Sodium-cooled fast reactor (SFR)

Main characteristics of the system

The primary mission for the SFR is the effective management of high-level wastes and uranium resources. If innovations to reduce capital cost and improve efficiency can be realized, the Generation IV SFR is an attractive option for electricity production. The Generation IV Technology Roadmap ranked the SFR highly for advances it offers towards sustainability goals. The fast reactor closed fuel cycle significantly improves the utilization of natural uranium, as compared to ~1% energy recovery in the current once-through fuel cycle. By recycling the plutonium and minor actinide spent fuel components, decay heat and radiotoxicity of the waste are minimized. The SFR is also highly rated for safety performance.

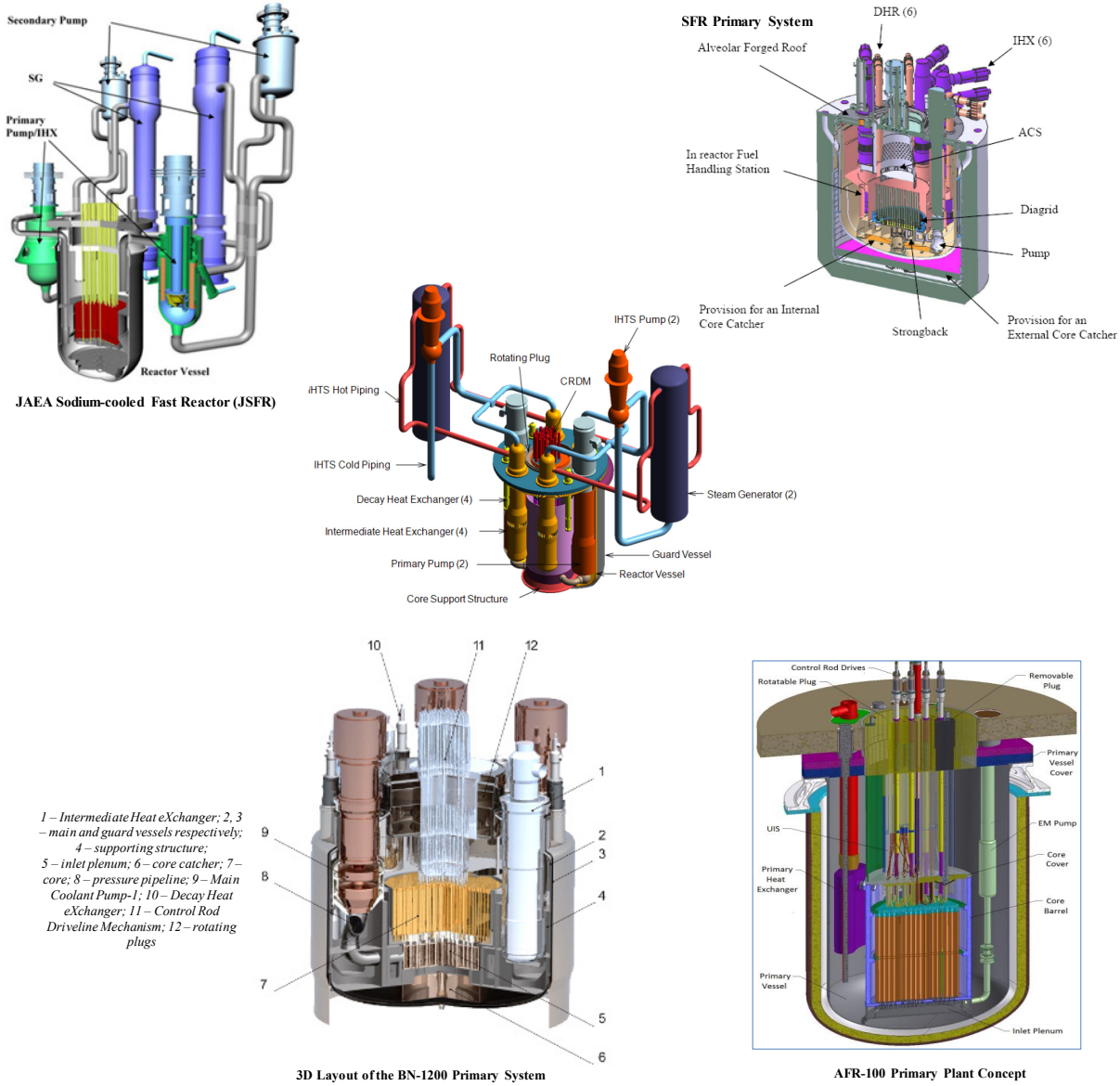
The SFR system uses liquid sodium as the reactor coolant, allowing high power density with low coolant volume fraction. Because of advantageous thermo-physical properties of sodium (high boiling point, heat of vaporization, heat capacity, and thermal conductivity) there is a significant thermal inertia in the primary coolant. While the oxygen-free environment prevents corrosion, sodium reacts chemically with air and water and requires a sealed coolant system. The primary system operates at near-atmospheric pressure with typical outlet temperatures of 500-550°C; at these conditions, austenitic and ferritic steel structural materials can be utilized, and a large margin to coolant boiling at low pressure is maintained. The reactor unit can be arranged in a pool layout or a compact loop layout. Typical design parameters of the SFR concept being developed in the framework of the Generation IV System Arrangement are summarized in **Table SFR 1**. Plant sizes ranging from small modular systems to large monolithic reactors are considered.

Table SFR 1. **Typical Design Parameters for the Generation IV SFR**

Reactor parameters	Reference value
Outlet Temperature	500-550°C
Pressure	~1 Atmosphere
Power Rating	50-2000 MWe
Fuel	Oxide, metal alloy, others
Cladding	Ferritic-Martensitic, ODS, others
Average Burn-up	150 GWD/MTHM
Breeding Ratio	0.5 -1.30

There are many sodium-cooled fast reactor conceptual designs that have been developed worldwide in advanced reactor development programs. In particular, the BN-800 Reactor in Russia, the European Fast Reactor in the EU, the Advanced Liquid Metal Reactor (PRISM) and Integral Fast Reactor Programs in the United States, and the Demonstration Fast Breeder Reactor in Japan, have been the basis for many SFR design studies. For Gen-IV SFR research collaboration, several *system options* that define general classes of SFR design concepts have been identified: loop configuration, pool configuration, and small modular reactors. Furthermore, within this structure several *design tracks* that vary in size, key features (e.g. fuel type) and safety approach have been identified with pre-conceptual design contributions by Gen-IV SFR Members: JSFR (Japan), KALIMER (Korea), ESFR (Euratom), BN-1200 (Russia), and AFR-100 (United States) (see **Figure SFR.1**). The Gen-IV SFR design tracks incorporate significant technology innovations to reduce SFR capital costs by a combination of configuration simplicity, advanced fuels and materials, and refined safety systems; thus, they are utilized to guide and assess the Gen-IV SFR R&D collaborations.

Figure SFR 1. **The five Gen-IV SFR design tracks**



Status of co-operation

The system arrangement for the Gen-IV international R&D collaboration for the SFR nuclear energy system became effective in 2006 and was extended for a period of ten years in 2016. Several new Members were added to the original agreement and United Kingdom was welcomed to the system arrangement in 2019. The present signatories are: Commissariat à l'énergie atomique et aux énergies alternatives, France; US Department of Energy; Joint Research Centre, Euratom; Japan Atomic Energy Agency; Ministry of Science and ICT, Korea; China National Nuclear Corporation; Rosatom, Russia; UK Department of Business, Energy and Industrial Strategy.

Based on international R&D plans, the Gen-IV SFR research activities are arranged by the SFR Signatories into four technical Projects: System Integration and Assessment (SIA), Safety and Operations (SO), Advanced Fuels (AF) and Component Design and Balance-of-Plant (CDBOP).

Three Project Arrangements (PAs) were signed in 2007: Advanced Fuel (AF), Component Design and Balance-of-Plant (CDBOP), and Global Actinide Cycle International Demonstration (GACID). The PA for Safety and Operation (SO) was signed in 2009, and the PA for System Integration and Arrangement (SIA) was signed in 2014. The Project Arrangements were agreed for a ten-year term with annual updates of the Member contributions. The PA for AF and the PA for GACID expired in 2017. A new PA (Phase II) for AF for next ten years was entered into force in 2018. The PA for CDBOP and SO extended for another ten years in 2017 and 2019 respectively.

R&D objectives

SFR designs rely heavily on technologies already developed and demonstrated for sodium-cooled reactors and associated fuel cycle facilities that have successfully been built and operated in several countries. Overall, approximately 400 reactor years of operating experience have been logged on SFRs, including 300 years on smaller test reactors and 100 years on larger demonstration or prototype reactors. Significant SFR research and development programs have been conducted in the United States, Russia, Japan, France, India[1] and the United Kingdom. The only SFR power reactors in operation are the BN-600 (Russia) which has reliably operated since 1980 with a 75% capacity factor, and the BN-800 which started commercial operations in 2016. Currently operating test reactors include BOR-60 (Russia), and CEFR (China). The JOYO (Japan) test reactor is in licensing process for restart. New SFR test reactors MBIR (Russia) and VTR (United States) are expected in the next decade. In addition, SFR technology development programs are being pursued by all members of the GIF SFR System Arrangement.

A major benefit of previous investments in SFR technology is that the majority of the R&D needs that remain for the SFR reactor technology are related to performance rather than viability of the system. Accordingly, the Generation IV collaborative R&D focuses on a variety of design innovations for actinide management, improved SFR economics, development of recycle fuels, in-service inspection and repair, and verification of favorable safety performance.

System integration and assessment project (SIA): Through systematic review of the Technical Projects and relevant contributions on design options and performance, the SIA Project will help define and refine requirements for Generation IV SFR concept R&D. The Generation IV SFR system options and design tracks are identified and assessed with respect to Generation IV goals and objectives. Results from the technical R&D projects will be evaluated and integrated to assure consistency.

Safety and operation project (SO): The SO project is arranged into three work packages (WPs) which consist of WP SO 1 “Methods, models and codes” for safety technology and evaluation, WP SO 2 “Experimental programmes and operational experience” including the operation, maintenance and testing experience in the experimental facilities and SFRs (e.g. Monju, JOYO, Phénix, BN-600, BN-800 and CEFR), and WP SO 3 “Studies of innovative design and safety systems” related to the safety technology for the Gen-IV reactors such as inherent safety features and passive safety systems.

Advanced Fuel project (AF: presently expired and phase II project is under preparation): The Advanced Fuel Project aims at developing and demonstrating minor actinide-bearing (MA-bearing) high burn-up fuel for SFRs. The R&D activities of the Advanced Fuel Project include fuel fabrication, fuel irradiation and core materials (e.g. cladding materials) development. The advanced fuel concepts include non-MA-bearing driver fuels for reactor start-up as well as MA-bearing fuels as driver fuels and targets dedicated to transmutation, in order to address both homogeneous and heterogeneous ways of MA transmutation as a long-term goal. Fuels considered include oxide, metal, nitride and carbide. Currently, cladding/wrapper materials under consideration include austenitic as well as ferritic/martensitic steels but aim to transition in the longer term to other advanced alloys, such as ODS steels.

1. India is not belonging to GIF.

Component design and balance-of-plant project (CD&BOP): The project includes the development of advanced Energy Conversion Systems (ECS) to improve thermal efficiency and reduce secondary system capital costs. The project also include R&D on advanced in-service inspection and repair (in sodium) technologies, small sodium leak consequences, and new sodium testing capabilities. The main activities in energy conversion systems include: (1) development of advanced, high reliability steam generators and related instrumentation; and (2) the development of advanced ECS based on a Brayton cycle with supercritical carbon dioxide or nitrogen as the working fluid. In addition, the significance of the experience that has been gained from SFR operation and upgrading is shared.

Main activities and outcomes

In this Section, recent Member contributions to the Gen-IV SFR collaboration are highlighted.

System integration and assessment (SIA) Project: In 2019, five trade and assessment studies were contributed. CIAE contributed a study that evaluates the CFR1200 design main heat transfer parameters. Key factors that significantly influence the thermal performance were identified (e.g. primary/secondary circuit temperatures). They performed sensitivity analyses for these main factors and quantified the impacts on system efficiency and component design.

During the design phases of the ASTRID demonstrator (2010-2019), CEA continuously assessed and improved the design of ASTRID to enhance its safety. It is a good example of how SFR safety can be improved by design with a core showing favorable natural behavior under multi-failure accident conditions, and with added devoted complementary safety devices to prevent or mitigate severe accidents. In 2019, based on the ASTRID design evolutions, CEA is providing its feedback experience on the SFR safety enhancement by design.

Within the ESFR-SMART project, various safety improvements for ESFR have been proposed, taking into account the safety objectives envisaged for Gen-IV reactors and the recommendations following the Fukushima Daiichi accident. The Euratom contributions provide overviews of the improved ESFR safety approach including the safety requirements for the evaluation of the innovative design options, the assessments of the proposed system safety measures and recommendations for further developments. Safety approaches assessments were performed using the GIF RSWG ISAM methodology relevant tools: Qualitative Safety features Review (QSR) and Objective Provision Tree (OPT). The focus of the contribution for this year is the use of ISAM QSR including a short description of the QSR approach for SFR and a checklist of recommendations developed for a generic SFR concept. The contribution discusses the assessment of the checklist for the ESFR-SMART considering compliance with defence-in-depth, safety objectives, ALARA principle and need for harmonization of safety and security architecture. Included are recommendations and conclusions on the QSR application.

JAEA contributed a study on countermeasures against sodium-water reactions. A single tube helical coil steam generator was evaluated as a design alternative to the JSFR double-wall type. Failure propagation and leak detection behavior was compared for the two concepts. Future work will include a detailed evaluation of sodium-water pressure and system impacts.

KAERI performed a deployment scenario study of large size TRU burners to estimate spent fuel accumulation from PWR operation and to evaluate radiotoxicity reduction of spent fuels by introducing TRU burners. The spent fuel accumulation from PWR operation was estimated based on domestic plans for long-term electricity demand and supply. The spent fuel accumulation of TRU recycle was compared to that of direct disposal, and the radiotoxicity of finally disposed high-level wastes of TRU recycle reaches natural uranium level after about 5 000 years.

Safety and Operations Project: As the topic of the SO project, the common project that consists of two benchmark analyses (EBR-II test and PHÉNIX Dissymmetric tests) have been started in the SO project last quarter of 2019. The first phase of the benchmark analysis ("blind phase") will take two years.

The SO project is structured in three work packages (WPs): WP SO 1 "Methods, models and codes", WP SO 2 "Experimental programmes and operational experiences" and WP SO 3 "Studies of innovative design and safety systems".

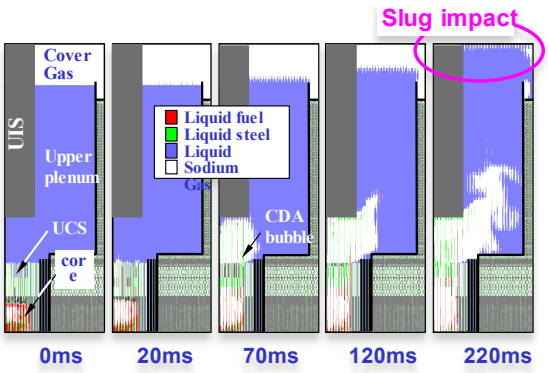
WP SO 1: Methods, models and codes

CIAE conducted benchmark analysis for EBR-II Shutdown heat removal tests SHRT-17 and SHRT-45R, as a part of joint work with ANL from 2017 to 2019.

In order to improve the simulation of an Unprotected Loss of Flow transient, CEA studied the way to calculate the reactivity coefficients to be used in the system code point kinetics module. MACARENa and APOLLO3 Codes were used for this study. As a result, it was confirmed that the point kinetics parameters (in particular the sodium-void reactivity worth) were strongly affected by 3D angular effects. Meanwhile, the transient simulation results were not very different. Hence, the most significant progress is probably to be found in improving the neutronic/thermo-hydraulic coupling. In order to demonstrate that a severe accident is sufficiently unlikely, CEA provided the deliverable which presents the safety demonstration methodology, measurement systems, and reactor protection sub-systems corresponding with each core meltdown initiating events.

JAEA developed the evaluation method of the consequence of energetics in Post-Disassembly Expansion (PDE) phase during an unprotected loss-of-flow accident. They performed the preliminary evaluation of mechanical energy and reactor vessel response using the developed method, see **Figure SFR 2**. As a result, no Slug Impact nor residual strain of the reactor vessel was predicted in the case of realistic temperature condition. Therefore, they obtained the perspective of the robustness of prototype SFR against the energetics in severe accident conditions.

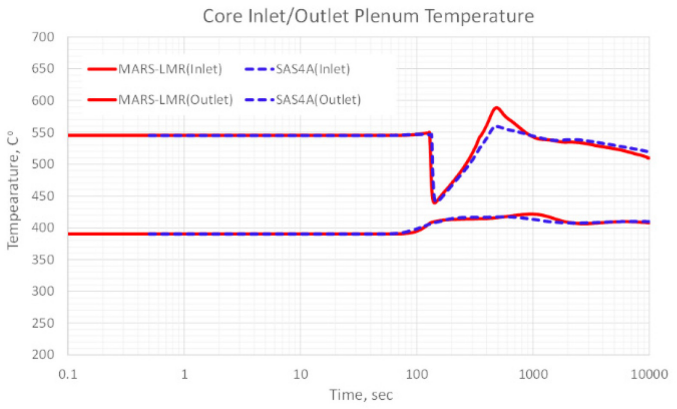
Figure SFR 2. **Material distribution calculated by JAEA (pessimistic temperature condition)**



In order to obtain licensing approval for the developed code (MARS-LMR), KAERI carried out the comparative safety analysis with SAS4A/SASSYS-1 code. In **Figure SFR 3**, sample results from the comparison of the Transient Over Power, Loss Of Flow, and Loss Of Heat Sink cases are shown. The safety analysis results from each code were found to be in good agreement.

IPPE (Rosatom) continued developing 3D severe accident analysis code COREMELT3D. The 3D model of the reactor gas system (from the gas volume under the sodium level in the reactor through the expansion tank up to the ventilation system) has been developed and implemented into the code. This model has been integrated with the primary circuit 3D thermo-hydraulic model, it is necessary for simulating transport of gaseous fission products from disintegrated fuel pins up to ventilation system, and consequently into the environment. IPPE has performed integral analysis of the consequences of the severe accidents in BN-1200. There have been used the following codes: COREMELT3D (core, primary and intermediate circulation loops, emergency system of heat removal, reactor gas system), KUPOL-BR (ventilation system), VYBROS-BN (transport of radioactive products in the environment under different meteorological conditions, doses). IPPE has performed preliminary experiments with thermite compositions to obtain melt of stainless steel with high temperature. This technique will be used in a facility (which is being designed now) to simulate transport of melt in the SFR conditions.

Figure SFR 3. **Comparison between MARS-LMR and SAS4A/SASSYS-1 codes on Loss Of Flow case**



Safety analysis results of core temperatures for LOHS accident

WP SO 2: Experimental programmes and operational experiences

CIAE conducted the experimental research and the code development for Heat Transfer Analysis of CEFR Damaged Spent Fuel Assemblies in Closed Space. The experiment simulated the spent fuel assemblies during transportation and the heat transfer characteristics were investigated.

The Project ESFR-SMART aims to evaluate the safety of a low-void Sodium Fast Reactor (SFR) core design, in particular the analysis of an unprotected loss-of-flow (ULOF) accident. Recent studies on the low-void SFR core show the occurrence of a stabilized chugging sodium boiling regime that can be classified as a new safety measure acting as a level of defence preventing severe accidents. In order to better understand and simulate the chugging boiling regime condition and to gather new experimental data, the ESFR-SMART project envisaged the construction of a new simple facility named CHUG (see **Figure SFR 4)**, designed using water as simulant. The Euratom contribution describes the pre-test calculation results, as well as the facility layout for the first phase of the test, including the main parts and the instrumentation. Preliminary results and main outcomes of first phase of experiments are summarized. Results of analytical simulations of the experiment conducted using the thermal hydraulics code TRACE to assess the validity of the code for the simulation of chugging boiling are shared.

Figure SFR 4. **Layout of the CHUG facility**

Euratom discusses design guidelines for sodium loops. Using liquid sodium at high temperatures in test facilities requires defining rules specific to this technology to ensure that operations are safe and reliable. The purpose of this contribution is to explain the safety rules to be incorporated by the designer during the definition of a project to build a facility implementing liquid sodium. The recommendations take into account European feedback on safety issues related to the design of sodium facilities. However, they do not under any circumstances replace the regulations in force applicable to each subject discussed.

WP SO 3: Studies of innovative design and safety systems

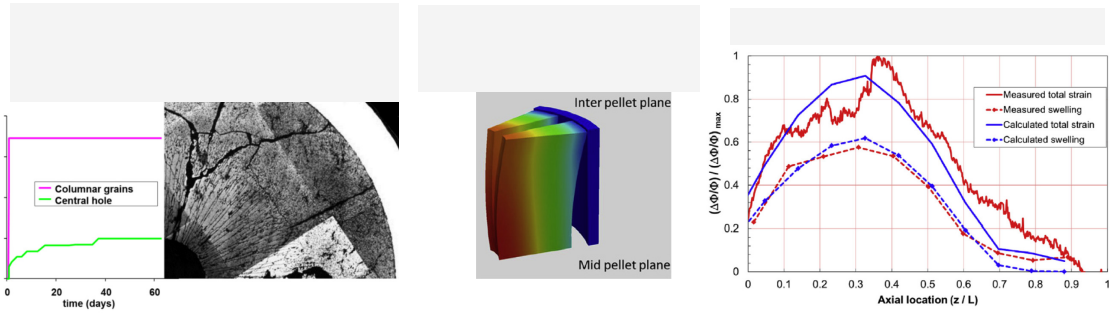
In order to confirm the applicability of Self-Actuated Shutdown System (SASS) in conditions of low power operation, JAEA carried out design modification and 3D thermal-hydraulic analysis and calculated the response time of SASS. By introducing the improvement of design and the required temperature difference, core damage was prevented by the SASS in the case of LOF type ATWS event from low power operation.

Advanced Fuels Project: The AF project consists of three work packages: WP 2.1 "SFR Non-MA-bearing Driver Fuel Evaluation, Opt. & Demo.", WP 2.2 "MA-bearing Transmutation Fuel Evaluation, Opt. & Demo." and WP 2.3 "High-burn-up Fuel Evaluation, Opt. & Demo."

WP 2.1: SFR Non-MA-bearing Driver Fuel Evaluation, Optimization & Demonstration

CEA presented the current capabilities of the GERMINAL fuel performance code, part of the PLEIADES simulation platform, used for (U, Pu) mixed oxide fuel pins calculations. The modelling of GERMINAL and its validation by comparison between calculations and measurements have been shared (see **Figure SFR 5**).

Figure SFR 5. **Examples of calculations and validation studies for GERMINAL code**



DOE continued to develop simulation tools for the evaluation of metallic fuel performance. Additional models were added and improved in the BISON fuel performance code to enhance its ability to model both U-Pu-Zr and MOX fuel for sodium fast reactors. Also, DOE successfully fabricated novel geometries of Pu bearing metal fuel to support accelerated testing. This includes both small diameter samples needed for high fission rate testing as well as more complex fuel geometries that can explore alternative methods to accommodate swelling that eliminate the need for sodium bonding.

JAEA measured the physical properties of non-stoichiometric (U, Pu)O_2 as function of Pu content, Am content, O/M ratio and temperature.

WP 2.2: MA-bearing Transmutation Fuel Evaluation, Optimization & Demonstration

CEA has performed a preliminary thermo-mechanical design of a MA-bearing oxide pin loaded with 10% of americium in UO_2 matrix. The behavior of the pin has been calculated with GERMINAL fuel performance code with specific developments for MA-bearing fuels.

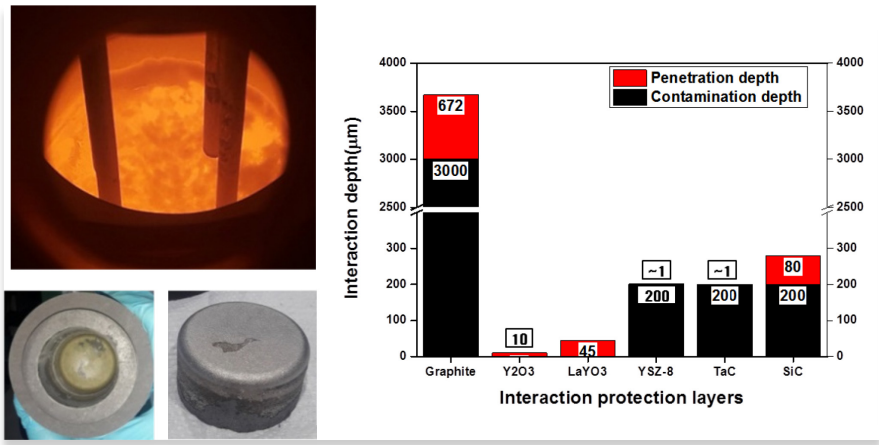
DOE investigated the effect of minor actinide additions to metallic fuel through post-irradiation examination (PIE) and micro-chemical analysis of minor actinide bearing transmutation fuel. Minor actinide bearing fuel irradiated in several different reactors including true fast spectrum reactors (EBR-II, Phenix) and pseudo-fast spectrum tests in the Idaho National Laboratory Advanced Test Reactor were all compared. The fuel performance across these different conditions was fundamentally the same, and the addition of minor actinides did not significantly change the performance of the fuel. The "Metallic Fuel Handbook" which documents the fundamental thermo-physical properties of metallic fuel alloys and their constituents was updated with a significant revision to the U-Zr and U-Pu-Zr system information.

Euratom JRC installed the Cold Finger Apparatus (KüFA) installed into the JRC hot cell facilities for out-of-pile safety transient testing. A temperature transient up to 1 800°C will be applied and the release of gaseous and solid fission products will be quantitatively determined as a function of time. Additionally, JRC studied on Synthesis of Am-bearing MOX fuel for the homogeneous recycling concept ($(U,Pu,Am)O_2$), containing circa 5% americium, 20% plutonium and 75% uranium. The proposed synthesis method synthesize $(U,Th)O_2$ nanopowder (particle size is about 5 nm).

JAEA evaluated effects of Am on MOX fuel temperature using an irradiation behavior analysis code, it was suggested Am-MOX fuel could be irradiated with the same conditions as conventional MOX fuel.

KAERI completed fuel rod fabrication for the 2nd Fuel irradiation test in HANARO (SMIRP-2 test), which will be started from 2020. They also conducted the development of reusable crucible and mold for metal fuel fabrication, see Figure 3.10. In order to reuse casting parts, various new materials were tested for the coating material, which confirmed the effectiveness of the Y_2O_3 coating.

Figure SFR 6. **Development of reusable crucible and mold**



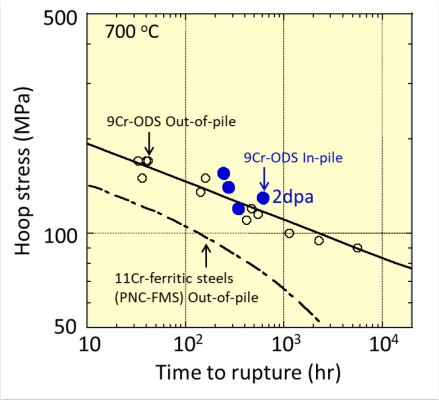
Rosatom created the experimental installation of low temperature sintering of mixed nitride uranium and plutonium fuel by the HVPC-process. Through the test using titanium nitride, they confirmed good repeatability of the HVPC-process. And they started the testing of the installation via uranium nitride.

WP 2.3: High-burn-up Fuel Evaluation, Optimization & Demonstration

CIAE updated the oxide fuel performance code, FIBER, to analyze up to 10at% burn-up fuel. They conducted the verification of the FIBER code and the benchmark analysis with the past CEFR calculation.

JAEA has developed 9Cr-ODS tempered martensitic steel (TMS) as prospective material for high burn-up fuel cladding tube. JAEA confirmed that there was no remarkable degradation in the in-pile creep mechanical strength of 9Cr-ODS TMS cladding tubes irradiated in JOYO (see **Figure SFR 7**). Furthermore, on the basis of knowledge on 9Cr-ODS TMS having prominent mechanical strength and irradiation resistance, JAEA started developing a new type of high Cr-ODS TMS: the 11Cr-ODS TMS for improving corrosion resistance.

Figure SFR 7. **Comparison of In-pile and Out-of-pile creep rupture strength of 9Cr-ODS TMS**



KAERI conducted the parametric study and sample manufacturing for Cr electroplating for nuclear cladding applications. And they also conducted several performance tests (Out-of pile diffusion couple test, mechanical test).

Rosatom have planned the Post-Irradiation Examination (PIE) for the specimens made of EP823 ODS steel in order to increase the burnout level of the fuel. Rosatom fabricated the samples for this PIE and conducted the pre-reactor tests. Additionally, the first stage of irradiation was completed and they are conducting the PIE.

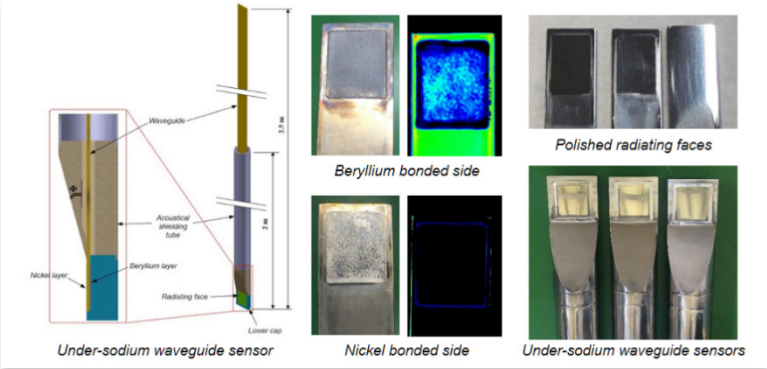
Component Design and Balance-Of-Plant Project: Activities within the CD&BOP project include experimental and analytical evaluation of advanced In-Service Inspection, Instrumentation & Repair technologies (ISI&R), development of Advanced Energy Conversion Systems (AECS), study of sodium Leakages and Consequences (SL), advanced Steam Generator technologies (SG), and study of sodium Operation technology and new sodium Testing Facilities (O&TF).

ISI&R technologies

CEA have studied the capability of the Leaky Lamb Waves on the non-destructive testing from outside of the main vessel. They developed the model to represent the behavior of Leaky Lamb Waves in plates, and validated it by comparing with the literature results. Additionally, for the further validation, they prepared experimental devices consisting of immersed plates, emitter and receiver.

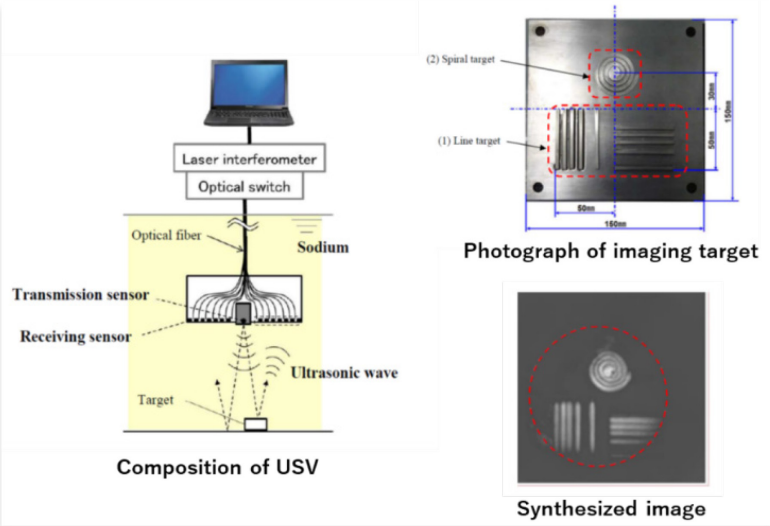
KAERI investigated the sodium-wetting property of the plate-type ultrasonic waveguide sensor under various conditions to improve the sensor performance. KAERI fabricated waveguide sensors with different surface roughness of radiating faces, and prepared a sodium-wetting test facility. They also constructed a new sodium test facility for the verification of under sodium viewing and ranging capabilities of the plate-type ultrasonic waveguide sensor as shown in Figure SFR.8.

Figure SFR 8. **Waveguide sensors fabricated for sodium-wetting tests**



JAEA developed an improved imaging under sodium viewer for a middle distance (see Figure SFR.9). The transmission sensor provided better profile of the wave, and the receiving sensor successfully reduced the noise of the wave profile. The imaging experiment in water showed that higher resolution can be obtained through the improvement of the imaging under sodium viewer.

Figure SFR 9. **Imaging experiment in water for the improved imaging under sodium viewer**

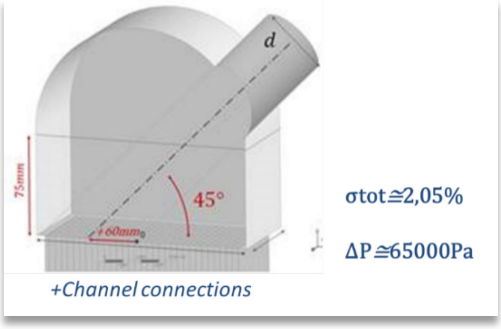


Supercritical CO_2 Brayton cycle

CEA is developing the method to detect the bubbles in the sodium flow by using eddy current flowmeter (ECFM). In 2019, CEA proposed a numerical representation of an ECFM and a bubble. Through the comparison study with experimental result, it was confirmed that this method can detect the bubble effectively.

CEA also conducted the parametric study and the optimization of the design of the heat exchanger's header. Their CFD models validated by the experimental result was used for this optimization. As the result of the optimization of the header and the channels bundle (see **Figure SFR 10**), the maldistribution level was reduced from 25% to about 2%, as compared to the design objective of 5%.

Figure SFR 10. **Design of heat exchanger's optimized header**



Sodium leakages and consequences

In 2019, no specific activity was conducted in this work package.

Steam generators

JAEA studied on the heat transfer coefficient inside tube in case of sodium-water reaction. The overheating tube rupture is one of the considerable failure mode derived from the sodium-water interaction. For the evaluation of the possibility of this mode, heat transfer coefficient of water side is important factor. JAEA conducted the rapid heating experiments for the tube containing water flow and estimated the heat transfer characteristics of inner surface. Based on this experimental results, correlations for RELAP5 code was modified and their conservativeness were confirmed.

KAERI has upgraded the signals analysis software as well as the combined SG tube inspection sensor and signal acquisition device. The upgraded software newly employs several signal transformation functions for MFL image processing, and an automatic defect detection algorithm. They conducted performance tests of the upgraded prototype combined SG tube inspection sensor system, and confirmed its defect detection performance.

Sodium operation technology and new sodium testing facilities

KAERI has completed the installation of the STELLA-2 test section in 2019, and remaining works for a cold test will be finished in early 2020. The first test data for sodium integral effect test using STELLA-2 will be obtained no later than the end of 2020 as well. Besides the sodium thermal-hydraulic test program, KAERI is constructing new test facilities for sodium leak detection and simulation, which are called WALSUM (Water-mock-up test for Advanced Leak Simulation and Upgraded Monitoring system) and SELAAD (Sodium Experimental Loop for Advanced Aerosol Detection). The objectives of the new facilities are to develop highly reliable sodium leak detection and monitoring system as well as performance evaluation of advanced sodium leak detectors.

Bob Hill

Chair of the SFR SSC and all Contributors