Sodium-cooled fast reactor

Main characteristics of the system

The primary mission of the sodium-cooled fast reactor (SFR) is the effective management of high-level waste and uranium resources. If innovations to reduce capital cost and improve efficiency can be realized, the Gen-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 closed fuel cycle significantly improves the use 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 the 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 used, and a large margin to coolant boiling at low pressure can be maintained. The reactor unit can be arranged in a pool layout or a compact loop layout. The typical design parameters of the SFR concept being developed in the framework of the Gen-IV system arrangement (SA) are summarized in Table SFR-1. Plant sizes ranging from small modular systems to large monolithic reactors are being considered.

Table SFR-1. Typical design parameters for the Gen-IV SFR

Reactor parameters	Reference value
Outlet temperature	500-550°C
Pressure	~1 atmosphere
Power rating	30-5 000 MWt (10-2 000 MWe)
Fuel	Oxide, metal alloy, and others
Cladding	Ferritic-martensitic, ODS, and 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. For example, 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 United States, as well as 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 the general classes of SFR design concepts have been identified: loop configuration, pool configuration and SMRs. Furthermore, within this structure several design tracks that vary in size, key features (e.g. fuel type) and safety approaches have been identified with pre-conceptual design contributions by Gen-IV SFR members: CFR1200 (China), the Japanese sodium-cooled fast reactor (JSFR, Japan), Korea advanced liquid metal reactor (KALIMER, Korea), ESFR (Euratom), BN-1200 (Russia) and AFR-100 (United States). Gen-IV SFR design tracks incorporate significant technology innovations to reduce SFR capital costs through a combination of configuration simplicity, advanced fuels and materials and refined safety systems. They are thus used to guide and assess Gen-IV SFR R&D collaborations.

Status of cooperation

The system arrangement for Gen-IV international R&D collaboration on 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 the United Kingdom was welcomed to the system arrangement in 2019. The present signatories are: the French Alternative Energies and Atomic Energy Commission (CEA), France; the Department of Energy, United States; the Joint Research Centre, Euratom; the Japan Atomic Energy Agency, Japan; the Ministry of Science and Information and Communication and Technology (ICT), Korea; the China National Nuclear Corporation, China; Rosatom, Russia; the Department for Business, Energy and Industrial Strategy, the United Kingdom.

Based on international R&D plans, Gen-IV SFR research activities are arranged by SFR signatories into four technical projects: system integration and assessment (SIA), safety and operations (SO), advanced fuel (AF) and component design and balance of plant (CD&BOP).

R&D objectives

SFR designs rely heavily on technologies already developed and demonstrated for sodium-cooled reactors, and for the 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: 300 years on smaller test reactors and 100 years on demonstration or prototype reactors. Significant SFR research and development programs have been conducted in

France, Japan, India,[1] Russia, the United States and the United Kingdom. The only SFR power reactors in operation are the BN-600 and the BN-800 (both in Russia). Currently operating test reactors include the BOR-60 (Russia) and CEFR (China). The Joyo (Japan) test reactor is in the licensing process for restart. New SFR test reactors, the MBIR (Russia) and VTR (United States), are expected in the next decade. In addition, SFR technology R&D programs are being pursued by all SFR GIF members.

A major benefit of the maturity of the SFR technology is that the majority of the remaining R&D needs are related to performance rather than the viability of the system. Accordingly, Gen-IV collaborative R&D focuses on a variety of design innovations for actinide management, improved SFR economics, the development of recycle fuels, in-service inspection and repair (ISI&R) and verification of favourable safety performance.

The system integration and assessment (SIA) project: through a systematic review of the technical projects and relevant contributions on design options and performance, the SIA project will help define and refine requirements for Gen-IV SFR concept R&D. The SFR system options are assessed with respect to Gen-IV goals and objectives. Results from the R&D projects will be evaluated and integrated to ensure consistency.

The safety and operation (SO) project: the SO project is arranged into three work packages (WPs): 1) WP SO1 "methods, models and codes" for safety technology and evaluation; 2) WP SO_2 "experimental programs and operational experience", including the operation, maintenance and testing experience in facilities and SFRs (e.g. Monju, Joyo, Phenix, BN-600, BN-800 and CEFR); and 3) WP SO3 "studies of innovative design and safety systems" related to safety technology, such as inherent safety features and passive systems.

The advanced fuel (AF) project: the AF project aims at developing and demonstrating minor-actinide-bearing (MA-bearing) high burn-up fuel for SFRs. The R&D activities of the AF project include fuel fabrication, fuel irradiation and core materials (cladding materials) development. The advanced fuel concepts include both non-MA-bearing driver fuels (reactor start-up) and MA-bearing fuels as driver fuels and targets (dedicated to transmutation). The fuels considered are oxide, metal, nitride and carbide. Currently, cladding/wrapper materials under consideration include austenitic and ferritic/martensitic steels, but the aim is to transition in the longer term to other advanced alloys, such as ODS steels.

The component design and balance-of-plant (CD&BOP) project: this project includes the development of advanced energy conversion systems (ECS) to improve thermal efficiency and reduce secondary system capital costs. The project also includes R&D on advances in sodium ISI&R technologies, small sodium leak consequences and new sodium testing capabilities. The main activities in ECS include: 1) development of advanced, high-reliability steam generators and related instrumentation; and 2) the development of advanced energy conversion systems (ECS) based on a Brayton cycle with supercritical carbon dioxide or nitrogen as the working fluid.

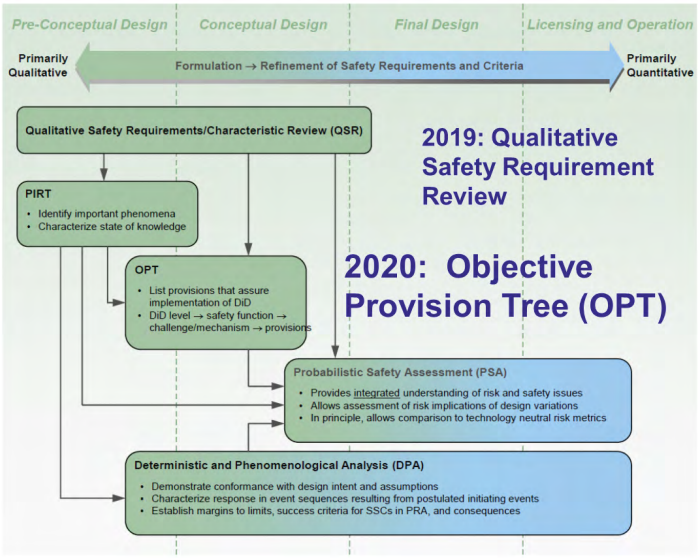


Figure SFR-1. Assessment of safety requirements for the European sodium fast reactor

Main activities and outcomes

SIA project

The China Institute of Atomic Energy (CIAE) contributed a study that evaluates the main heat transfer parameters of the CFR1200 design. 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.

The CEA has developed the "CADOR" core concept, which reduces the volume power and adds some moderators in order to eliminate severe accidents scenarios induced by unprotected events. As a result of safety analysis studies, the CADOR core has demonstrated good natural behavior in the case of unprotected transients through improvements inDoppler feedback. However, the volume of the core becomes larger than the classical core design (ASTRID CFV core).

The JRC conducted on an assessment of safety requirements for the European sodium fast reactor (ESFR) using Integrated safety assessment methodology (ISAM) tools. The Horizon 2020 European Sodium Fast Reactor Safety Measures

1. India is not a member of GIF.

Assessment and Research Tools (ESFR-SMART) project, launched in 2017, aims at enhancing further the safety of Gen-IV SFRs, and in particular the commercial-size ESFR in accordance with the European Sustainable Nuclear Industrial Initiative (ESNII) roadmap. Within the project, Euratom applied the ISAM tools developed by GIF RSWG to the ESFR in order to assess safety requirements. This contribution is multi-annual, and 2019 deliverables dealt with the application of the ISAM qualitative safety feature review (QSR). In 2020, Euratom released deliverables on the application of the ISAM objective provision tree (OPT) to ESFR-SMART studies.

The JAEA has been reconsidering advantages gained from previous innovative technologies because significant changes were required in the design of the JSFR after the Fukushima Daiichi NPP accident. The R&D load, the risk of each innovative technology and updated lists of innovative technologies for loop-type reactors were reviewed. This review is based on development easiness and preparation of design standards. Additionally, the total mass and safety of the design are being considered.

KAERI is developing a steam generator concept to minimize sodium-water reaction. A copper bounded steam generator (CBSG) was selected as an alternative steam generator concept. Triple isolation wall structure with steel tube/copper matrix/steel tube layered geometry and eliminated welds were introduced to achieve a very low probability of sodium-water reaction, and the heat exchanger modules were designed by sizing and CFD analysis. Manufacturing tests of a small-scale module through hot isostatic pressing (HIP) diffusion bonding, tension and contact thermal resistance tests of HIP bonding materials, and thermal fluid visualization tests and structural analyses for CBSG were performed.

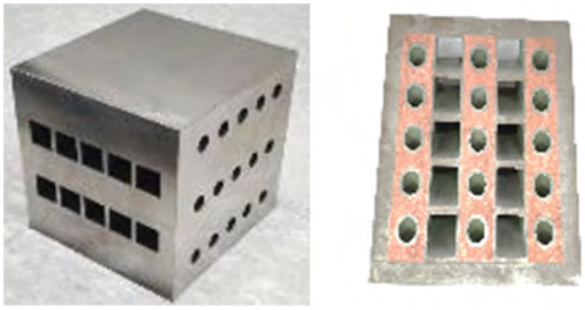


Figure SFR-2. Manufacture of a small-scale copper bonded steam generator module using the hot isostatic pressing process

Safety and operations project

On the topic of the safety and operation (SO) project, the common project that consists of two benchmark analyses (the EBR-II test and Phenix dissymmetric tests) started from the last quarter of 2019. The first phase of the benchmark analysis is a "blind phase", which will take two years to complete. Argonne National Laboratory (ANL), the JAEA and KAERI completed the blind phase of the EBR-II test study at the end of 2020.

WP SO 1: Methods, models and codes

The CEA has analyzed debris bed cooling after a severe accident that followed an unprotected loss of flow (ULOF). In this analysis, one in-vessel DHX was considered and one ex-vessel decay heat removal system. Two types of models have been used for CFD calculations: the laminar model and the K-Ω SST turbulence model. The CEA has demonstrated the calculation results on core catcher temperatures and hot pool temperatures. In terms of the core catcher, the temperature of the insulator (i.e. a few centimetres of ZrO_2) was less than 1 000°C. That value was verified at less than the fusion temperature of ZrO_2; 2 750°C during the simulations. The temperature of the debris, taking into account the conservative assumptions performed, were less than 1 500°C. This result provides a wide margin for a non-desired scenario disruption.

The CEA also presented detailed calculations for flow patterns in the lower plenum, which requires a full 3D-calculation approach. The heat removal trend up until 35 000 seconds was calculated through CFD, modified by the porous media

Figure SFR-3. Local debris and insulator temperatures

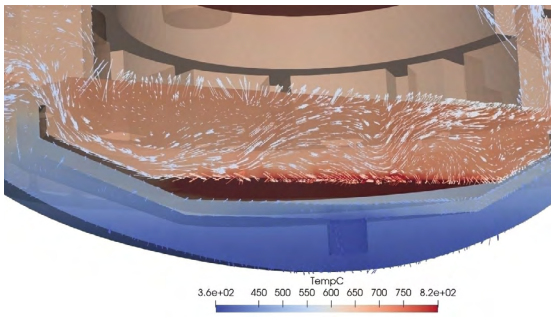
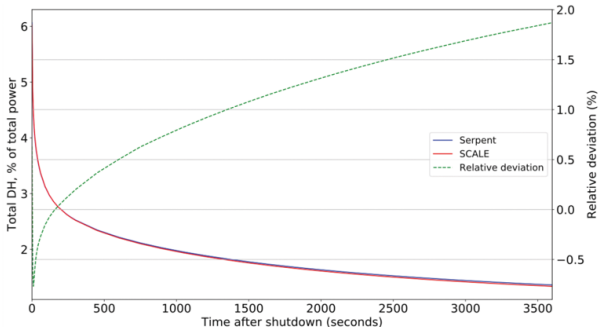


Figure SFR-4. Total decay heat normalized to nominal power



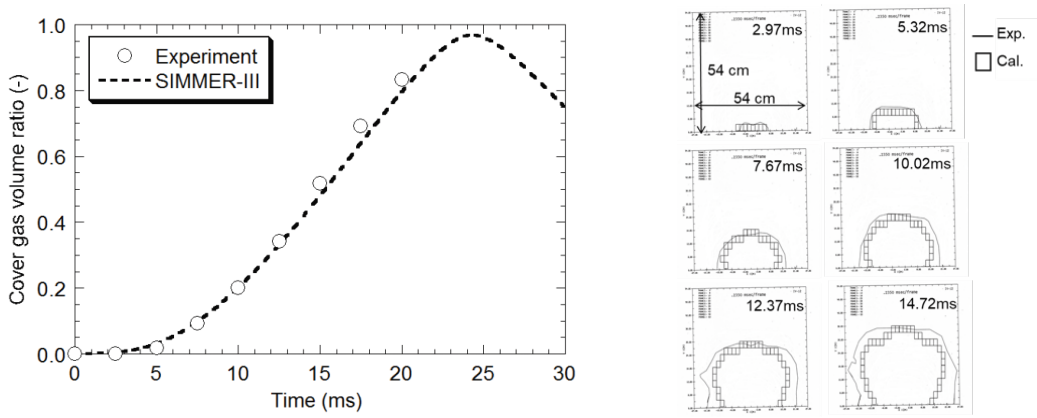


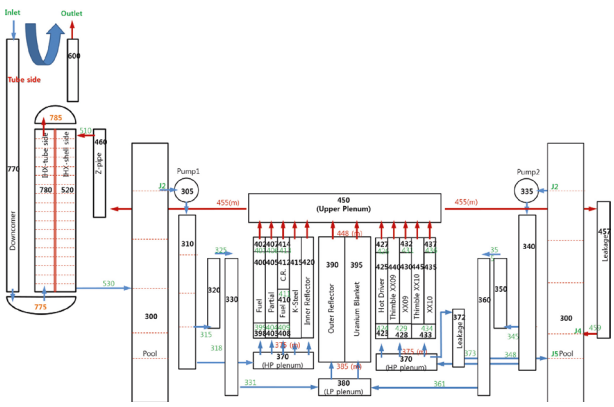
Figure SFR-5. Examples of SIMMER-III analysis results for the OMEGA test

model. After 35 000 seconds of transient, the power balance between heat decay and power removed had not yet been reached. The power gap decreases suggest a convergence after a few days.

Euratom has assessed safety parameters related to the end of cycle (EOC) loading of the ESFR-SMART core, including full core and local effect analysis. The EOC state of the core presents the most limiting case for safety analysis. The deliverable discusses estimation safety parameters, such as control rod insertion S-curve, sodium void reactivity, thermal expansion, Doppler constant and void worth. In addition, sensitivity and uncertainty analyses of different safety parameters related to nuclear data have been performed. The spatial and time-dependent decay heat characteristics normalized to nominal power were also estimated.

The JAEA has been developing an advanced computer code, SIMMER-III/IV, for the analysis of a core disruptive accident (CDA). For the validation of the SIMMER-III code, the JAEA presented a comparison of analysis results for material expansion dynamics, with experimental results. The JAEA selected two experiments for the validation study: the VECTORS test carried out by JAEA to focus on the phenomena of multi-phase flow in structure, and the OMEGA test undertaken by Purdue University to focus on the phenomena of huge vapour bubble expansion dynamics. SIMMER-III successfully simulated both the VECTORS test and the OMEGA test, and has proven to be practical and useful for SFR severe accident analysis.

Figure SFR-6. MARS-LMR modelling of the EBR-II BOP test



KAERI performed a preliminary benchmark analysis of EBR-II BOP-301/302R tests using MARS-LMR. Typical behaviors and significant increases in the core inlet temperature in unprotected loss of heat sink (ULOHS) were investigated in the EBR-II BOP-301/302R. Similar trends in the BOP301 and 302R results were observed and compared with ANL calculations.

The Institute of Physics and Power Engineering (IPPE, Rosatom) continued to develop the 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, and up to the ventilation system) has been developed and implemented into the code. This model has been integrated into the primary circuit 3D thermo-hydraulic model to simulate the transport of gaseous fission products from disintegrated fuel pins to the ventilation system, as well as potential releases into the environment.

The IPPE has performed integral analysis of the consequences of severe accidents in the BN-1200. The following codes have been used: 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). The IPPE has performed preliminary experiments with thermite compositions to obtain the melt of stainless steel with high temperatures. This technique will be used in a facility (which is currently being designed) to simulate transport of melted core in SFR conditions.

WP SO 2: Experimental programs and operational experiences

The CIAE conducted experimental research and code development for a heat transfer analysis of the China experimental fast reactor (CEFR) damaged spent-fuel assemblies in a closed space. The experiment simulated the spent-fuel assemblies during transportation, and the heat transfer characteristics were investigated.

Euratom contributed to the standard procedure for the sodium loop operation and measurement treatment. The deliverable presents the results of the selection of review elements for sodium technology. Aspects considered include procedures used for testing prototypical components at large facilities, procedures for calibration of sensors and signal treatment for the measuring system, methodologies for treatments of the measurement for subsequent use as input data for codes and the conservation of a facility in appropriate conditions enable restart in safe conditions.

WP SO 3: Studies of innovative design and safety systems

The JAEA attempted to identify accident sequences against severe accidents, referring to the rule of new regulations for LWRs. The JAEA carried out an internal event probabilistic risk assessment (PRA) in order to identify the accident sequences to be evaluated. Measures against the accident sequences (anticipated transient without scram [ATWS] and loss of heat removal system [LOHRS]) were studied and developed (see Figure SFR-7). Regarding external events, earthquakes and tsunamis were studied as the most prioritized initiating events. The accidents initiated with these events can be categorized into the accident category identified through internal event PRA (i.e. protected loss of heat sink [PLOHS]). By evaluating identified accident sequences, the JAEA can confirm that most of the SA (i.e. core damage and control valve failure) can be prevented.

The CEA has been studying the design of a small modular fast reactor. It has been considering constraints such as the plutonium content and maximal linear heat rate. The CEA has shown that criticality, which is an issue for small cores, could be achieved by adjusting the number and design of the assemblies. Further studies are planned (in 2021) to investigate other dimensioning parameters for a small modular sodium-cooled fast reactor (SMSFR).

Advanced fuels project

The AF project consists of three work packages (WPs): WP2.1 "SFR non-minor-actinide (MA)-bearing driver fuel evaluation, optimization and demonstration"; WP2.2 "MA-bearing transmutation fuel evaluation, optimization and demonstration"; and WP2.3 "high-burn-up fuel evaluation, optimization and demonstration".

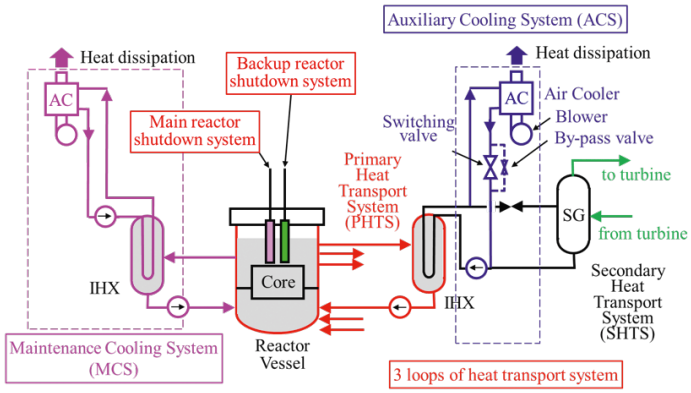


Figure SFR-7. MARS-LMR Modelling of the EBR-II BOP test

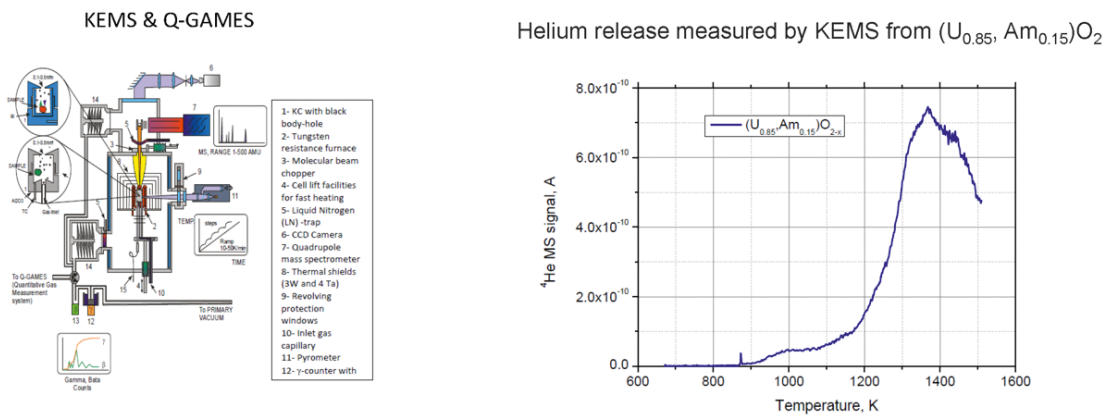
WP2.1: SFR non-MA-bearing driver fuel evaluation, optimization and demonstration

The CIAE is preparing to undertake some irradiation tests. It has finished the fabrication of dummy irradiation assemblies and out-of-pile hydraulic tests (hydraulic characteristic experiments). In 2020, more out-of-pile hydraulic and mechanical tests were conducted to ensure the future safety of in-pile irradiation assemblies.

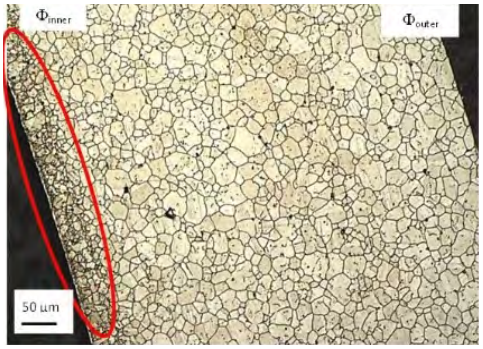
The CEA characterized a PAVIX-8 axially heterogeneous pin irradiated in the Phenix SFR at intermediate linear heat rate (LHR) to extend the validation basis of the GERMINAL V2 fuel performance code. Compared to high LHR irradiated fuels, major differences resulting from the lower fuel operating temperature have been observed. The GERMINAL V2 code underestimated the fuel swelling because this code does not consider gaseous swelling. The implementation of a new fuel swelling model in the GERMINAL code is underway.

The JAEA developed a plutonium and uranium mixed oxide powder adhesion prevention technology, applying nanoparticle coating on the acrylic panels

Figure SFR-8. Helium transport and release behavior in $(U,Am)O_2$



Cladding tubes made of 15-15Ti AIM1 steel



Grain size reduction on inner side of cladding tubes due to nitrogen contamination

Figure SFR-9. Characterization of the 15-15Ti AIM1 cladding tubes

of the glove box to minimize retention of nuclear fuel materials in glove box components and curtail the external exposure dose.

Rosatom manufactured three experimental nitride fuel assemblies for the BN-600 reactor.

WP2.2: MA-bearing transmutation fuel evaluation, optimization and demonstration

Euratom has also contributed to the work of MA-bearing oxide fuel performance evaluation. (U,Am) O_2 mixed dioxides are promising candidate fuels for the transmutation of americium (Am) in fast reactors in the heterogeneous recycling concept. One of the major differences in the irradiation performance of these fuels, compared to conventional MOX or uranium dioxide (UO_2), is their large He production, which can have a significant impact on safety-related phenomena, such as fuel swelling and pressure build-up inside the fuel pin. However, knowledge of its behavior in fuel is limited. Therefore, separate effect tests were performed on He generated in situ by alpha decay in (U,Am) O_2, and He introduced by ion-beam bombardment in $(U,La)O_2$ simulant materials. He transport and release mechanisms were then investigated by Knudsen Cell effusion mass spectrometry for both sample types, with complementary experiments on their microstructure evolution by transmission electron microscopy (TEM).

The JAEA is developing a simplified pelletizing process for MA-bearing MOX fuel fabrication. As part of this project, the JAEA evaluated performance of the granulation system in actual scale with simulated powder, which is composed of modernized a wet granulator, sizing machine, dryer and other auxiliary equipment.

KAERI developed metal fuel for the prototype Gen-IV sodium-cooled fast reactor (PGSFR). The fuel assembly was designed to satisfy requirements for the core performance and safety. The PGSFR fuel assembly consists of the handling socket, upper/lower reflector, hexagonal duct, fuel rods and nose piece. The structural characteristics and design features of PGSFR fuel assembly and its components have been described.

KAERI also analyzed the interaction between casting parts and U-10 wt.% Zr alloy containing rare-earth (RE) elements through the sessile drop test. Candidates for alternative crucibles and moulds was demonstrated using a casting of the U-Zr-RE alloy. Interaction behaviors and defects of the casting parts were evaluated after casting.

Rosatom developed technological processes for the manufacture of americium-burning elements within the framework of the "heterogeneous" scenario of the nuclear fuel cycle closing (NFC). Rosatom manufactured an experimental batch of mixed nitride of uranium and plutonium (MNUP) fuel samples through the method of high-voltage electric pulse consolidation and control of their characteristics.

WP 2.3: High-burn-up fuel evaluation, optimization and demonstration

The CIAE will conduct a program to do some CN-1515 and CN-FMS material irradiation tests in the CEFR in the coming years. The R&D and fabrication of CN-1515 and CN-FMS has been completed. The design of the irradiation rig and the fabrication of irradiation assemblies has also been finished. The mechanical properties of CN-1515 and CN-FMS have been tested and have been used to evaluate the irradiation assemblies.

The CEA has characterized the cladding tubes of 15-15Ti AIM1 from two different fabrication routes. The AIM1 is a titanium (Ti)-stabilized austenitic stainless steel treated as the reference cladding material for Phenix and ASTRID SFRs. In the results of the glow discharge mass spectrometry (GDMS), nitrogen contamination is observed on the inner surface of the cladding tubes of one of the two batches. The CEA carried out tensile property measurements on the two batches. Based on these results and past experience with AIM1 cladding irradiated in Phenix, the CEA concluded that a good behavior in pile can be foreseen for these AIM1 cladding tubes.

The JAEA carried out high- and ultra-high-temperature creep rupture tests, internally pressurized creep rupture and ring rupture tests, and temperature-transient-to-burst tests of 9Cr-ODS steel claddings. For comparison, the

transient burst strength of 11Cr-ferritic/martensitic steel (PNC-FMS) cladding was also evaluated. The obtained data was used to investigate the applicability of the life fraction rule to rupture life prediction of 9Cr-ODS steel and PNC-FMS claddings in various load-time temperature histories.

KAERI developed technology for a barrier cladding tube to suppress fuel-cladding chemical interaction (FCCI) for the use of MA-bearing metal fuel. Cr plating was applied at the inner surface of the cladding tube to achieve 20 μm thickness of Cr at the 500 mm length of HT9 cladding. Optimization of Cr plating to enhance layer property, such as pulse plating and surface treatment through nitriding process, has been reported.

Rosatom developed and manufactured BN-600 irradiation assemblies for testing fuel elements up to extreme parameters.

Component design and balance-of-plant project

ISI&R technologies

The CEA has studied the capability of the leaky Llmb waves in view of inspection from the outside of the main vessel. In 2020, the CEA conducted this experiment using devices that consisted of several austenitic steel plates immersed in water, an ultrasonic emitter and receiver. The experimental results with one plate were compared to simulation results.

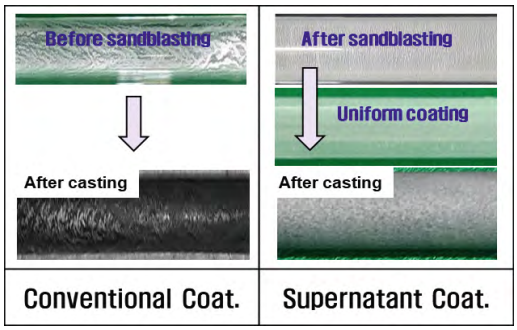


Figure SFR-10. Development of coating technology for the development of reusable mould

KAERI demonstrated the performance of the plate-type ultrasonic waveguide sensor in a sodium environment. KAERI fabricated under-sodium waveguide sensors and then conducted several under-sodium tests for viewing and ranging performance verification.

The JAEA performed the imaging test under-sodium viewer for medium distance in a sodium environment, and a performance test for long distance in actual plant configuration.

Supercritical CO_2 Brayton cycle

The CEA investigated the development of sensors to examine this heat exchanger. For this, eddy current probes were developed after defining the

Figure SFR-11. Under-sodium performance demonstration of the ultrasonic waveguide sensor

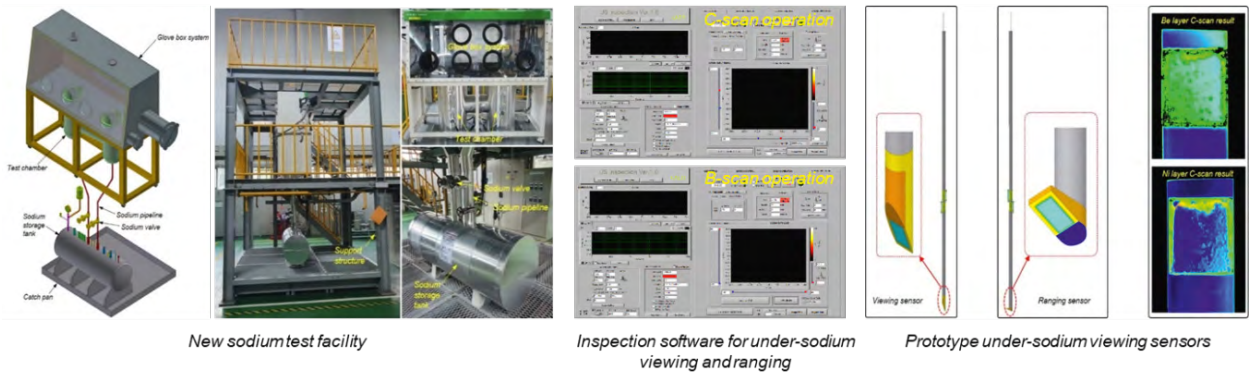


Figure SFR-12. Imaging experiments in sodium environment

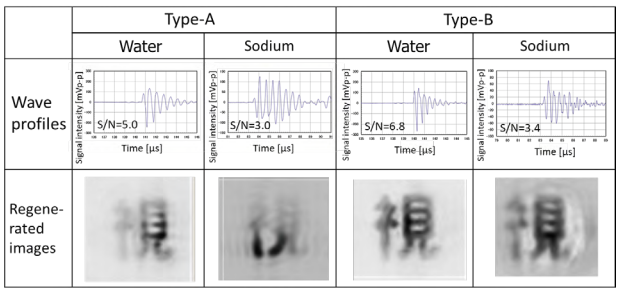


Figure SFR-13. Eddy current technique for NDT within small channels

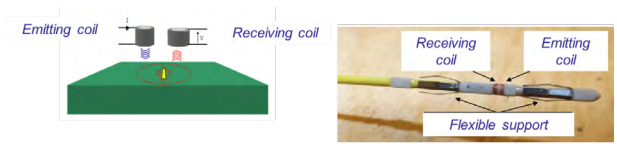


Figure SFR-14. Sodium-water reaction in a semi-open volume

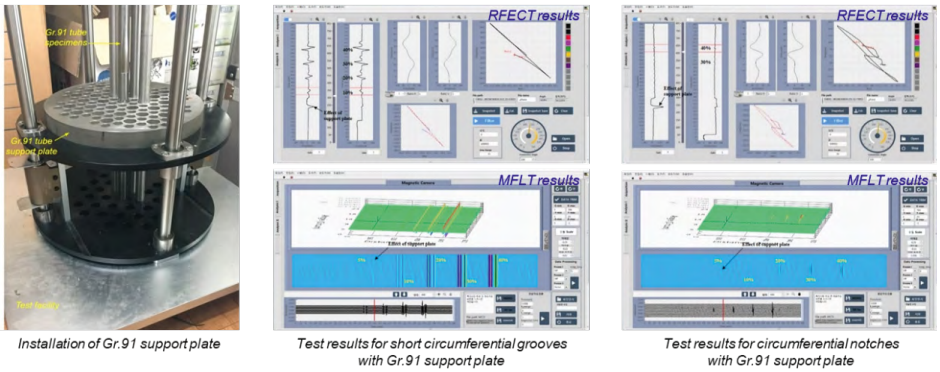


Figure SFR-15. Inspection system for a single-walled tube of a Rankine-type steam generator

specifications. A very small (3 mm by 3 mm, as an order of magnitude) probe was designed and manufactured.

The ANL prepared a report summarizing what has been learnt about sodium-CO_2 interactions and sodium-CO_2 reaction products based mainly upon data and results from the SNAKE sodium-CO_2 interaction experiments carried out at ANL.

Sodium leakages and consequences

No specific activity was conducted this year in this work package.

Steam generators

The CEA started a new activity in the field of the sodium-water reaction. It studied the sodium-water reaction (SWR) in specific conditions, such as open or semi-open volume (see Figure SFR-14). For this, after a review of the existing knowledge, the CEA defined and performed SWR in a dedicated facility (MININANET). The first experimental campaign results were then presented.

The JAEA performed the investigation on the applicability of mechanistic sodium-water reaction analysis code for steam generator performance and safety evaluation.

KAERI continued its performance demonstration of the upgraded prototype combined steam generator tube inspection system. More specifically, KAERI investigated the effects of the tube support plate and neighbouring tubes on measured remote field eddy current testing (RFECT) and magnetic flux leakage signals.

Sodium operation technology and new sodium testing facilities

KAERI finished constructing the sodium thermal-hydraulic integral effect test facility (STELLA-2), and completed its shake-down test and start-up operation with liquid sodium. A couple of sets of sodium integral effect test databases were collected and were used for some computational verification and validation (V&V) codes. KAERI continued the study of specific sodium technologies to get some useful measurements of process variables,

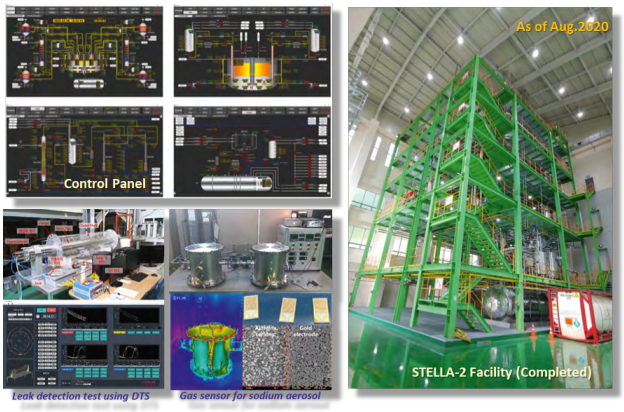


Figure SFR-16. Progress of the sodium integral effect test (STELLA-2) and development of advanced instrumentation techniques for liquid sodium applications

as well as for better operation of large-scale sodium facilities covering sodium feeding, draining, accident prevention, high-temperature operation and measurements.

The DOE conducted the design and construction of an intermediate-scale sodium test facility for the purpose of testing systems and components (mechanisms) in prototypical sodium environments. This facility consists of four experimental test vessels (of two sizes). Sodium has been fed to these test vessels from a main loop. The test vessels were designed to provide an independent testing environment, if isolated from the main loop. In addition, the test vessels were designed to allow for independent draining to the main dump tank without impacting the sodium environments in the other test vessels.

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