SAFIR2022 & KYT2022 Project abstracts

SAFIR2022 RG1 Overall safety and organization:

Building operational readiness of control room crews – BORS
The BORS project investigates operator work practices from the perspective of human-system interfaces, procedures, operators' resources for action and skills training. Our aim is to advance and deepen our understanding of resilience skills, operators' work practices and cognitive processes in complex incidents and severe accidents. To that aim, we conducted simulator tests that are unique in a sense that they are either performed in a new kind of environment (i.e., virtual control room) or that they address topics that have not been widely studied (e.g., stress management in complex incidents). The results suggest that some signs of operational resilience were identified in these studies supporting the view that operators are able to mobilize new work practices to promote task execution in demanding operational situations. Based on our results, methods, tools, guidance, and training activities can be developed for the promotion of operators' resilience skills that expand their resources for action in case of severe accidents and other demanding situations.

Development of framework for justification of overall safety – OSAFE
The general objective of the OSAFE project (2019-2022) is to advance (the understanding of) nuclear power plant safety and security, i.e., overall safety by applying a set of methods (risk-informed, graded approach, safety culture, defence-in-depth, institutional strength-in-depth, system modelling) and improving of these methods for the purposes of safety assessment and safety justifications e.g. in the context of operating plant's electric systems and the new technologies, such as SMRs. OSAFE project focused in 2020 on two different tasks: Small Modular Reactors (SMRs) from the perspective of design, and semantic modelling in the context of non-baseload operations. We will present here the research on SMRs.
The specific goal regarding SMRs was to study the safety systems of U.S. EPR and NuScale SMR design and compare these with each other based on the analysis of their Design Control Documents. The results of this assessment show that U.S. EPR implements more safety systems as compared to a simple and compact NuScale design. NuScale Power Module design implements an integrated primary circuit, which utilizes natural circulation to cool the reactor core. U.S. EPR design requires forced circulation with Main Coolant Pumps to create sufficient coolant flow to remove heat from the reactor core. Additionally, emergency heat removal systems are passive in NuScale design as opposed to more complex active heat removal systems in U.S. EPR design.

Participative development for supporting human factors in safety – PARSA
New ways to commit and motivate personnel, and to develop competence, work practices and new learning are necessary for safety in the nuclear domain. The change forces facing nuclear industry are aging personnel and technology, new technologies and new ways of organizing work at nuclear power plants. This further emphasizes the relevance of art and practice of human factors, which aims for participative development (PD) with human-centered frameworks and models.
PARSA project develops, applies and evaluates measures and tools to improve working practices, work process knowledge and mutual co-operation with shared situational awareness across organizational levels and units, aligning PD approach. Case studies are conducted in the context of nuclear maintenance.
PARSA uses video-based reflection and collaborative work process analysis as methods, and conducts a critical inquiry to nuclear specific human performance tools, with applicative targets. The aim is synergy among theoretical frameworks and practical implications, to have a shared view of needs, measures and prerequisites of PD in the nuclear industry, to improve safety.

Effective improvement of leadership and safety culture – EPIC

EPIC develops knowledge and approaches that support effective improvement of nuclear safety through leadership and safety culture. The project is divided into two work packages: methodical safety culture improvement, and characterization of leadership activities. The first work package systematically models how safety culture improvement activities are performed in nuclear power companies, focusing on the work of the safety culture experts. The second work package identifies good leadership practices within a selection of managerial or supervisory contexts at nuclear power companies.

RG2 Plant level analysis:

Co-simulation model for safety and reliability of electric systems in a flexible environment of NPP – COSI

Essentially, a co-simulation model has been developed that interfaces a Nuclear Power Plant (NPP) with its internal electrical network, which has already producing some interesting results, some of which have already been reported and published. A scaled down first version of the Finnish transmission network has been modelled that has interfaced successfully with a simplified version of the internal electrical network, and some preliminary simulation results have been attained. The main challenges are implementing changes in the Apros modelling of the NPP, including endowing it with a time resolution commensurate with that required to model faults in the power system and getting the full co-simulation model running, i.e., the Apros modelling of the NPP and the Matlab Simscape models of the full internal NPP electrical network and a sufficient model of the external Finnish transmission network.

The authors are mindful of the computational burden and lack of efficacy in over-developing the transmission model. The main goal of the third year of the project is to improve the existing model to be fit for purpose and utilize it to investigate events in the internal network and transmission network that are most likely to adversely impact the NPP. To this end a simulation matrix is being developed so that we can most efficiently harness the resources we have available in the third year of this project. Preliminary simulations produced in a report at the end of 2021 show promise, although it would be unwise to over-interpret these preliminary results.

New developments and applications of PRA – NAPRA

NAPRA aims at generating new knowledge about various topical issues in probabilistic risk assessment (PRA).

The Finnish seismic PRA practices have been compared with international ones.

A simulation-based event tree model has been developed for a cable room fire scenario; the model offers several advantages over a probabilistic fire model. The event tree includes an operation time model, which is used for gaining timing information of fire detection, fire brigade actions, and other human actions. The reliability of sprinkler systems has been studied concerning both available literature and a case study.

The PRA of long time windows has been considered from many viewpoints in the Nordic PROSAFE project. A Finnish pilot model of spent fuel pool, utilizing simulation-based event trees, has been developed and compared with other advanced PRA methods used by Swedish research partners; the resulting fuel damage frequencies are quite comparable. A human reliability analysis (HRA) model for the pilot that handles both recoveries and repairs has been constructed.

Human reliability in dynamic contexts has been studied through a literature review, operator interviews and a stakeholder survey.

PRA of systems involving digital subsystems has been studied together with five foreign research partners in an OECD/NEA WGRISK initiated benchmark study called DIGMAP. Each participant has modelled the same reactor protection system (RPS) with their own approach. The results demonstrate the importance of identification of common cause failure groups, and of the diversity (or lack of it) of the RPS.

A stakeholder survey concerning the methods, uses and applications of failure tolerance analysis (FTA) has been conducted. The survey clarified the present status of FTA in Finland, the role and purpose of FTA, and the connections between FTA and PRA. An FTA example analysis has been carried out, illustrating the FTA process and highlighting how some parts of FTA could be conducted.

Predicting extreme weather and sea level for nuclear power plant safety – PREDICT
The safety management over the life cycle of a nuclear power plant requires probability estimates of exceptional weather and sea level conditions in the current and future climate. We studied the occurrence of sea-effect coastal snowfall and high sea level as a single event or jointly with heavy precipitation. High wind speeds as an object of probabilistic weather forecasting were also considered. Atmospheric conditions favouring sea-effect snowfall were detected to occur most often in the western coast of Finland. Passing extratropical cyclones (low air pressure systems) were found to induce compound heavy rain and high sea level events. Synthetically generated low-pressure systems appeared as a promising tool for studies of extreme sea levels. Bayesian hierarchical modelling of sea level extremes reduced the range of uncertainty in their return level estimates. Probabilistic weather forecasts bring more confidence on decision making in case of nuclear emergency.

Safety and security assessment of overall I&C architectures – SEARCH
The objective of SEARCH (2019-2022) is to develop methods and tools for assessing technical design solutions related to overall instrumentation and control (I&C) system architectures. A particular focus is on trade-offs between requirements related to safety, dependability and security. We also aim to broaden the scope in which formal methods can effectively be used to verify properties on how the I&C systems operate as a whole.
In the first two project years, SEARCH has developed models and methods for early hybrid assessment of safety and security of overall I&C architectures. In addition, we have further developed methods of performing software model checking, accounting for hardware failures and communication delay. We have also developed a practical tool for explaining counterexamples

Uncertainty management in fire risk analyses – URAN
A pyrolysis modelling approach was expanded to full-scale simulation of a cable tray fire. Earlier experimental thermogravimetric analysis (TGA) and cone calorimeter results were used to construct the pyrolysis model. For model validation, data form the CFSS-1 experiment of the OECD/NEA PRISME 2 project was used.
A state-of-the-art methodology for the simulation of the cable cone calorimetry with FDS wass proposed in this study. Cable geometry was modelled with cubic obstructions using the new 3D heat transfer (HT3D) and 3D pyrolysis (PYRO3D) routines of the FDS.
Effects of ageing on the thermal decomposition process of polyethylene using united-atom MD simulations were studied. Qualitatively, ageing had a very little effect on the shape and position of the simulated TGA curves, suggesting essentially no effect of ageing on pyrolysis of polyethylene.

RG3 Reactor and fuel:

Coupled analysis of transient scenarios – CATS
System code TRACE and reactor analysis code PARCS were adopted into use, including transient analyses with coupled TRACE-PARCS and participation of two young experts in a PARCS training course. Master's thesis on sensitivity and uncertainty analysis of transient calculations was completed in 2019. In 2020, these developments were combined by adapting the VTT's uncertainty and sensitivity analysis tool Sensla for use with the newly adopted TRACE-PARCS.
A PWR core with a blocked fuel assembly was analysed with porous-CFD simulations. Due to the blockage,

coolant exceeded its saturation point in a small part of the core. Thus, handling conditions where steam bubbles form is necessary for modelling blocked assemblies and an important future research topic.

Interdisciplinary fuels and materials – INFLAME

The INFLAME project focuses on nuclear fuel behaviour during its irradiation in a nuclear reactor, including steady-state, transient and design basis accident conditions. The behaviour of nuclear fuel is studied both by means of modelling and experiments. In the first part of the project, the modelling items of fuel behavior in reactivity-initiated accident (RIA) conditions has included CABRI international program pre-test simulations, and an RIA sensitivity analysis with Sobol' variance decomposition method. Code development work has continued with the in-house fuel performance module FINIX.

In the experimental work packages of the project, both cladding and pellet experimental studies have been performed. On the part of cladding, cladding creep testing, autoclave testing and steam furnace testing capabilities have been developed and some of the material characterization and performance tests for conventional and accident tolerant cladding concepts done. On the part of the pellet, experimental capabilities at the VTT Centre for Nuclear Safety have been developed, and iodine release experiments performed for CsI doped CeO_2 pellets.

One DSc thesis was finalized in 2020 based on the fuel thermochemical behavior analyses made in this and the preceding SAFIR fuel projects, and one Master's thesis work is ongoing.

Developing the working arms of Kraken, the next generation computational framework for reactor design and licensing analyses – LONKERO

The aim of LONKERO a four year excellence project is to build a next generation computational framework for reactor analysis that can be used for deterministic safety analyses when the current generation of Finnish tools based on e.g. the TRAB3D and HEXTRAN sequences are eventually retired.

LONKERO started from individual new generation Finnish solver modules for reactor core physics, such as the Serpent Monte Carlo neutronics code, the Ants nodal neutronics code, the FINIX fuel behavior module and the Kharon porous medium thermal hydraulics solver.

In two years LONKERO has nurtured Kraken from these individual solvers into a respectable reactor core simulator capable of modelling multiple operating cycles of large light water reactors (LWRs) and small modular reactors (SMRs) while automatically evaluating licensing relevant data such as power peaking, control rod reactivity worths, reactivity feedback coefficients and shutdown margins.

Modularity and the internal cross validation of reduced order solvers with high fidelity have been two large design goals that have threaded their way through the first two years of Kraken's life. This has been demonstrated in SMR core level problems where the solution can be obtained both with continuous energy Monte Carlo and few-group nodal diffusion based calculation sequences.

The initial results for the two-operating-cycle experimental benchmark BEAVRS show that Kraken can predict detector maps for operating reactors at an accuracy comparable to current industry leaders. The project moves to transient modelling for its second half.

Radiation shielding and criticality safety analyses – RACSA

The scope of the RACSA project contains applications related to particle transport outside of the actual reactor core. More accurately, it covers photon transport for radiation shielding applications, reactor dosimetry and criticality safety. The focus with the photon transport has been on the validation process of the previously developed functionalities in the Serpent Monte Carlo code. The capability of Serpent to model complicated geometries, such as laboratory facilities, has also been further tested.

Reactor dosimetry capabilities are being continuously extended and improved. Validation has been a major part of dosimetry efforts too, as an ambitious verification and validation effort has been undertaken by calculating both computational and experimental benchmark problems in accordance with the guidelines of the U.S. NRC Regulatory Guide 1.190. A total of 2 calculational and 3 experimental problems were

modelled, yielding very satisfactory results. Modelling of these problems required copious amounts of coding work, in combination with the use of Serpent's built-in variance reduction techniques. Moreover, work was started on the development of a brand-new transport solver specially conceived for the needs of practical reactor dosimetry.
Considering the criticality safety, the long-time process to construct a validation package for fresh fuel criticality safety analyses has continued. In addition to increasing the number of modelled critical experiments, a similarity evaluation tool based on sensitivity and uncertainty analysis methodology has been implemented for Serpent. It should facilitate the selection of suitable cases for each validation run. The suitability of Serpent for burnup credit applications has also been improved.

Enhanced multi-physics calculation capabilities for fuel behaviour and reactor analyses – EMBER
The recently developed coupled neutronic and fuel performance calculation capabilities at LUT University are extended within EMBER to obtain a comprehensive, yet light-weight calculation framework for multi-physics reactor analyses. During the planned duration of three years, the project aims to deepen the level of the current coupling between neutronics, burnup and fuel behaviour, complete the framework via integration of a core thermal hydraulic solver and finally establish transient analysis capabilities. During the first year of the project, coupling between the Monte Carlo neutronics solver Serpent and the fuel behaviour solver TRANSURANUS was extended with the possibility to use nuclide concentrations calculated by Serpent in TRANSURANUS as part of the coupled calculation. Necessary modifications and additions were implemented to the coupled calculation driver program Kytkin and a new nuclide data interface was developed and implemented in TRANSURANUS to allow reading-in external nuclide information. The functioning of the nuclide interface was verified with a test calculation.

Fuel microstructure and radium solubility – PORA
In this project, the scope of the microstructure modeling is to gain information of the changes in the microstructural cracking evolution in the fuel pellet caused by different outer dimensions of the pellet and estimate the implications on macroscopic behavior and modeling. Especially different radius in a cylindrical fuel pellet changes potentially the fuel centre line temperature and temperature gradient across the pellet altering the microstructural cracking evolution in the pellet. The project concentrates on topics common to both the SAFIR and KYT research programmes.

RG4 Thermal hydraulics:

CFD methods for reactor safety assessment - CFD4RSA
The overall objective of the CFD4RSA project is to improve the usability and reliability of Computational Fluid Dynamics (CFD) calculations in nuclear Reactor Safety Assessment. The work consists of four Work Packages, where topics that are important in safety assessment are studied. First, the reliability of coupled CFD-Apros calculations is investigated in the modelling of postulated accident scenarios. Second, uncertainty quantification methods in CFD calculations are tested and taken into use. Third, coarse-mesh CFD models are developed for reactor pressure vessel. Fourth, thermal stratification in pressure suppression pools of BWRs is studied.

Passive heat exchanger experiments – PAHE
The objective of the project was to improve understanding of the AES-2006 design PHRS-C passive heat removal system and to generate data for code validation. Carefully designed experiments are the most reliable way to obtain fundamental understanding and reliable data of the phenomena. Data can be used in the development and validation of system and CFD codes for the safety analyses of nuclear power plants.

PWR PACTEL tests – PATE

The objective of the project is to improve the understanding of thermal hydraulic system behavior of EPR type PWRs by performing integral effects tests with PWR PACTEL (Kouhia et al., 2019). Carefully designed experiments are the most reliable way to obtain fundamental understanding and reliable data of the phenomena. Data can be used in the development and validation of computer codes for the safety analyses of nuclear power plants. Performing experiments not only requires the hardware and programs controlling the devices and gathering data, but also the knowledge of the system behavior. Computer analyses are needed in the planning of the experiments as well as in post analyses to help understanding the physics in the experiments.

Sparger separate effect tests – SPASET

The SPASET project has increased knowledge of small-scale phenomena affecting the effective heat and momentum sources during steam injection through spargers. The effect of pool water sub-cooling, a transition from sub-sonic to sonic flow conditions, injection nozzle chamfer, and multi-hole injection on the effective momentum have been studied. The experiment results have been utilized in the development and validation of the simplified effective heat source (EHS) and effective momentum source (EMS) models proposed by KTH. They are also being used at LUT and VTT in the improvement of the computational fluid dynamics (CFD) models related to direct contact condensation (DCC).

Safety through thermal-hydraulic analyses and cooperation – THACO

The first two years of THACO project have been heavily influenced by the OECD/NEA's new Rod Bundle Heat Transfer (RBHT) project. In 2019, as the new experimental data was not yet released, the work on this subject started by calculation of older RBHT experiments with both Apros and TRACE. In 2020, the new RBHT open-phase experiment data was available and it focused on the impact of rod peak power, inlet subcooling and most notable, different reflooding rates and types such as oscillatory reflooding. These cases were also calculated with Apros and TRACE. Machine learning and BEPU analysis were used in the Apros analysis and the model was made ready for the 2021 blind phase code benchmark.
Additionally, PASI (LUT), PKL and FIX-II facility experiments have been calculated within the first two years. International cooperation in various OECD/NEA's and U.S. NRC's programs is heavily represented in the project.

RG5 Mechanical integrity:

Advanced materials characterisation for structural integrity assessment – AMOS

Advanced materials characterisation is required for fracture mechanical assessment of reactor pressure vessel and other safety class 1 components. One practical example is surveillance programs and the extension of the programs to cover modern requirements and long-term operation. In AMOS, these challenges are investigated through developing miniature testing techniques for C(T) specimens and arrest toughness, techniques for fracture mechanical testing during thermal transients and methods to account for the transferability. During 2019-2020, we carried out experimental testing and numerical analysis to develop quality criteria for miniature C(T) specimens, developed a new method to investigate the effect of thermal transients on fracture toughness in the ductile regime and reviewed the LTO performance of RPVs. During the upcoming years, the results will be post processed, in addition to performing more measurements to better understand the uncertainties and the connection to the structural integrity of the reactor pressure vessel.

Effect of long-term operation on aging and environmentally assisted cracking of nuclear power plant component materials – ELIAS

During the first two years of the ELIAS project the main focus has been on reactor pressure vessel embrittlement mechanisms (2019) and reactor pressure vessel repair welding techniques and approaches (2019 – 2020). The reactor pressure vessel repair welding research work has been performed in close collaboration with the FEVAS Project.

Extended lifetime of structural materials through improved water chemistry – ELMO
Water chemistry applied in primary and secondary circuits have significant effect to the lifetime of the NPP components. Several water chemistry related research topics have been identified and included in the ELMO project. Hydrazine replacement study, enrichment of impurities and corrosion products in steam generators, lead assisted stress corrosion cracking of steam generator tubing and surveying on the existing information on SMR water chemistries have been included in the project. Useful results related on alternative scavenging chemicals for hydrazine have been obtained, as it was shown that there are sufficiently efficient scavenging chemicals available to be used during power operation. First estimates on the deposition and release of impurities on a simulated steam generator tube could be calculated from the unique experimental setup developed at VTT.

Fatigue and evolving assessment of integrity – FEVAS
The project FEVAS focuses on the structural integrity aspects of the pressure vessel and primary circuit with special emphasis on developing novel models, tools and techniques that help to ensure safe plant operation over the whole plant lifetime. The project has focused on the development of a F_{en} model to take into account the effect of the environment and temperature on the fatigue endurance of the primary circuit, conducted a study of surface crack behavior under cyclic thermal loads followed development of experimental and numerical techniques to capture it and developed techniques needed for the assessment of pressure-pulsation induced piping vibrations. Additionally, the project concentrated on the development of surface crack repair welding techniques through mock-up manufacturing, its material characterization and numerical simulations of the welding process.

Non-destructive examination of NPP primary circuit components and reliability of inspection – RACOON
Non-destructive testing is vitally important in ensuring the safe and economic operation of aging nuclear power plants and other industrial systems. Reliable detection of service induced damage as early as possible allows for improved safety and effective maintenance. At the same time, finding flaws early pushes the NDT methods to the limits of detectability. Thus, effective estimation of NDE reliability is vital to successful application of NDE. While much progress has been made with NDE qualification, quantitative estimation of NDE reliability (namely, probability of detection, POD) is still not commonly used in the nuclear industry. The inspections vary, and providing sufficient empirical data for POD has remained infeasible.
In the RACOON project, inspection reliability and POD is studied with new tools (eFlaw's) that allow better empirical data gathering. Based on previous work, in the current project a first of a kind virtual round robin (VRR) with world-wide participation was conducted in collaboration with an international project. The target was the ultrasonic inspection of dissimilar metal welds (DMWs). The VRR demonstrated the capability and efficiency of the approach to determine NDT reliability and performance.
In addition to theoretical reliability, the consistency and repeatability of inspections are crucial to actual field reliability. In a short project ANDIE, now merged to the RACOON project, the application of machine learning to automated data analysis and defect detection was studied. Machine learning (ML) powered ultrasonic inspection has proven highly feasible approach to increase reliability, repeatability and efficiency of mechanized ultrasonic inspection. Originally short project ANDIE demonstrated the capability to use virtual flaws for training the ML models. In year 2020 the effect of different flaw types and sizes used in training the ML model was studied to have impact on detectability.

Fatigue Management for LTO – FATIMA
The general objective of FATIMA is to go beyond existing state-of-the-art in fatigue management for long-

term operation (LTO), accounting for environmental effects on fatigue life. As an alternative to existing practices, improved methods can be adopted through scientifically solid justification and international acceptance, which is the long-term goal of the project.
During the first year of FATIMA, international codes, standards and practices were reviewed for an up-to-date understanding of the state-of-the-art. International activities were started by participation in a collaboration group on environmentally-assisted fatigue issues. Experimental fatigue research was initiated in small scale during the first project year and will be scaled up during following years with material batches representative of Finnish nuclear power plants.

RG6 Structures and materials:

Additive manufacturing in nuclear power plants - AM-NPP
The general objective of AM-NPP is to increase the knowledge of finnish stakeholders on the use of Addtive Manufacturing (AM), in particular Laser Powder Bed Fusion (L-PBF), therefore ensuring the safe use of additively manufactured metallic components in the nuclear sector. It is a technology which is showing a lot of applicability potential, e.g. in dealing with obsolescence in Nuclear Power Plants, but for which there is still little exposure to for both licensees and regulator. AM-NPP aims at closing this gap. The work developed during the different work packages aims at: Expanding standard procedures and deepen understanding of material-process-property relationships; contributing with scientific based facts to the introduction of AM in nuclear design codes; and, identifying safe ways of replacing obsolete components and realize new designs using AM. During the first two years of the project the focus has been on creating a roadmap of the use of AM in Finnish nuclear sector. Also, applicability of AM components and methods of quality control have been studied.

Critical studies in support of the ageing management of NPP concrete infrastructure – CONAGE
The research topics being addressed in the CONAGE project are strongly linked by key aspects of ageing management, where critical input is needed to support decision-based actions, whether related to inspection and maintenance actions on existing concrete infrastructure, or the design of new concrete infrastructures.
The research topics are divided into three work packages, addressing i) the non-destructive evaluation of NPP concrete infrastructure (WP1); ii) the risk of internal expansive reactions for NPP concrete infrastructure (WP2); and iii) steel liner and anchor corrosion in contact with concrete (WP3).
In the following sections, the research undertaken during the first two years of the project in each of the three work packages is briefly presented.

Modelling of aged reinforced concrete structures for design extension conditions – CONFIT
Two different types of constitutive material models have been developed during the two first years of this project. The first one is Concrete Damaged Plasticity (CDP) model that has been available in Abaqus Finite Element (FE) code and used at VTT for many years. It was further extended and calibrated in the project in 2019. This user-extended CDP model is especially suitable for hard impact simulations. The second one is so-called tensorial damage model that captures the effect of crack initiation and growth on the continuum response. Beside the stress-state, the mechanical response of cracked material depends strongly on the opening and closure of micro-cracks, called unilateral effect. The effect is of importance in the vibration problems. This is the only material model type that describes damage-induced anisotropy and has not been available in any FE code. It was further developed and implemented to Abaqus during 2020. The user-extended CDP model has now been partly validated against different types of material tests. The same calibration and validation process will be conducted for the other model in the future.
The implications of material ageing for structural performance assessment and especially for numerical

modelling of reinforced concrete structures has been studied. This information has been summarized, focusing on stressors, degradation mechanisms, potential failure modes, and in-service inspection methods. Calculation models have been collected for the estimation of the ageing and deterioration effects on the key mechanical properties. These provide inputs for mechanical properties in structural models that take into account the ageing and deterioration of concrete in NPP structures

Safety criteria and improved ageing management research for polymer components exposed to thermal-radiative environments – SAMPO
In SAMPO project several research topics related to polymeric components used inside NPP containments are included. The project contains three different work packages, which comprise of task with similar topics. The first work package is related to acceptance criterion and safety margin assessment. Here, work is focused on providing improved estimation for lifetimes of critical polymer components, studying the sensitivity of polymer properties to additive content and methods to verify polymer quality and setting up safety margins for O-rings. The results of this WP indicate the importance on setting acceptance criterion based on the functional property. Also improvements in polymer quality management is developed as techniques for on-site quality analysis are provided. The second work package focuses on improving ageing management of polymer components. Here within important research topics are online condition monitoring techniques as well as sensitive analysis methods and improving the interpretation of non-destructive testing data. Promising results have been obtained for the use of online monitoring methods such as microcalorimetry and permittivity measurements. Further development have been made in developing non-destructive cable condition monitoring methods and analysing the ageing of such cables by the means of molecular dynamics simulations. The third work package is solely on international cooperation, improving the knowledge transfer between the research community and industry and following international trends on polymer based components used in NPPs.

RG7 Severe accidents:

Analytical severe accident research – ANSA
In the frame of this project, national competence in the area of severe accidents has been further improved and the tools and methods in use have been validated in their intended purposes.

Third versions of VTT's MELCOR models for the Fukushima unit 2 and unit 3 accidents were developed. The reactor water level measurement systems were added to the models to have more information on the progress of the accidents. Measured pressures are reproduced well.
Models were developed to assess the hydrogen behaviour in the containment on more profound level. Especially was focused on developing a model for the combustion of ultra-lean hydrogen mixtures as those are more probable in the containment than richer ones. The computational results are mainly in a good agreement with experimental results.
To have an in-depth understanding of the phenomena affecting heat transfer in a crucible type core catcher, a CFD model was developed to simulate heat transfer in an externally cooled, homogeneous molten corium pool. The results agreed well with SIMECO-2 pre-test analysis conducted by Li (2016).
Pool scrubbing experiments for CsI aerosol and CH_3I considering the effect of NaOH and $Na_2S_2O_3$ were analysed with ASTEC. For CsI aerosols, ASTEC results seem to be very sensitive to particle size and the effect is larger on globular regime. At jet regime, the behaviour of analytical DF results is reverse to experimental results when considering the effect of inlet flow rate.
An international code benchmark was participated to compare VTT's in-house codes VALMA and ARANO with well-known international code packages for off-site dispersion and dose assessment. The results were well comparable wih the results of other participants.

Mitigation and analysis of fission products transport – MANTRA

The aim in the MANTRA project during 2019-2022 is to investigate the transport and chemistry of gaseous and particulate fission products in severe accident conditions. In 2019-2020, the main focus was on the behaviour of iodine, caesium and tellurium, which are highly radiotoxic and the mitigation of their possible source term is of utmost importance. It was observed that the fission product deposits on the reactor coolant system (RCS) surfaces act as an important source of gaseous iodine, which can enhance the iodine source term. The tellurium transport was related to the oxidizing or reducing conditions in the RCS. The containment spray system was efficient in removing the airborne tellurium species. The pool scrubbing of fission product aerosols was notable and the decontamination factor increased significantly in the jet regime (high flow rates). Further actions to consider the long-term severe accident management issues were also taken. The follow-up of OECD/NEA STEM-2, BIP-3 and ESTER projects was carried out.

RG8 Research infrastructure:

Barsebäck reactor pressure vessel material used for true evaluation of embrittlement – BRUTE

The objective of the BRUTE Excellence SAFIR2022 project is twofold, i.e., to perform mechanical and microstructural investigations of Barsebäck 2 boiling water reactor (BWR) reactor pressure vessel (RPV) materials in irradiated and thermally aged conditions. These results are used for determination of the compatibility of the surveillance program, used to assess the influence of ageing and assessment of the structural integrity of the RPVs, with the results of the real RPV material.

The other main objective is to pioneer the new infrastructure in the Centre for Nuclear Safety, CNS, VTT. Building a new national infrastructure is a huge undertaking, and has been supported by several SAFIR and VTT projects. BRUTE is taking the CNS infrastructure into final use, and lessons learned are used also in future projects.

The pioneering and validation work has resulted in accreditation of all mechanical test methods at CNS, after overcoming smaller and bigger challenges. The material investigations have resulted in high-quality results on mechanical properties and microstructure, improving the understanding of irradiation induced embrittlement and factors affecting brittle fracture initiation.

Infrastructure development at LUT safety research laboratory – IDEAL

The general objective of the IDEAL project (2019–2022) is to develop the experimental thermal hydraulic infrastructure at LUT University nuclear safety research laboratory. The project comprises maintenance of the existing thermal hydraulic test facilities, development and upgrade of the instrumentation and data acquisition capabilities, as well as development of the new modular integral test facility, MOTEL. The motivation for the project is in providing state-of-the-art experimental thermal hydraulic capabilities, which benefit the whole Finnish nuclear community. The maintained and upgraded facilities are used in other important SAFIR-, industrial- and EU-funded safety research projects. Enhancement of the measurement capabilities enables production of high-quality data for understanding the thermal hydraulic phenomena and for the development and validation of computational tools, such as CFD codes. Procurement and implementation of novel measurement techniques promote the growth of expertise on the field of experimental thermal hydraulics and offer topics for master's and bachelor's theses. Maintenance of the facilities and expertise enables rapid solution of problems that arise in the Finnish nuclear power plants. Upgrade of the process control, computational and data storage systems enables better handling and filing of the large amounts of experimental data.

Participation in the Jules Horowitz Reactor Project - JHR2022

During the first two years of the JHR2022 project, the main focus has been on participation in the Jules Horowitz Reactor Project through the three working groups (fuel, materials and technology), upgrading

MeLoDIE devices (mechanical loading device for irradiation experiments) and the participation in related international collaboration and research program, including the OECD/NEA Framework FIDES Program, the Horizon2020 coordination and support action JHOP2040 and a secondee exchange at CEA-Cadarache.

Pre-emptive reduction of radiological laboratory legacy waste – LABWAST

The infrastructure renewal of the radiological research infrastructure hosted by VTT is embodied in the new VTT Centre for Nuclear Safety. Initiated a decade ago, the renewal has proceeded well, and valuable assets have been built-up to enable high-level nuclear safety research in Finland. The renewal work reached a crescendo in the SAFIR and KYT 2018 programmes, where the focus was on constructing, equipping and licensing the new facility. The LABWAST project in the first year of the current SAFIR and KYT programs focused on taking the new facilities into full operation, with commissioning and ramp-up of operations for both reactor safety and final repository research. The LABWAST project in particular already looked to the future and eventual decommissioning of the facilities, with a particular focus on efforts to better utilize existing research materials, and developing effective means of handling the radioactive waste generated over the long term during the operation of the facilities. Simultaneously, the investment aid project RADINFRA carried out the last remaining procurements over the 2019-2020 period, mainly related to supporting facilities for handling, storage, and transport of specimens and waste.

KYT2022 SR1 Buffer / rock interface:

BROCTIO - Bentonite-Rock Interaction (Bentoniitti-kallio –vuorovaikutus)

The project investigates experimentally clay components of the KBS-3 disposal system (bentonite buffer and tunnel backfill) and the surrounding bedrock. Particular attention is paid to the interface between bentonite and bedrock. The research topics are 1) bentonite-rock interface transport phenomena, 2) the effect of bedrock mechanical and thermal behavior of bentonite; and 3) rock fractures and materials characterization. The result of the study is detailed and comprehensive information that enhances the understanding of the phenomena involved and that can be used for development and parameterization of models.

KYT2022 SR2 Canister:

KAPSELI - Canister performance assessment

The KAPSELI project aims to study the performance of the canister through various mechanisms and interactions. The different sub-projects of the research project are briefly described below.

KAPSELI/BECOLT - Behaviour of Copper under Load Transients

The project is focused on the effects of the stresses experienced by the canister in the initial phase of disposal, when the load experienced by the capsule changes and the effects on the creep mechanisms. The effect of grain size on creep is also studied experimentally. The interaction between creep and corrosion is tested. Methods for detecting copper creep damage at the earliest possible stage when grain boundary cavities and other defects are very small will be developed.

The results and models of load history dependence, relaxation, and the effect of grain size are combined, resulting in a more reliable estimate of the stresses and deformations experienced by the capsule. The results also provide a basis for the authorities to assess the safety criteria.

KAPSELI/MECAN - Mechanical strength of the copper canister and its cast iron interior (Kuparikapselin ja sen valurautaisen sisäosan mekaaninen lujuus)

The aim of the project is to investigate the mechanical properties of different parts of the copper canister and to understand the macroscopic and microscopic plastic deformation in its inhomogeneous structures. In this way, deformation and possible rupture of the capsule in long-term use can be predicted and its behavior can be modeled reliably. The project also studies the absorption of hydrogen and its effect on the mechanical properties of copper, as well as the mechanism of stress corrosion of copper under both oxidizing and reducing conditions.

The research of the properties of the cast iron inner part are also included and the fragility mechanisms related to the properties of the cast iron will be elucidated. The result is the effects of capsule discontinuities (defects and geometric discontinuities) and inhomogeneous microstructure on the localization and fracture of the deformation. The study is very important in defining the quality requirements for the manufacture of the capsule, in assessing the criticality of the discontinuities (tolerances and defect sizes and their acceptance criteria), and in particular in the safety analysis of the canister.

KAPSELI/CRYCO - Validated advanced modelling and prediction of long term deformation and damage of copper

The aim of the project is to create an accurate, reliable, physically justified and validated method for assessing the structural integrity and copper material behavior of a disposal canister. The expected result is an accurate, reliable, physically validated and validated method based on crystal plastic modeling of the structural behavior of the disposal canister, research results on the mechanical behavior of the copper material, and advanced material characterization methods utilizing machine learning. The results are useful for assessment of the behavior of the canister under disposal conditions or the assessment of the design of the disposal canister.

KAPSELI/OXCOR - The effect of oxide layer on copper corrosion in repository conditions

The aim of the project is to determine the effect of oxide layers formed on the surface of copper corrosion rate in synthetic bentonite pore water and groundwater. Oxide layer is formed when the capsule is in intermediate storage after refueling and the surface of the capsule heats up. The oxide layer can accelerate general corrosion and cause localized corrosion. The research is a continuation of the KYT2018 program's REPCOR project. The project presented will do long-term experiments to confirm the results of the REPCOR project.

KAPSELI/SUCCESS - Sulphide induced stress corrosion in copper (Sulfidin aiheuttama jännityskorroosio kuparissa)

The aim of the project is to investigate experimentally and through modeling whether copper is prone to stress corrosion in sulphide containing groundwater. The study applies a test method that has been shown to be able to predict stress corrosion susceptibility with stainless steels. Preliminary experiments have shown that the method is also suitable for copper (CuOFP) under oxidizing conditions. The project will give information on whether CuOFP is prone to stress corrosion in sulphide-containing groundwater. The results also provide information on whether the copper is passivated sufficiently in a sulfide-containing environment to cause stress corrosion at all possible.

KYT2022 SR3 Release barrier interactions and microbiology:

MoToPro - Multibarrier System Performance - Microbiological and Chemical Processes (Moniestejärjestelmän toimintakyky - Mikrobiologiset ja kemialliset prosessit)

The goal of the coordinated MoToPro project is to gain new knowledge about the effect of microbes the performance of a multi-barrier system in the geological disposal of high-level nuclear waste In Finnish conditions. The different sub-projects of the research project are described below.

MoToPro/VaVu - Interactions of the release barriers (Vapautumisesteiden vuorovaikutukset)

The aim of the project is to launch a long duration experiment related to the multibarrier system and the interactions between its components in co-operation with experts from the other MoToPro subprojects. The subproject also includes the coordination of the entire MoToPro project.

MoToPro/KUKO - Interactions of the release barriers and their impact on copper canister corrosion (Vapautumisesteiden vuorovaikutukset ja niiden merkitys kuparikapselin korroosioon)

The aim of the project is to evaluate the effect of changes in release barriers on copper corrosion and the possible impact of copper corrosion on the performance of other release barriers. Copper corrosion is studied under disposal conditions, taking into account the entire disposal period. Exposure tests are performed at relevant temperatures and in the presence of different microbial groups and under the same conditions without microbes.

Based on the results, it is possible to analyze the corrosion mechanisms and corrosion rate of copper in Finnish disposal conditions. Information is also obtained on the properties and compositions of the corrosion product layers formed under different conditions. The project will develop new methods for monitoring microbiological activity and its impacts on corrosion.

MoToPro/MiBe - Microbial impacts on bentonite (Mikrobien vaikutukset bentoniitissa)

The aim of the project is to determine whether the activity of microbes poses a risk to the structure and performance of bentonite in "worst case" conditions. It has been previously observed that bentonite has microbial activity and the activity has effects on bentonite, but the significance of the effects remained unclear and this issue will be further studied in the project.

The MiBe subproject focuses on microbiological and chemical changes in bentonite in a multibarrier system. The MiBe project will provide information on whether microbes may pose risks to bentonite performance that should be further investigated, taking into account the long time spans of nuclear waste management.

MoToPro/BIKES - Biogeochemical scenarios (Biogeokemialliset skenaariot)

The aim of the project is to describe the significant biogeochemical reactions and processes involved in the transport and retention of substances in rock groundwater using three scenarios, the sulphate scenario, the hydrogen scenario and the salinity scenario. In addition, the aim is to examine the thermodynamics of biogeochemical reactions in natural analogy sites and in a laboratory experiment, and to compare the results with those obtained by microbiological methods.

The project includes biogeochemical sampling of deep boreholes, thermodynamic modeling and mineralogical and isotope research. The results of the project can be used to assess the advantages and disadvantages of biogeochemical processes for the long-term safety of nuclear waste disposal and the applicability of different methods (microbiological, geochemical, thermodynamic) to determine the parameters required for safety justification.

MoToPro/MIMOSA - Diverse metabolic pathways of microbial communities in deep pressurized bedrock (Mikrobiyhteisöjen monimuotoiset aineenvaihduntareitit syvässä paineellisessa kallioperässä)

The aim of the subproject is to find out site-specific differences in metabolic strategies, metabolic activity and type of the Finnish deep biosphere in extreme. The project will develop equipment for pressure sampling and maintaining the pressure of the sample during sampling, as well as a high-pressure growth method. The aim is to study the metabolism of deep biosphere microbes in situ under pressure, because many metabolic reactions of microbes adapted to high pressure are only possible at high pressure.

The project uses the latest power sequencing methods. The result is information on the metabolic processes of extreme microbial communities and new research methods for pressure sampling and microbial culture, which can also be used to examine in more detail the effect of certain conditions (e.g. nutrients) on microbial growth.

KYT2022 SR4 Safety factors:

SYSMET - Systematic Scenario Methods in Overall Safety Estimation (Systemaattiset skenaariomenetelmät kokonaisturvallisuuden arvioinnissa)

Research in SYSMET contributes to the overall safety assessment by providing insightful information on how to scenario methods have been applied in various fields to identify and analyze uncertainties. The SYSMET project conceptualizes systematic reference frameworks and procedures to support the overall safety assessment of a nuclear waste facility and to develop decision analytical scenario selection methods.

OMT - Overall safety, multibarrier system and transient phase

The overall safety of nuclear waste is a broad and interdisciplinary research package, which also includes cooperation between research institutes, authorities and nuclear waste management actors. At present, there is no general approach to assessing long-term safety, quality and operational safety and the performance of organizations in terms of overall safety. The goal of the OMT project was to initiate a framework for overall safety assessment, starting with the concept of a state-of-the-art safety case that benefits all parties involved in nuclear waste management. In the first year of the project, the main focus was on understanding the overall security concept and communicating the perspectives of experts in different fields. The project included interviews and a seminar to identify more in-depth research needs.

KYT2022 SR5 Host rock studies:

MIRA-3D - 3D Modeling of microstructures (Mikrorakenteiden 3-D –mallinnus)

The aim of the MIRA-3D project is to identify and characterize fissure systems on a cm scale. The project utilizes the new “3D-grinder” equipment developed at the University of Turku, in which accurate 3D models can be made from samples with a maximum size of 50x60 mm. With the help of the method, it is possible to examine the micro level and to produce reference material in relation to KARIKKO project.

The project provides information on the 3D geometry and properties of fracturing at the micro level and evaluates the relationship between transplants and associated secondary fracturing and the kinematics of transplants, evaluates lithology properties at the micro level and the orientation of the rock and the relationship between orientation and the stress field. The aim is to compare the findings with the materials produced in the KARIKKO project in assessing possible scalability. The method can also potentially be utilized in modeling the small-scale migration properties of samples.

KARIKKO - Bedrock Fractures (Kallioperän Rikkonaisuus)

The aim of the project is to evaluate the properties and quality of bedrock cracking at different scales in the region of Southern Finland using large-scale lineament interpretations based on geophysical and topographic data, orthophotos and 3D models produced by drone photogrammetry and crack maps based on them. To enable the collection of large amounts of data from digital materials, the project will explore and develop semi-automated / automated methods for collecting and analyzing slit or lineament data using either machine learning or algorithms. By increasing and analyzing the crack data, it is possible to reduce the uncertainties currently associated with 3D modeling of brittle structures and gap network modeling. The project also evaluates the utilization of other geological information and interpretation, as well as the use of stress state modeling in predicting fracture properties.

KARIKKO produces material on the basis of which it is possible to compare the cracking of Olkiluoto and the created crack network models with the wider regional geological picture and properties of cracking.

RAKKA - Water Conductivity of Fractured Rock (Rakoilleen kalliomassan vedenjohtavuus)

The primary goal of the RAKKA research project is to determine the flow properties of the cracks through photogrammetry, mechanical numerical modeling and coupled hydromechanical modeling. The results of the project will be validated through large-scale laboratory experiments.

KYT2022 SR6 Other safety research:

RASK - In-situ Experiments - Radionuclide Transport on Cement and Rock Interface (In-situ tutkimukset - radionuklidien kulkeutuminen sementin ja kallioperän rajapinnalla)

The aim of the project is to determine the retention and transport of radionuclides in cementitious materials, crystalline rock and especially at the interface between the two by in situ experiments. This is achieved by studying the structural and mineralogical change in these materials caused by alkaline cement water. The work also investigates the behavior of radium in the multi-barrier system. The aim is to produce concrete baseline data for long-term safety analysis.

RABIO - Better Radioecology for Biosphere Modeling (Parempaa radioekologiaa biosfäärimallintamiseen)

The objective of the project is to refine the radioecological modeling applicable to Finnish (and other boreal) aquatic ecosystems and its use to assess the potential risks of nuclear waste management. The study is expected to produce more sophisticated, empirically advanced biosphere modeling to assess the risks of spent fuel disposal and other radioactive waste. In addition, the results can be used to assess the environmental impact of uranium ore exploration and mining of uranium or uranium-bearing ores that may be established in Finland, to develop radioecological models in general and to assess the risks of other contaminants (e.g. heavy metals).

PORA - Fuel microstructure and radium solubility
The PORA project is a joint KYT2022 and SAFIR2022 project. In this project, the scope of the microstructure modeling is to gain information of the changes in the microstructural cracking evolution in the fuel pellet caused by different outer dimensions of the pellet and estimate the implications on macroscopic behavior and modeling. Especially different radius in a cylindrical fuel pellet changes potentially the fuel centre line temperature and temperature gradient across the pellet altering the microstructural cracking evolution in the pellet.

KÄRÄHDE - Spent Fuel Characterization and Source Term (Käytetyn polttoaineen karakterisaatio ja lähdetermi)

The aim of the project is to identify how the uncertainties affecting the computational characterization of spent nuclear fuel studied in this project affect the spent fuel source term (heat generation, nuclide inventory and other essential properties) and to write instructions on how the spent fuel source term should be determined in Serpent taking these uncertainties into account.

KYT2022 SR7 Low and intermediate waste management:

SURFACE - Near Surface Repositories

The aim of the project is to find out the suitability, conditions and requirements of soil / surface placement for Finnish conditions. The work covers the design criteria, waste types suitable for soil disposal, disposal concepts, environmental conditions in Finland, technical emission barriers, monitoring and the principles of safety analysis as well as conduct laboratory studies on the waste and emission barriers.

TERKOR - Corrosion of low and intermediate level steel wastes under in-situ disposal conditions (Teräksisten matala- ja keskiaktiivisten jätteiden korroosio loppusijoituksen in-situ olosuhteissa)

The aim of the project is to assess the corrosion of pressure vessel steels and other power plant and demolition waste steels under natural or simulated groundwater conditions, also taking into account the possibility of microbiological corrosion and to determine whether activation accelerates corrosion effect. The aim is to extend and compare the already existing corrosion measurement data with the test results made with activated samples and with the groundwater environments of several final disposal sites.

As a result of the project, the effect of the activity on the corrosion of demolition waste and the structures used for its storage can be assessed. The study of new groundwater environments extends the applicability of the results of previous projects to assess the safety of disposal in geographically different environments.

DEMONI - Decommisioning Material characterization and final disposal studies

The aim is to develop methods for measuring activity, to study the structural structure of concrete strength phenomena in aging reactors and certifies decommissioning waste models for assessing long-term safety under disposal conditions. The samples studied in the project are real nuclear facility materials and in that respect are usually difficult to obtain public research.

KaMu - Effect of existing conditions to gas formation in low level waste repositories (Olosuhteiden vaikutus kaasujen muodostumiseen matala-aktiivisen huoltojätteen loppusijoituksessa)

The aim of the KaMu project is to assess the factors influencing the formation of gases in the final disposal of low-level waste in Finnish conditions. The project utilizes the in-situ gas development experiment in the Olkiluoto VLJ cave, which was launched more than 20 years ago. The project causes process disturbances (e.g., an increase in pH) that reflect probable scenarios for disposal. After causing the disturbance, changes in gas formation and chemical and microbiological parameters are analyzed.

KYT2022 SR8 Alternative technologies:

ALES - Actanide-Lantanide Separation (Aktinidi-lantanidi erotus- ja separointimateriaalit)

The project aims to develop a material and method for actinides / lanthanides nuclear fuel used for group separation. In addition, the goal is advanced monitoring and disseminating information on international research into fuel cycles separation techniques. The project synthesizes a hybrid material with Ac / Ln group resolution and seeks to understand the causes of Ac / Ln group resolution. The aim is also to understand the inorganic and organic role of the hybrid in the selectivity of sorption as well as to understand the sorption mechanism.

The project aims to control the Ac / Ln group elution of the hybrid material so that the radiotoxicity of nuclear waste is reduced and practical transmutation of actinides is possible - more than 99.5% separation percentage for Ac / Ln groups.

KYT2022 Social license research:

SOLID - Acquiring Social License for Disposal: trust and acceptance

The SOLID project aims to increase understanding of the building of trust and social acceptability in nuclear waste management. The aim is to compare disposal decision-making security argument in Finland and France for trust and social acceptance from view. The project provides information on differences in security arguments, which helps to assess mm. issues related to the monitoring debate. The project has also organized a seminar on trust and social licensing.