Final Seminar of the SAFIR2022 and KYT2022 Research Programmes 23.-24.1.2023

Project abstracts of Kaleva meeting room in alphabetical order

SAFIR2022 - The Finnish Research Programme on Nuclear Power Plant Safety 2019 - 2022

KYT2022 - Finnish Research Programme on Nuclear Waste Management 2019 - 2022

Other abstracts can be found on the Lumituuli meeting room abstracts file

AM-NPP - Additive manufacturing in nuclear power plants

Alejandro Revuelta, Tuomas Riipinen, Konsta Sipilä, Tuomas Koskinen

VTT Technical Research Centre of Finland

AMNPP has increased the knowledge of Finnish stakeholders regarding the use of Additive Manufacturing (AM), in particular Laser Powder Bed Fusion (PBF-LB), thus supporting the safe use of additively manufactured 316L components in the nuclear sector. As a relatively new manufacturing process, the first stages of the project focused on creating a roadmap on the use of the technology in Finland based on the landscape of existing standards as well as improving the understanding of the stakeholders on design issues and limitations. Additionally, the project has studied the expected level of quality which can be achieved for the reference material and demonstrated different quality control methods which can support the certification of AM parts for critical applications. Finally, the effect of novel higher temperature solution annealing heat treatments on the mechanical and stress corrosion cracking properties has been studied.

AMOS - Advanced materials characterisation for structural integrity assessment

Laura Sirkiä[1], Sebastian Lindqvist[1], Juha Kuutti[1], Pentti Arffman[1], Antti Forsström[1], Timo Veijola[2]

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The AMOS project focuses on development of numerical and experimental testing methods in fracture mechanics for the purpose of structural integrity assessments. Five main topics were identified: 1) development of quality criteria for miniature C(T) specimens, 2) development of improved methods for assessing the constrain effect on transition temperature, 3) development of verified and validated methods for determining the effect of transient loads in the upper shelf region to fulfil YVL requirements, 4) validation of crack arrest testing techniques for small specimens, and 5) methods for applying fracture toughness in structural integrity analyses and aging management.

ANSA - Analytical severe accident research

Anna Korpinen, Tuomo Sevón, Sara Ojalehto, Veikko Taivassalo, Mikko Ilvonen

VTT Technical Research Centre of Finland

The objective of the project was to develop national competence in the area of severe accidents and ensure that the analytical tools and methods in use are validated in their intended purposes and developed further if necessary.

Fukushima accident provides a unique opportunity to get more information on the progress of severe accidents. Adding water level measurement system to the models indicated a leak of superheated steam to the drywell. Stratification of the suppression pool has major effect on the containment pressure. Sectoral nodalisation, with at least three layers, produced good results for annular geometry.

Hydrogen combustion constitutes a risk to the containment integrity. VTT has developed a model for the flame wrinkling factor that improves the modelling reliability. Pool scrubbing is an important mechanism in mitigating fission product release. Analytical results indicate need of model development especially for small particles, in jet regime and with gaseous iodine retention.

Understanding of dispersion and exposure phenomena is needed for all-encompassing nuclear safety. Sensitivity studies provide further information on how input data and modelling assumptions affect the results.

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

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We investigated the Joint Cognitive System (JCS) between an operator and nuclear power plant process systems from the perspective of human-system interfaces (HSIs), procedures, operators' resources for action and skills training. Our goal was to advance and deepen our understanding of operator practices and cognitive processes in complex incidents and severe accidents and develop tools and methods for the analysis of simulator data and/or promotion of resilience skills assessment and skills training. The aim was also to better understand how the JCS is shaped and evolved, and how cognitive readiness and resilience skills are acquired. We found among others that skillful use of operating procedures is a key to successful performance in simulated incidents, field operator training has to be better tailored to their needs, virtual training environments are especially suitable for training of rarely performed operator tasks, and some new types of Human and Organizational Factors' challenges may emerge in flexible operation of nuclear power plants. Methods (e.g., the Functional Resonance Analysis Method, FRAM) and tools (e.g., part-scope simulator) were introduced and tested, and they showed to be valuable for HSI and procedure design and simulator training.

BRUTE - Barsebäck RPV material used for true evaluation of embrittlement

Sebastian Lindqvist, Noora Hytönen, Jari Lydman, Laura Sirkiä, Jussi Leporanta, Jarmo Saarinen, Mika Jokipii, Johanna Lukin, Pentti Arffman

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The objective of BRUTE is twofold; 1) to perform mechanical and microstructural investigations of Barsebäck 2 BWR reactor pressure vessel (RPV) welds and surveillance weld, enabling validation of embrittlement predictions based on the RPV surveillance programme 2) to pioneer the new infrastructure in the Centre for Nuclear Safety, CNS, VTT. The B2 plant was operated for 27 years.

During the project, the new CNS infrastructure key-processes were accredited, round robin investigations were performed, the personal got experience with standardized testing of irradiated materials, specimen preparation inside hot-cells and the whole material flow, best practice guidance was established, and tools to ease specimen handling were prepared.

The results confirm that the surveillance results describe the mechanical behavior of the reactor pressure vessel. There are no significant changes in mechanical properties after 27 years of operation. The axial beltline weld has higher toughness and lower strength than the other investigated welds which is possibly caused by variations in thermal histories.

CATS - Coupled analysis of transient scenarios

Rebekka Komu, Riku Tuominen, Jussi-Pekka Ikonen, Ville Valtavirta, Ville Hovi, Veikko Taivassalo

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System code TRACE was adopted into use for transient analyses by calculating several benchmarks alone and together with a 3D neutronics code. TRACE was integrated with VTT's computational framework Kraken for future safety analyses, and couplings with nodal neutronics code Ants and fuel solver SuperFINIX were implemented and tested. VTT's tool Sensla was utilized for uncertainty and sensitivity analyses, and it was transferred to Kraken to be used within the framework.

A PWR core with a blocked fuel assembly was analysed with a two-phase porous CFD simulation. The results demonstrate the usefulness of porous CFD in coupled transient analysis applications. The methodology was also used within the Kraken framework with a coupled Ants-SuperFINIX-OpenFOAM simulation.

CFD4RSA - CFD methods for reactor safety assessment

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The overall objective of the CFD4RSA project has been to improve the usability and reliability of Computational Fluid Dynamics (CFD) calculations in nuclear Reactor Safety Assessment. The work consisted of five Work Packages, where topics important in safety assessment were studied. First, coupled CFD-Apros calculations were validated against experiments performed at the LUT University. Second, uncertainty quantification methods in CFD calculations were tested and taken into use. Third, coarse-mesh CFD models were developed for reactor pressure vessel and validated against international benchmark results. Fourth, CFD models for fuel rod bundles of boiling water reactors were validated against available experimental results. Fifth, thermal stratification in pressure suppression pools of BWRs were studied and Apros models for stratification were investigated.

CNS - VTT Centre for Nuclear Safety

Wade Karlsen

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The infrastructure renewal of the radiological research infrastructure hosted by VTT is embodied in the new VTT Centre for Nuclear Safety. Funded in large part by SAFIR and KYT, in the ten years since the decision was made to renew the research infrastructure for radioactive materials at VTT, the new facility has been designed, built, equipped and successfully taken into use in support of nuclear safety research in Finland. While the facility can be considered fully underway, the investment aid provided through RADCNS is stated to continue through 2025. As a research laboratory, the facility continues to evolve going forward as well, thanks to continuous additional investments in equipment and facility improvements being made by VTT.

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

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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). The main findings for each of these work packages are: i) WP1 – NDT of thick-walled structures continues to challenge the industry. Basic and advanced Non-Destructive Testing (NDT) testing methods were used to evaluate the characteristic properties of the reinforced concrete and identify/quantify sub-surface defects. Research shows that the combination of several NDT techniques can mutually strengthen individual assessment, however, the diagnosis of the results requires additional expertise to interpret NDT combined measurements; ii) WP2 – when identifying critical areas of NPP SSCs where ASR has the possibility to occur, based on concrete compositions and exposure conditions, results show that a large part of the concrete infrastructure show medium to high potential of ASR occurrence, if aggregate were to be reactive (due to high alkali content and high relative humidity). Furthermore, the extensive testing of aggregates reactiveness used in construction of Finnish NPPs has contributed significantly to our increasing understanding of Finnish aggregate performance.; and iii) WP3 – The study of liner corrosion concluded that: 1) in the used experimental set-up, the presence of the delamination gap did not significantly change the corrosion behaviour of the steel liner in comparison to the corresponding flat surface of a normal concrete; and, 2) the loss of passivity explains the corrosion of steel liner embedded in concrete and enables an active corrosion cell to be developed over long periods of time. This was observed both in the case of low-pH concrete and in the presence of the foreign matter: piece of wood. For the study of anchor corrosion, the studied variables of post-installed anchors were corrosion type, steel grade, anchor type and testing exposure. Research results show that the type of the anchor and material affect the corrosion based on the accelerated tests conducted. Corrosion risk varied from low (stainless steel) to severe (carbon steel). Failure mechanisms were dependant of the amount of corrosion occurring in the anchors.

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

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The SAFIR2022 CONFIT project uses a multi-disciplinary approach to investigate the various physical and chemical degradation mechanisms and how they affect the mechanical load bearing capacity of concrete in long term operation. Reinforced concrete structures are, indeed, of safety relevance in nuclear power plants due to the containment function of the reactor building and load bearing functions of the control building and shielding functions of specific concrete structures. During the project, it was investigated how various external chemical and physical stressors affect the mechanical concrete properties as a material [Ferreira, M. and Fülöp, L., 2020, ”Understanding the effect of ageing and deterioration of reinforced concrete on its durability and mechanical performance”, VTT research report VTT-R-01115-20] and in particular how corrosion of the reinforcement affects the load bearing capacity of a concrete structure [Calonius K. et al, 2022, “Artificial ageing of impact slabs”, VTT-R-01061-22]. For the simulation of full-scale loading scenarios on reinforced concrete structures involving physically, chemically or mechanically deteriorated concrete, specific material models for concrete were developed during the project. One of the advantages of such advanced concrete models is the ability to respond to anisotropic behaviour, which is inherent in damaged concrete [Vilppo, J. et Al, 2021, ”Anisotropic damage model for concrete and other quasi-brittle materials” Int. Journal of Solids and Structures]. Since the calibration of the model parameters requires measurements of anisotropy in concrete under controlled multiaxial loading, a specific method using ultrasonic wave velocity measurement was developed [Calonius et al, 2022, ”Determination of concrete stiffness tensor at different levels of damage”, VTT research report VTT-R-00692-22]. This method enables the computation of the damaged stiffness matrix components from the ultrasonic pressure and shear wave velocity measurements on the concrete sample in different directions. As a result, the project has generated important findings in the domain of nuclear safety, some of which present novelty value of academic importance.

CONTSA - Containment safety research

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The general objective of CONTSA has been to study important thermal-hydraulic containment safety issues and validate analysing methods to improve their reliability for containment deterministic safety analyses, and hence, to increase containment safety. CONTSA has mainly relied on the test program of the international OECD/NEA HYMERES-2 projects by using selected tests as code validation cases. CONTSA has also followed closely the OECD/NEA PANDA project results and delivered the main findings to the SAFIR2022 organisations.

The analysing work focused to multi-nozzle spray and containment cooler safety system effects by analysing the OECD/NEA HYMERES-2 project test series H2P5 and H2P6 performed at Panda facility in Switzerland. CONTSA rounded up the containment know-how across relevant research areas such as DBA thermal-hydraulics, severe accidents and CFD method. Capability of Apros, MELCOR and CFD Fluent codes to simulate the safety system behaviour was investigated. The calculation results were compared with measurement data and also code benchmark has been conducted. Suitable modelling approach and nodalisation concepts with related sensitivity studies were also investigated.

The essential goal of the project was also to exchange knowledge from the older to younger experts and educate new experts in the area of containment safety. One Master's thesis was finalized in this two-year project.

The Finland's participation fee for the international co-operation program OECD/NEA PANDA was paid through CONTSA. The agreement of the PANDA project was signed by VTT with the authorization gained from TEM.

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

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The general objective of the COSI project (2019-2022) is to develop a co-simulation platform for the analysis of thermomechanical and electrical behaviour of a nuclear power plant (NPP) and its supporting electrical grid infrastructure, both on- and off-site. The platform was developed in MATLAB, and connects detailed simulations of the NPP behaviour in Apros Nuclear with grid models developed in Simulink and DIgSILENT PowerFactory.

This platform was tested with the evaluation of impacts on NPP operations and safety caused by disturbances in the on-site and in the off-site electric grids, as well as with the evaluation of NPP operations during future grid scenarios of large-scale renewable energy integration. A detailed model of the Finnish transmission system was developed in Simulink to support these studies, and is another important output of the project.

CRITFLO - Critical flow separate effect test facility

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CRITFLO was a one-year project conducted in 2022. In the original project proposal, LUT proposed to design and commission a separate-effect-test facility that would enable critical flow experiments in a primary-to-secondary guillotine break scenario. Motivation for the critical flow experiments were result of LUT's road show conducted in 2021 for the future research needs for Finnish nuclear safety community. Talks included all the Finnish power companies (TVO, Fortum, Fennovoima), VTT, and STUK. In the talks the general conclusion was that more phenomena based research was preferred. SET facilities provide easier approach on experimental study of the phenomena where it can be isolated compared to integral facilities. Due to funding reduction of the original proposal, a literary review of recent critical flow experimental work was conducted instead.

The literary review: "state-of-the-art report on experiment work on critical flow" included introduction of the theoretical background and short description of TPCF models used in SYS-TH codes. The experiment work presented was divided into two parts. First part focused on critical flow experiments conducted with key large-scale experimental facilities and second part to more recent experiments. The recent work was divided between experiments in pipes, cracks and valves.

As a part of CRITFLO, SILENCE meetings considering TPCF model development were attended where valuable information about experimental needs for the future TPCF model development was acquired.

ELMO - Extended lifetime of structural materials through improved water chemistry

Konsta Sipilä, Essi Jäppinen, Timo Saario, Aki Toivonen

VTT Technical Research Centre of Finland

Optimizing the water chemistry to mitigate corrosion and maximize the lifetime of the structural components is essential as the lifetime of nuclear power plants are considered. In ELMO project several topics related to this field were studied.

Regarding hydrazine replacement chemicals, their high temperature oxygen scavenging efficiency were determined. Carbohydrazide seemed to be the most efficient chemical in this sense. Decomposition products and effects on corrosion of structural materials were studied as well. The decomposition analysis indicated that erythorbic acid interferes the acidic conductivity measurement so severely that it does not seem to be a good candidate for the plant trials.

Impurity deposition is steam generator conditions were studied by validating the use of "Hot loop" facility. The facility enables online chemistry monitoring and performance of electrochemical measurements. It was shown that the effects of boiling to corrosion are only moderate and in a large extent reversible. Similarly to the validation of the "Hot loop" facility, high temperature zeta potential measurement setup based on streaming potential technique was validated. In addition, the dissolution of cobolt in simulated primary chemistry was estimated.

Finally, the stress corrosion cracking susceptibility of A690 in alkaline and Pb containing crevice chemistry was assessed. It was shown that in these kind of crevice environments the presence of Pb can be beneficial for A690 as it improved the protectivity of the oxide layer.

EMBER - Enhanced multi-physics calculation capabilities for fuel behaviour and reactor analyses

Ville Rintala, Heikki Suikkanen

LUT University

EMBER ultimately aims at reducing uncertainties in reactor and fuel analyses via development of advanced multi-physics calculation capabilities that will account for the reactor core neutronics, thermal hydraulics and fuel behavior. Existing solvers are used in the coupling. The Monte Carlo type Serpent is used for the reactor physics solution, the subchannel code SUBCHANFLOW is used as the thermal-hydraulic solver, and TRANSURANUS is used as the fuel thermo-mechanics solver.

The goal is to add a subchannel thermal-hydraulic solver to an already functioning coupling of Serpent and TRANSURANUS and to implement transient version of the coupling. In addition to these, the purpose is to improve the overall accuracy of the coupled solution for example by adding support for the predictor-corrector method used in burnup calculations and by using Serpent to calculate nuclide compositions also for the fuel code.

EPIC - Effective improvement of leadership and safety culture

Kaupo Viitanen, Merja Airola, Nadezhda Gotcheva

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The general objective of the project was to develop knowledge and approaches that support effective improvement of leadership and safety culture. First, the project examined how methodical safety culture improvement has been implemented in Finnish nuclear power companies, and what the experts' experiences were. Good practices for implementing effective safety culture improvement were summarized according to the framework based on collected empirical data. Second, the project examined safety leadership in three different contexts: the operational decision-making process, activities of middle managers, and safety walks. Third, the project examined effectiveness assessment of safety culture improvement from a systemic, multilevel perspective. Two complementary approaches were identified: phenomenon-based approach and process-based approach. Four practical tools were developed to illustrate the practical application of these two approaches.

FATIMA - Fatigue management for LTO

Tommi Seppänen, Jussi Solin, Esko Arilahti, Jouni Alhainen

VTT Technical Research Centre of Finland

The general objective of FATIMA was to go beyond existing state-of-the-art in fatigue management for long-term operation, including accounting for environmental effects on fatigue life. In the longer term, improved methods can be adopted as alternatives to existing practices through a combination of scientifically solid justification and international acceptance.

The FATIMA project aimed to understand the current state-of-the-art in codified practices for environmentally-assisted fatigue. The project team joined an international collaboration group on the topic for visibility and active discussions.

Experimental project work consisted of fatigue studies on a stainless steel pipe manufactured to Olkiluoto 3 specifications. The results, which are limited to air environment, fully align with expectations and highlight benefits of material-specific understanding and application. The planned experimental campaign in simulated reactor water could not be realized during the project timeframe. A modernized research facility for this purpose was designed and procured, but commissioning continues beyond the project.

IDEAL - Infrastructure development at LUT safety research laboratory

Joonas Telkkä

LUT University

The general objective of the IDEAL project (2019–2022) was to develop the experimental thermal hydraulic infrastructure at LUT University nuclear safety research laboratory. The project comprised maintenance of the thermal hydraulic test facilities, development and upgrade of the instrumentation and data acquisition capabilities, as well as the implementation of the new modular test facility, MOTEL, which models a SMR. 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 projects. Enhancement of the measurement capabilities enables production of high-quality data for better understanding of the thermal hydraulic phenomena and for the development and validation of computational tools. Procurement and implementation of novel measurement techniques promote growth of expertise 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 archiving of the large amounts of experimental data.

INFLAME - Interdisciplinary fuels and materials

Janne Heikinheimo, Asko Arkoma, Jussi Peltonen, Markus Hirvensalo, Rami Pohja, Konsta Sipilä, Juha-Matti Autio, Yangling Ge, Unto Tapper

VTT Technical Research Centre of Finland

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 behaviour 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 and SuperFINIX with performance optimization and MPI parallelization.

In the experimental work packages of the project, experimental studies have been performed for both, cladding and pellet materials. For the cladding samples, creep testing, autoclave testing and steam furnace testing capabilities have been developed. Tests and material characterizations have been done for conventional and accident tolerant cladding concepts. For the pellet materials, experimental capabilities at the VTT Centre for Nuclear Safety have been developed for iodine release experiments and scanning electron microscopy analyses.

One DSc thesis was finalized in 2020 based on the fuel thermochemical behaviour analyses made in this and the preceding SAFIR fuel projects. One Master's thesis work has been completed within the INFLAME project.

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

Ville Tulkki, Jussi Peltonen, Pekka Moilanen, Seppo Hillberg, Caitlin Huotilainen, Petri Kinnunen

VTT Technical Research Centre of Finland

Over the past forty years, materials testing reactors in Europe have provided essential and invaluable support to the nuclear power plant community and nuclear industry. Jules Horowitz Reactor (JHR), a new European material testing reactor, is currently under construction at CEA Cadarache research centre in France. JHR2022 project has provided a way for Finnish stakeholders to be involved in preparation of future test series preparation.

JHR2022 has consisted of participation in the JHR working groups preparing future experimental plans as well as preparation and start of OECD/NEA's FIDES framework for irradiation experiments. Finnish secondees have studied new non destructive testing techniques and irradiation modelling analyses at CEA in Cadarache. The second iteration of a biaxial creep device Melodie has been worked on, with the intention of testing the device in LVR-15 reactor in Czech Republic in preparation for future use in JHR.

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

Ville Valtavirta

VTT Technical Research Centre of Finland

A modern Finnish reactor analysis framework, Kraken was developed for light water reactor analyses. The capabilities of Kraken were demonstrated by validating the framework for VVER-1000 fuel cycle and coolant transient analyses using international benchmarks with measured data from real world reactors. Kraken development provides the tools and expertise for future safety analyses of Finnish reactors.

In addition to developing Kraken as a whole, the nodal neutronics program Ants was extended in the project from static neutronics calculations to fuel cycle and transient simulations considering feedback effects from thermal hydraulics and fuel behaviour solvers. Ants is now also able to predict spent fuel nuclide compositions and the activation of in-core structural materials.

Small modular reactor models were routinely utilized in the testing and demonstration of modelling capabilities, including the evaluation of licensing relevant data during operating cycle analyses.

MANTRA - Mitigation and analysis of fission products transport

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The aim in the MANTRA project during 2019-2022 was to investigate the transport and chemistry of gaseous and particulate fission products in severe accident conditions. 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). Fission product deposits on containment building surfaces may react with the surfaces and form volatile species. 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.

NAPRA - New developments and applications of PRA

Ilkka Karanta, Kim Björkman, Atte Helminen, Terhi Kling, Timo Korhonen, Marja Liinasuo, Tero Tyrväinen

VTT Technical Research Centre of Finland

Groundwork for PRA with long time windows has been laid in the Nordic PROSAFE project; NAPRA pilot model of a spent fuel pool integrates deterministic behavior and probabilistic analysis, and takes into account repairs and recoveries. Dynamic features of human reliability, and human error possibilities and circumstances conducive to it have been identified for nuclear power plant planned outages in general, and for a heavy lifting case in particular. Modelling issues of digital I&C PRA have been clarified in the international DIGMAP and DIGMORE benchmark projects. The DIGMAP study showed that the same results can be produced with very different modelling approaches. A fire PRA model has been created that integrates results of deterministic fire analyses, fire brigade behavior and probabilistic aspects; it is robust and straightforward to apply on the PRA part.

OSAFE - Development of framework for justification of overall safety

Juhani Hyvärinen[1], Juhani Vihavainen[1], Kim Björkman[2], Essi Immonen[2]

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The general objective of the OSAFE project was to advance the understanding of nuclear power plant safety and security, i.e., overall safety by applying a set of methods and improving of these methods for the purposes of safety assessment and safety justifications.

We have worked to reach the objective by improving methodologies for creating a framework for evaluating overall safety and by developing of overall safety framework, security and safeguards and implementing them into Overall Safety Conceptual framework (ORSAC). Three Master's thesis have been performed related, e.g., to small modular reactors and evaluating risks in non-nuclear areas of society. Annual Overall Safety seminars have been organized during 2015 – 2022.

PARSA - Participative development for supporting human factors in safety

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The PARSA project developed and examined measures and tools to foster participative development (PD) of safety in the nuclear industry. We developed and applied collaborative, cross-organisational work process analysis (CWPA), video-based method for collaborative learning, and analysis of documentation practices related to specific human performance tools in case studies conducted in the nuclear maintenance.

Results showed that applying PD methods has benefits and their application may respond to current needs in the nuclear industry such as skills renewal, organisational learning, or cross-organisational collaboration at critical work processes.

However, challenges to implement to current safety practices, cultural features or resources may hinder PD applications.

PATE - PWR PACTEL tests

Vesa Riikonen, Virpi Kouhia, Joonas Telkkä, Lauri Pyy, Antti Räsänen, Eetu Kotro

LUT University

The objective of the project was to improve the understanding of thermal hydraulic system behavior of EPR type PWRs by performing integral effects tests with PWR PACTEL. The project enhanced the Finnish nuclear safety assessment capability for solving future safety issues as they appear. It maintained and extended the research expertise needed for the experimental work and produced data for the validation of computational tools. Computer analyses were needed in the planning of the experiments as well as in post analyses to help understanding the physics in the experiments. The project had a significant international connection through the OECD/NEA PKL Phase 4 and OECD/NEA ETHARINUS projects training new researchers and familiarizing them with international networking.

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

Kirsti Jylhä, Ulpu Leijala, Milla M. Johansson, Marko Laine, Terhi K. Laurila, Anna Luomaranta, Taru Olsson, Meri Virman, Mika Rantanen, Jenni Rauhala, Olle Räty, Jani Särkkä

Finnish Meteorological Institute (FMI)

The safety management over the life cycle of a nuclear power plant requires probability estimates of single and compound weather and sea level events in the changing climate. Sea-effect snowfall typically results in larger accumulation of snow than non-convective winter precipitation. Severe convective wind storms due to derechos last longer and cause damage over wider areas than other major convective wind events. Characteristics of both historical and synthetic low-pressure systems causing extreme sea levels in Finland were studied. In most cases, coastal flooding events require more than one cyclone to pass by. Hierarchical extreme value models for calculating return levels of annual maximum sea level in Finland were found to have less uncertainty in the calculated estimates than tide gauge specific reference extreme value models.

RACSA - Radiation shielding and criticality safety analyses

Pauli Juutilainen, Eric Dorval, Silja Häkkinen, Mika Jokipii, Petri Kotiluoto, Topias Kähkönen, Jaakko Leppänen, Antti Räty, Lauri Vaara, Ville Valtavirta

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The main project target was to produce validated tools for radiation shielding, dosimetry and criticality safety analyses. New radiation shielding cases provided increased confidence in the Serpent Monte Carlo Code by validating its photon transport functionality. The dosimetry capabilities were improved in two ways: by validating Serpent 2 against eight benchmarks, and by developing the new deterministic code REMS as a reduced-order solver. The validation package for criticality safety analyses for fresh fuel was extended with a few new experimental series. Sensitivity and uncertainty-based methods were adopted to identify those experiments most neutronically similar to the target application. A Total Monte Carlo calculation environment was established to study the impact of nuclear data uncertainties from a burnup credit preparedness perspective.

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

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The objective of SEARCH (2019-2022) was to develop methods and tools for assessing technical design solutions related to overall instrumentation and control (I&C) system architectures, based on Model-Based Systems Engineering (MBSE) practices. We developed (1) tools for analysing Defence-in-Depth properties, (2) methods for the hybrid assessment of safety and security, as well as (3) data models to support conformity assessment and design iteration.

A particular focus was on broadening the scope in which a formal verification method called model checking could be used. We developed methods for software verification, where we also account for failures and communication delay in the underlying hardware architecture. We also developed tools for finding the root cause of counterexamples, and our capabilities to verify infinite-domain models.

SMRSIMA - Small Modular Nuclear Reactors (SMRs), Siting and Waste Management

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SMR Siting and Waste Management is a coordinated project between VTT and Geological Survey of Finland (GTK) focusing on waste management of spent nuclear fuel produced in a SMR and in siting and societal acceptability of a SMR plant and repository. In 2022, SMRSiMa received funding from both SAFIR2022 and KYT2022 programmes. Based on the study, the characteristics of the spent nuclear fuel (SNF) play a crucial role in adapting the current disposal methods (KBS3V) for SNF produced in a SMR. Considering siting of a SMR plant from geological point of view, the same principles apply for SMR plants as for any NPP. Considering an SMR that would be utilised for district heating purposes, the location of the site would be closer to a city than normal NPPs. This brings challenges considering the siting from the geological perspective, but also from societal acceptability point of view. Early engagement of stakeholders, civil society and public is required for successful SMR deployment and waste management.

THACO - Safety through thermal-hydraulic analyses and cooperation

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The goal of THACO is to improve nuclear safety through increasing the reliability of thermal-hydraulic deterministic safety analyses. Main part of the analysis work has been carried out with, and related to, the system-scale safety analysis tool Apros that has been developed in Finland in cooperation between VTT and Fortum. Participation in international thermal hydraulics field research projects has formed an essential part of the project. The most notable international cooperation task is the participation in OECD/NEA Rod Bundle Heat Transfer (RBHT) benchmark in all of its phases. Domestically, cooperation ties have been tightened with LUT University. The project has also included representation in multiple international OECD/NEA projects and in U.S. NRC CAMP program.

URAN - Uncertainty management in fire risk analyses

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Full-scale CFD simulations of cable fires were conducted in order to achieve better accuracy for predicting the heat release rate of these fires, and to understand and quantify the sources of uncertainty involved. Experimental data used for validation was collected in the OECD/NEA PRISME3 project.

Combustion of cylindrical fuels was studied by high-resolution CFD simulations to gain new and detailed information on the heat fluxes and pyrolysis rates of cables in a cone calorimeter experiment. Modelling involved detailed 3D heat transfer calculations inside the cable, taking into account the internal structure of the cable.

Ageing mechanisms of a typical cable jacket material, XLPE, were examined both numerically and experimentally in order to understand the effect of aging to the general fire safety of the NPP. Validation simulations of thermogravimetry and cone calorimeter experiments on aged polymers were conducted.

VALERI - Valorisation of the recent probabilistic seismic hazard projects and results

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Probabilistic seismic hazard analysis (PSHA) is the standard method to assess seismic hazard for nuclear power plants (NPPs). The regulatory status in Finland, given in the guide YVL B.7 and its Explanatory Memorandum, is that the median confidence seismic hazard at annual frequency (AFE) of 10^{-5} is used for design basis earthquake (DBE), with a minimum horizontal peak-ground acceleration threshold of 0.1g. Exceptional earthquake effects for design extension conditions (DEC C) correspond to the occurrence frequency of 10^{-7}/year, or approximately twice the DBE value. The use of median hazard, as basis for the design values appears to be a minority position among regulators. In this work we outline the PSHA and its consequent use in risk assessment and risk-informed decision-making, reviewing arguments about the mean and median hazards. Drawing particularly on the outcomes of the SENSEI project, conducted under the auspices of STUK in 2019–2020, we discuss possible confidence options for DBE and DEC C. Since the use of median hazard has a long tradition in Finland, an update is no trivial undertaking.