe prototype design, as part of this system there are sampling tubes for each FSA placed as annular tubes concentric to thermo-wells of core temperature monitoring system. Each sampling sleeve admits sodium sample from a specific subassembly, through an annular channel within a sampling sleeve.The sampling end at the bottom of sampling sleeve is at axial separation of about 130 mm from respective subassembly outlet.

The aim of thepresent study is to estimate dilution in concentration of a delayed neutron precursor suffered during sampling of (contaminated) sodium for FFLM system. This is necessitated due to the complex hydraulics with multiple interacting jets at varied temperatures and flow rates emanating from numerous SA outlets. Due to the configuration of reactor assembly, outlet jets from SA top attain a significant radial component at the cost of their axial components. The prevailing velocity and temperature fields lead to complex hydraulics within hot pool of a fast reactor. The presence of diverse scales (large domain with large number of small structures) makes this study highly challenging. Towards this, a detailed three dimensional CFD model of a 90° sector of hot pool of reactor has been developed with inner vessel, control plug and connected structures, intermediate heat exchanger and pump standpipes. Core outlet has been modeled accurately with individual outlets for fuel and blanket subassemblies. All other subassemblies have been grouped together appropriately. Each discrete FFLM sampling sleeves are modeled along with shrouds for control rods. Such detailed modeling approach allows estimation of dilution for flow from individual subassemblies. The complete paper would summarize the dilution estimates from numerous runs for each monitored subassembly. It is seen from the studies that species dilution is insignificant for all fuel subassemblies with the maximum dilution predicted being 0.02 %. This ensures reliable and accurate sample collection for failed fuel localization.

Country/Int. organization

India

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Presenter: Mr MAITY, Ram Kumar (Indira Gandhi Center for Atomic Research)

Session Classification: 6.6 Fuel Performance and Material Modelling

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **460** Type: **POSTER**

Integrated thermal hydraulic analysis of Hot and Cold Pools of a liquid sodium cooled 600 MWe fast reactor

Friday, April 22, 2022 10:30 AM (2 hours)

A comprehensive CFD model of reactor pool of liquid sodium cooled pool type 600 MWe fast reactor design along with immersed reactor components is developed for detailed thermal hydraulic studies. Hot and cold pools along with immersed components represent the primary heat transport system. The two pools are physically separated by inner vessel, which completely envelopes the hot pool. Cold pool along with inner vessel is enveloped by main vessel. Inner vessel is in contact with both hot and cold pools, having widely different temperatures. Apart from this, the complex flow patterns in hot and cold pools introduce circumferential and axial temperature asymmetry on both inner and main vessels. The combination of complex computational domain and flow physics necessitates a detailed three dimensional CFD study. Towards this, a three dimensional CFD model that includes both hot and cold pools along with all major immersed components is developed. Development of a three dimensional model is a challenging task due to the large dimensions, several immersed solid structures and requirement of modelling components with widely different scales. Further inclusion of internal structures like spherical headers, primary piping etc. complicates the task of mesh generation. The model developed for the present study is a 180° sector model to take advantage of inherent computational symmetry. The main focus of this work is on resolving temperature distributions of important structural components, viz., inner vessel, main vessel, primary piping, pump and heat exchanger standpipes, headers etc. during full power operating conditions of reactor. Heat bypass from hot to cold pool through inner vessel is another important quantity estimated from this study. Other important aspects predicted include (i) cross flow velocity patterns in cold pool, (ii) free surface velocity profile in hot pool and (iii) velocity & and temperature distributions at IHX inlet and outlet windows. The results from this study are necessary for thermo-mechanical analysis of reactor assembly components.

Country/Int. organization

India

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Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **461** Type: **ORAL**

Thermal hydraulic assessment of the performance of secondary sodium system based decay heat removal circuit

Wednesday, April 20, 2022 11:04 AM (12 minutes)

Country/Int. organization

India

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Session Classification: 2.2 Safety Design and Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **462** Type: **POSTER**

Thermal Hydraulic Simulation of Loss of Flow Without Scram Test in FFTF using DYANA-P code

Wednesday, April 20, 2022 1:40 PM (2 hours)

Country/Int. organization

India

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Presenter: GOVINDARAJAN, Vikram (IGCAR)

Session Classification: Poster Session

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **464** Type: **ORAL**

DEVELOPMENT OF COOLANT VOIDING MODEL FOR FAST REACTOR CORE

Wednesday, April 20, 2022 2:16 PM (12 minutes)

Unlike thermal reactors, LMFBR core is not in the most reactive configuration. Any undesirable event may raise the reactivity in the core and can result in increase of the reactor power. Liquid metals used in LMFBRs have high boiling points and there is considerable margin between normal operating temperatures and their boiling points. By design, liquid metals used as coolant are not expected to boil under any normal operating conditions of the reactor. However, in case of loss of flow accidents due to pipe rupture, pump failure along with the failure of reactor shutdown systems, boiling of the coolant in reactor core is possible. Such accidents are termed as Unprotected Loss of Flow Accidents (ULOFA). Boiling of coolant in reactor core can also be caused by uncontrolled increase in power. Such accidents are termed as Transient Overpower Accidents (TOPA).

Improved understanding of the mechanics of sodium ejection in case of ULOFA or TOPA is of critical importance for an LMFBR safety. A reduction of sodium density can result in either a positive or a negative reactivity, depending on the location and extent of the vapour void. Therefore, an accurate description of the voiding process with respect to space and time is necessary. NaBOIL, which stands for Natrium Boiling Onset Influence in LMFBRs, is a code developed for predicting boiling behaviour of liquid metals in channels of a reactor core. It calculates the heat transfer from the fuel pin to the coolant until at some location (in coolant channel) and time the coolant reaches a specified superheat. At this point, onset of boiling occurs. The stages of bubble growth are approximated by a thin bubble assumed to occupy the entire coolant channel area, except for the liquid film remaining at the clad wall. The coupled solution of the energy and hydrodynamic equations of the coolant, and the heat transfer equations of the fuel pin are then continuously solved during the voiding process. The main purpose of this model are to predict the extent and rate of voiding that can be used for voiding reactivity calculations and to predict the heat removal from the cladding surface after the onset of boiling, for fuel and cladding temperature calculations.

Country/Int. organization

India

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Presenter: GANATRA, Dhrumil

Session Classification: 6.2 Thermal Hydraulics

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **465** Type: **ORAL**

DESIGN, MANUFACTURING AND IN-SODIUM TESTING OF AM350-WELDED DISC BELLOWS FOR FBTR CONTROL ROD DRIVE MECHANISM

Friday, April 22, 2022 1:54 PM (12 minutes)

Control Rod Drive Mechanisms (CRDM) along with their control rods are used for control and safe shut down of Fast Breeder Test Reactor (FBTR). Lower part of CRDMs which is partially immersed in sodium and nested ripple type welded disc bellows are used to prevent entry of sodium to the annular spaces between the concentric tubes in the lower part. Translation bellows are used to prevent the entry of sodium between translation tube and tube sheath. Welded disc bellows are selected for this application because of requirements of large stroke to length ratio and low stiffness. During SCRAM, the bellows are compressed from their free length at a peak speed of 2.4 m/s. Design and manufacturing of the welded disc bellows is not addressed in standard design codes for bellows such as standards of Expansion Joint Manufacturers Association (EJMA). Design of the bellows was carried out by detailed inelastic analysis. The material data required for design and analysis of the bellows was generated in house. Detailed study for selection of the materials for bellows was carried out and AM350 (precipitation hardened stainless steel) was selected as one of the material of construction of the bellows. Manufacturing of the large stoke welded disc bellows in AM350 which can withstand the loads experienced during SCRAM is first of its kind in India. During this work, various challenges in design, analysis, manufacturing and quality assurance of the bellows were addressed. Procedure and special tooling required for heat treatment of the bellows assembly was established. Indigenously developed translation bellows were subjected to extensive testing including 500 cycles of fast drop testing (Simulating the movements during SCRAM) in air at room temperature and 170 cycles in sodium at 530oC. Subsequently, ten numbers of translation bellows were manufactured for service in the reactor.

Country/Int. organization

India

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Session Classification: 5.3 Experimental Programs II

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **466** Type: **ORAL**

Estimation of mean charge on sodium metal aerosol in the argon and nitrogen gas environment during external gamma irradiation

Thursday, April 21, 2022 10:40 AM (12 minutes)

The cover gas region of sodium cooled fast reactors is always being subjected to intense ionization radiation field apart from radioactive aerosols and gases. The radiation produces significant ionization of the medium resulting in large amount of bi-polar ions. The acquisition of electrical charge by sodium aerosols in cover region under bipolar ionic atmosphere draws special attention as it modifies the dynamics of aerosol transport, deposition and process inside the cover gas space. Towards this, a study has been conducted to characterize the sodium aerosols present in the cover gas region using sodium loop facility (SILVERINA loop) with and without the presence of gamma radiation field using argon as a cover gas and the experiment is repeated with nitrogen gas. The experiments demonstrated that the size of sodium aerosol is found to be relatively higher and mass concentration is lower in the presence of gamma field as compared to the condition without gamma. In order to address the behavior sodium aerosols in the presence of radiation, it is customary to understand the charge acquired by the aerosols under the radiation field. The charge acquired by aerosol is defined by the modified Boltzmann theory and determined as a function of ion mobility of argon and nitrogen gases, aerosol diameter and temperature of cover gas region. The average elementary charge is found to be negative and charge number increases with increase of aerosol size which in turn depends on the sodium pool surface temperature for both the gases. The average charge on sodium aerosol is more in argon gas as compared to the nitrogen and the difference increases with sodium pool temperature. The increase in sodium aerosol charge in argon gas is due to the higher mobility ratio of positive to negative ions for nitrogen relative to the argon. The theoretically determined charge distribution is more asymmetric in argon gas compared to the nitrogen gas. Since the charged aerosols promote more coagulation and enhance surface deposition which are indirectly indicated by the changes in the measured aerosols characteristics. Finally the mean charge can be used for calculation of coagulation and deposition rates which is important for the realistic determination of aerosol characteristics in the cover gas region. The sodium aerosol characteristics and behavior in cover gas gives significant insight into the heat and mass transfer across the cover gas space, cover gas purification, roof slab and handling machines for the fuel sub-assembly.

Country/Int. organization

India

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Session Classification: 5.2 Experimental Programs I

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **468** Type: **ORAL**

LFR Design and Technologies Development at ENEA: Status and Perspectives

Friday, April 22, 2022 2:54 PM (12 minutes)

The next generation of nuclear energy systems, also known as Generation-IV reactors are being developed to meet the highest targets of safety and reliability, sustainability, economics, proliferation resistance and physical protection, with improved performances with respect to plants currently operating or presently being built. Among the proposed technologies, Lead-cooled Fast Reactors (LFRs) have been identified by nuclear industries and Member States among the optimal Generation IV candidates.
Since 2000, ENEA is supporting the core design, safety assessment and technological development of innovative nuclear systems cooled by heavy liquid metals (HLM), and most recently fully oriented on LFRs, developing world-recognized skills in the fast spectrum core design and one of the largest European fleets of experimental facilities aiming at investigating HLM thermal-hydraulics, coolant chemistry control, corrosion behavior for structural materials and material properties in HLM environment, as well as at developing corrosion-protective coatings, components, instrumentation and innovative systems, supported by experiments and numerical tools.
Efforts are also devoted in developing and validating numerical tools for the specific application to HLM systems, ranging from neutronics codes, system and core thermal-hydraulic codes, computational fluid dynamics (CFD) and fuel pin performance codes, including their coupling.
The present work aims at highlighting the capabilities and competencies developed by ENEA so far in the framework of liquid metal technologies for GEN-IV LFRs. In particular, an overview on the ongoing R&D experimental program will be depicted considering the actual fleet of facilities: CIRCE, NACIE-UP, LIFUS5, LECOR, BID-1, HELENA, RACHEL and Mechanical Labs. An overview on the numerical activities performed so far and presently ongoing is also reported.
Finally, an overview of the ENEA contribution to the ALFRED Project in the frame of the FALCON international consortium is reported, mainly addressing the activity ongoing in terms of core design, technology development and auxiliary systems design.

Country/Int. organization

Italy

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Session Classification: 5.3 Experimental Programs II

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **469** Type: **ORAL**

Fuel handling Experience of FBTR

Tuesday, April 19, 2022 4:22 PM (12 minutes)

Country/Int. organization

India

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Session Classification: 8.1 SFR Commissioning, Operation, and Decommissioning

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **475** Type: **ORAL**

Comparisons of Feedback under UTOPA with In Pin Fuel Motion Dynamics in Fast Reactors

Friday, April 22, 2022 11:18 AM (12 minutes)

Safety studies of fast reactors are carried out on a medium sized core and found that, under Unprotected Transient Over Power Accidents (UTOPA) there is fuel melting and there is feedback due to in pin fuel motion. From the UTOPA analyses it is found that, the in pin fuel motion feedback reduces the peak reactor power and hence it reduces the hot spot clad and coolant temperatures at the end of the transients. Under such transients, when there is change in reactor power, for the given primary & secondary systems, the balance of plant gets affected. This results in change in inlet coolant temperature, which affects the temperature profile and has a considerable contribution in shaping the power profile and its respective feedbacks. To study the effect of UTOPA with different boundary conditions, analyses has been carried out with constant inlet coolant temperature (CICT), Time Dependent Inlet Coolant Temperature (TDICT) profile, with and without In Pin Fuel Motion (IPFM) feedback, etc. Comparisons of those results give a better understanding of their respective feedbacks & calculation methodology with respect to different boundary conditions. Considered UTOPA studies with different boundary conditions are,
1. The uncontrolled withdrawal of control rod with CICT, without IPFM. It is based on the assumption that, the molten fuel is hypothetically stationary (available in the same location) even after melting.

1. The uncontrolled withdrawal of control rod with TDICT, without IPFM.
2. Uncontrolled withdrawal of control rod with CICT and IPFM after the initiation of fuel melting.
3. Uncontrolled withdrawal of control rod with TDICT and IPFM feedback after the initiation of fuel melting.

From the comparison of results, it is learnt that the IPFM is significant in bringing the reactor to a safe state under UTOPA. The involved conservatism of the result is observed in the analyses where the IPFM feedback is not considered as compared to the best estimate analyses of considering IPFM feedback. From the study, it is concluded that if a reactor is found to be safe without IPFM feedback, it can actually converged to a safe state with IPFM feedback. Thus the results without IPFM feedback are conservative.

Country/Int. organization

India

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Session Classification: 2.4 Severe Accidents

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **476** Type: **ORAL**

AN EXPERIMENTAL STUDY ON SECONDARY SODIUM SYSTEM BASED DECAY HEAT REMOVAL CIRCUIT OF A SODIUM COOLED FAST REACTOR

Thursday, April 21, 2022 12:04 PM (12 minutes)

Decay heat removal is an important safety function of a nuclear power plant and failure of the same needs to be practically eliminated. Various concepts for decay heat removal have been adopted in different designs of sodium cooled fast reactors (SFRs) depending upon the size and type of reactor. Some of the pool type designs adopt safety grade decay heat removal system (SGDHRS) which consists of three coupled natural convection loops. SGDHRS removes decay heat through immersed decay heat exchangers in the hot sodium pool and sodium to air heat exchanger kept at a higher elevation at the bottom of a tall stack. These systems are capable of functioning under possible extreme conditions affecting the plant. In order to improve reliability of the decay heat removal function, an additional decay heat removal system capable of functioning under emergency conditions which is connected to secondary sodium system may be considered. One typical design configuration for this system adopts a forced cooling type of sodium to air heat exchanger, operating in parallel to steam generators in the secondary sodium circuit of the reactor. The secondary sodium system based decay heat removal circuit could also be considered to function as a normal shutdown cooling system if its controllability under various operational conditions is established. Accordingly, the controllability of the system under varying decay power scenario and utilization of this system for long term maintenance of cold shutdown condition in the plant are some of the important aspects to be established. Towards this, experimental studies have been carried out using 2 MWt capacity sodium to air heat exchanger in the Steam Generator Test Facility (SGTF) at Indira Gandhi Centre for Atomic Research, Kalpakkam, India under varying heat source conditions (simulating the decay power evolution in the reactor core). These studies ensure controllability and operability of the system at desired operating conditions respecting various thermal hydraulic design constraints under various simulated transient conditions.

Keywords: sodium cooled fast reactors, safety grade decay heat removal system, secondary sodium system based decay heat removal system, steam generator test facility.

Country/Int. organization

India

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Session Classification: 5.2 Experimental Programs I

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **477** Type: **ORAL**

DESIGN & ANALYSIS OF A NOVEL ARRANGEMENT FOR COUPLING AND DECOUPLING OF ROTATABLE PLUGS IN PFBR

Friday, April 22, 2022 10:54 AM (12 minutes)

Large and Small Rotatable Plugs (LRP & SRP) form part of top shield and are used to position transfer arm over any required subassembly location during fuel handling. The annular gap between the Roof Slab and LRP as well as between LRP and SRP is sealed with the help of two types of elastomeric seals - Primary inflatable seals and secondary back up seal. These seals have a design life of 10 years and need to be replaced after every 10 years. The procedure to be adopted for replacement of seals involves removal of the hexagonal socket head screws connecting top & middle ring to support ring of LRP/SRP, lowering of the LRP/SRP onto roof slab/LRP respectively and lifting of top & middle ring to gain access to the seals. Similar procedure is applicable for carrying out maintenance of large diameter bearings also. To remove top & middle ring, load on the screws connecting the top & middle ring to rotatable plugs (400t / 200 t for LRP & SRP, respectively) needs to be relieved and then the plug needs to be lowered onto the Roof Slab. To relieve the load and lower the plug, a novel concept of multiple tie rod & nut arrangement, with each one operating in a sequential way was conceived & implemented. This dedicated arrangement was designed taking into account the space constraints over the Top Shield and the eccentric nature of the loading of the plugs. In this paper, the design aspects of arrangement conceived for lowering / lifting of LRP/SRP are discussed along with the results of structural analysis carried out to confirm the design. The maximum allowable torque and nut turn during each operation in the tie rod is estimated and presented.

Country/Int. organization

India

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Session Classification: 1.3 System Innovations

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **478** Type: **ORAL**

Progress in the Design and R&D for future FBRs

Tuesday, April 19, 2022 1:12 PM (12 minutes)

Country/Int. organization

India

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Session Classification: 1.1 Overviews and Fundamentals of Fast Reactors

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **481** Type: **POSTER**

Optimization of Ruthenium concentration in PUREX Process during Fast reactor fuel Reprocessing

Thursday, April 21, 2022 1:40 PM (2 hours)

In Purex process, Ruthenium is one of the troublesome fission products due to its complex chemistry and presence of multiple oxidation states &some extractable stable complexes in nitric acid medium. The higher concentrations of both stable and radioactive ruthenium isotopes, pose many challenges. During the reprocessing of FBR spent fuel, the tri and tetra nitrato complexes of ruthenium get extracted in the first cycle in to TBP. The daughter product of Ru106 is Rh106 which is a hard gamma emitted with a half-life of only 30 s. This leads to the degradation of the solvent due to the extraction of radioactive ruthenium. The degraded solvent in turn holds high plutonium in to the organic phase rendering it unstrippable. In addition, the radioactive Ruthenium pick up will contribute significantly to the residual activity of the product stream. Also, the extractable ruthenium species pose problems in the treatment of the lean organic. Hence it is necessary to limit the co-extraction of Ruthenium in to the solvent. In order to accomplish this, a scheme to scrub the plant stream with concentrated acid has been tested in the plant.

In the CORAL plant during the reprocessing of Pu rich mixed carbide spent fuel discharged from FBTR, the first cycle extraction was carried out at an acidity of 5.5 M. This resulted in the reduction of ruthenium activity in the loaded organic by more than 50%. This reduction was due to the fact that the distribution ratio of ruthenium bears an inverse relation to the concentration of nitric acid in the aqueous phase in equilibrium with the organic in contact with the acid. Analysis of the plant data shows about 30-40% retention of ruthenium activity in the loaded organic. This could be due to the presence of tri nitrato nitrosyl complex of ruthenium $RuNO(NO_3)_3$ under the operating conditions which is extracted by TBP. In the loaded organic phase, the extracted ruthenium is initially expected to be existing as an outer sphere complex, which is slowly gets converted to an inner sphere complex which renders the complex un-scrubbable even with a high acid stream.

Country/Int. organization

India

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Presenter: KRISHNAMURTHY, Ananthasivan (IGCAR, DAE)

Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **484** Type: **POSTER**

Assay of Waste drum based on Passive Neutron Counting Technique

Thursday, April 21, 2022 1:40 PM (2 hours)

Characterization of alpha emitting nuclide and other fission products in the radioactive waste generated in reprocessing plants is a regulatory requirement for their disposal. The assay of plutonium in the solid radioactive wastes could be carried out either using gamma spectrometry or neutron counting, depending mainly on the surface dose of the container. Presence of large amount of fission products renders the use of gamma spectrometry inappropriate due to the increased background radiation. In order facilitate the detection of plutonium in such instances, a passive neutron based assaying system for alpha bearing solid wastes has been designed and developed. The detector system has been fabricated in a semi-circular shape to assay the alpha bearing solid wastes in a 200 L capacity SS drum. This detector employs eight numbers of 3He neutron detectors embedded in High Density Poly Ethylene (HDPE). All the detectors were identical with an active length of 900 mm and a diameter of 50 mm filled with a gaseous mixture of 75% 3He and 25% Kr at 3 bar pressure. For deploying this system in an environment with reasonably high beta gamma background radiation, 25 mm lead shield in front of the detector was used.

Studies reveal that, the rotation of the drum during the assay improves the accuracy of the results. However, only rotation by supplementary angles was found to yield results with minimum error which was found to be in the range of ± 20%. Hence two measurements at 180° apart were found to be sufficient for satisfactory assay. This system was calibrated with plutonium source typically handled in fast reactor fuel reprocessing facilities. The minimum detection limit of typical of research reactor grade plutonium has been estimated to be 35 mg with a 99.9 % confidence level for the back ground prevailing in the site. It has been determined that a gamma radiation background of up to 5 mSv/h could be tolerated without loss of accuracy.

Country/Int. organization

India

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Presenter: Dr P, Sivakumar

Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **487** Type: **POSTER**

Mitigation of Sloshing Effects in High level Liquid Waste (HLW) Storage Tank for Nuclear Spent Fuel Applications

Friday, April 22, 2022 1:30 PM (2 hours)

High level liquid waste (HLLW) is often stored in large capacity horizontal cylindrical tanks especially in fast reactor fuel reprocessing plants. However, these huge tanks when partially filled, pose safety concerns due to seismicity. Violent sloshing during an earthquake-induced Fluid-Structure Interaction (FSI) can lead to catastrophic effects such as structural failures, gas entrainment and roof impact buckling. Therefore, it is important to ensure safe design margins and develop methodologies to overcome a wide range of possible scenarios during design.

In this context, a Computational Fluid Dynamics (CFD) based numerical study is proposed to understand the liquid sloshing dynamics in horizontal cylindrical tank subjected to harmonic excitations. For the purpose of tracking the free surface during the simulation, Volume of Fluid method (VOF) is to be employed. Verification and validation of the proposed numerical model will be presented in detail. An optimum baffle configuration will be recommended to suppress the free surface fluctuations and the associated slosh forces. The sloshing induced free surface height and hydrodynamic pressures will be measured at different locations inside the tank under these conditions. Furthermore, the effectiveness of the baffle geometry is to be tested for a seismic excitation in mitigating the free surface fluctuations and thereby forces on tank walls.

Country/Int. organization

India

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Presenter: KRISHNAMURTHY, Ananthasivan (IGCAR, DAE)

Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **488** Type: **ORAL**

Design Studies Towards Raising FBTR to Full Power

Wednesday, April 20, 2022 11:52 AM (12 minutes)

Country/Int. organization

India

Primary authors: S., RAGHUPATHY (Indira Gandhi Centre for Atomic Research, Kalpakkam); Mr R, Devan (IGCAR); Mr JOHN, Arul (IGCAR); Mr K, Natesan (IGCAR); Mr G.S., Srinivasan (IGCAR); Mr D, Naga Sivayya (IGCAR); Mr V.L, Anuraj (IGCAR); Mr D, Sunil Kumar (IGCAR); Mr JANWADE, Niraj Ganesh (IGCAR)

Presenter: S., RAGHUPATHY (Indira Gandhi Centre for Atomic Research, Kalpakkam)

Session Classification: 2.2 Safety Design and Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **490** Type: **POSTER**

Design, manufacturing and transportation of high capacity High Level Liquid Waste Storage tanks

Thursday, April 21, 2022 10:40 AM (2 hours)

High level liquid Waste (HLLW) storage tanks with large storage capacity weighing a few tens of MT are proposed to be used in Fast Reactor Fuel Reprocessing Plant (FRP) of Fast Reactor Fuel Cycle Facility (FRFCF), Kalpakkam to store the HLLW. Six years storage capacity is envisaged for allowing Ru106 to decay sufficiently before sending the HLLW for vitrification.These tanks have lot of internals such as cooling coil (2 Km piping with about 600 welded Joints) arrangement with ballast tanks positioned inside the vessel to remove decay heat. Presence of highly radioactive fields coupled with highly concentrated nitric acid and remote chance of maintenance after commissioning demands well proven design, material selection, fabrication, meticulous QA practices and transportation methodology to be employed. This presentation dwells upon the design, manufacturing aspects and transportation methodology used for the fabrication of such critical high capacity tanks.

HLW tank, weighing around 67 MT, is a Horizontal cylindrical tank with capacity of 212 m3 fabricated using torispherical formed heads with 4.7m OD and an overall length of 13 m. Material of construction for all the components of the tank is AISI 304L with tailor made chemical composition and other supplementary requirements. The welding process used was manual & semi-automatic GTAW with ER308L filler wire. Prior to fabrication, a detailed study was made to decide the simultaneous fabrication of individual components & the assembling sequence from the point of fabricability & inspectability in tandem meeting project delivery schedule. Based on the above a manufacturing sequence cum quality assurance plan. Accordingly, the individual shell courses, Baffle plates, ballast tanks, saddle supports, cooling coils & air spargers, dished ends with outlet piping, machining of nozzles, manholes, deep feed piping of all tanks were fabricated simultaneously. Production test coupon has been put in place in qualifying all special processes such as Post forming heat treatment. Assembly of individual components including assembly of closure dished end to shell were carried out as sub-assemblies in sequence and the final assembly was completed & tested. Because these were over dimensional consignments, these tanks were transported & unloaded after a detailed road survey, as per approved methodology.

A well designed, thought provoking and proven design, manufacturing sequence & QAP and its effective implementation resulted in the successful fabrication of such critical and complex tanks.

Country/Int. organization

India

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Presenter: Mr DHANANJEYA, Kumst

Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **491** Type: **POSTER**

Evaluation of EPDM and Silicone rubber compounds for application in Reprocessing Plant

Thursday, April 21, 2022 10:40 AM (2 hours)

Many elastomers seals are used in the nuclear industry. Among these elastomers, ethylene propylene diene monomer (EPDM) and silicone rubbers have excellent radiation stability. Both the rubbers can be used for gasket and O-ring application in Reprocessing Plants. To study the suitability of these rubbers for application in the plant, EPDM rubber compound and silicone rubber compounds were prepared and test slabs were fabricated. These rubber compounds were tested for their mechanical properties. Two test slabs of EPDM rubber compound were taken. First test slab was irradiated in gamma chamber followed by exposure to nitric acid (6 M) and the second test slab was exposed to nitric acid followed by irradiation in gamma chamber. Samples were taken out of gamma chamber at regular intervals of time and mechanical properties were tested. Similar procedure was adopted for silicone rubber compound also. The mechanical properties of both the rubber compounds were found to degrade with radiation. Elongation at break of the EPDM rubber compound decreased to 50% of its initial value at a dose of 1 MGy. For silicone rubber, an identical decrease was found even at a dose of 0.1 MGy. When silicone rubber compound was exposed only to radiation, elongation at break decreased to 50 % of its initial value at a dose of 1 MGy. Hence, for seal application in radiation atmosphere alone, silicone rubber can be used up to a dose of 1 MGy and for seal application in combined radiation and acid atmosphere, EPDM rubber can be used up to a dose of 1 MGy.

Country/Int. organization

India

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Presenter: KRISHNAMURTHY, Ananthasivan (IGCAR, DAE)

Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **492** Type: **ORAL**

Design, Experimental trials and Qualification of explosive welding technique for plugging of degraded PFBR Steam Generator tubes

Tuesday, April 19, 2022 4:46 PM (12 minutes)

Country/Int. organization

India

Primary author: PADMANABHAN, Visweswaran (Reactor Design and Technology Group, Indira Gendhi Centre for Atomic Research)

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Session Classification: 8.1 SFR Commissioning, Operation, and Decommissioning

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **493** Type: **ORAL**

Advanced Flow-Sheet for Partitioning of Trivalent Actinides from Fast Reactor High Active Waste

Thursday, April 21, 2022 2:28 PM (12 minutes)

The processes developed for partitioning of trivalent actinides (TA) from high-level liquid waste (HLLW) generated during reprocessing are all focused on single-cycle approaches for waste minimization. In this process the formation of third phase has to be avoided. Hence, a phase modifier is often employed in most of the processes in vouge, even though the use of the later is more desirable. To avoid these complications advanced symmetrical and unsymmetrical diglycolamides were developed in our laboratory and systematically studied for the group separation from fast reactor simulated high-level liquid waste (SHLLW), and then subjected to single-cycle separations. This method involved the separation of trivalent actinides and chemically similar lanthanides, as a group, from SHLLW followed by mutual separation of lanthanides and actinides from the loaded organic phase using aqueous soluble complexing agents.

The potential solvents identified for the group separation of trivalents from HLLW were 1) 0.2 M TODGA (N,N,N'N'-tetraoctyldiglycolamide) + 5% octanol / n-DD, 2) 0.2 M TODGA + 0.5 M TBP (tri-n-butylphosphate) / n-DD, 3) 0.1 M TODGA + 0.25 M HDEHP (di-(2-ethylhexyl) phosphoric acid) / n-DD, 4) 0.2 M TDDGA (N,N,N'N'-tetradecyldiglycolamide) / n-DD, 5) 0.2 M D3DODGA (N,N-didodecyl-N'N'-dioctyldiglycolamide) / n-DD, 6) 0.4 M DOHyA (N,N-dioctyl-2-hydroxyacetamide) / n-DD. The selective stripping of Am (III) from the loaded organic phase containing trivalent lanthanides was investigated using aqueous soluble bis-1,2,4-triazine derivatives such as SO3-Ph-BTP, SO3-Ph-BTBP and SO3-Ph-BTPhen in dilute nitric acid solution. The results revealed that the SF of Eu (III) over Am(III) decreased with increase in the concentration of nitric acid in all cases and separation factor (SF) decreased in the order SO3-Ph-BTP >SO3-Ph-BTBP >SO3-Ph-BTPhen. The co-stripping of lower lanthanides (La, Ce, Pr, Nd) was also observed during the recovery of Am(III). The distribution ratio of Am(III) and Ln(III) in all the organic phases were quite similar. However, TODGA system requires 1-octanol phase modifier for preventing the third phase formation, which is undesirable for safety concerns, whereas the other ligands were modifier-free reagents. Therefore, the other ligands (TDDGA and D3DODGA) developed in our laboratory offered a significant advantage over the TODGA. D3DODGA stripped Am(III) better. As a result the SF for Eu(III) over Am(III) was significantly high (>400) indicating the possibility of using them for MA partitioning. This presentation describes, the summary of our research and development activities carried out at IGCAR towards the development of advanced flow-sheet for TA separation from HLLW generated during fast reactor fuel reprocessing.

Country/Int. organization

India

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Presenter: NARASIMHAN, desigan (IGCAR, DAE)

Session Classification: 3.3 Reprocessing, Partitioning, and Transmutation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **494** Type: **ORAL**

Advanced in-situ Calibration and Probe Release Mechanism for PFBR SG Inspection System (PSGIS)

Tuesday, April 19, 2022 3:46 PM (12 minutes)

Country/Int. organization

India

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Presenter: JOSE, Joel

Session Classification: 8.1 SFR Commissioning, Operation, and Decommissioning

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **495** Type: **POSTER**

Development of Artificial Intelligence through PLC & SCADA to predict process related failure and abnormality in a Reprocessing Plant

Thursday, April 21, 2022 1:40 PM (2 hours)

Programmable Logic Controllers offer complete automation solution and flexibility to control in a plant like the nuclear fuel reprocessing plant. To ensure the safety of both the plant and personnel, continuous monitoring and diagnostics of plant parameters are implemented through various means. Audible and visual alarms are provided to alert the operator in case of process abnormality. But all the mechanisms are available only to report to the operator after an abnormal event, so that corrective action can be taken by the maintenance personnel. This increases the plant down time and involves tedious investigation of all related data in the operator's log and historical trend of the data.

The current work aims to develop an artificial intelligence (AI) based diagnostic tool within the existing HMI application, which can continuously monitor the plant data, review all associated process conditions and then predict the possibility of occurrence of any abnormal event. This AI based tool doesn't limit to event prediction, but also alerts the operator and suggests suitable corrective action that can be taken to avoid the event. An AI algorithm has been developed within the existing PLC/SCADA, thus requires no additional diagnostics tool and ensures smooth operation of plant.

Methods and materials: The system comprises PLC modules and SCADA servers & clients. The PLC modules continuously communicate with the field instruments to read inputs, execute logics and send control outputs to perform various liquid transfers, damper operations in ventilation/offgas systems and other utilities management. The SCADA server communicates with PLC to monitor and control various process, ventilation and radiation data. An AI based application/utility is developed which continuously monitors data in a plant scale to predict the next failure of a particular process or a system, so that predictive maintenance can be taken up by maintenance personnel. The tool predicts any unintended process events like process parameter deviation (tank level or tank pressure) due to impulse tubing leaks /failure of compressed air supply, unintended transfer of radioactive liquid, volume imbalance between source tank and destination tank, reduction in area/containment box vacuum due to exhaust system failures, temperature sensor failures and so on.

Country/Int. organization

India

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Presenter: KRISHNAMURTHY, Ananthasivan (IGCAR, DAE)

Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **498** Type: **POSTER**

Development of a 15 kg servo manipulator for remote handling applications

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

India

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Presenter: D, Jagadishan (IGCAR)

Session Classification: Poster Session

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **499** Type: **POSTER**

Novel Electrical, Electronics and Instrumentation systems for Fast Reactor Fuel Reprocessing Plants

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

India

Primary authors: Mr BHANU, Prakash; Mr SYED, Imran Ali; Mr PADI, Srinivas Reddy; Mr SANDEEP, P C; AVIK, Kumar Saha; Mr R V, Satheesh Kumar; Mrs SWATILEKHA, Bhattacharje; Mr J W, Reuben Daniel; Mr A, Dhanasekaran; Mr JOHN, Swamidoss; Mr AMUDHU, Ramesh Kumar; Mr M, Santhanam; Mr M S, Gopi Krishna; Mr GEO, Mathews; KRISHNAMURTHY, Ananthasivan (IGCAR, DAE)

Presenters: Mr BHANU, Prakash; Mr GEO, Mathews

Session Classification: Poster Session

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **500** Type: **ORAL**

Reactor Core Viewing System for the pre-commissioning stage inspection of reactor core components of Prototype Fast Breeder Reactor

Tuesday, April 19, 2022 3:10 PM (12 minutes)

Country/Int. organization

India

Primary author: RAMALINGAM, Chellapandian (Reactor Design and Technology Group, Indira Gandhi Centre for Atomic Research)

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Session Classification: 8.1 SFR Commissioning, Operation, and Decommissioning

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **501** Type: **POSTER**

DESIGN & DEVELOPMENT OF CUSTOM SHAPED BACK-UP SEAL IN SILICONE FOR PFBR

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

India

Primary authors: AITHAL, Sriramachandra (Indira Gandhi Centre for Atomic Research, Kalpakkam); MOTE, Jagruti (Reactor Design & Technology group, Indira Gandhi centre for Atomic Research); Mrs R, Vijayashree (IGCAR); S., RAGHUPATHY (Indira Gandhi Centre for Atomic Research, Kalpakkam)

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Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **502** Type: **ORAL**

Design of secondary sodium based decay heat removal system for future fast breeder reactors

Friday, April 22, 2022 11:18 AM (12 minutes)

Fast Breeder Reactor -1&2 (FBR-1&2) is a sodium cooled, pool type, Mixed Oxide (MOX) fuelled reactor with two sodium loops (primary and secondary). The design of this reactor is based on experience from Fast Breeder Test Reactor (FBTR) and prototype Fast Breeder Reactor (PFBR). Decay Heat Removal (DHR) system removes decay heat from the reactor after shutdown to ensure adequate cooling of core sub-assemblies. PFBR has two diverse paths for decay heat removal namely, Safety Grade Decay Heat Removal System (SGDHRS) and Operation Grade Decay Heat Removal System (OGDHRS).
OGDHR system requires at least one secondary loop, steam water circuits and off-site power supply for decay heat removal. SGDHR system is operated when OGDHR system is not available. In order to improve reliability of DHR system, it is planned to have an additional DHR system operating on secondary sodium, thus reducing the dependency on SGDHR system. The design of Secondary Sodium based Decay Heat Removal System (SSDHRS) for FBR-1&2 was carried out after, reviewing the design and operational experiences of BN 800, SUPERPHENIX and MONJU available in various forums.
SSDHRS is a part of Secondary Sodium Main Circuit (SSMC), it operates only during shutdown condition for decay heat removal. This system is designed for a heat removal capacity of 15MW. It is provided with an Air Heat Exchanger (AHX) with hot sodium flow in tube side by forced circulation using Secondary Sodium Pump (SSP) and air flow over the tubes by forced circulation using blower.
Heat removal capacity of the system with passive operational mode was also studied and found to be about 60% of the active capacity. System optimization was carried out to arrive at the sizing of various equipment of SSDHRS (Dimensions of AHX, blower capacity, height of stack and circuit design). Parametric studies have been carried out to analyse the effects of sodium temperature and flow rate on heat removal capacity of SSDHRS. SSDHR system is envisaged to cater to fuel handling and other maintenance conditions instead of relying on OGDHR system which requires external power supply, recirculation pumps, condenser cooling fans and steam generators to function.

Country/Int. organization

India

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Presenter: Mr PATEL, Pritam Kumar (Indira Gandhi Centre for Atomic Research)

Session Classification: 1.3 System Innovations

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **503** Type: **ORAL**

Design of metal fuel pin for test irradiation in FBTR and for future reactors.

Thursday, April 21, 2022 11:04 AM (12 minutes)

In India, a structured R&D program on the development of metallic fuel and associated fuel cycle for Fast Breeder Reactors (FBRs) is undertaken so as to realize commercial metal fuel FBRs in the future. Towards this, initially test irradiation of sodium bonded metal fuel pins in Fast Breeder Test Reactor (FBTR) core was proposed and hence the pin design for various compositions of metal fuel was carried out and they are currently being irradiated in FBTR. The compositions include Natural U- 6%Zr, Enriched U -6%Zr, Natural U-19%Pu-6%Zr and Enriched U-23%Pu-6%Zr. Three pins of each type are being irradiated in FBTR and their current burn-up levels are 2.26, 15, 19.5 and 3.75 GWd/t respectively. For a typical test pin irradiation (Enriched U-23%Pu-6%Zr), three sodium bonded metal fuel pins of length 531.5 mm are arranged inside a capsule which is kept inside an ISZ 100 special SA. The thermal and mechanical design of the pin was carried out and the safe operation of fuel pin is ensured for a peak Linear Heat Rating (LHR) of 318 W/cm and for a target burn-up of 100 GWd/t. Also during transients, the maximum allowable flow reduction in the ISZ100 SA was found out to arrive at the blockage limits. During manufacturing of sodium bonded pin, bubbles get trapped inside the bond sodium and hence analysis was carried out to determine the allowable bubble size in a pin.
For the design basis transients, the design safety limits for the metal fuel pin (fuel and clad) have been arrived at by analysis. Also, a 2-D transient mathematical model has been developed for predicting fuel melting and movement of melt interface with respect to time. It was observed that fuel melting starts when the reactor power reaches 1.45 times the nominal power.
Based on the above inputs, for a power reactor fuel composition (Natural U-19%Pu-6%Zr), the thermal and mechanical design of sodium bonded metal fuel pin was also carried out. Thus, this paper details about the design aspects of sodium bonded metal fuel pin which includes arriving the size of fuel pin, fission gas plenum length, allowable linear power, allowable bubble size in the bond sodium and the safety limits for transient events.

Country/Int. organization

India

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Presenter: THIRUNAVUKKARASU, RAJKUMAR (IGCAR)

Session Classification: 3.2 Development of innovative fuels: design and properties irradiation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **504** Type: **ORAL**

Root Cause Analysis of FBTR Failed Fuel Pin

Thursday, April 21, 2022 11:52 AM (12 minutes)

The Fast Breeder Test Reactor (FBTR) at Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, is a loop type, sodium cooled fast reactor. Its main aim is to provide experience in fast reactor operation, large scale sodium handling and to serve as a test bed for irradiation of fast reactor fuels & materials. India has been operating FBTR with Mixed Carbide Fuel as the driver fuel since 1985. Mixed Carbide was chosen as the 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. In the year 2011, MK-1 fuel SA that reached 148.3 GWd/t burnup in III ring of FBTR core had a single pin failure which was identified by both cover gas detectors as well as bulk DND detectors. Subsequently, Post Irradiation Examination (PIE) was carried out on the Failed Fuel Subassembly. Various possible causes of fuel pin failure in the SA were postulated.

One of the initial causes of failure was identified as flow reduction through the SA which was studied and ruled out by a detailed analysis. Also, deformation caused in the fuel pin geometry due to high irradiation dose, results in only 4 % reduction in flow through the SA. Subsequently, a detailed analysis of the failed fuel pin has been carried out for the estimation of Fission gas pressure & FCMI induced stress, clad strains, Clad Cumulative Damage Fraction etc. at different axial levels as a function of burnup. Studies were also carried out to find out the reasons for the ovality of the pins after irradiation. Above parameters are analyzed for the Failed fuel SA and the results are compared with first ring MK-1 fuel SA so as to assess whether failed fuel SA has experienced any abnormalities compared to first ring SA. Also, an attempt has been made to bridge the gap areas between the PIE observations and the analysis results of the failed fuel SA to ascertain the reasons for the failure.

Country/Int. organization

India

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Session Classification: 3.2 Development of innovative fuels: design and properties irradiation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **509** Type: **ORAL**

Over three decades of radiological protection experience at Fast Breeder Test Reactor (FBTR)

Wednesday, April 20, 2022 2:40 PM (12 minutes)

Country/Int. organization

India

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Session Classification: 2.3 Accident Analysis

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **510** Type: **ORAL**

Tensile testing of sub-sized T91 and 316L steel specimens in liquid lead

Wednesday, April 20, 2022 11:52 AM (12 minutes)

Country/Int. organization

European Commision

Primary authors: TUCEK, Kamil (European Commission, Joint Research Centre); Dr SZARAZ, Zoltan; Dr NOVOTNY, Radek (European Commission, Joint Research Centre); Dr NILSSON, Karl-Fredrik (European Commission, Joint Research Centre); Mr NOVAK, Michal (European Commission, Joint Research Centre); FAZIO, Concetta (JRC)

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Session Classification: 4.2 Structural, Novel, and Large Components Materials

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **511** Type: **ORAL**

VERSATILE TEST REACTOR: CONCEPTUAL CORE DESIGN OVERVIEW

Wednesday, April 20, 2022 3:16 PM (12 minutes)

Country/Int. organization

United States of America

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Session Classification: 5.1 Experimental Reactors and Facilities

Track Classification: Track 5. Test Facilities and Experiments

Contribution ID: **513** Type: **ORAL**

Experience in Preheating of PFBR Reactor Assembly

Tuesday, April 19, 2022 3:22 PM (12 minutes)

Country/Int. organization

India

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Session Classification: 8.1 SFR Commissioning, Operation, and Decommissioning

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **514** Type: **ORAL**

Commissioning and Operating Experience for Secondary Sodium Systems and is Auxiliaries of PFBR

Tuesday, April 19, 2022 3:34 PM (12 minutes)

Country/Int. organization

India

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Session Classification: 8.1 SFR Commissioning, Operation, and Decommissioning

Track Classification: Track 8. Commissioning, Operation, and Decommissioning

Contribution ID: **516** Type: **ORAL**

An e-learning tool on Fast Reactors and their Fuel Cycles

Thursday, April 21, 2022 1:52 PM (12 minutes)

The United Kingdom was one of the countries pioneering fast reactor development and demonstration with the construction and operation of two fast neutron spectrum reactors at the Dounreay site; multiple support facilities including zero-power and thermal hydraulic facilities; and demonstrated several sustainable and closed fuel cycles. Since the end of the UK Fast Reactor Programme in 1994 much of the knowledge acquired has been lost or at best archived. The UK National Nuclear Laboratory (NNL), as part of the Department for Business, Energy and Industrial Strategy's (BEIS) £505m Energy Innovation Programme –which includes the biggest nuclear fission investment in a generation, is leading on the Advanced Fuel Cycle Programme (AFCP). This work has begun activities aimed at ensuring this globally unique and highly valuable resource is not lost, through a Fast Reactor Knowledge Capture project.

A key feature of the AFCP is to support the next generation of technical experts, especially the development of these skills in early career individuals from across the nuclear sector. Using information captured from the UK's historic fast reactor programme, a number of online e-learning training modules are to be developed in partnership with the International Atomic Energy Agency (IAEA). These resources are being developed as part of the recently announced NNL-IAEA Collaborating Centre on the Advanced Fuel Cycle. Through the Centre, the IAEA, NNL and UK partners will collaborate on a number of topics relating to the development of advanced fuels and fuel cycles required to power the reactors of the future. The collaboration will place particular emphasis on the exchange of technical expertise between the UK, IAEA and Member State representatives, and in supporting the development of the next generation of experts.

The proposed approach to develop the e-learning module is discussed, focusing on the modular nature to allow the addition to incorporate further topics as they are produced as well as the interactive nature. This will be followed by an overview of the content that provides part of the of the "Background and Introduction"section on Fast Reactors and the planned fuel manufacture and reprocessing content. In summary, the development of future fast reactor experts is a timely and costly endeavour. To support and accelerate the realisation of this aim, modern and interactive training techniques befitting the 21st century should be sought and developed that utilise the historic knowledge developed from research and reactor operations.

Country/Int. organization

United Kingdom of Great Britain and Northern Ireland

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Presenter: Dr BARRON, Nicholas (National Nuclear Laboratory)

Session Classification: 9.1 Education, Profesional Development, and Knowledge Management

Track Classification: Track 9. Education, Profesional Development, and Knowledge Management

Contribution ID: **521** Type: **ORAL**

Passive Heat Removal System Analysis for the Westinghouse Lead Fast Reactor

Friday, April 22, 2022 11:18 AM (12 minutes)

Westinghouse continues to develop its Next Generation high-capacity nuclear power plant (NPP) based on Lead-cooled Fast Reactor (LFR) technology. By leveraging its long experience in NPP commercialization as well as strategic domestic and international partnerships established to most effectively complement capabilities, Westinghouse is progressing the plant's design and its business and delivery model. With a power output of approximately 950 MWt, the Westinghouse LFR is a competitive, medium-size, simple, scalable and passively safe plant harnessing a liquid lead-cooled, fast neutron spectrum core operating at high temperatures in a pool configuration reactor encompassed by a passive heat removal system (PHRS). The paper focuses on the overall design of the Westinghouse LFR PHRS, and the analytical modeling and heat removal capabilities of the PHRS analyzed using the GOTHIC code. The PHRS design encompasses the LFR vessel and relies on a pool of water to effectively remove heat away from the reactor vessel during accident scenarios. During beyond design basis accidents, the PHRS pool water will eventually boil and steam off into the atmosphere, transitioning the pool to air-only cooling. To demonstrate the effectiveness of this system, an evaluation model of the PHRS was created with the GOTHIC computer program. GOTHIC is an integrated, general purpose thermal-hydraulics software package for design, licensing, safety and operating analysis of nuclear power plant containments, confinement buildings and system components. The code can model multiple forms of heat transfer including two-phase and boiling correlations. To simulate a real time response of the entire LFR PHRS, the GOTHIC code is coupled, using internal subroutines, to the LFR system code SAS4A/SASSYS 1. This paper will provide the details associated with the PHRS analysis for the Westinghouse LFR.

Country/Int. organization

United States of America

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Presenter: Mr WISE, Daniel

Session Classification: 6.5 Integrated Analysis and Digitalization

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **528** Type: **ORAL**

The Westinghouse Lead Fast Reactor: overview and progress in development

Wednesday, April 20, 2022 1:40 PM (12 minutes)

Country/Int. organization

United States of America

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Presenter: Dr FERRONI, Paolo

Session Classification: 1.2 Innovative Design Advances

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **534** Type: **ORAL**

BLIND PHASE RESULTS FOR TRANSIENT SIMULATIONS OF THE FFTF LOSS OF FLOW WITHOUT SCRAM TEST #13

Tuesday, April 19, 2022 2:00 PM (12 minutes)

Country/Int. organization

United States of America

Primary author: SUMNER, Tyler (Argonne National Laboratory)

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Presenter: MOISSEYTSEV, Anton (Argonne National Laboratory)

Session Classification: Special Session: IAEA Coordinated Research Projects

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **536** Type: **ORAL**

Blind-Phase Results of the FFTF Neutronic Benchmark

Tuesday, April 19, 2022 2:12 PM (12 minutes)

Country/Int. organization

United States of America

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Presenter: STAUFF, Nicolas (Argonne National Laboratory)

Session Classification: Special Session: IAEA Coordinated Research Projects

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **537** Type: **ORAL**

Sodium coolant: interaction with its environment and coolant processing

Tuesday, April 19, 2022 2:00 PM (12 minutes)

Country/Int. organization

France

Primary author: Dr LATGE, Christian (CEA)

Presenter: Dr LATGE, Christian (CEA)

Session Classification: 4.1 Advanced Reactor Cladding and Core Material, Coolants, and Related Chemistry

Track Classification: Track 4. Fast Reactor Materials (Coolants, Structures) and Components

Contribution ID: **538** Type: **ORAL**

EXAMPLES OF AREAS OF NOVELTY IN LIQUID METAL FAST REACTORS TO CONSIDER IN THE REVIEW OF APPLICABILITY OF THE IAEA SAFETY STANDARDS: FISSION PRODUCT RETENTION BARRIERS: DIFFERENCES BETWEEN LIQUID METAL FAST REACTORS AND LIGHT WATER REACTORS

Tuesday, April 19, 2022 3:22 PM (12 minutes)

Country/Int. organization

IAEA

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Session Classification: 2.1 General Safety Approach

Track Classification: Track 2. Fast Reactor Safety

Contribution ID: **540** Type: **ORAL**

Overview of IAEA Fast Reactor Related Technology Development Activities

Thursday, April 21, 2022 2:40 PM (12 minutes)

The International Atomic Energy Agency (IAEA) supplements and supports nuclear research and development with many efforts to improve and make nuclear data more accessible. Through technical community building research projects, and tool development, the Nuclear Power Technology Development section (NPTDS) provides many services and opportunities to amplify the research of member states'experts. The Fast Reactor (FR) team is guided by the TWG-FR (Technical Working Group on Fast Reactors), the longest operating technical group at the IAEA, by identifying topics for Technical Meetings, providing data for CRPs (Coordinated Research Projects) and proposing other opportunities to further develop fast reactors technology internationally. This paper details several of the projects and work that has been recently completed and the projects currently ongoing.
One of the most visible products of NPTDS are the CRPs. Each project typically lasts for four years and aims to produce a high-quality publication detailing the work of all contributors. This method provides an avenue for information and dating sharing, as well as an opportunity to compare and contrast different simulation tools in use around the world. This paper discusses the CRP process and explain how to contribute to upcoming projects. The Fast Reactor team of NPTDS currently has two ongoing FR CRPs: Neutronics Benchmark of CEFR Start-Up Tests, and Benchmark Analysis of FFTF Loss of Flow Without Scram Test. Recently completed in 2020, Source Term Estimation for the Prototype Sodium Fast Reactor (PSFR) is currently preparing for publication.
In addition to leading and facilitating CRPs, the IAEA focuses on capacity building and reducing barriers to entry of nuclear power. Most recently, this has been accomplished through tool development and open source code community building. In 2020-2021, the IAEA conducted several studies on particular topics such as passive shutdown systems for fast reactors, benefits and challenges of small modular fast neutron systems, interaction of structural materials and liquid heavy metals. By leveraging the vast historical data preserved at the IAEA, NPTDS is engaging in building new resources for analysis. One new project is the Sodium Properties Calculator web application, which demonstrates the initiative for future module development as an additional output of CRPs. ONCORE, the open source code initiative aims to build support resources and communities around analysis tools.
This paper provides an overview of the projects and initiatives of the IAEA in the area of fast reactors technology.

Country/Int. organization

IAEA

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Presenter: MAHANES, Joseph (IAEA)

Session Classification: 9.1 Education, Profesional Development, and Knowledge Management

Track Classification: Track 9. Education, Profesional Development, and Knowledge Management

Contribution ID: **541** Type: **POSTER**

Reprocessing of nitride and metallic spent nuclear fuel using molten salts

Thursday, April 21, 2022 1:40 PM (2 hours)

In recent years, several countries, including Russia, have been developing a pyrochemical (anhydrous) method for spent nuclear fuel (SNF) reprocessing. Molten salts have several advantages, such as thermal and radiation stability, a wide electrochemical window, etc. They can be used practically at all technological stages of SNF processing.

The first stage of pyrochemical reprocessing of nitride spent nuclear fuel can imply its dissolution in the molten LiCl-KCl eutectic containing a chlorinating agent. It is proposed to use CdCl2 or PbCl2 as a chlorinating agent. We have studied in detail the interaction between UN and molten LiCl-KCl eutectic, containing cadmium chloride, depending on the temperature and CdCl2 concentration. It was found that at temperatures below 750 ℃, the interaction proceeds through several parallel reactions and, along with UCl3, a precipitate, consisting of UNCl, nonstoichiometric nitrides UN1.59, UN1.69, U4N7, and several other compounds, is forms. At 750 ℃ and above, all intermediate uranium nitrides dissolve in excess CdCl2 to form UCl3. The conditions, under which the100% conversion of UN ⟶ UCl3 is possible, are provided. The use of lead chloride as a chlorinating agent has also been studied. The chlorination proceeds according to the same mechanism, but the use of PbCl2 allows the process temperature to be reduced by 100 degrees maintaining the 100% UCl3 yield.

The interaction between metallic uranium and its alloys and noble metals with the LiCl-KCl eutectic melt, containing PbCl2, was studied. The dissolution of uranium in such melt is very intense. For samples weighing ~ 15 g, the reaction is completed in 15-20 minutes. The interaction between U-Pd and U-Ru alloys proceeds much more slowly and according to a more complex mechanism. A high temperature and a large excess of PbCl2 are required to complete the reactions. Thermodynamic modeling of the interaction reactions was carried out. The kinetics was studied and the reaction products were identified.

It is shown that pyrochemistry methods may be successfully used for reprocessing of both nitride and metallic spent nuclear fuel. Virtually all processing operations can be performed using molten salts as a process medium.

Country/Int. organization

Russian Federation

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Presenter: Prof. POTAPOV, Alexei (Institute of High Temperature Electrochemistry)

Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **542** Type: **POSTER**

Thermodynamic simulation of the oxidation processes at the reprocessing of spent nuclear fuel in the LiCl-KCl melt

Friday, April 22, 2022 1:30 PM (2 hours)

To date spent nuclear fuel (SNF) reprocessing is a promising field of study. More than 370 thousand tons of SNF have been accumulated in the world and 10-12 thousand tons are added to this amount annually. In Russian Federation, ~25000 tons of SNF were accumulated according to the data of 2018. Pyrochemical technology of SNF processing, which is supposed to substitute aqueous technologies, is currently developed in several countries.

The aim of the present work is the thermodynamic modeling of the processes that take place at the pyrochemical nitride SNF processing.
Spent nuclear fuel is a complex multicomponent system, which is difficult to study because of the composition complexity and high radioactivity. Methods of thermodynamic modeling are indispensable in the process of evaluation of the SNF properties and development of the SNF treating processes.

In the framework of the present work:
- the material composition of the nitride SNF was calculated. The obtained results are in good agreement with the literature data;
- the oxidation processes of UN and some fission products by cadmium and lead chlorides in the molten LiCl-KCl eutectic were modeled. In particular, the formation of the deposit containing UNCl and non-stoichiometric uranium nitrides is explained. The optimal chlorination temperature is suggested;
-the oxidation process of metallic uranium and some U-noble metals alloys are modeled;
- the amount of missing thermodynamic data is evaluated. In particular, the thermodynamics of nitrides of several rare-earth metals as well as americium and curium nitrides was evaluated.

The modeling was performed using the software HSC Chemistry 9.9.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 6. Modelling, Simulations, and Digitilization

Contribution ID: **543** Type: **POSTER**

Electrical conductivity of multicomponent chloride melts, containing ions of mono-, di-, and trivalent metals

Thursday, April 21, 2022 1:40 PM (2 hours)

Melts based on the LiCl-KCl eutectic are becoming attractive in various industrial fields, including nuclear industries. However, their transport characteristics have not yet been sufficiently studied.

The purpose of this work is to study the electrical conductivity of melts similar to those formed during the dissolution of real nitride spent nuclear fuel in (LiCl-KCl)eut., and also to develop a model that would allow us to evaluate the electrical conductivity of multicomponent melts of arbitrary compositions based on the conductivity of 2-3 component mixtures.

To achieve this goal, we measured the electrical conductivity of the molten (LiCl-KCl)eut. mixtures with various mono-, di- and trivalent metal chlorides (CsCl, CdCl2, SrCl2, CeCl3, NdCl3, UCl3) in a wide temperature range. Also, the electrical conductivity of several multicomponent mixtures (LiCl-KCl)eut. - CsCl + MeCl2 + MCl3 with various combinations and concentrations of components was measured. In the present work, a capillary quartz cell with platinum electrodes and the AC-bridge method at the input frequency of 10-75 kHz were used. The density of the melts and their molar electrical conductivity was calculated.

Electrical conductivity is a non-additive property. For example, the conductivity deviations of the (LiCl-KCl)eut. + NdCl3 molten mixture from additive values reach ~ 80-90%. The stronger the interaction (complexation) between the ions in the melt, the greater the deviations from additivity. Mixtures composed of (LiCl-KCl)eut. and CeCl3, NdCl3, etc., showed such strong interactions. The results were interpreted in terms of the coexistence and mutual influence of the complexes formed by mono-, di-, trivalent cations, and counter anions in these molten mixtures. When UCl3 or LnCl3 are dissolved in the molten LiCl-KCl eutectic the nearest U3+ - Cl- or Ln3+ - Cl- distance is reduced, as well as the coordination number of the trivalent cation. In all cases, coordination number(CN) ≥ 6. This leads to a decrease in the concentration of electricity carriers Li+, K+ and, especially, Cl-, and, thus, to a decrease in the electrical conductivity of the melt, as we observed experimentally.

Country/Int. organization

Russian Federation

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Presenter: Prof. POTAPOV, Alexei (Institute of High Temperature Electrochemistry)

Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **544** Type: **ORAL**

Investigation of the anodic processes on the ceramic anode in the oxide-chloride melts

Thursday, April 21, 2022 3:04 PM (12 minutes)

Alkaline halide melts and alkaline earth metals are used for the electrochemical reduction of metal oxides to their metallic forms. In practice, fluoride, chloride, and mixed chloride-fluoride melts of alkali and alkaline earth metals are used most often. Graphite is usually applied as an inert anode material in these media. However, during the electrolysis of oxide-halide melts, carbon is not an inert anode. In recent years, metals and their alloys, metal oxides, and cermets have been considered to be the candidate inert anode materials for the electrolysis of oxide-halide melts. Almost all investigated metal anodes made of individual metals such as iron, nickel, copper, chromium, and their alloys, except some noble metals are unstable in an oxygen atmosphere and oxidize at high temperatures.

At present, oxide-chloride melts based on LiCl is used for the electrolytic reduction of uranium and plutonium oxides with lithium formed at the cathode. Platinum metal is usually used as the anode material for oxygen evolution in these melts. However, platinum is highly susceptible to corrosion in these melts and therefore is not the inert anode material. Also, platinum is an expensive metal, which makes it difficult to use as an anode material in the industry. We have carried out studies of anodic processes and electrolysis tests on the ceramic anode NiO-Li2O in LiCl-KCl-Li2O melts. Studies have shown that ceramics NiO-Li2O is the inert anode material for electrolysis of LiCl-KCl-Li2O melts at temperatures of 550-650˚C. Voltammetric studies have shown that two electrode processes can occur on the NiO-Li2O anode: 1) oxidation of oxide ions with the formation of gaseous oxygen up to potentials of 2.8-2.9 V vs E (Li+/Li) and 2) chlorination of the anode material at potentials more positive than 3.0-3.1 V vs E (Li+/Li). Experiments carried out in the process of electrolytic reduction of UO2 and UO2 with additions of rare earth oxides shown that NiO-Li2O is found to be the inert anode material. The anode current efficiency of oxygen evolution at this anode is close to 100%. As a result of electrolysis experiments during 35 h, the diameter and length of the anode sample did not decrease. Thus, ceramics NiO-Li2O can be used as the inert anode material for electrolysis of oxide-chloride melts based on LiCl.

Country/Int. organization

Russian Federation

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Session Classification: 3.3 Reprocessing, Partitioning, and Transmutation

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **545** Type: **POSTER**

Determination of the metallic and oxide compounds in models based on metallic uranium containing uranium dioxide, metallic neodymium, cerium as well as neodymium and cerium oxides

Thursday, April 21, 2022 1:40 PM (2 hours)

To determine uranium in the metallic phase in the presence of uranium oxide there is a reliable, so-called "bromine method", which implies a metallic-oxide mixture treating in the bromine ethyl acetate solution. However, analogous manipulations with rare earth metals and their oxides do not provide such reliable data. Reduction melting of oxides in a graphite crucible with the melt composed of additional metals is another method, which allows determining the total amount of oxygen bonded in the sample. Together with the common chemical analysis of desired elements and using two aforesaid methods we obtain an algorithm of definite manipulations that provide the relation of metallic and oxide phases of different metals in the samples under study. In the present case, the "bromine"method provides reliable data on the metallic uranium and uranium dioxide, i.e. oxygen bonded to uranium. At the same time, the reduction melting method provides information on the total oxygen concentration in the sample under study, which allows calculating the amount of oxygen on every atom of the rare-earth metal considering the data on the amount of oxygen bound to uranium.

The suggested algorithm of the chemical operations was verified using model mixtures, in which various combinations of metals (U, Nd, Ce, and metallic Pd) and their oxides (U, Nd, Ce) were used. The mixtures composed of the known amounts of various substances were used as samples. Metallic uranium served as a basic component (85-95 wt.%). The experimental results were in good agreement with the theoretically obtained values on the concentrations of metallic uranium, neodymium, cerium, and their oxides.

Country/Int. organization

Russian Federation

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Presenter: Prof. ZAIKOV, Yury (Institute of High Temperature Electrochemistry of the Ural Branch of the Russian Academy of Sciences)

Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **546** Type: **POSTER**

Electrolytic reduction of the simulated oxide spent nuclear fuel in LiCl-Li2O melt

Thursday, April 21, 2022 1:40 PM (2 hours)

A pyrochemical technology for reprocessing spent nuclear fuel (SNF) and fast reactors is being implemented. One of the redistributions of pyrochemical technology is the electrochemical reduction of uranium dioxide (actinide oxides) with lithium in a LiCl - Li_2O melt (1-2 wt.%) uranium dioxide and rare earth oxides at 650 ℃. To test the technological regimes of the reduction process, we used a model nuclear fuel (MNF). It was a mixture of uranium dioxide and rare earth oxides. Nickel oxide ceramics were used as the anode, and a stainless steel basket, into which MNT pellets were loaded, served as the cathode. The electrolysis process was carried out at a cathode potential more positive than the separation of the liquid phase of metallic lithium. The total amount of electricity consumed for the reduction of MNF in one cycle did not exceed 160% of the theoretical value required for the electrolytic production of lithium for the reduction of uranium dioxide.

The UO_2 + 5-10 wt. % tablets (La_2O_3, CeO_2, Nd_2O_3 in a ratio of 1:1:1) were used as samples for reduction. To determine the degree of reduction of the cathode product to metals, we proposed a combined approach for the determination of the metal phase in the reduced product.

The first, "bromine" method consists of dissolving the reduced product in a solution of bromine in ethyl acetate. The metal fraction of uranium goes into the liquid phase, and the remaining uranium oxides remain in the precipitate. This method is generally accepted for determining the conversion of uranium dioxide to metal. It is possible to accurately determine the amount of uranium metal and its dioxide and, consequently, the oxygen associated with uranium in the test sample.

The second method is the reduction melting of metals and oxides in a graphite crucible using a molten metal bath at a high temperature, carried out by us on a Metavak-AK device. This method allows determining the total oxygen content of the sample. The combination of these two methods and general chemical analysis for the elements of interest to us allows us to determine the amount of oxygen per atom of a rare earth element (lanthanide).

It has been shown experimentally that executing the reduction process, subject to the above conditions, makes it possible to obtain a product with the reduction to metallic uranium by 98-99%, while lanthanum, cerium, and neodymium remain in the form of oxides.

Country/Int. organization

Russian Federation

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Session Classification: Poster Session

Track Classification: Track 3. Fuels, Fuel Cycles and Waste Management

Contribution ID: **548** Type: **POSTER**

EXPORT OF RBN WITH SNCD AND NUCLEAR PROLIFERATION RISKS

Tuesday, April 19, 2022 3:10 PM (2 hours)

Country/Int. organization

Russian Federation

Primary authors: KUCHINOV, Vladimir; Mr GORIN, Nikolai (FSUE RFNC - VNIITF named after Academ. E.I. Zababachin, Snezhinsk); KUZNETSOV, Evgeni (FSUE RFNC - VNIITF named after Academ. E.I. Zababachin, Snezhinsk); Mr ADAMOV, Evgeni (JSC «Proryv», Moscow); Mr CHEBESKOV, Aleksandr (JSC «SSC RF - IPPE», Obninsk); Mr SHIDLOVSKIY, Vladimir (JSC «Proryv», Moscow)

Presenter: KUCHINOV, Vladimir

Session Classification: Poster Session

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **550** Type: **ORAL**

The INPRO project studies on the double-component nuclear power systems with the closed fuel cycle and fast reactors: past and future

Tuesday, April 19, 2022 5:10 PM (12 minutes)

Country/Int. organization

IAEA

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Presenter: BYCHKOV, Alexander (Senior Nuclear Engineering Expert - INPRO)

Session Classification: 7.1 Sustainability: Economics, Environment, and Proliferation

Track Classification: Track 7. Sustainability: Economics, Environment, and Proliferation

Contribution ID: **551** Type: **POSTER**

LARGE-SCALE HYDROGEN PRODUCTION; Fast-neutron Reactors Coupled to Thermochemical Copper-Chlorine Hydrogen Plant

Tuesday, April 19, 2022 1:00 PM (2 hours)

Country/Int. organization

Canada

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Presenter: EL-EMAM, Rami (Ontario Tech University)

Session Classification: Poster Session

Track Classification: Track 1. Innovative Fast Reactor Designs

Contribution ID: **552** Type: **not specified**

China Key Note

Tuesday, April 19, 2022 10:30 AM (12 minutes)

Country/Int. organization

Presenter: Mr YANG , Hongyi

Session Classification: Plenary 1. Keynotes from Member States

Contribution ID: **553** Type: **not specified**

France Key Note

Tuesday, April 19, 2022 10:42 AM (12 minutes)

Country/Int. organization

Presenter: Mr SERRE, Frederic

Session Classification: Plenary 1. Keynotes from Member States

Contribution ID: **554** Type: **not specified**

India Key Note

Tuesday, April 19, 2022 10:54 AM (12 minutes)

Presenter: Mr VENKATARAMAN, Balasubramaniam

Session Classification: Plenary 1. Keynotes from Member States

Contribution ID: **555** Type: **not specified**

Japan Key Note

Tuesday, April 19, 2022 11:06 AM (12 minutes)

Presenter: Mr KAMIDE, Hideki

Session Classification: Plenary 1. Keynotes from Member States

Contribution ID: **556** Type: **not specified**

Republic of Korea Key Note

Tuesday, April 19, 2022 11:18 AM (12 minutes)

Presenter: Mr CHAE YOUNG, Lim

Session Classification: Plenary 1. Keynotes from Member States

Contribution ID: **557** Type: **not specified**

Russian Federation Key Note

Tuesday, April 19, 2022 11:30 AM (12 minutes)

Presenter: Mr PERSHUKOV, Vyacheslav

Session Classification: Plenary 1. Keynotes from Member States

Contribution ID: **558** Type: **not specified**

United States Key Note

Tuesday, April 19, 2022 11:42 AM (12 minutes)

Presenter: Ms CAPONITI, Alice

Session Classification: Plenary 1. Keynotes from Member States

Contribution ID: **559** Type: **not specified**

European Commission (EC) Key Note

Wednesday, April 20, 2022 9:30 AM (12 minutes)

Presenter: Ms BETTI (TBV), Maria

Session Classification: Plenary 2. International Organizations and YGE Winners

Contribution ID: **560** Type: **not specified**

Generation IV International Forum Key Note

Wednesday, April 20, 2022 9:42 AM (12 minutes)

Presenter: Mr HILL, Bob

Session Classification: Plenary 2. International Organizations and YGE Winners

Contribution ID: **561** Type: **not specified**

OECD/Nuclear Energy Agency Key Note

Wednesday, April 20, 2022 9:54 AM (12 minutes)

Presenter: Ms IVANOVA, Tatiana

Session Classification: Plenary 2. International Organizations and YGE Winners

Contribution ID: **562** Type: **not specified**

International Atomic Energy Agency (IAEA) Key Note

Wednesday, April 20, 2022 10:06 AM (12 minutes)

Presenter: Ms DES CLOIZEAUX, Aline

Session Classification: Plenary 2. International Organizations and YGE Winners

Contribution ID: **563** Type: **not specified**

Conference Chair Address

Tuesday, April 19, 2022 9:40 AM (10 minutes)

Presenter: Mr BHADURI, Arun Kumar

Session Classification: Opening Session

Contribution ID: **564** Type: **not specified**

IAEA Opening Address

Tuesday, April 19, 2022 9:30 AM (10 minutes)

Presenter: Mr GROSSI, Rafael

Session Classification: Opening Session

Contribution ID: **565** Type: **not specified**

Administrative Remarks

Tuesday, April 19, 2022 9:50 AM (15 minutes)

Presenter: Ms GONZALEZ-ESPARTERO, Amparo (Scientific Secretary)

Session Classification: Opening Session

Contribution ID: **566** Type: **not specified**

YGE Winner: Advanced Functional Materials for Next-Generation Fuel Reprocessing

Wednesday, April 20, 2022 10:18 AM (5 minutes)

Country/Int. organization

Presenter: Mr KUNTAL KUMAR PAL

Session Classification: Plenary 2. International Organizations and YGE Winners

Contribution ID: **567** Type: **not specified**

YGE Winner: Production of Mo-99 isotope in the BN reactor by beryllium blocks

Wednesday, April 20, 2022 10:23 AM (5 minutes)

Country/Int. organization

Presenter: Ms KUCHERYAVYKH, Oksana

Session Classification: Plenary 2. International Organizations and YGE Winners

Contribution ID: **568** Type: **not specified**

YGE Winner: Small Modular Fast Reactors for the ASEAN Region: Implementation of the TRISO Fuel Particle Concept as a Regional Variant of the Fast Reactor

Wednesday, April 20, 2022 10:28 AM (5 minutes)

Country/Int. organization

Presenter: Mr CHUAN, Tan Zhe

Session Classification: Plenary 2. International Organizations and YGE Winners

Contribution ID: **569** Type: **ORAL**

Chair of Conference International Advisory Committee Closing Remarks

Friday, April 22, 2022 3:40 PM (20 minutes)

Presenter: Mr BHADURI, Arun Kumar

Session Classification: Closing Session

Contribution ID: **570** Type: **ORAL**

Conference General Co-Chair Closing Remarks

Friday, April 22, 2022 4:00 PM (20 minutes)

Presenter: Mr CHUDAKOV, Mikhail

Session Classification: Closing Session