RESEARCH REPORT

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SAFIR2022 Annual Plan 2019

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Summary

The mission of the National Nuclear Power Plant Safety Research programme 2019-2022 (SAFIR2022) is derived from the stipulations of the Finnish Nuclear Energy Act, concerning ensuring of expertise. The programme is continuation to a series of earlier national research programmes that have proven their worth in developing and maintaining expertise in nuclear power plant safety.

SAFIR2022 Management Board is responsible for steering of the research programme and consists of representatives of the Radiation and Nuclear Safety Authority (STUK), the Ministry of Economic Affairs and Employment (MEAE), Fennovoima Oy, Fortum, Teollisuuden Voima Oyj (TVO), Technical Research Centre of Finland Ltd (VTT), Lappeenranta-Lahti University of Technology (LUT), and Aalto University (Aalto).

In 2019 the planned volume of the research projects part in the SAFIR2022 programme is 6,7 M€ and 43 person years. Main funding organisations are the Finnish State Nuclear Waste Management Fund (VYR) with 4,4 M€ and VTT with 1,3 M€. Research is carried out in 32 projects.

This report consists of a summary of the research plans of the projects and financial and administrative issues of the programme in 2019.

Confidentiality	Public

Espoo 14.6.2019

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1. Introduction

In accordance with Chapter 7a of the Finnish Nuclear Energy Act, the objective of the National Nuclear Power Plant Safety Research programme 2019-2022 SAFIR2022 is to ensure that should new matters related to the safe use of nuclear power plants arise, the authorities possess sufficient technical expertise and other competence required for rapidly determining the significance of the matters. High scientific quality is required of the research projects in the programme and the results must be available for publication.

The SAFIR2022 programme's planning group, nominated by the Ministry of Economic Affairs and Employment in November 2017, defined the following mission for national nuclear safety programmes:

> *National nuclear safety research aims at high national nuclear safety assessment capability. It develops and creates expertise, experimental facilities as well as computational and assessment methods for solving future safety issues in close cooperation with competent international partners.*

The vision of SAFIR2022 was defined as follows:

> *The SAFIR2022 research community is a vigilant, internationally recognised and strongly networked competence pool that carries out research on topics relevant to the safety of Finnish nuclear power plants on a high scientific level and with modern methods and experimental facilities.*

The Framework Plan [1] describes the research to be carried out in SAFIR2022. The new programme essentially covers the themes of the preceding SAFIR2018 programme [2].

SAFIR2022 Management Board was nominated in August 2018. It consists of representatives of the Radiation and Nuclear Safety Authority (STUK), the Ministry of Economic Affairs and Employment (MEAE), Fennovoima Oy, Fortum, Teollisuuden Voima Oyj (TVO), Technical Research Centre of Finland Ltd (VTT), Lappeenranta-Lahti University of Technology (LUT) and Aalto University (Aalto).

A public call for research proposals for 2019 was announced on the 24th of August 2018. After the closure of the call on the 19th of October 2018, SAFIR2022 Management Board, taking into account the evaluations made by the steering groups, prepared a proposal for the MEAE regarding the projects to be funded in 2019. The funding decisions were made by the Finnish State Nuclear Waste Management Fund (VYR) in March 2019. In 2019, the programme consists of 32 research projects and a project for programme administration.

VYR funding is collected from the Finnish utilities Fennovoima Oy, Fortum and Teollisuuden Voima Oyj based on their MWth shares in Finnish nuclear power plants (units in operation, under construction, and in planning phase according to the decisions-in-principle). In addition to VYR, other key organisations operating in the area of nuclear safety also fund the programme.

SAFIR2022 programme consists of the research part covering the funding of the research projects and an infrastructure part. In 2019 the infrastructure funding is allocated to the VTT Centre for Nuclear Safety research equipment and laboratory facility support. The volumes of the research part and infrastructure part are 6,7 M€ and 6,2 M€, respectively. Thus the total volume of SAFIR2022 with the infrastructure funding projects RADINFRA and RADCNS is 12,9 M€. The above infrastructure projects are steered in SAFIR2022 (SG4 and RG8, see chapters 3 and 4) and also include 1,5M€ funding from the KYT2022 programme.

This annual plan summarises the plans of the research projects (Chapter 2) and provides statistics of the research part of the programme (Chapter 3). Administrative issues are summarised in Chapter 4. Appendix 1 contains a list of SAFIR2022 Management Board members.

This report has been prepared by the programme director and project coordinator in cooperation with the managers and staff of the individual research projects.

2. Research areas and research projects in 2019

The SAFIR2022 programme is divided into four research areas:

1. Overall safety and systemic approach to safety
2. Reactor safety
3. Structural safety and materials
4. Research infrastructure.

The research areas are presented with more detailed descriptions of their research needs for the programme period 2019-2022 in the SAFIR2022 Framework Plan [1].

In 2019, the research is carried out in 32 research projects. The planned total volume of the research projects is 6,7 M€. The volumes as well as the research execution organisations of the research projects are listed in Table 2.1. The projects marked as “Excellence project” have been granted funding for several years (funding for 2019 is shown Table 2.1).

Table 2.1 SAFIR2022 projects in 2019.

Research area	Project	Acronym	Organisation(s)	Total funding (k€)
1. Overall safety and systemic approach to safety				
	Building operational readiness of control room crews: preparing for the unexpected	BORS	VTT, Finnish Institute of Occupational Health (FIOH)	135,0
	Co-simulation model for safety and reliability of electric systems in flexible environment of NPP	COSI	VTT, Aalto	195,4
	New developments and applications of PRA	NAPRA	VTT	217,9
	Development of framework for justification of overall safety	OSAFE	VTT, Risk Pilot, LUT	85,0
	Participative development for supporting human factors in safety	PARSA	FIOH, VTT	121,4
	Predicting extreme weather, sea level and atmospheric dispersion for nuclear power plant safety	PREDICT	Finnish Meteorological Institute (FMI)	178,6
	Safety and security assessment of overall I&C architectures	SEARCH	VTT, Aalto - *Excellence project 2019-2022*	368,0
	Uncertainty management in fire risk analyses	URAN	VTT, Aalto - *Excellence project 2019-2022*	220,7
2. Reactor safety				
	Analytical severe accident research	ANSA	VTT	264,0
	Coupled analysis of transient scenarios	CATS	VTT	200,0
	CFD4RSA - CFD methods for reactor safety assessment	CFD4RSA	VTT	194,0
	Interdisciplinary fuels and materials	INFLAME	VTT	243,0
	Developing the working arms of Kraken, the next generation computational framework for reactor design and licensing analyses	LONKERO	VTT - *Excellence project 2019-2022*	254,3

	Mitigation and analysis of fission products transport	MANTRA	VTT	161,8
	Passive heat exchanger experiments	PAHE	LUT	126,0
	PWR PACTEL tests	PATE	LUT	233,0
	Radiation shielding and criticality safety analyses	RACSA	VTT	180,0
	Sparger separate effect tests	SPASET	LUT	165,0
	Safety through Thermal-Hydraulic Analyses and co-operation	THACO	VTT	227,7
3. Structural safety and materials				
	Additive manufacturing in nuclear power plants	AM-NPP	VTT, Aalto, LUT	93,0
	Advanced materials characterisation for structural integrity assessment	AMOS	VTT	137,0
	Critical studies in support of the ageing management of NPP concrete infrastructure	CONAGE	VTT, Aalto	151,4
	Modelling of aged reinforced concrete structures for design extension conditions	CONFIT	VTT, Tampere University of Technology (TUT)	115,7
	Effect of long-term operation on aging and environmentally assisted cracking of nuclear power plant component materials	ELIAS	VTT, Aalto, TUT	136,0
	Extended lifetime of structural materials through improved water chemistry	ELMO	VTT - *Excellence project 2019-2022*	193,0
	Fatigue and evolving assessment of integrity	FEWAS	VTT, Aalto	180,0
	Non-destructive examination of NPP primary circuit components, machine learning and reliability of inspection	RACOON	VTT, Aalto	128,6
	Safety criteria and improved ageing management research for polymer components exposed to thermal-radiative environments	SAMPO	VTT, RISE Research Institutes of Sweden - *Excellence project 2019-2022*	184,0
4. Research infrastructure				
	Barsebäck RPV material used for true evaluation of embrittlement	BRUTE	VTT - *Excellence project 2019-2022*	366,8
	Infrastructure development at LUT safety research laboratory	IDEAL	LUT	448,0
	Participation in Jules Horowitz Reactor project - towards first criticality in 2022	JHR2022	VTT	136,0
	Pre-emptive reduction of radiological laboratory legacy waste	LABWAST	VTT	306,0
Programme administration				
	SAFIR2022 administration	ADMIRE	VTT	373,0

The costs of ADMIRE are for the period 1.1.2019-31.3.2020. The costs include the small projects and value-added tax 24%.

2.1 Overall safety and systemic approach to safety

In 2019 the research area “”Overall safety and systemic approach to safety” includes eight projects:

1. Building operational readiness of control room crews: preparing for the unexpected (BORS)
2. Co-simulation model for safety and reliability of electric systems in flexible environment of NPP (COSI)
3. New developments and applications of PRA (NAPRA)
4. Development of framework for justification of overall safety (OSAFE)
5. Participative development for supporting human factors in safety (PARSA)
6. Predicting extreme weather, sea level and atmospheric dispersion for nuclear power plant safety (PREDICT)
7. Safety and security assessment of overall I&C architectures (SEARCH)
8. Uncertainty management in fire risk analyses (URAN).

2.1.1 BORS - Building operational readiness of control room crews: preparing for the unexpected

The project will focus on operator work in the control room context. Some of the main objectives of SAFIR2022 is to maintain and develop know-how on conventional and modern main control rooms, develop modern approaches to support and evaluate control room design, and in more general level deepen understanding of HOF (human and organizational factors) and HFE (human factors engineering) aspects. It is also stated that expertise in HOF themes that have been studied in the previous SAFIR programmes, such as control room behaviour and validation, should be further developed and strengthened.

The project aims to study the joint cognitive system (JCS) from the perspective of human-system interfaces (HSIs), procedures, operators’ resources for action and skills training. First, the project aim is to identify the key functions and activities (i.e., resilience skills, resources for action and constraints) and their dependencies for the proposed work. Second, the project aim is to advance and deepen general understanding of resilience skills and develop specific tools and methods for the analysis of simulator data and/or promotion of resilience skills assessment and skills training. This work is partly based on existing methods and tools that have been developed in earlier studies (e.g., see Laarni, 2018; Pakarinen et. al., 2018). Third, the project aim is to further increase understanding of operator practices and cognitive processes in complex incidents and severe accidents. To that aim, the project will conduct 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 a novel digital control room) or that they address topics that have not been widely studied (e.g., complex troubleshooting). Fourth, the project aim is to better understand how the JCS is shaped and evolved, and how cognitive readiness and resilience skills are acquired. To that aim, the project develops a training program for the advancement of cognitive readiness and resilience skills in operator work.

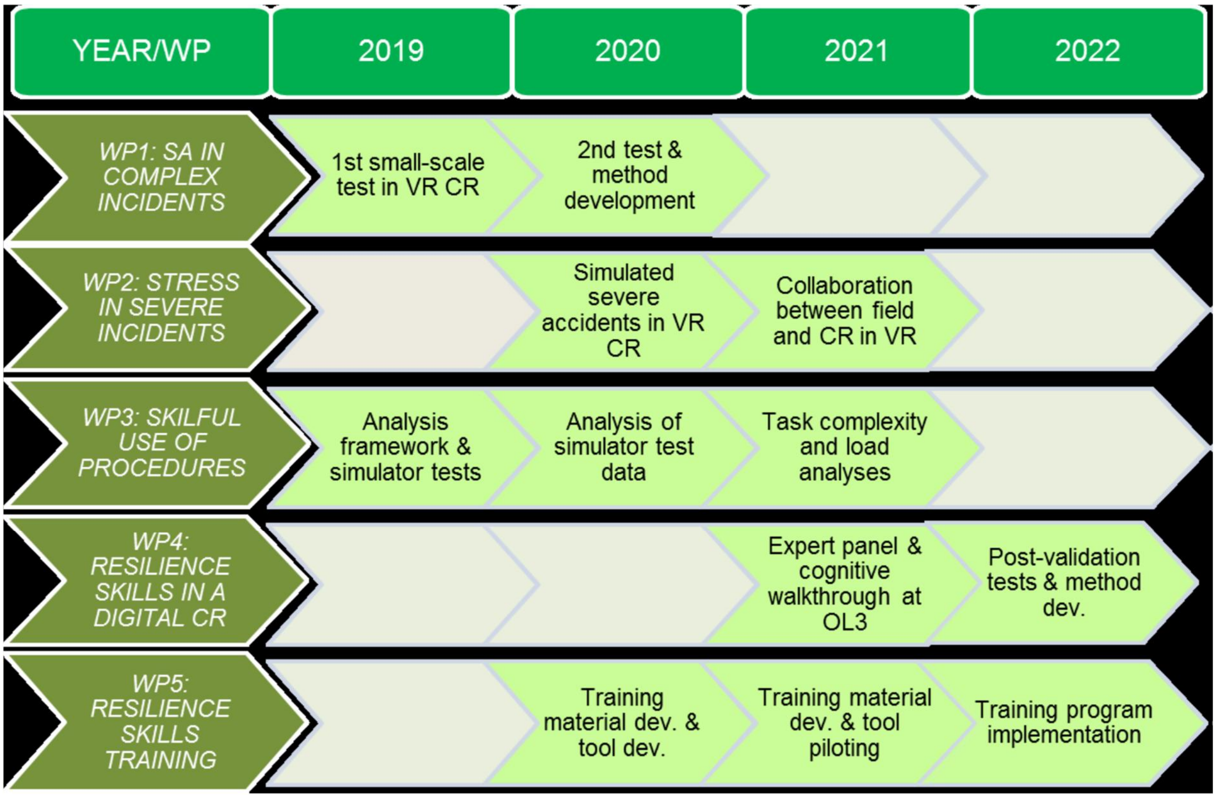

Figure 1. BORS project activities throughout the SAFIR2022 programme period.

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

The project will study safety design principles of electrical systems focusing on selected topics considered most relevant for the stakeholders on the three levels of design: 1) plant level safety design, 2) systems safety design, and 3) component safety design. The objective is to reach a general understanding on the set of electrical system initiating events that should be included in a safety case for a nuclear power plant. This includes identifying, classification and simulation of types of plant external and internal events related to electrical systems. Another relevant issue at plant level is to analyse how well redundancy, diversity and separation principles are actually accomplished for electric systems
.
To provide supporting analyses for safety cases, COSI project will develop digital twin for off-site electric power system and NPP. A detailed multi-physic simulation model for on-site electric power system of NPP interfaces to the off-site high voltage power system model and to thermal, reactor-physical and automation models like f.ex. APROS. This holistic model covers the whole chain of electrical systems enabling a structured approach for evaluation of possible common cause failures and design principles of electrical systems in the existing and future nuclear power plants and in small modular reactors including flexible operation. The simulation platform will be utilized for evaluation the adequacy and balance of safety requirements of the electrical systems in NPP in the cases of faults and disturbances.

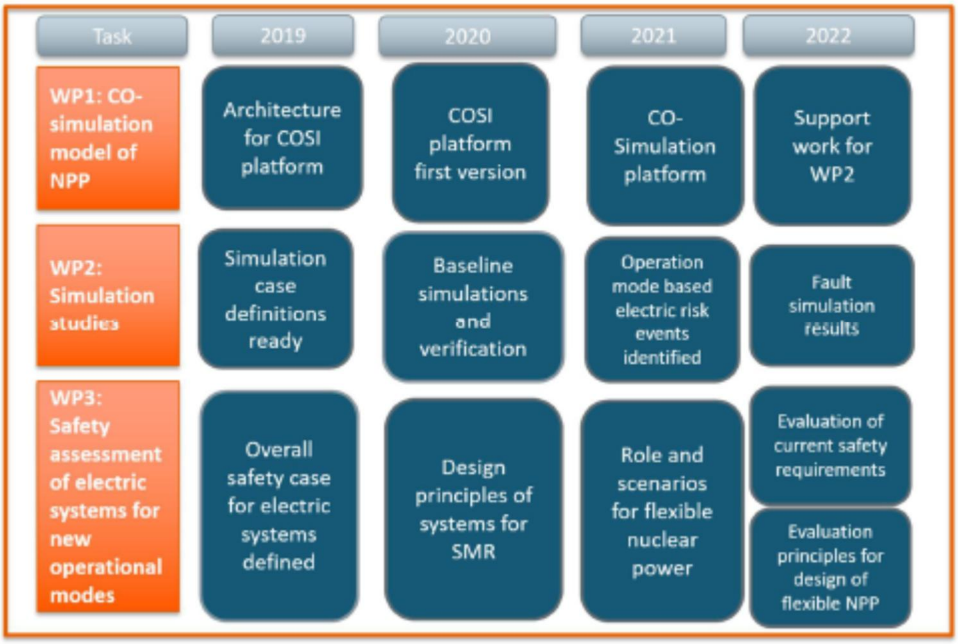

Figure 1. Expected results of COSI project.

The authority (STUK) can exploit the results of this project for assessment of the adequacy and balance for safety requirements of electrical system and control.

Power companies (NPP) can exploit the results for assessing and developing the design principles of electrical systems and getting information for developing resilience and protection against electrical faults and disturbances.

Network operator can exploit the simulation results for analyzing what kind of impacts the faults in high voltage grid have to NPP operation and safety, and vice versa what kind of impacts the faults in NPP electrical system have on grid stability and cross-border connection especially on operation of HVDC links.

2.1.3 NAPRA - New developments and applications of PRA

The project develops methods and analyses for PRA of nuclear facilities. It addresses various topics of interest in contemporary PRA, relevant to nuclear safety: PRA in long time window scenarios, PRA related to systems containing digitalized subsystems (e.g. automation systems of an NPP), internal and external hazards (seismic phenomena and fires), human reliability analysis, PRA-related quantification and computation, and severe accidents. The project also aims to improve PRA knowledge and expertise in Finland.

The main objectives of the NAPRA project are:

- to develop Seismic Probabilistic Risk Assessment (SPRA) and harmonise Finnish practices with international ones
- to develop more realistic probabilistic analysis of fires in nuclear power plants by dynamic modelling
- to improve the quality of PRA with respect to modelling of time, long time windows, dynamic success criteria and component repairs
- to define realistic or slightly conservative human error probability estimates and to identify the most relevant human failure events in hybrid control rooms, also taking into account the dynamic nature of PSFs (Performance shaping factors)
- to participate in an international benchmark study on digital I&C modelling

- to develop an integrated probabilistic and deterministic approach for failure tolerance analysis
- to study PRA for SMRs
- to better implement containment integrity and ageing related phenomena in level 2 PRA

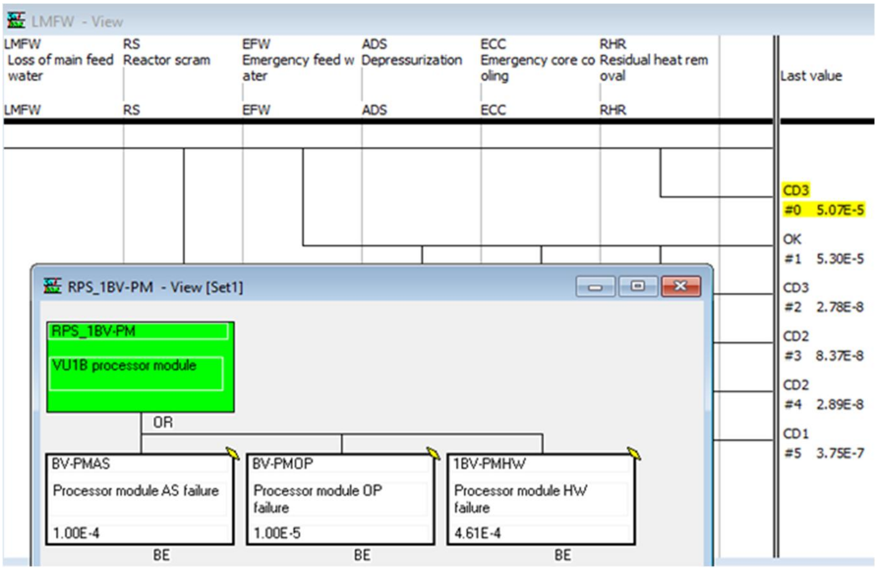

Figure 1. A fault tree and an event tree used in an international OECD WGRISK benchmark study of PRA for digital I&C systems, as represented in VTT's PRA code FinPSA.

2.1.4 OSAFE - Development of framework for justification of overall safety

The general objective of the project 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 justification, safety culture, institutional streght-in-depth, system modelling) and improving of these methods for the purposes of safety assessment and safety justifications in the context of operating plant's electric systems, decommissioning and the new technologies, such as SMRs. Thus, the project focuses on both the current challenges in the operating plans and the projections to the futures in terms of decommissioning and SMRs.

The regulator, nuclear power companies, researchers and possible new actors in the nuclear field are end users. The results of this project:

- Provide means for efficiently documenting and communicating the aspects of overall safety
- Enable defining objectives, context, constraints, assumptions, and criteria for acceptance of different aspects of overall safety
- Provide base for measuring the ”overall safety level”
- Help understanding the meaning of, and interaction between different aspects of overall safety.

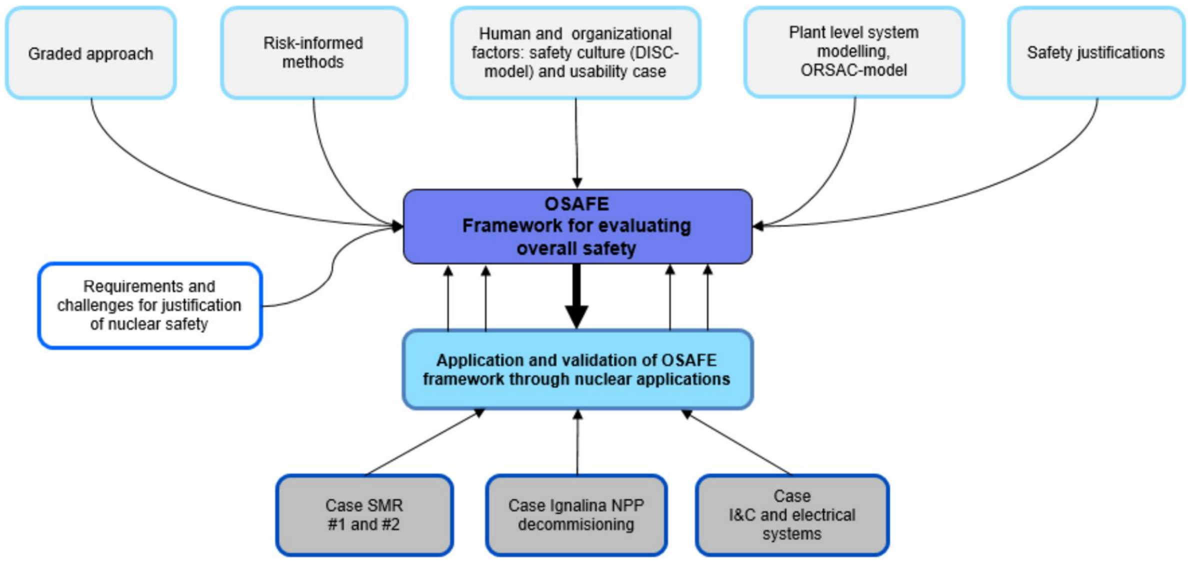

Figure 1. Framework for evaluating overall safety in OSAFE.

2.1.5 PARSA - Participative development for supporting human factors in safety

The project aims to improve participative development for supporting human factors as a part of safety management at nuclear power plants. Especially, the project evaluates, if and how currently used ways, models and tools support safety and efficiency of work procedures, processes and practices in nuclear power plants (NPP). Based on evaluation, the project aim to improve models and tools for human focused safety management. The project is targeting on reframing and improving new ways of action, guidance, operational practices and other practical solutions that promote organizational learning, modelling tacit knowledge, and conceptual mastery of work processes in nuclear operations. The case studies are scoped and implemented in maintenance operations of nuclear power plants.

The main objectives of the PARSA project are the following:

- Maintaining collaboration among contributors at Finnish nuclear HF/HOF and safety culture R&D field (WP1)
- Building synergy among the four PARSA work packages (WP1)
- Create a video based developmental training method for new learning and verbalizing embodied knowledge for nuclear maintenance work (WP2)
- Provide a task-analysis of nuclear maintenance work (WP2)
- Visualizing tacit knowledge of maintenance work processes. (WP2 and WP3)
- Conducting work process analyses at selected maintenance tasks. (WP3)
- Evaluating and disseminating development needs raised by the work processes (WP3)
- Reflect the underlying assumptions, limits and possibilities of human performance programmes and facilitate their participative development (WP4)
- Develop and test novel human performance tools (WP4).

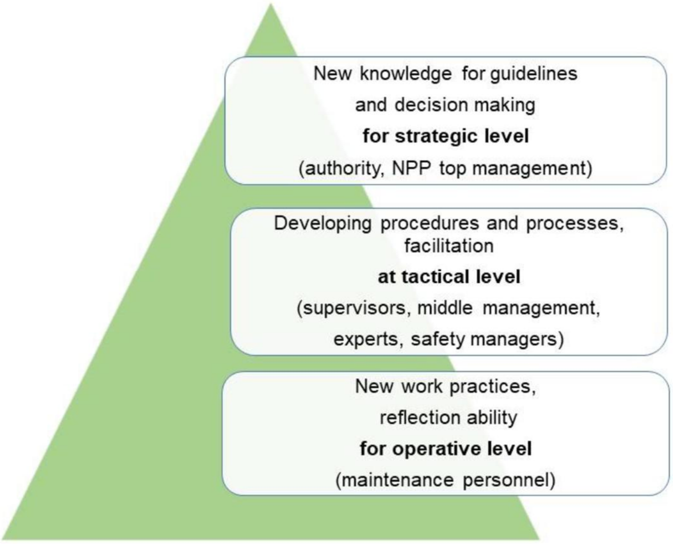

Figure 1. Exploitation and usefulness of the PARSA results, related to nuclear industry needs for new knowledge, methods and tools.

2.1.6 PREDICT - Predicting extreme weather, sea level and atmospheric dispersion for nuclear power plant safety

Improving nuclear safety requires a better understanding of and preparedness against external weather and marine events. The main objective of the project is to develop and maintain research expertise and methods needed for assessing probabilities of occurrence of safety-relevant single and compound extreme weather and marine events, both in the range of 0-15 days ahead and in decadal time scales of recent past and future climatic changes. An important goal is also to strengthen expertise on statistical and physical models, analysis steps and processes involved when assessing frequencies of external events. Related activities within OECD/NEA and elsewhere are followed and also contributed to.

The work of extreme weather aims to improve the reliability of estimates about the likelihood of exceptional single and co-occurring events in the surroundings of the NPP sites. More specifically, first goal is to clarify how frequently intense coastal snowfall cases have occurred in the past and whether the warming climate will increase or decrease their frequency and magnitude in the future. Second, the project will assess joint probabilities of intense precipitation and high sea level, taking into account their potential mutual dependencies in different seasons. The third goal is to provide estimates of the occurrence of downbursts, i.e., hazardous gusts of winds related to thunderstorms. The project will also follow new research elsewhere about climate change impacts on large-scale windstorms and may later in SAFIR2022 specify a NPP-relevant research objective for them.

The objective of the sea level studies is to produce accurate and up-to-date flood risk estimates for the NPPs. In order to do this, the project will follow closely international studies on sea level and waves and new developments on sea level rise scenarios and location-specific and multivariate flood risk analyses. The project will also actively conduct sea level and wave research to improve scientific understanding and methods related to these phenomena. Based on the views and needs of the NPPs, the PREDICT project will focus on studies to benefit the flood risk assessments and safety of the NPPs in the most relevant way.

The goal of the work related to forecasting extreme events is to exploit the potential of state-of-the art numerical weather prediction (NWP) systems for the benefit of safe and economical nuclear power production. To this end, a weather service for nuclear power production will be designed, taking into account the expertise of NPP-operators and NWP-developers alike. The latest developments in the services of leading actors such as the European Centre of Medium Range Weather Forecasts or the ALADIN-HIRLAM consortium for short range NWP will be applied.

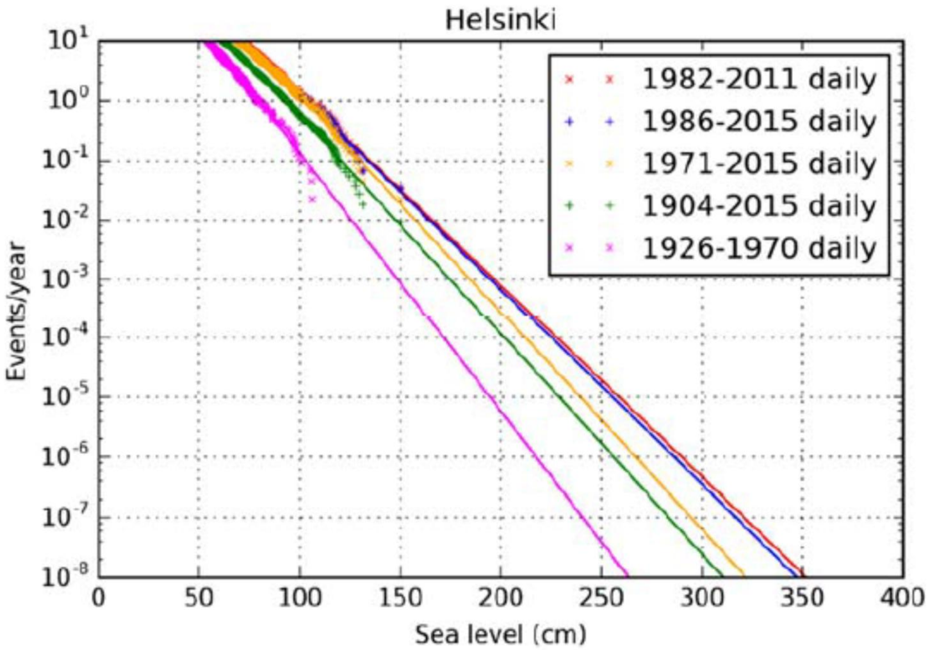

Figure 1. Extrapolations made on different sea level time series at the Helsinki tide gauge.

2.1.7 SEARCH - Safety and security assessment of overall I&C architectures

The overarching four-year objective of the project is to develop an **integrated design and assessment framework for overall I&C architectures**. More specifically, the aim is to: (1) verify I&C system and architecture level properties related to **Defence-in-Depth**, and (2) broaden the scope in which **formal methods** can be used for deterministic assessment of I&C system properties related to the plant as a whole.

The project aims to achieve the objective in the following ways:

1. SEARCH builds upon the multidisciplinary DiD models developed in the SAUNA SAFIR2018 project (VTT 2017), and develop methods to assess how the technical design solutions related to overall I&C architectures fulfil DiD specific requirements related to, e.g., failure tolerance, and separation of safety classes, DiD levels, and security zones. Avoiding architectural flaws in the early, functional design stage—before the architecture is used as input for more detailed system design—is particularly crucial for design process success.

2. SEARCH develops tool-based ways for requirement management, with particular focus on how to resolve potential conflicts between requirements related to safety, security, safeguards, society and sustainability (5S). Addressing the trade-offs paves the way for justifying the design in the licensing process.

3. SEARCH broadens the scope in which model checking—a proven formal verification method for I&C software assessment—can be used to verify properties related to how

the plant operates as a whole. Model checking can provide conclusive proof of properties related to software, and pushing the boundaries in terms of addressing also non-functional properties helps promote a deterministic, comprehensive approach for overall I&C architecture assessment. SEARCH also builds methods and tools to automate the overall model checking work process, and on a more practical level, trains experts on a topic that is recognized important in the Finnish nuclear industry.

4. SEARCH aims at developing a framework for efficient co-use of different modelling and analysis tools, so that any change made in the I&C system design could immediately facilitate an update and reassessment in all the relevant, connected tools, including on the overall architecture level. Such a framework supports continuing re-evaluation through the plant development process, as details become available or change.

5. SEARCH promotes Model-based system engineering (MBSE) and the use of open standards for plant data representation and exchange. The shift from a document-based to a domain model-based information exchange is a precondition for an integrated design and assessment framework. Open standards enable technological interoperability, and the opportunity to modify data interchange formats to suit tool-specific needs. Throughout the four years and across research tasks,

6. SEARCH uses practical—and preferably *visual*— demonstrations based on actual industrial data as a way ensure that (1) the proposed tools and methods are sound and feasible, (2) the utilities are engaged in the research, and (3) the added value from the methods (as well as the necessary steps for further development) become apparent and communicable.

The project aims at producing methods and tools that are practically applicable in the work of utilities, regulators, and TSOs for assessing I&C architectures. In addition to contributing to safety, such tools also have the potential to reduce costs in design and licensing.

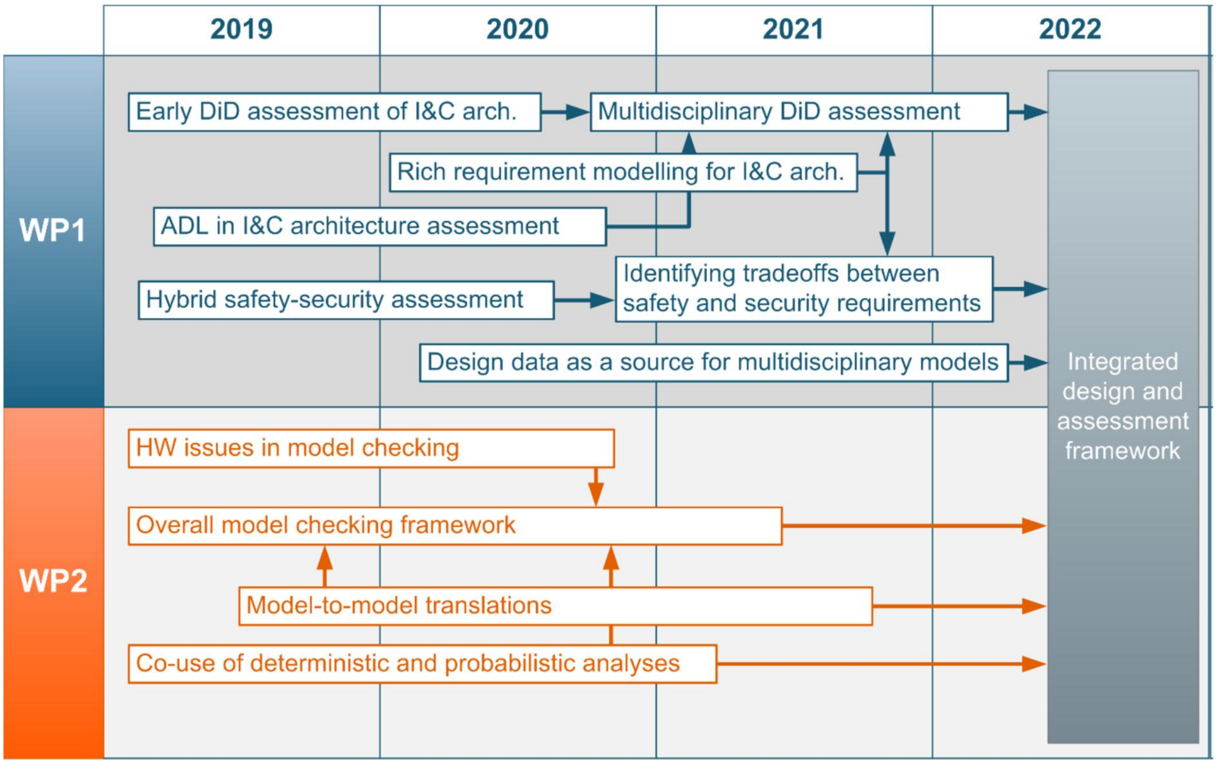

Figure 1. SEARCH work plan for 2019-2022

2.1.8 URAN - Uncertainty management in fire risk analyses

The **overall goals** of this project are **to quantify the uncertainties** related to the fire growth predictions, and **to manage them** by developing better models that can serve the safety assessments of nuclear facilities and installations with different lifespans. The objectives of URAN during the whole duration of the project are summarized in Table 1. Real scale model validation has a significant role in this, and in URAN it will be done in several levels:

- "Traditional" model validation using large-scale experimental results (OECD PRISME 3 cable fire experiments).In this the experimental set-up, input data and the target results are available to the modeller from the beginning.
- Semi-blind exercise where the target results are not available to the modeller.
- Blind exercise which corresponds to a real case in a NPP. Only limited information available to the modeller.

Significant uncertainty in the NPP fire analyses is related to the ageing of the materials. In particular, the ageing mechanisms of the new flame retardant chemicals, and the flame retardant materials are not known. **The goal in URAN is to investigate the ageing mechanisms both numerically and experimentally, and to understand the mechanisms and their effect to the general fire safety of the NPP.**

Table 1. Objectives of URAN during four years.

Objective	Verification
Model uncertainty for fire spread predictions quantified and reduced	1. FDS validation simulations of OECD PRISME 3 cable fire experiment (article). 2. New generation of cable models developed and validated using literature and PRISME data (FDS development and guides updated).
User effects quantified for fire spread simulations	1. Participation in OECD/PRISME open and semi-blind benchmark (PRISME AWG Report).
Relative roles of different uncertainty types quantified in a real NPP fire event	1. Participation in OECD/PRISME 3 blind benchmark of real NPP fire (PRISME AWG Report).
Understanding the relevant effecs of ageing on flame retardant polymers	1. Validation simulations of pyrolysis and reaction-to-fire experiments on aged polymers.
Model uncertainty can be taken into account in fire-PRA	1. Proposal for the improvements in code validation and probabilistic simulation procedures (report).
Dissemination of PRISME 3 results in Finland	1. Participation in PRISME AWG, PRG and MB meetings, reporting in SAFIR RG and ad hoc meetings.
Education of new fire safety experts	1. Doctoral dissertation. 2. Journal articles. 3. New expert to the VTT's fire research group.

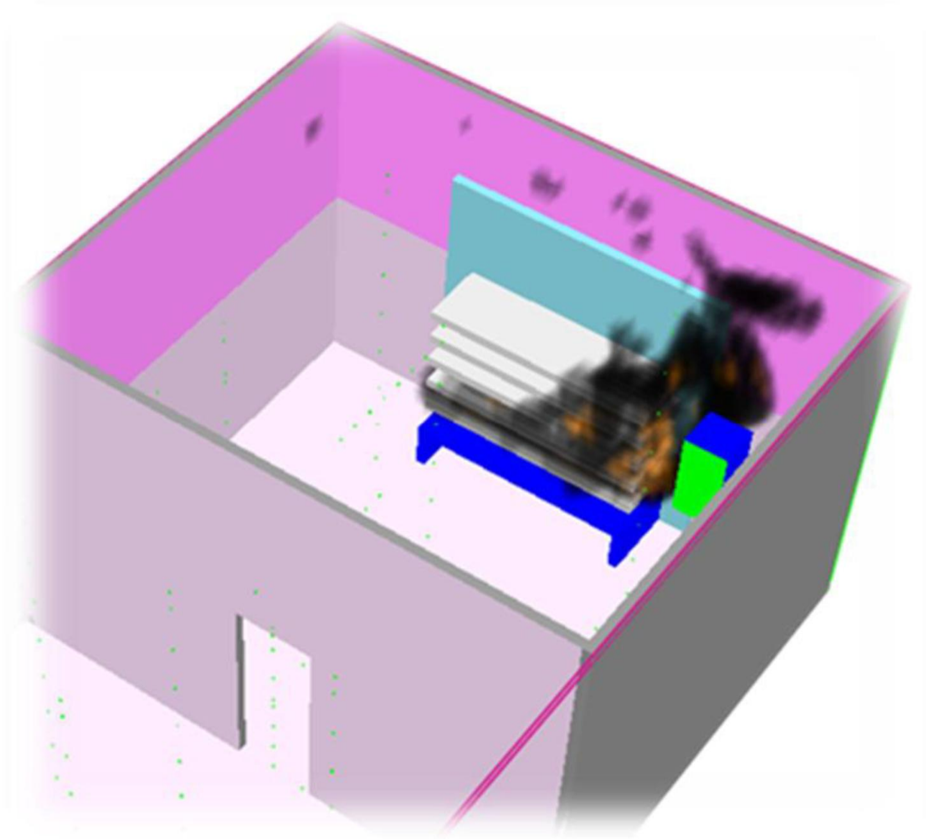

Figure 1. Simulation of Cable Fire Spread test - possible target for Phase 1 of PRISME3 AWG.

2.2 Reactor safety

In 2019 the research area “Reactor safety” includes eleven projects:

1. Analytical severe accident research (ANSA)
2. Coupled analysis of transient scenarios (CATS)
3. CFD methods for reactor safety assessment (CFD4RSA)
4. Interdisciplinary fuels and materials (INFLAME)
5. Developing the working arms of Kraken, the next generation computational framework for reactor design and licensing analyses (LONKERO)
6. Mitigation and analysis of fission products transport (MANTRA)
7. Passive heat exchanger experiments (PAHE)
8. PWR PACTEL tests (PATE)
9. Radiation shielding and criticality safety analyses (RACSA)
10. Sparger separate effect tests (SPASET)
11. Safety through thermal-hydraulic analyses and co-operation (THACO)

2.2.1 ANSA - Analytical severe accident research

The overall objective of the project is to ensure that a sufficient national competence exists in the area of severe accidents and that the tools and methods in use are properly validated. On more detailed level it is expected that more information will be received from the Fukushima accidents and that this information can be taken into use in the form of improved models and more capable experts in modelling.

Proper functioning of the passive safety features under all circumstances should be ensured. This is possible only after the models have been validated against suitable experiments and reliable modelling practices for the passive safety features have been developed.

Reliable modelling of the phenomena inside the containment is essential in determining the timing and characterisation of radioactive releases accurately. Models to simulate hydrogen combustion in low concentrations will be developed to assess the risk of containment or reactor hall failure followed by such event. More complete understanding on the behaviour of different iodine species in pool scrubbing will be achieved in collaboration with the planned experimental and analytical activities. The pool scrubbing models will be developed together with the integral code developers.

Work packages and tasks of the ANSA project are:

- Participation in international programmes:
  - OECD/NEA ARC-F
  - U.S. NRC CSARP
  - OECD/NEA THAI-3
- Fukushima
  - Accident progression
  - Suppression pool stratification
- Passive safety features
  - Core Catchers
- Containment phenomena
  - Hydrogen combustion
  - Pool scrubbing

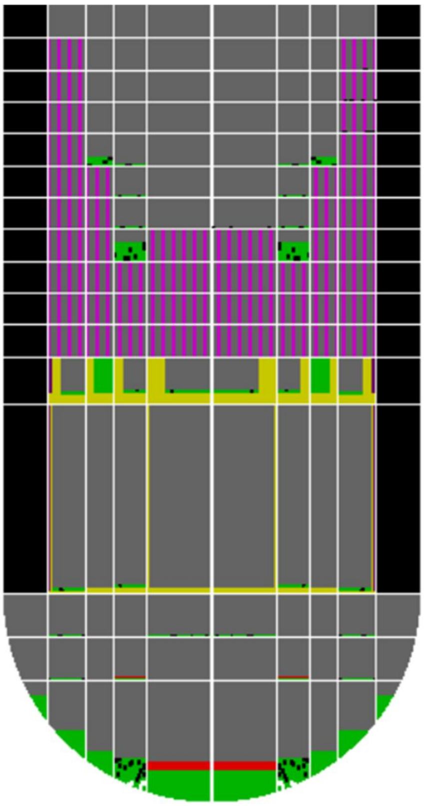

Figure 1. MELCOR analysis result for the Fukushima Unit 1 reactor when about half of the fuel rods have collapsed, and a lot of material has been discharged to the containment through the failed lower head penetrations. The green color shows particulate debris and red molten materials.

2.2.2 CATS - Coupled analysis of transient scenarios

Aim of the CATS project is to achieve and extend the comprehensive understanding of the phenomena occurring in transients and accidents in NPPs and the capability to simulate them with models embedded in computer codes in such a way, that we can give more reliable answers to the safety requirements set in the YVL guides. Project supports both the short-term objective that safety authority has to have, continually available, a validated and applicable tools together with an ability for independent safety analysis as well as the long-term goal of a new validated code set. Therefore, the results of CATS are exploitable by the safety authority and other parties that require transient safety analyses for their operation.

The project focuses particularly on tools and expertise that are necessary for the modelling of events where neutronics, thermal-hydraulics in primary and secondary circuit, fuel behaviour and plant systems all play important role. One significant objective is to ensure sufficiency of experts also after 2022 by recruiting new experts and facilitating transfer of existing knowledge and expertise such that new experts will be able to perform complete safety analyses.

Work packages and tasks of the CATS project are:

- Plant modelling with neutronics coupling
    - Plant modelling with TRACE code
    - BWR steam line dynamics
- Safety analyses methodology
    - Hot channel methodology
    - Uncertainty and sensitivity analysis
- Multi-physics simulations

   - o Blockage in a fuel assembly
- International co-operation

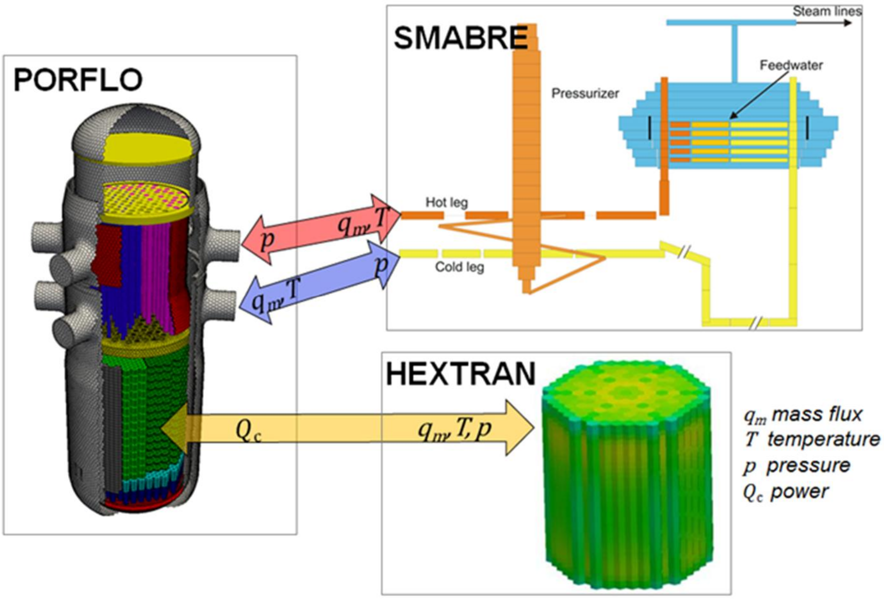

Figure 1. Coupling scheme for CFD, reactor dynamics and system code.

2.2.3 CFD4RSA - CFD methods for reactor safety assessment

Computational Fluid Dynamics (CFD) methods will be developed and validated for Nuclear Reactor Safety (NRS) Assessment. The overall objective of the CFD4RSA project will be to improve the usability and reliability of CFD calculations in NRS assessment. In 2019, the project will consists of four work packages:

- In WP1, the reliability of coupled CFD-Apros calculations in the modelling of passive safety systems was proposed to be studied and improved. In 2019, a journal article on coupled CFD-Apros calculations will be written.

- In WP2, Uncertainty Quantification (UQ) methods in CFD calculations will be tested and taken into use. Use of UQ methods will be an important step in the acceptability of CFD calculations for NRS assessment.

- In WP3, coarse-mesh OpenFOAM models will be developed for reactor pressure vessel. Coupling of the CFD model with neutron transport solver makes possible the calculation of many transient scenarios of the pressure vessel. In particular, the coarse-mesh formulation makes possible three-dimensional analysis of the mixing of the coolant in the complicated geometry of reactor core.

- In WP4, thermal stratification in pressure suppression pools of BWRs will be studied. Effective Heat Source and Effective Momentum Source (EHS/EMS) methods will be implemented and validated in CFD code. In addition, implementation of an EHS/EMS based model in Apros will be tested. Such models make possible the CFD calculation of thermal stratification of the pool during postulated accident scenarios.

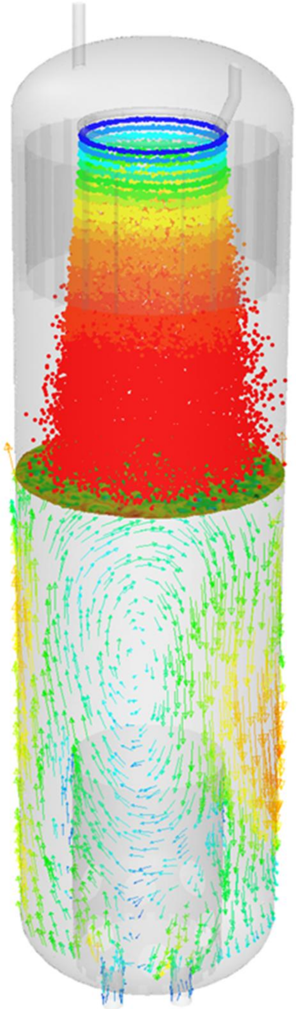

Figure 1. CFD model for VVER-440 pressurizer.

2.2.4 INFLAME - Interdisciplinary fuels and materials

The INFLAME project consists of modelling and experiments current and advanced nuclear fuel, both in normal operation and accident conditions. The main part of the project focuses on nuclear fuel modelling, as data from integral experiments is obtained from international research programmes. The fuel behaviour module FINIX will be developed further in the project, validated and application range broadened. On the experimental side, advanced claddings will be investigated on their mechanical and oxidation behaviour. As a first step in Finland, experimental studies on the behaviour fuel pellets in in-reactor conditions will be performed with simulant materials.

The project consists of three work packages:

- Integral fuel behaviour
  - FINIX
  - RIA
  - Advanced and accident tolerant fuels
- Advanced cladding
  - Mechanical properties
  - Hot loop
  - Steam furnace oxidation testing
- Fuel pellet materials

- o Equipment construction
- o Microscopy

As nuclear fuel provides the first safety barriers against the spread of fission products, the understanding of its behaviour is important for the nuclear safety. The applicability of FINIX in LOCA conditions will be improved and codes used for safety analysis of Finnish reactors will be validated further as well as models developed for these codes. New information on the performance of advanced claddings will be obtained and experimental capability on fuel pellet studies at VTT will be increased.

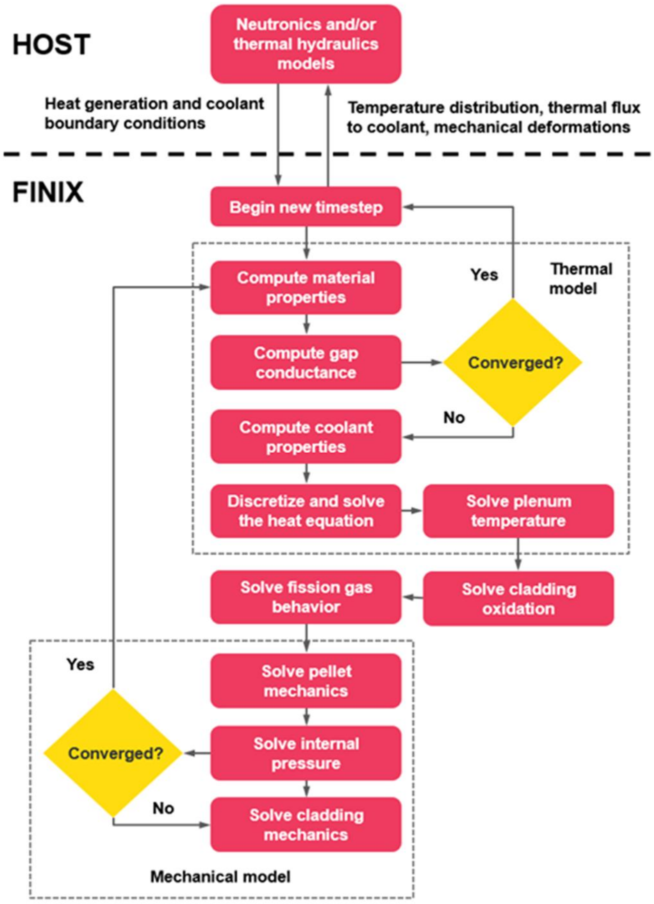

Figure 1. Flowchart of the solution method of the FINIX fuel behavior module.

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

The objective is to produce an internationally benchmarked and validated reactor analysis framework that can be used both as a workhorse in day-to-day reactor design and in various novel and detailed analyses that benefit from the flexible high-fidelity simulation capabilities of the framework.

The LONKERO project ties in several recent reactor analysis code development projects conducted in the SAFIR programmes into a novel reactor analysis framework called Kraken. The project builds on the excellent experience gained in the development of novel solvers such as the Serpent Monte Carlo neutronics code, the FINIX fuel behaviour module and the Ants nodal neutronics solver.

LONKERO aims to develop both the new reactor analysis toolchain (Kraken) and source code level expertise on the different solvers as well as their application to licensing relevant analyses.

Kraken will be developed in a way that allows the user to conduct analyses required by the YVL-guides and international regulations (e.g. NUREG-0800).

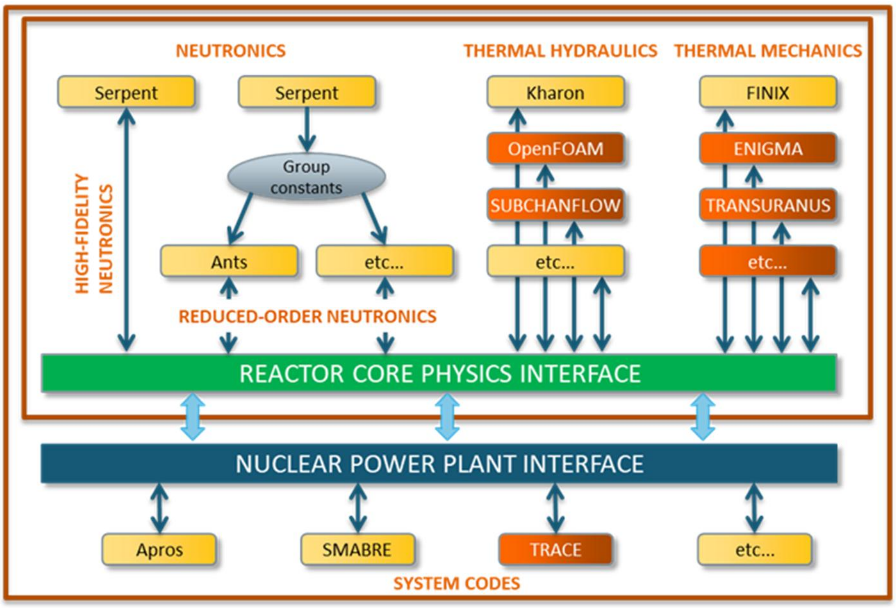

Figure 1. A schematic representation of the future Kraken framework. Codes developed at VTT are marked in yellow and potential third-party solvers in orange.

2.2.6 MANTRA - Mitigation and analysis of fission products transport

MANTRA project is focused on to reduce the uncertainties associated with the behaviour and speciation of FPs in a severe accident. **The objective** is to experimentally study the chemistry and transport of fission products, especially iodine, caesium and tellurium, in primary circuit and containment conditions. **Second objective** is to experimentally study the mitigation of gaseous and aerosol fission products releases into the environment by pool scrubbing phenomenon and validate the related models. **Third objective** is on long-term severe accident management study, which address physical and chemical processes in a severely damaged nuclear power plant after the period of 72 hours or after the plant has reached the safe stable state.

As a result, new models based on the experimental data will be derived. The models of phenomena, which could not be previously considered in the accident analysis, can be included in SA analysis codes.

The project consists of four work packages:

- Primary circuit chemistry of I, Cs, Te
    - Iodine and Caesium chemistry
    - Tellurium chemistry - NKS collaboration (PhDs)
    - ASTEC/Sophaeros training and work
- Pool scrubbing (containment and FCVS)

    - CsI, I2 and CH3I retention
    - Tellurium retention spray/pool - NKS collab. (PhDs)
- Long-term SA management
    - Literature review and discussions with end-users
- OECD/NEA follow-up projects
    - OECD/NEA STEM-2
    - OECD/NEA BIP-3

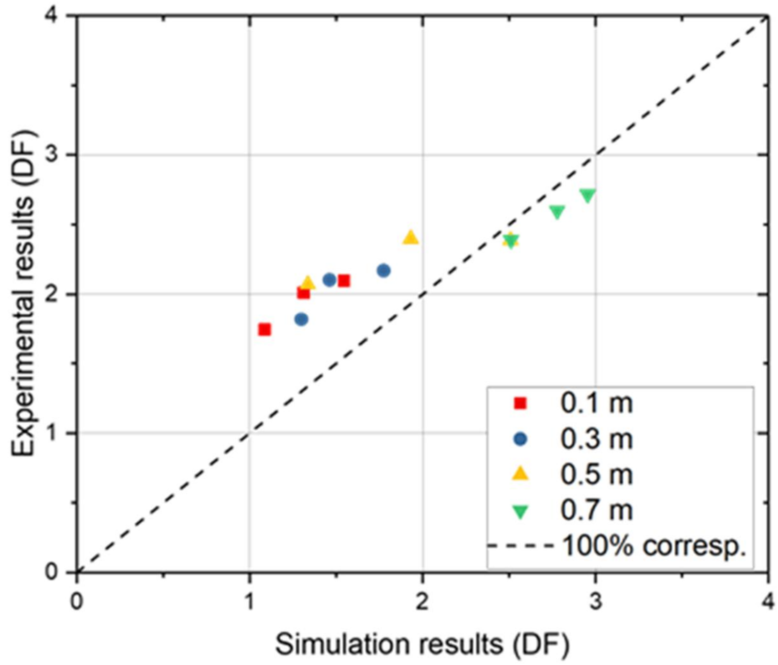

Figure 1. Comparison of experimental (MANTRA project) and simulated (ANSA project) decontamination factor (DF) results. Pool scrubbing experiments were performed with CsI aerosol at 20 °C varying the pool depth from 0.1 to 0.7 m and non-condensable N_2 flow rate through the pool from 8, 14 to 27 l/min. ASTEC simulations were in good agreement with the experimental results.

2.2.7 PAHE - Passive heat exchanger experiments

The main aim of the project is to ensure the operation and efficiency of the AES-2006 design PHRS-C passive heat removal system in accident and transient situations of nuclear power plants and to generate data for code validation. The ultimate goal is to identify physical mechanisms that can reduce performance or prevent the functioning of the loop, to help recognizing conditions in which the functioning of the system could be endangered and to suggest ways assuring the operation. Especially, flow oscillations in two-phase conditions and the effects of aerosols on the outer surface of the heat exchange tubes are topics of interest. The experiments also fulfil the experiment needs of the APROS-Fluent projects proposed by VTT.

The project consists of two work packages:

- Natural circulation tests
  - Experiments
  - TRACE model
  - Conference / journal article
- Tests for VTT projects
  - Experiment planning
  - PASI modifications
  - Experiments

Different organizations can use data in the development and validation of system and CFD codes for the safety analyses of nuclear power plants. Computer analyses are needed also in the planning of the experiments as well as in post analyses to help understanding the physics in the experiments. The results of the flow oscillations in two-phase conditions are available by the end of 2019. The other results of the project can be fully exploited after 2020.

Figure 1. PAHE focuses on improving understanding of the AES-2006 design PHRS-C passive heat removal system and to generate data for code validation.

2.2.8 PATE - PWR PACTEL tests

The main aim of the project is to ensure the operation of safety related systems or the efficiency of the procedures in accident and transient situations of nuclear power plants. An integral test facility, such as PWR PACTEL, offers a good possibility to carry out tests that supplement test campaigns in the other facilities (PKL) or make independent tests to study phenomena relevant to the safety of nuclear power plants. As a result, counterpart-like tests give information of parameter effects such as a smaller scaling ratio or a higher pressure level (PWR PACTEL/PKL) when certain operator actions or system activation set points are used. For LUT it would be beneficial to participate in also the analytical work of the experiments with the PKL facility for educational purposes using the TRACE model made in the SAFIR2018 INTEGRA project in 2018.

The main goal of the nitrogen effect tests is to map the full range of the pressures at which the decoupling of the primary and secondary side pressures takes place and to generate data for the code validation of system thermal hydraulic codes. With the tests of inadvertent opening of the pressurizer pilot operated safety valve with a simultaneous full opening of the main steam relief valves can find out if a full guillotine break can cover this event.

The project consists of two work packages:

- Participating the OECD/NEA PKL projects
  - PWR PACTEL experiments
  - Analytical work of the experiments with the PKL facility
- PWR PACTEL integral tests
  - Nitrogen effect tests
  - Inadvertent opening of SV and MSRT tests

Figure 1. PATE focuses on improving understanding of thermal hydraulic system behavior of EPR type PWRs by performing integral effects tests with PWR PACTEL

2.2.9 RACSA - Radiation shielding and criticality safety analyses

The projects aims at improving the computational tools and human expertise in radiation shielding and criticality safety problems. In the scope of the project, the concept of radiation shielding is not only related to human exposure to ionizing radiation but also to the assessment of damage caused to structural elements and electronic device by fast neutrons and electrons. The analysis capabilities are improved by developing electron transport mode to the Monte Carlo code Serpent and proper validation of previously implemented photon transport mode. Also the applicability of Serpent for dosimetry studies is improved. The capabilities to perform criticality safety analyses are improved through further validation of the codes that are to be used for the purpose.

The project consists of three work packages:

- Radiation Shielding
  - Radiation transport in Serpent
  - Variance reduction methods

- Validation of Serpent for radiation shielding
- Applications
- International co-operation

- Dosimetry Tools
  - Validation of Serpent for reactor dosimetry
  - Method development
  - International co-operation
- Criticality Safety
  - Validation package
  - Burnup credit
  - International co-operation

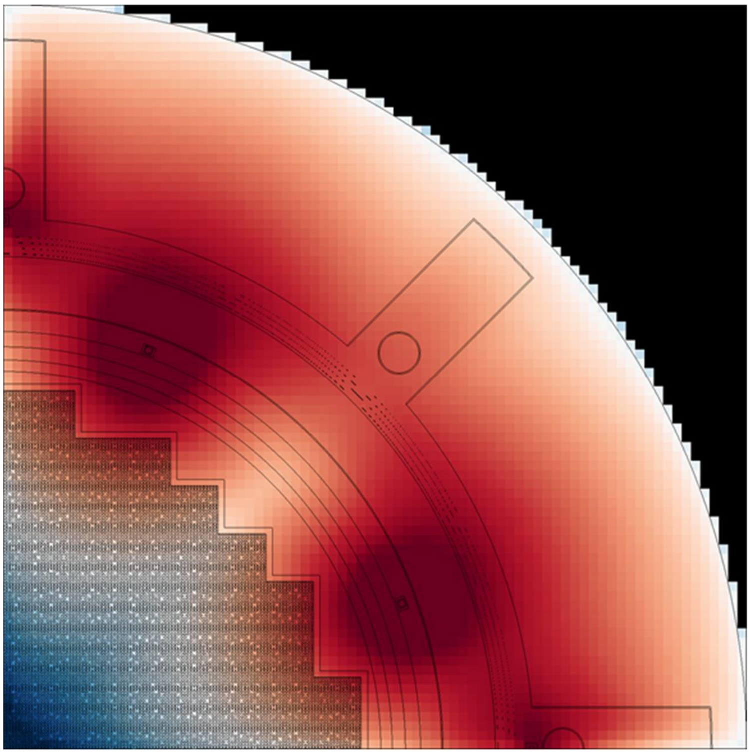

Figure 1. Fast neutron importance mesh created for a Serpent calculation assessing the fast neutron irradiation against the reactor pressure vessel.

2.2.10 SPASET - Sparger separate effect tests

To achieve the project objectives a combined experimental/analytical/computational program is carried out by LUT, KTH and VTT. LUT is responsible for developing an experimental database to be used in the validation of the simulation tools. In 2017-2018, a separate effect test facility, SEF-POOL, for studying steam discharge through a sparger pipe and directly measuring the associated force has been designed, built and taken into use at LUT.

Experiments in the SEF-POOL facility at LUT will provide necessary data to understand which characteristics of small-scale phenomena affect the effective heat and momentum sources and will thus help in the validation of the simplified EHS/EMS models as well as CFD models for steam injection through spargers.

The project consists of three work packages:

- SELF-POOL tests
  - Modifications of SELF-POOL facility
  - Tests with SELF-POOL facility

- Testing of DCC models of CFD codes
    - OpenFOAM simulation model for SEF-POOL facility
    - Pattern recognition analysis and analysis of NEPTUNE_CFD results of SEF-POOL tests
- Project management
    - Project management, Nordic co-operation and publications

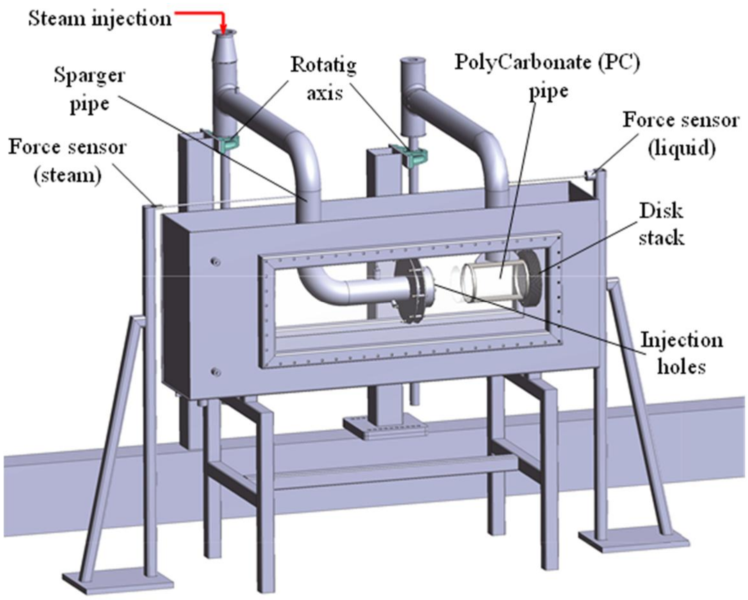

Figure 1. A small-scale separate effect test facility for sparger studies.

2.2.11 THACO - Safety through thermal-hydraulic analyses and co-operation

The main objectives of the THACO project are the improved validation and consequently increased reliability in simulation results obtained with Apros, successful competence transfer, participation in international research projects and education of new experts. Most of these objectives are achieved through thermal-hydraulic analyses performed by younger scientists in the guidance of senior experts of the field.

International and domestic cooperation forms an essential part of the project. Multiple OECD Nuclear Energy Agency's (NEA) experimental and theoretical research projects are participated. Also U.S. NRC's Code Applications and Maintenance Program (CAMP) is followed.

The project consists of two work packages:

- Thermal-hydraulic analyses
    - Master's thesis
    - RBHT characterizing experiments
    - FIX-2 reflooding analysis
    - PASI Apros-CFD coupling
    - TRACE comparison analysis
- International cooperation
    - Updating the Apros validation matrices
    - OECD/NEA HYMERES Phase 2
    - OECD/NEA PKL-4
    - OECD/NEA WGAMA
    - OECD/NEA RBHT

- o USNRC CAMP

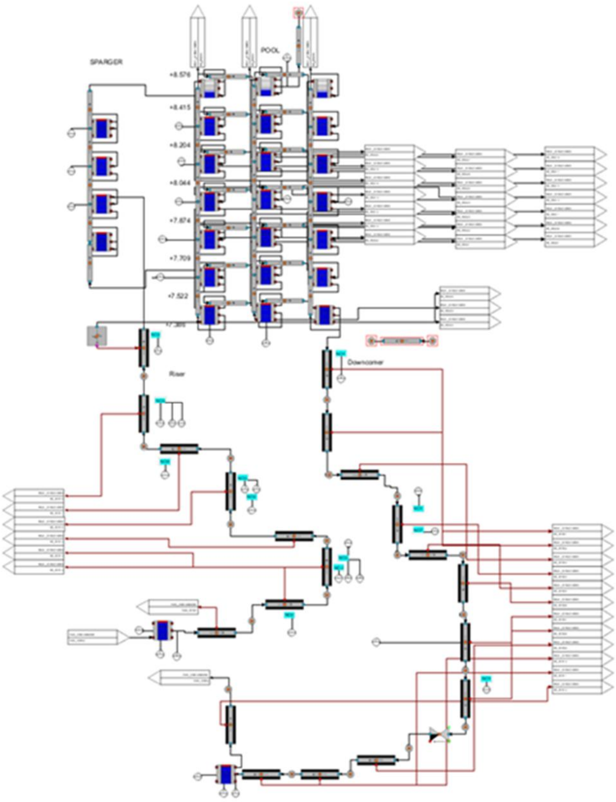

Figure 1. An Apros model was created for the new PASI experimental facility, located in Lappeenranta University of Technology. Characterizing tests have been calculated and the model is ready for the analyses of experiments that follow in the coming years.

2.3 Structural safety and materials

In 2019 the research area “Structural safety and materials” includes nine projects:

1. Additive manufacturing in nuclear power plants (AM-NPP)
2. Advanced materials characterisation for structural integrity assessment (AMOS)
3. Critical studies in support of the ageing management of NPP concrete infrastructure (CONAGE)
4. Modelling of aged reinforced concrete structures for design extension conditions (CONFIT)
5. Effect of long-term operation on aging and environmentally assisted cracking of nuclear power plant component materials (ELIAS)
6. Extended lifetime of structural materials through improved water chemistry (ELMO)
7. Fatigue and evolving assessment of integrity (FEWAS)
8. Non-destructive examination of NPP primary circuit components, machine learning and reliability of inspection (RACOON)
9. Safety criteria and improved ageing management research for polymer components exposed to thermal-radiative environments (SAMPO)

2.3.1 AM-NPP - Additive manufacturing in nuclear power plants

Additive manufacturing (AM) of metals has taken a huge step towards industrialisation over the past few years and currently the field is growing rapidly. The industrialization started with dental products and medical implants, but was soon followed by aerospace and other demanding industrial sectors. In some areas resembling the nuclear sector with extremely high quality and qualification requirements, e.g. aerospace, full-scale industrial production is close or already available. All modern aeroplanes contain parts made using AM. All the research partners affiliated to this proposal, have carried out research related to additive manufacturing for many years, having also the critical mass of researchers for cumulative knowledge.

As a digital, tool-less technology, Additive Manufacturing could improve the availability of spare parts to old equipment and installations. In addition to spare parts, additive manufacturing has also other potential benefits, e.g. repair of components and better performance of parts based on design freedom.

The objective of this project is to establish a roadmap for the use of additively manufactured components in Nuclear Power Plants in Finland with an especial emphasis on safety aspects. In addition, its applicability to spare parts will be studied.

The results from the AM-NPP project can be used by all stake-holders for an increased understanding on what the possibilities of AM are, on how quality assurance is performed using AM, where AM is currently used in the nuclear sector and what steps has been taken to approve AM parts in the nuclear sector considering different safety classes. This information is needed for the safety authority to prepare for applications by the licensees to adopt AM in repair or spare parts, for the licensees to understand which requirements shall be put on additively manufactured parts and components, and for the research community to understand the behavior of AM materials and components and to assess what additional research is needed in the future.

Figure 1. AM component example: VTT Pipe: Manifold pipe case study designed for assessment of printability and design principles. Material is AISI 316L stainless steel printed using Laser Powder Bed Fusion (L-PBF) and final connecting surfaces finished by conventional machining.

2.3.2 AMOS - Advanced materials characterisation for structural integrity assessment

The structural analysis of safety class 1 components shall be based on fracture mechanics. An emerging trend is that safety analyses are conducted without sufficient understanding of the material behavior and fracture properties. AMOS project focuses on development of analytical and experimental testing methods for fracture mechanical assessment of safety class 1 components. The goal is also to educate new experts in a field.

As a result of the project, methods and mechanistic understanding will be developed that enable more reliable life time predictions and characterisation of fracture toughness from small material volumes.

Furthermore, this enables life time predictions for NPPs operating in the long term operation regime. Plants in LTO regime have restricted amount of material available.

The goal is also to increase the knowledge and improve the tools/methods for assessing the significance of the results on a component level. Without sufficient knowledge, the safety of components cannot be assessed.

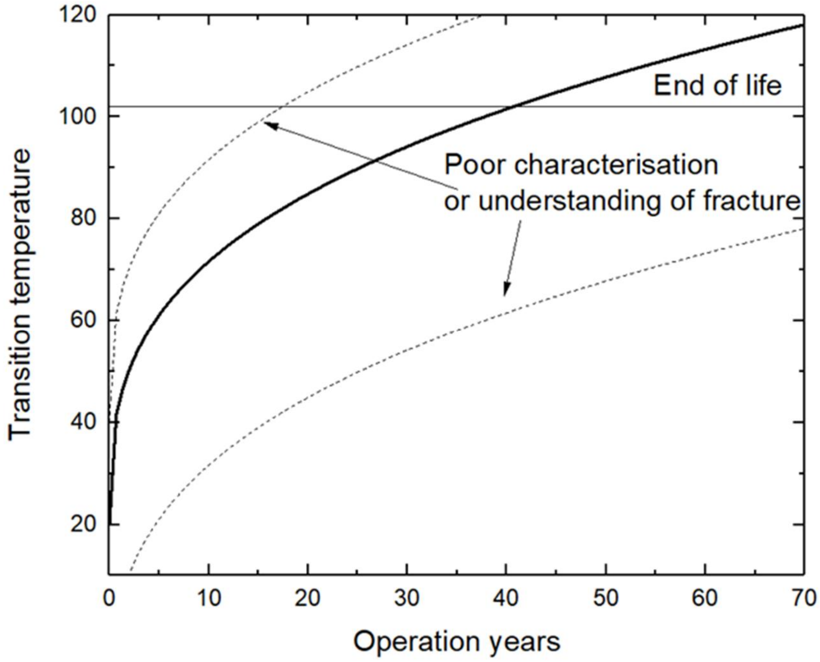

Figure 1. Significance of advanced characterization techniques on end of life predictions.

2.3.3 CONAGE - Critical studies in support of the ageing management of NPP concrete infrastructure

The main goal of the CONAGE project is to form a broader view of plant life cycle management, with particular interest in securing the necessary knowledge and expertise to support the existing Gen II plants in operation and conduct methods of validation. With respect to the use of new plants, a significant goal is to implement life cycle management programmes so that they are already complete before the plants are commissioned. Based on experiences from the currently operating plants, this has been deemed important, as ageing phenomena and the possibilities for their management must be identified already during the design stage

The project addresses special degradation related issues that affect long-term operation of NPP concrete structures. For NPP concrete structures performing multiple safety related functions, it is neither technically nor economically feasible to have them replaced. Therefore, a comprehensive understanding of all possible ageing mechanisms, their degradation processes, and the consequences for the safety function of the structure must be achieved.

The CONAGE project addresses potential ageing mechanisms and age related degradation of concrete structures, focusing on three main areas:

- Non-destructive evaluation of NPP concrete infrastructure – addresses condition assessment and inspection of concrete structures.
- Assessing the risk of internal expansive reactions for NPP concrete infrastructure – addresses degradation mechanisms such as alkali aggregate reactions (AAR) and delayed ettringite formation (DEF) of concrete structures.
- Assessing steel liner and anchor corrosion – addresses age related degradation of steel compo-nents embedded or in direct contact with concrete structures.

Figure 1. Mock-up of containment structure for NDE R&D

2.3.4 CONFIT - Modelling of aged reinforced concrete structures for design extension conditions

The protective walls, containment and civil structures of Nuclear Power Plants are mainly reinforced concrete structures. Long term operation of NPPs requires structural integrity assessment of aged concrete structures in YVL design extension conditions e.g. external hazards like earthquakes and wide body aircraft crashes. There is a need for a universal material model which is firmly based on physical phenomena and adequate for different types of numerical simulations.

The objective of the project is to develop understanding of the material modelling of concrete in the nonlinear domain by improving the use of existing material models, developing new material models and using calibration tests to stand at the basis of the theoretical model development.

The expected main result is a validated user-friendly analysis tool for reinforced concrete structures under design extension conditions, which takes the effect of ageing into account and gives more reliable results than the previous analysis tools.

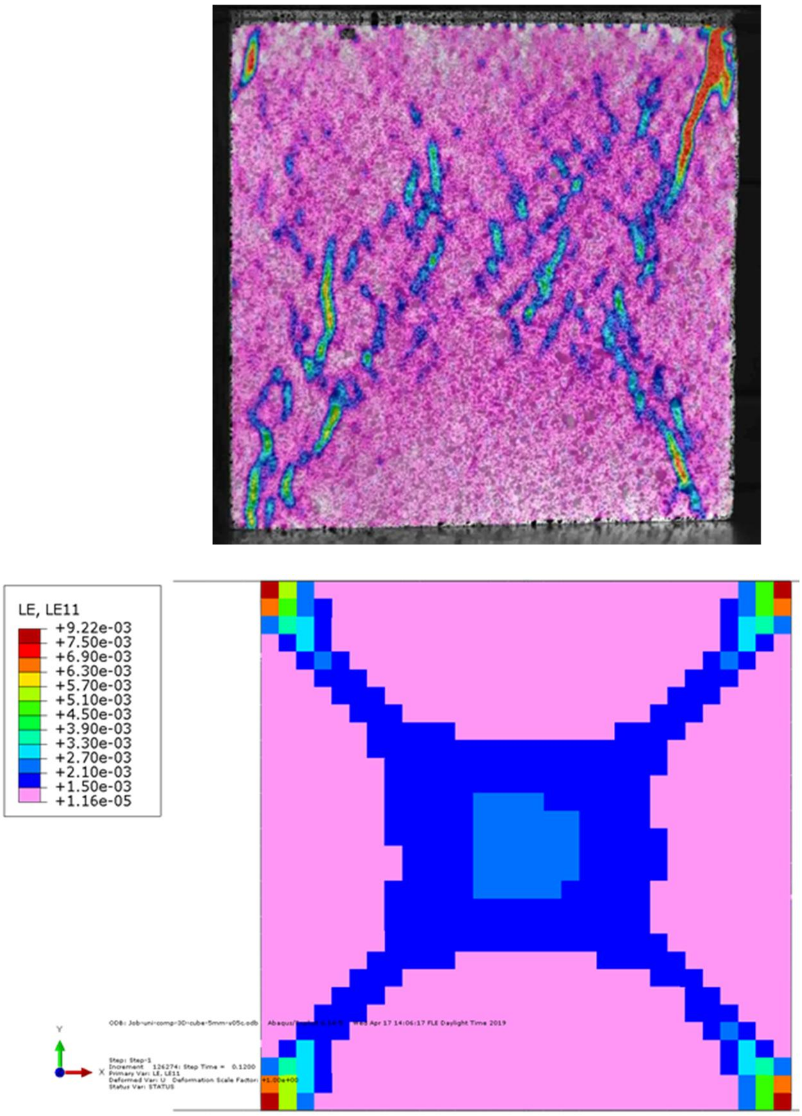

Figure 1. Horizontal strain of a compressed cube measured with Digital Image Correlation (LaVision) and simulated with Finite Element Method (Abaqus).

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

The long-term behavior of metallic nuclear power plant (NPP) components is of particular interest in Finland, as there are NPPs reaching the planning phase for lifetime extensions. Understanding materials' aging phenomena and degradation mechanisms are of the utmost importance to ensure safe long-term operation (LTO) and effective lifetime management. Some of the most important metallic material groups in NPPs are the low-alloy steels (e.g. reactor pressure vessel (RPV) steel), austenitic stainless steels and nickel-base alloys. Irradiated RPV steel and internals, environmental effects on components, Ni-base alloy stress corrosion cracking (SCC) and thermal aging of cast austenitic stainless steels have been identified as key gap areas for LTO. The same areas are mostly relevant for the Finnish nuclear power plant fleet as well. The ELIAS project will address the key material topics relevant to reliable plant operation and LTO.

The results obtained in the project will provide the necessary information to support the evaluations for the safe and long-term operation of nuclear power plants from the structural materials' performance point of view. The results will also be exploited by VTT in root cause analyses performed for the licensees and in expert statements on materials' behavior in nuclear environments for both STUK and the licensees.

The project consists of five work packages:

- Thermal aging
- Repair welding
- Austenitic stainless steels in NPPs
- Identification of embrittlement mechanisms in RPV steel
- International activities

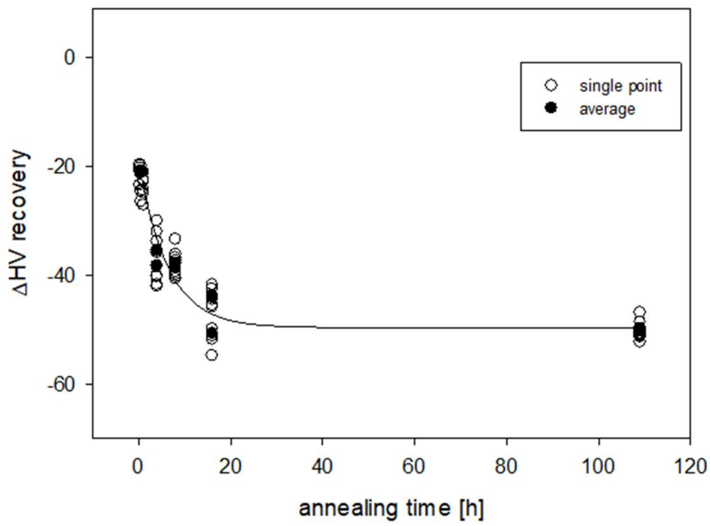

Figure 1. Effect of annealing on measured hardness of an irradiated RPV weld material.

2.3.6 ELMO - Extended lifetime of structural materials through improved water chemistry

Mitigation of different corrosion phenomena of the structural materials of nuclear power plants is essential as safe use of nuclear power is considered. Since corrosion is based on the electrochemical interaction of the environment and the structural material, the most convenient way for the operating plants is to optimize the water chemistry to minimize the effects of different corrosion phenomena. Several water chemistry related issues have been identified within this project that require more detailed study, including replacement chemicals for hydrazine, impurity enrichment and release in steam generators (SG), lead assisted stress corrosion cracking (PbSCC) and small modular reactor (SMR) water chemistries.

The main objective of the project aim at developing knowledge on optimal PWR/VVER water chemistry programs. The expected outcome is to improve the knowledge basis on which decisions on advanced water chemistries are made, e.g. plant life extension management and safety programs.

The project consists of six work packages:

- Hydrazine replacement in PWR/WWER
- Impurity enrichment in steam generator
- Optimal water chemistry for pre-passivation of AES 2006/VVER 1200 plants
- Mitigation of PbSCC
- Small and modular reactors (SMR) - water chemistry
- International cooperation

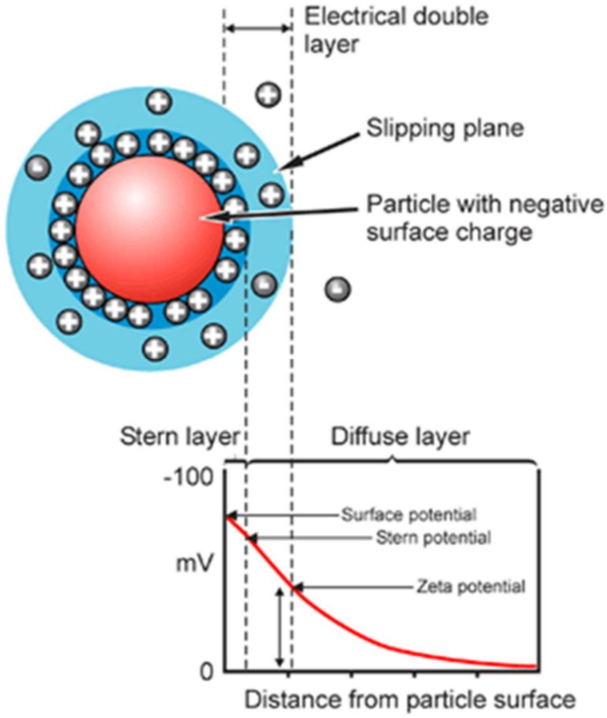

Figure 1. Formation of deposits may cause several different corrosion related issues.

2.3.7 FEWAS - Fatigue and evolving assessment of integrity

The emphasis of the FEVAS project is the research and development of the methods to assess the structural integrity of the primary circuit and the uncertainties and conservatisms included in these methods.

The project aims to improve the mechanical understanding of three phenomena:

§ The effects of primary water environment on the fatigue usage of the primary circuit

§ Development and optimization of repair welding techniques and calibration of welding simulations

§ Methods to assess the piping response of pressure pulsations and flow mixing loads

The project consists of three work packages:

- Piping assessment
  - Acoustic and structural vibrations
  - Thermally induced cracking
- Fatigue usage of primary circuit
  - EAF
  - Fen model
- Repair welding
  - Qualification of modelling
  - Planning of repair weld mock-up
  - Welding procedure tests and mock-up manufacturing
  - Residual stress measurements

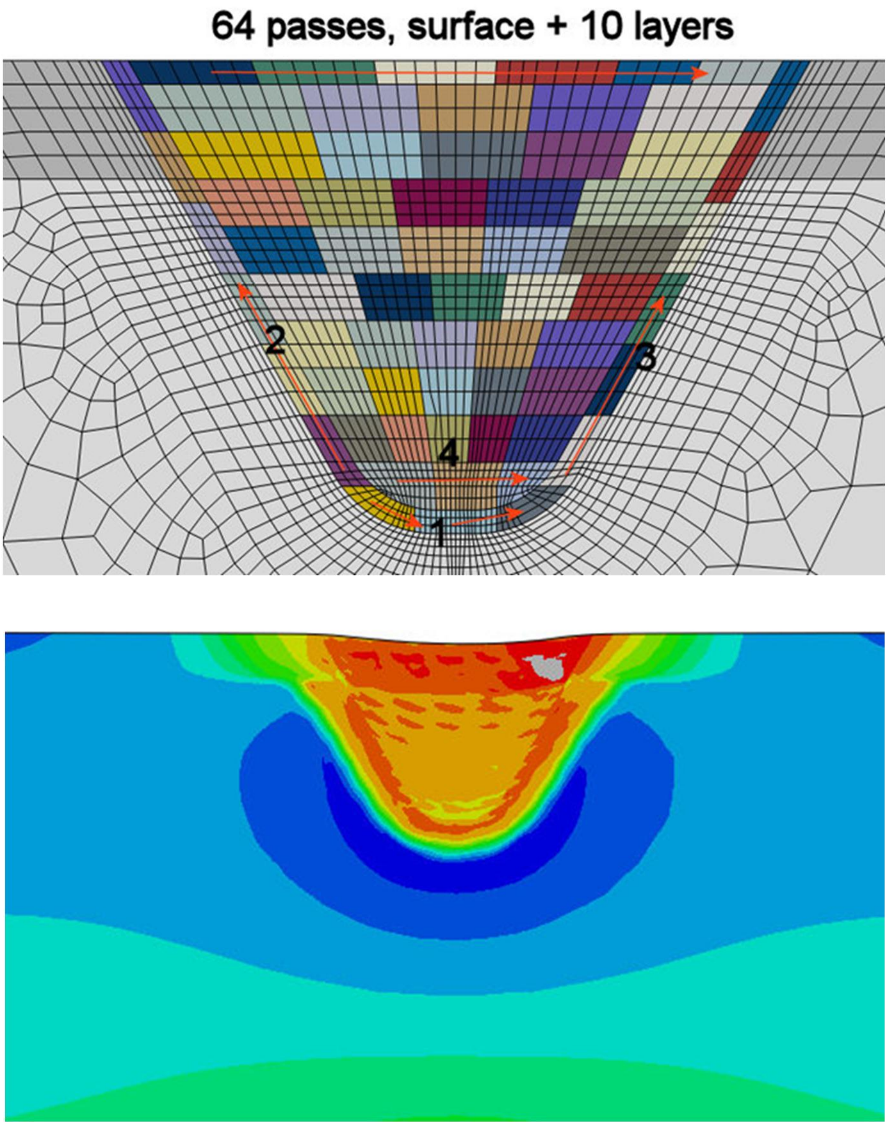

Figure 1. Bead placement and simulated residual stress distribution in a RPV repair weld mock-up.

2.3.8 RACOON - Non-destructive examination of NPP primary circuit components, machine learning and reliability of inspection

Long-term operation (LTO) of a NPP requires the periodic renewal of the operating license. In addition, the initial 40-year design life has been based on economic and antitrust considerations – not on limitations of nuclear technology itself. In-service inspection (ISI) programs play an important role in demonstrating sufficient structural integrity of materials and components and guarantee the structural safety, to ensure continued NPP operation in a reliable and safe manner. ISI has an important role in identifying adverse environmental loadings or ageing factor effects before they potentially deteriorate structures compromising the safety of an NPP. To date, no proper quantitative method to evaluate the effectiveness and the reliability of an NDT method in ISI of NPPs. When the exact capability of an NDT technique is known, ISIs can be planned even more reliably and cost efficiently.

The project consists of two work packages:

- Probability of Detection
    - POD augmented data

- Simulation of novel NDE
    - Defect response simulation
    - Preliminary study on NDT opportunities for material degradation

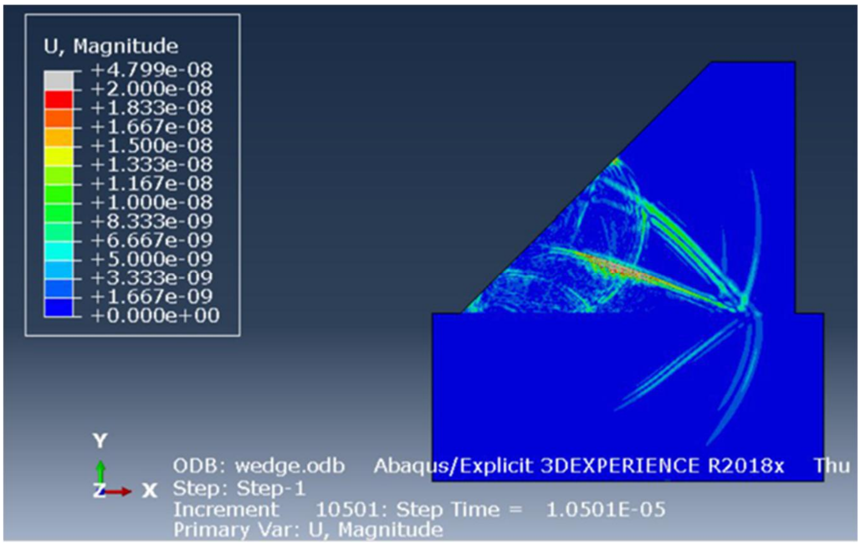

Figure 1. FEM simulation of ultrasound propagation.

2.3.9 SAMPO - Safety criteria and improved ageing management research for polymer components exposed to thermal-radiative environments

SAMPO project focuses on safe long-term use of polymer components and improving their ageing management. This is done by studying ageing mechanisms in thermal-radiative environments, determining how to set acceptance criterion properly and providing robust tools for condition monitoring. These topics become more relevant as the original planned lifetime of the plants is approaching and extension is considered. Safety criteria assessment and ageing management needs to be at sufficient level in order to prevent premature component breakdown and avoiding endangering the overall safety. In this project both experimental data and modelling are combined to achieve the project objectives.

The project consists of three work packages:

- Acceptance criteria and safety margin assessment
    - Improved estimation for lifetimes of critical polymer components
    - Sensitivity of polymer properties to additive content and methods to verify polymer quality
    - Setting up safety margins for O-rings
    - Ageing mechanisms of polymers in NPP containments
- Improvements in ageing management of polymer components
    - Online condition monitoring techniques
    - Sensitive analysing techniques
    - Improved interpretation of non-destructive testing data
- International Cooperation
    - Polymer ageing in NPP applications – event organization

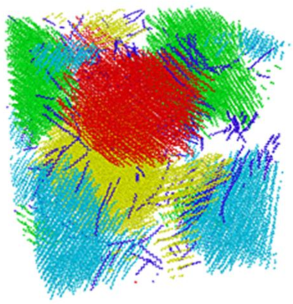

Figure 1. Partial crystallization at isothermal conditions.

2.4 Research infrastructure

In 2019 the research area “Research infrastructure” includes four projects:

1. Barsebäck RPV material used for true evaluation of embrittlement (BRUTE)
2. Infrastructure development at LUT safety research laboratory (IDEAL)
3. Participation in Jules Horowitz Reactor project - towards first criticality in 2022 (JHR2022)
4. Pre-emptive reduction of radiological laboratory legacy waste (LABWAST)

2.4.1 BRUTE - Barsebäck RPV material used for true evaluation of embrittlement

The BRUTE excellence project pioneers the new infrastructure of the VTT Centre for Nuclear Safety (CNS) hot cells, and determines the properties of Reactor Pressure Vessel (RPV) material after thermal ageing and neutron irradiation. All methods used are verified and commissioned in the new CNS hot cell environment.

Mechanical tests, e.g. impact, fracture toughness and tensile tests are performed for determination of the thermal and neutron embrittlement in high-Ni reactor pressure vessel weld metal, typical for Nordic RPVs.

The microstructure is characterised to get an improved understanding on factors affecting brittle fracture and embrittlement.

The results are used to compare the results from surveillance programs, used for assessment of the embrittlement using specimens manufactured at the time of the RPV manufacturing with results from material from the component itself.

The data is used to update the existing prediction curve(s) for embrittlement and improve the understanding of both neutron and thermal embrittlement.

The assessment of the RPV integrity, ensuring that the embrittlement is within acceptable range, is a corner stone for nuclear safety.

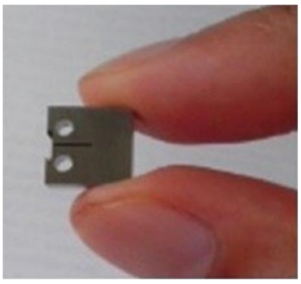

Figure 1. Test specimen cut from the Barsebäck Reactor Pressure Vessel.

2.4.2 IDEAL - Infrastructure development at LUT safety research laboratory

The objective of the IDEAL project is to develop and maintain the experimental thermal hydraulic infrastructure at LUT University nuclear safety research laboratory. This comprises the development of advanced instrumentation, maintenance of the existing test facilities, assembly and development of the new modular integral test facility, MOTEL, and upgrading of the process control and computational systems.

As a result of the project, LUT will have a safety research laboratory equipped with well-maintained test facilities, state-of-the-art thermal hydraulic measurement capabilities, upgraded process control systems, as well as a new integral test facility to be used for various integral and separate effects tests.

The project consists of five work packages:
- Development of instrumentation
- Maintenance of test facilities
- Process control and computational systems
- Development of the MOTEL test facility
- International co-operation

Figure 1. The first configuration of the MOTEL test facility.

2.4.3 JHR2022- Participation in Jules Horowitz Reactor project - towards first criticality in 2022

Over the past forty years, materials testing reactors (MTR) in Europe have provided essential and invaluable support to the nuclear power plant (NPP) community. As the existing fleet of MTRs continues to age, they will face an increasing probability of shut-down due to outdated safety standards and experimental capabilities that are no longer able to respond to today's increasing demands and requirements.

The Jules Horowitz Reactor (JHR) is currently under construction. The JHR will become a major part of European nuclear research infrastructure (NRI) in the coming years. Finland is participating in the construction with a 2 % in-kind contribution. The JHR is designed to (i) provide a high neutron flux, (ii) run highly instrumented experiments, (iii) support advanced modeling needs and (iv) operate experimental devices capable of simulating the environment, in terms of coolant chemistry, pressure, temperature and neutron flux, and (v) respond to the experimental needs of water reactors, gas cooled thermal or fast reactors, sodium fast reactors, etc. As successful irradiation campaigns require planning, sample preparation, transfer of the samples and rig configuration prior to performing the actual test, the first experimental projects must be planned and prepared for during the span of SAFIR2022.

The JHR MTR is an invaluable infrastructure for the international nuclear community. The results obtained in this project from will be used as guidelines for preparing and planning in-core nuclear fuel and materials test campaigns of national interest.

The project consists of three work packages:

- JHR Working Groups
- Materials Investigations
- Nuclear Fuel

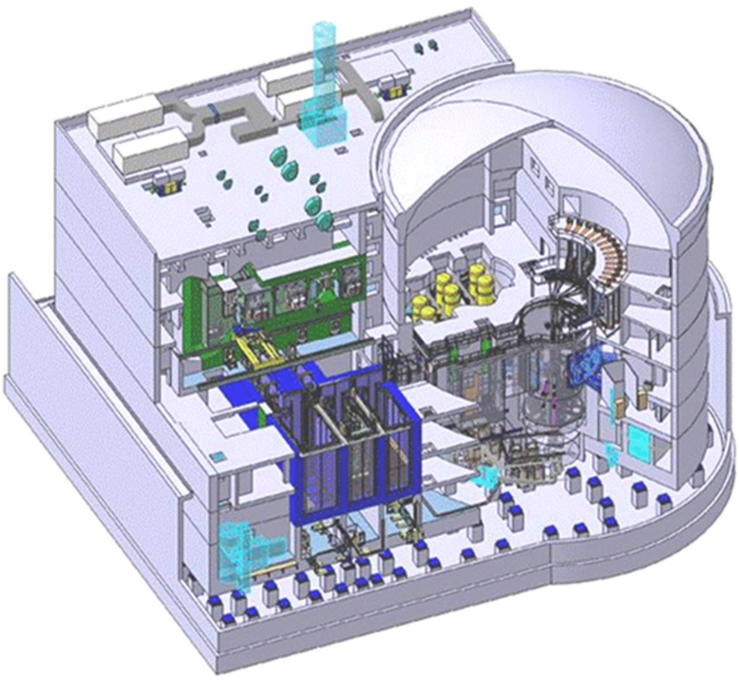

Figure 1. The Jules Horowitz Reactor, currently under construction at the CEA (France).

2.4.4 LABWAST - Pre-emptive reduction of radiological laboratory legacy waste

The radioactive materials research infrastructure renewal, initiated a decade ago, has proceeded well during the RADLAB project. The new VTT Centre for Nuclear Safety and its new hot cell facilities are licensed for operation, and the functionality of in-cell equipment has been demonstrated. An important aspect in utilizing the infrastructure for research and testing of radioactive materials, is the inevitable production of waste containing activated material. In the current state-of-the-art, waste is either returned to the “owner” (i.e. one of the domestic NPPs), or else transferred to interim storage.

The LABWAST project is comprised of two major aspects: conclusion of work from the RADLAB project in the infrastructure renewal, and taking actions to minimize the accumulation of legacy waste. The long-term sustainability of conducting research and testing of radioactive materials without creating a legacy waste problem is possible by improving methods for (re)utilizing smaller specimens of radioactive materials for research purposes, and by developing advanced waste handling methods to substantially reduce the volume of waste needing to be sent out of the facilities for interim storage.

The project consists of three work packages:

- Legacy waste mitigation
- Advancements in hot cell testing
- Advancements in hot radiochemistry

Sustainable rad waste handling enables a much broader nuclear safety research portfolio. This is particularly important for international cooperation utilizing the new hot cells. Finland benefits from being very active in international nuclear safety programmes such as the IAEA, OECD NEA, Euratom and NKS. When the Jules Horowitz research reactor is commissioned the VTT hot cell facilities is one of the key places for conducting post irradiation examinations (PIE) of irradiated structural materials.

Figure 1. The new VTT Centre for Nuclear Safety has hot cells for research and testing of radioactive structural materials from nuclear power plants. The LABWAST project develops methods to minimize the risk of a legacy waste problem emerging in the new facilities.

3. Financial and statistical information

The planned total budget of the research projects part in 2019 is 6,7 M€. The major funding organisations are VYR with 4,38 M€, VTT with 1,32 M€, LUT with 0,23 M€, Aalto with 0,14 M€, Halden Reactor Project with 0,07 M€, NKS with 0,16 M€, and SSM with 0,08 M€ (Figure 3.1). Funding from the KYT2022 programme is 0,09 M€ (LABWAST project) and from other organisations 0,34 M€ (including direct funding of some projects by TVO, Fennovoima and Fortum). The volumes of SAFIR20122 research projects in 2019 are shown in Table 2.1 (Chapter 2). The total volume of the programme in 2019 is planned to be 43 person years. The personnel costs make up the major share of the planned expenses (Figure 3.2).

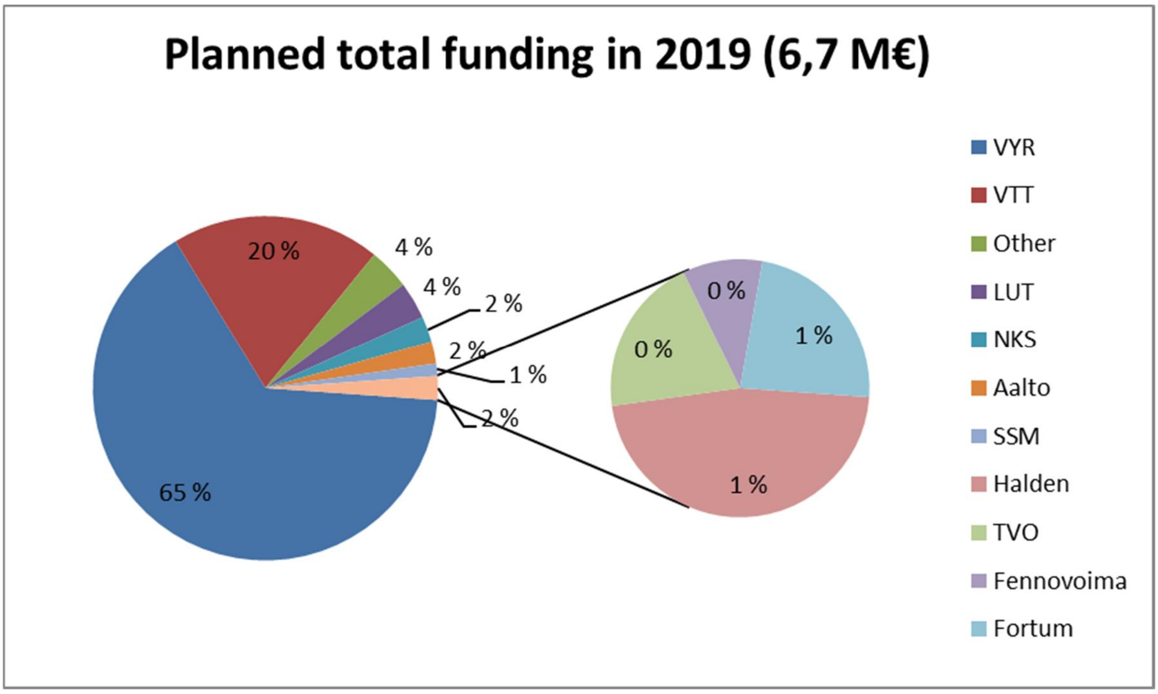

Figure 3.1. Funding of the SAFIR2022 programme in 2019.

Figure 3.3 shows the VYR and total funding for the projects of the research areas of the SAFIR2022 steering groups SG1-SG4:

SG1	Overall safety and systemic approach to safety
SG2	Reactor safety
SG3	Structural safety and materials
SG4	Research infrastructure.

Figures 3.4 and 3.5 illustrate the distribution of the total and VYR funding for the above research areas, respectively.

Figure 3.6 shows the planned person years in 2019 of the projects in the research areas SG1-SG4 and Figure 3.7 in the reference group areas RG1-RG8:

RG1	Overall safety and organisation
RG2	Plant level analysis
RG3	Reactor and fuel
RG4	Thermal hydraulics
RG5	Mechanical integrity

RG6	Structures and materials
RG7	Severe accidents
RG8	Research infrastructure.

The costs related to experimental equipment, materials and external services are reflected in the smaller shares of person years versus shares of funding in the research areas SG3 and SG4 (Figures 3.4 and 3.6). Membership fees for several international projects are funded in SG2 area.

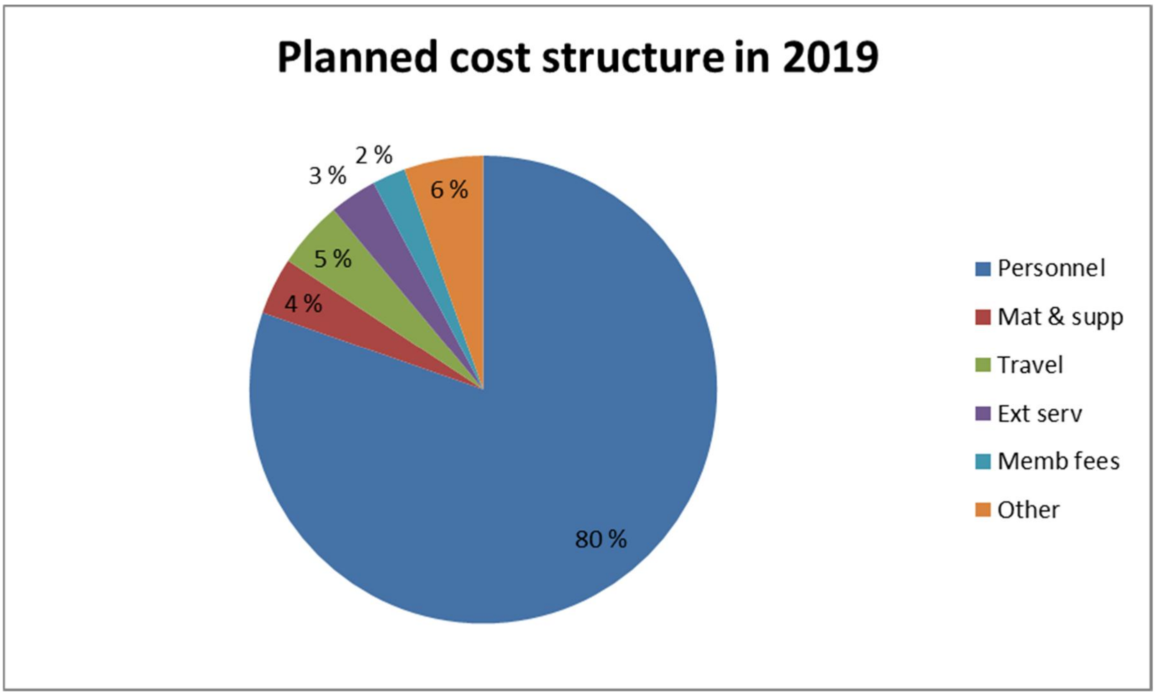

Figure 3.2. Cost structure of the SAFIR2022 programme in 2019.

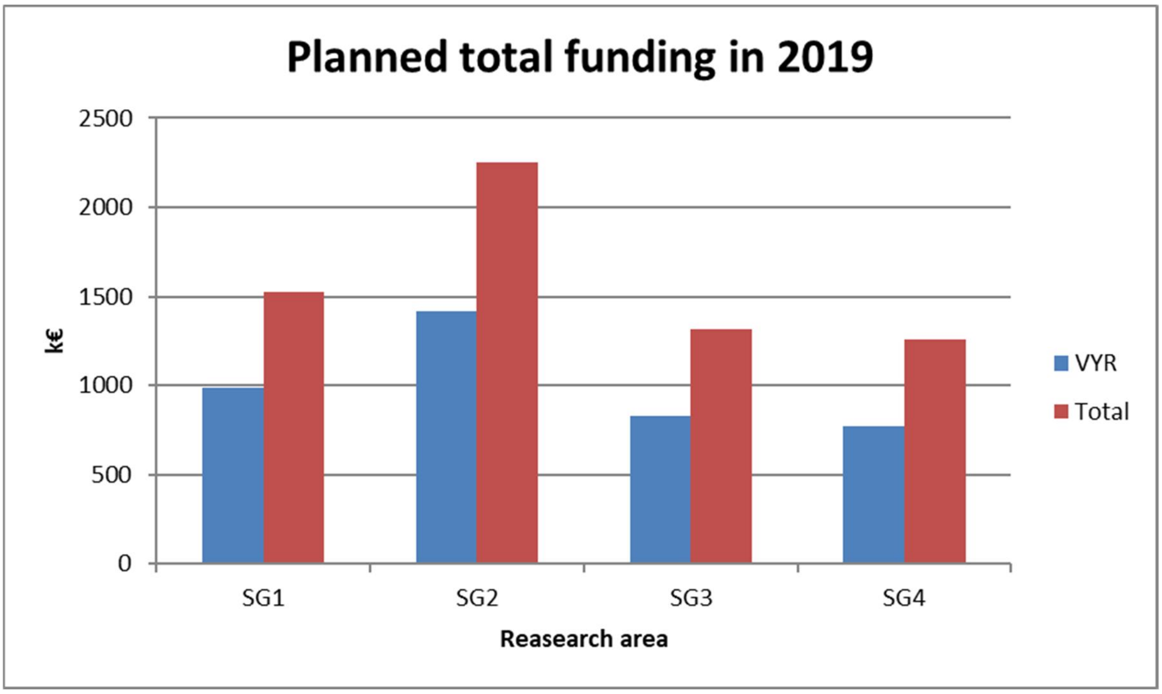

Figure 3.3. VYR and total funding for SAFIR2022 research areas in 2019.

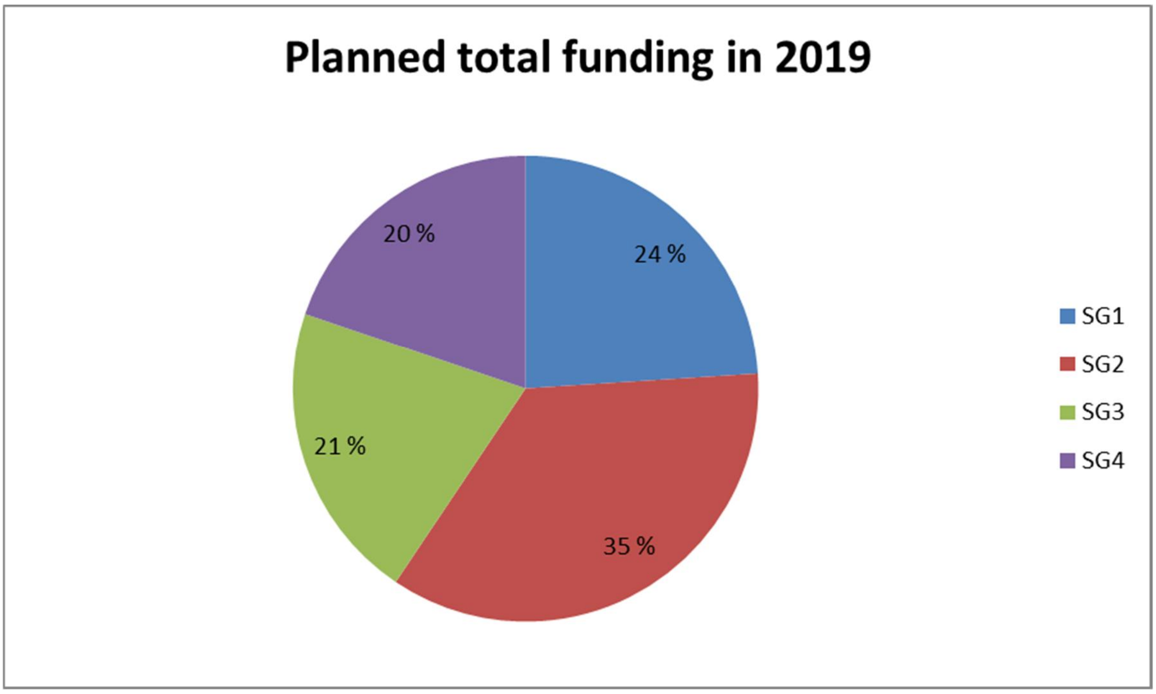

Figure 3.4. Total funding for SAFIR2022 research areas in 2019.

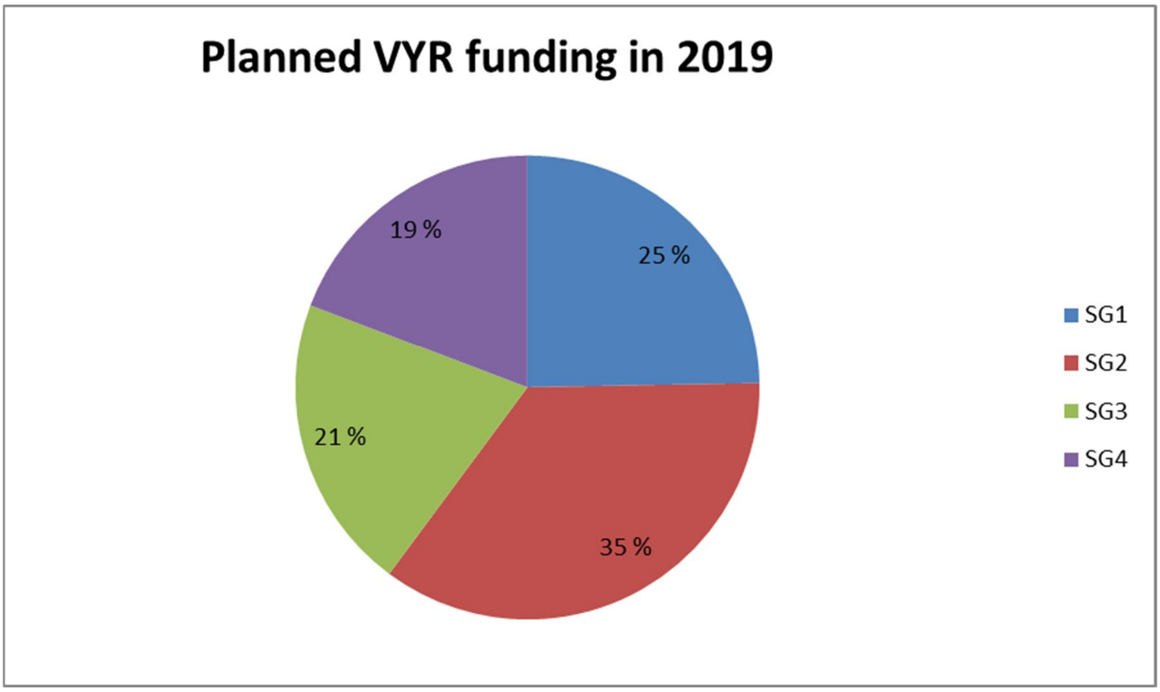

Figure 3.5. VYR funding for SAFIR2022 research areas in 2019.

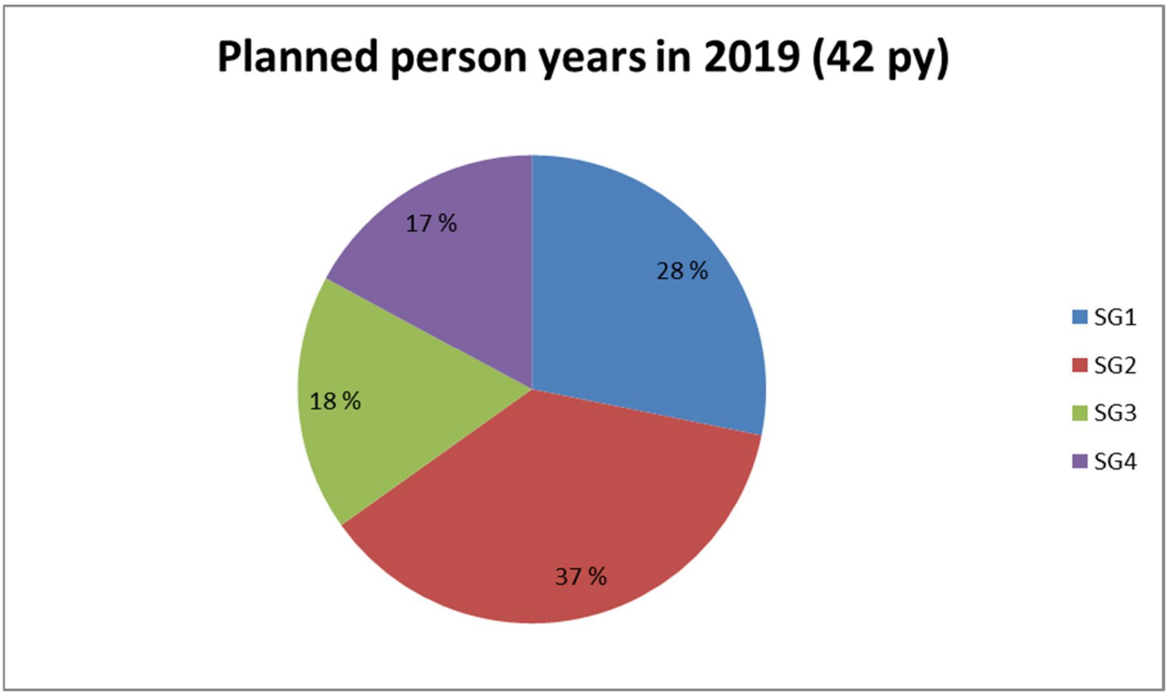

Figure 3.6. Planned person years in SAFIR2022 research areas in 2019.

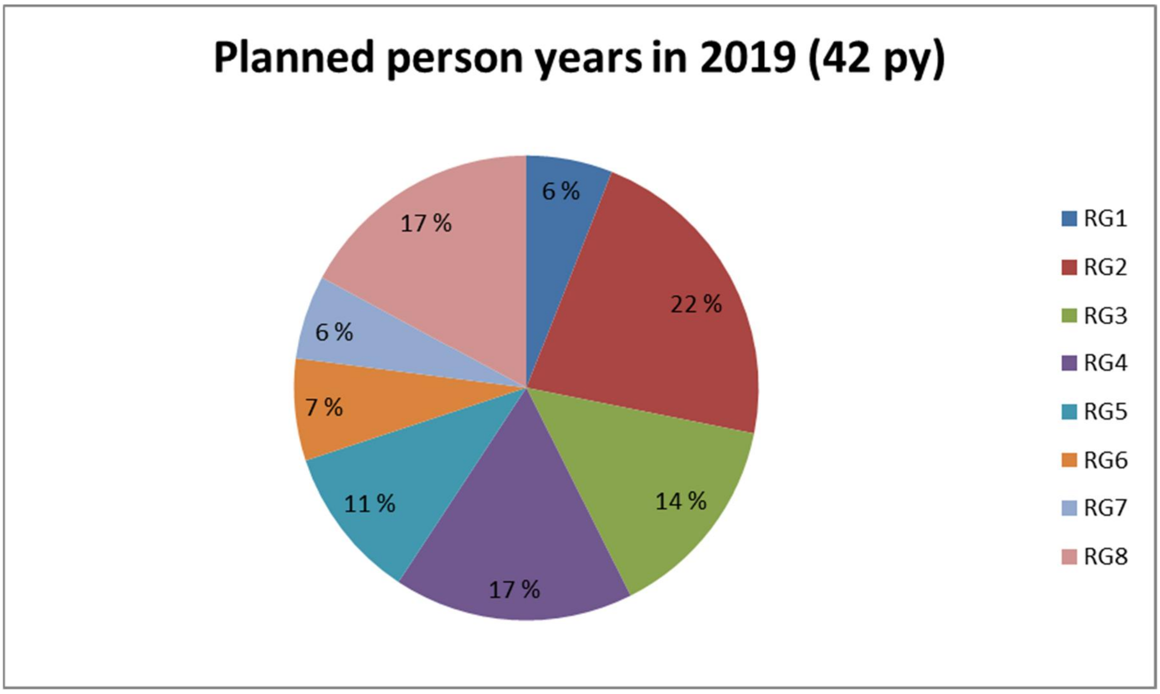

Figure 3.7. Planned person years in SAFIR2022 reference group areas in 2019.

4. Organisation and management

SAFIR2022 organisation is shown in Figure 4.1 and its function described in the SAFIR2022 Framework Plan [1]. It is also summarised on SAFIR2022 website: http://safir2022.vtt.fi/ .

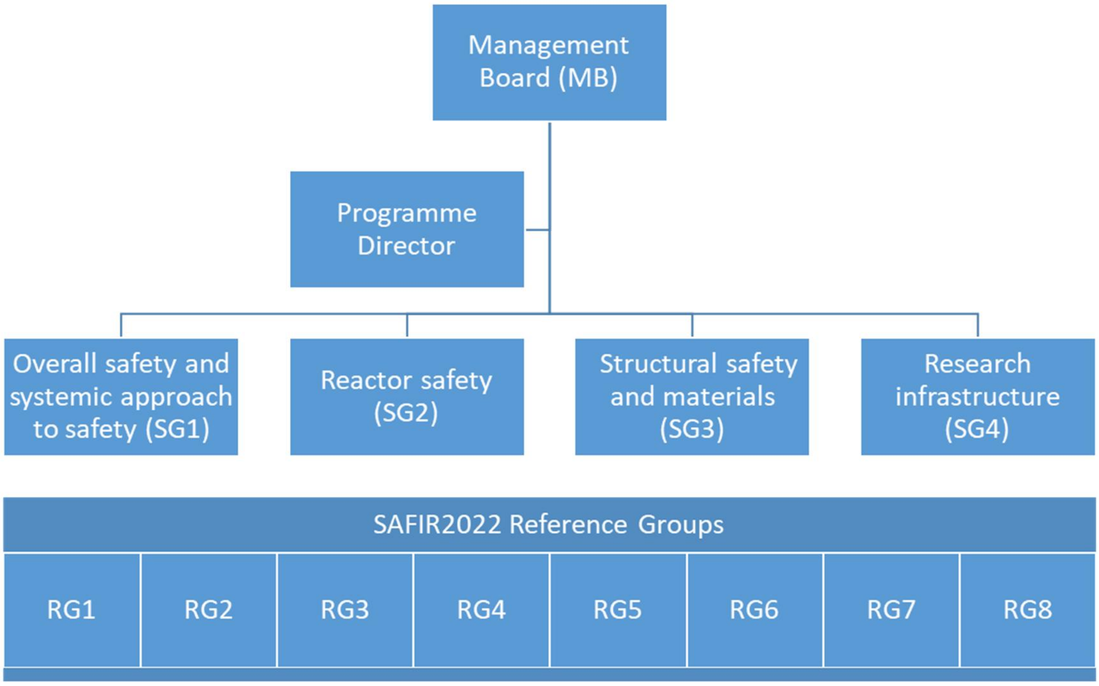

Figure 4.1. Structure of SAFIR2022 organisation. Each project belongs to one of the research areas SG1-SG4 and is quided by one of the reference groups (RG1-RG8).

The programme management bodies, the Management Board (MB), the steering groups (SG) and the reference groups (RG), have meetings on a regular basis. The steering and reference groups have at least three meetings annually for guiding the projects and following the progress of work. The MB also has at least three meetings during the year. Project ad hoc meetings are encouraged for discussing specific topics of the projects. The persons involved in the MB are listed in Appendix 1.

The MB can annually initiate small preliminary type studies with the order procedure. Decisions on the small projects are made after the funding decisions for the actual call for proposals. The small projects support the implementation of the SAFIR2022 Framework Plan in topics where actual research projects have not been started and they can also introduce new topics. The costs of the small projects are included in the budget of the administration project (ADMIRE). In 2019, four small project will be realised:

1. Artificial intelligence powered non-destructive inspection and evaluation (ANDIE)
2. CLADS Experimental facility
3. Leadership and safety culture (LESCU)
4. Thermal aging of alloy 690 and high-Ni alloys (TAAN).

Information about research carried out in SAFIR2022 is communicated by the progress reports of the projects for the reference group meetings, the annual plans and reports of the programme, and SAFIR2022 website (public site and protected extranet). Additional information is trasferred in the reference group and other meetings and seminars organised by the projects. The detailed scientific results are published as articles in scientific journals, conference papers, and research reports.

References

1. National Nuclear Power Plant Safety Research 2019-2022. SAFIR2022 Framework Plan. Publications of the Ministry of Economic Affairs and Employment 22/2018. http://safir2022.vtt.fi/

2. Hämäläinen, J. & Suolanen, V. (eds.) SAFIR2018 – The Finnish Research Programme on Nuclear Power Plant Safety 2015-2018. Final Report. VTT, Espoo, 2019. VTT Technology 349. 498 p. ISBN 978-951-38-8682-0. http://safir2018.vtt.fi/

Appendix 1 – SAFIR2022 Management Board Members in 2019

Organisation	Member
STUK	Marja-Leena Järvinen (Chair) Tomi Routamo (Vice chair)
Aalto	Simo Hostikka, Filip Tuomisto
Fennovoima	Pekka Viitanen, Tuire Haavisto
Fortum	Matti Kattainen, Satu Sipola
LUT	Heikki Purhonen, Heikki Suikkanen
MEAE	Jorma Aurela, Jaakko Louvanto
TVO	Arto Kotipelto, Antti Tarkiainen
VTT	Petri Kinnunen, Eila Lehmus
SAFIR2022 (Secretary)	Jari Hämäläinen