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Linde, Christian; Andersson, Kasper Grann; Magnússon, Sigurður M.; Physant, Finn

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Review Article

Nordic research and development cooperation to strengthen nuclear reactor safety after the Fukushima accident

Christian Linde ^{a, b, *}, Kasper G. Andersson ^{a, c}, Sigurður M. Magnússon ^{a, d}, Finn Physant ^{a, e}^a NKS, Roskilde, Denmark^b Swedish Radiation Safety Authority, Stockholm, Sweden^c Technical University of Denmark, DTU Nutech, Roskilde, Denmark^d Icelandic Radiation Safety Authority, Reykjavik, Iceland^e FRIT, Roskilde, Denmark

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ABSTRACT

A comprehensive study of photon interaction features has been made for some alloys containing Pd and Ag content to evaluate its possible use as alternative gamma radiations shielding material. The mass attenuation coefficient (μ/ρ) of the present alloys was measured at various photon energies between 81 keV – 1333 keV utilizing HPGe detector. The measured μ/ρ values were compared to those of theoretical and computational (MCNPX code) results. The results exhibited that the μ/ρ values of the studied alloys are in same line with results of WinXCOM software and MCNPX code results at all energies. Moreover, Pd75/Ag25 alloy sample has the maximum radiation protection efficiency (about 53% at 81 keV) and lowest half value layer, which shows that Pd75/Ag25 has superior gamma radiation shielding performance among the compared other alloys.

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1. Introduction

NKS (Nordic Nuclear Safety Research) is a forum for Nordic cooperation and competence sharing within nuclear safety and emergency preparedness, serving as an umbrella for Nordic initiatives and interests. This initiative dates back to the earlier days of Nordic nuclear research in the 1950's. It was built on the foundation of a common cultural and historical heritage and a long tradition of collaboration between the Nordic countries, i.e. Denmark (including the Faroe Islands and Greenland), Finland, Iceland, Norway and Sweden. The work in NKS is divided into one program area for reactor safety (NKS-R), which is presented in this article, and the other for emergency preparedness (NKS-B). The NKS programs are financed and supported by Nordic authorities, companies and other organisations. Activities are focused towards practical and directly applicable scientific results and competence building of interest for the financing organisations. NKS promotes participation of young scientists in the activities, and also has a

dedicated budget for travel support for young scientists. All activities are documented in technical reports. The results from the activities within NKS are also made available through seminars, exercises, scientific articles and other types of reference material. More than 500 project reports are available for free download from the NKS website (www.nks.org) together with, e.g., presentation material from two large joint NKS-R and NKS-B seminars held in Stockholm in 2013 and 2016 on Nordic developments after the Fukushima accident.

Among the Nordic countries, nuclear power plants are operated in Sweden and Finland, some of which are boiling water reactors (BWRs) of a common design. As a result, the majority of the activities in the NKS-R program are led by Swedish or Finnish organisations but there is also significant participation from Norway and Denmark where research reactors have been operated for many years.

The activities of the NKS-R program cover a wide variety of topics that are funded by NKS through annual calls from an average budget of 0.5 MEUR per year. A requirement for funding from NKS is that the activity is co-funded from other sources on at least the same level as the requested amount. The research areas 'thermal hydraulics' and 'severe accidents' have been funded steadily by

* Corresponding author. NKS, Roskilde, Denmark.
E-mail address: christian.linde@ssm.se (C. Linde).

NKS, each area with a total amount of nearly 1 MEUR since 2008. After the Fukushima accident in 2011, ‘risk analysis and probabilistic methods’ together with ‘organisational issues and safety culture’ have appeared as two areas that have attracted increasing funding from NKS, both areas now having received an accumulated funding of nearly 1 MEUR during 2008–2017. In recent years activities within the areas of ‘plant life management and extension’ as well as ‘decommissioning’ have become of growing interest. The areas of ‘reactor physics’ and ‘automation and control room’ also receive occasional funding from NKS.

The Swedish Radiation Safety Authority (SSM) recently published a report on the evaluation of the Swedish participation in the NKS collaboration during 2008–2015 [1]. The report contains a thorough investigation of the added value from NKS to nuclear safety and emergency preparedness in Sweden and in the Nordic countries. It is concluded that the NKS funding should not be viewed as basic funding for national research environments. Instead, the high additionality lies in the support to smaller R&D projects and pilot projects, which result in valuable collaborations between multiple Nordic experts.

The NKS projects in the areas of ‘thermal hydraulics’ and ‘severe accidents’ often involve collaborations between experimental and analytical activities at technical research organisations and universities. In the COPSAR project, thermal hydraulic experiments were conducted at Lappeenranta University of Technology in Finland (LUT) and the collected data were used at Royal Institute of Technology in Sweden (KTH) and VTT Technical Research Centre of Finland (VTT) in their analytical work. In the DECOSE project, experiments on different topics within ‘severe accidents’ were conducted at both VTT and KTH and analytical work was performed at both sites. The projects in the area of ‘risk analysis and probabilistic methods’ also cover model development and modelling of an analytical tool and involve a number of consultant companies together with industry as partners, see below e.g. the L3PSA project led by Lloyd’s Register Consulting in Sweden (LRC).

Workshops and case studies are often used as methods in the projects in the area of ‘organisational issues and safety culture’. These are often led by technical research organisations, e.g. LESUN led by Institute for Energy Technology in Norway (IFE) and SC_AIM led by VTT, which are both done in collaboration with industry. The projects within the area of ‘decommissioning’ have a broad range of partners from technical research organisations, authorities, industries and consultant companies, both in the ongoing NORDEC project and the latest NKS seminar on decommissioning that was held in 2013.

In this article, examples from some activities that have been conducted between 2012 and 2016 are presented in order to illustrate the forms of collaborations that exist and to give some examples of the safety topics that have been covered since the Fukushima accident in 2011, see Table 1.

2. Thermal hydraulics

2.1. Containment pressure suppression systems analysis for boiling water reactors (COPSAR)

The BWR containment is a complex system that includes many elements, which affect each other’s operation. The elements are e.g. the pressure suppression pool, spray and containment venting systems for containment pressure control, blowdown pipes for rapid steam condensation in case of a LOCA (Loss-of-Coolant Accident), in which the core is insufficiently cooled. There are also spargers for the pressure vessel relief valves, strainers for water supply to emergency core cooling and spray systems, nozzles and strainers of the residual heat removal (RHR) system, vacuum breakers, etc. There are a number of safety important scenarios, where containment pressure suppression function operation can be affected by (i) stratification and mixing phenomena, (ii) interactions with emergency core cooling system (ECCS), spray, RHR system, filtered containment venting system (FCVS), and (iii) overall water distribution between containment compartments. Such scenarios include (i) interplay between pool behaviour, diagnostics and operator procedures that can affect activation and performance of ECCS and containment spray systems; (ii) small LOCA; (iii) station blackout; (iv) leaking safety relief valve; (v) LOCA with broken blowdown pipe; (vi) severe accidents; and (vii) steam line breaks inside the radiation shield.

In the COPSAR project, the need is addressed for a validated modelling tool for simulation of realistic accident scenarios with interplay between phenomena, safety systems, operational procedures, and overall containment performance. Investigations of the phenomena that can affect pressure suppression function due to the operation of the other equipment and systems in the BWR containment are made. The experimental work is performed in the PPOOLEX facility at Lappeenranta University of Technology (LUT). PPOOLEX is a downscaled model of a BWR containment, which consists of a closed stainless steel vessel divided into two compartments, drywell and wetwell, that can be pressurized and equipped with different spray installations. Pre-test analysis and simulations for selection of operational regimes and test procedures is performed at KTH. Post-test analyses are made both at KTH and at VTT.

In a BWR, steam released from primary coolant system is condensed in the pressure suppression pool. Thermal stratification in the pool affects pressure suppression capacity of the pool. Heat and momentum sources generated by the steam condensation define pool behaviour. Direct Contact Condensation (DCC) of steam presents a challenge for contemporary modelling tools. In previous work, the Effective Heat Source (EHS) and Effective Momentum Source (EMS) models were proposed to simulate development of thermal stratification or mixing induced by steam injection into a large pool of water.

The experimental results from several tests in the PPOOLEX

Table 1
Overview of NKS projects discussed in this article.

Research area	2011	2012	2013	2014	2015	2016	2017	2018	Funded networks
Thermal hydraulics		ENPOOL				COPSAR			LUT, VTT, KTH
Severe accidents		DECOSE					SPARC		KTH, VTT, LRC CTH, VTT
Risk analysis & probabilistic methods				ATR	L3PSA				LRC, VTT, ÅF, Risk Pilot, Vattenfall VTT, SEI, ÅF, AAU, UU, GEUS
Organisational issues & safety culture					ADdGROUND LESUN				IFE, VTT, RAB VTT, KTH
Decommissioning						SC_AIM		NORDEC	IFE, VTT, ÅF, Fortum, Vattenfall, SSM, STUK, NRPA, SIS
Plant life management and extension						BREDA-RPV		BREDA-RPV	KTH, VTT, CTH

facility have been reported in the NKS report series; mixing tests with a residual heat remover (RHR) nozzle [2], sparger tests [3], spray tests [4], single spray nozzle tests [5] and sparger tests with reduced number of injection holes [6]. For the RHR nozzle tests particularly the effects of nozzle orientation, ΔT in the pool, injection water temperature and injection water mass flow rate were studied. The tests verified that orientation of an RHR nozzle plays an important role in the success of the mixing process of a thermally stratified pool. The injection water temperature and flow rate at the nozzle and ΔT in the pool have an effect on the mixing process but it is not as dominant as the nozzle orientation.

The analytical work by VTT and KTH is reported in several reports; e.g. pre-calculation of a PPOOLEX spray experiment [7] and CFD calculations on the thermal stratification and mixing in the sparger tests [8], modelling of pool behaviour and the Direct Contact Condensation (DCC) phenomena [9,10].

LUT, KTH and VTT have studied suppression pool phenomena also in the ENPOOL project (2011–2014), see e.g. Refs. [11,12].

High pressure in the containment due to reduced pressure suppression function is a safety concern for the containment integrity since it is the last barrier to the surrounding environment. The COPSAR and ENPOOL projects have provided important experimental data on pool behaviour addressing stratification and mixing issues that enable validation of computer models and realistic evaluation of safety margins in BWRs.

3. Severe accidents

3.1. Debris coolability and steam explosions (DECOSE)

Lower drywell flooding is adopted in some Nordic BWRs as a means of mitigation of core melt accident consequences. It is assumed that upon reactor pressure vessel failure the melt will fragment, quench and form a debris bed coolable by natural circulation of water in a deep pool under the vessel. However, if the debris bed is not coolable, or if there is an energetic steam explosion when molten material meets water as it is released from the vessel, containment integrity can be threatened, potentially leading to release of radioactive materials into the environment.

VTT and KTH have collaborated in the DECOSE project to develop experimental facilities at both sites in order to produce data necessary for development of new models and codes for severe accident analysis. These codes and models can be used to address the long-standing technical issues of ex-vessel debris coolability and steam explosions in Nordic BWRs. The goal of the project was to reduce uncertainties in debris coolability and steam explosion impact. The project had significant co-funding, VTT through the SAFIR2018 program (The Finnish Research Program on Nuclear Power Plant Safety 2015–2018) and KTH through the APRI-MSWI program (Accident Phenomena of Risk Importance – Melt-Structure-Water Interactions).

A summary of the results of the experimental studies of debris bed coolability in the COOLOCE program at VTT is presented in Ref. [13]. The experiments addressed the effects of the debris bed geometrical shape, which is a result of the melt jet fragmentation and solidification in a water pool. Six variations of the debris bed geometry with different flooding modes were examined in the experiments, including a top-flooded cylinder and five beds with more complex, heap-like geometries. The experimental results in Ref. [14] suggest that the heap-like shape of the debris bed is favourable to coolability.

A study and comparison of data on coolability of particulate beds packed with irregular multi-size particles is presented in Refs. [15,16]. The data from COOLOCE are compared with that of POMECO-FL and POMECO-HT facilities at KTH. The effective

particle diameters obtained from the experiments are discussed, which is followed by the dryout heat flux comparison. It is suggested that the observed coolability variations are due to variations in porosity of the particulate beds resulting from the filling processes that were used in the experiments. It also includes the discussion over the possible factors (heaters' geometry and porosity) affecting the coolability of the bed.

The DECOSIM code has been developed for analysis of porous debris coolability and further validated against COOLOCE data [17]. Debris bed cooling in post-dryout regime was addressed. Analytical models for prediction of the maximum temperature of the debris and relative size of the dry zone as a function of overheating parameter was proposed and validated against DECOSIM simulations. In the study, the DECOSIM code was extended to in-vessel coolability analysis.

In case of a core melt accident where the reactor pressure vessel fails, the melt ejected from the lower head into the water pool may be in the form of a continuous jet. A series of DEFOR-A (Debris Bed Formation – Agglomeration) tests was carried out in order to clarify the effect of the melt jet velocity on the particle size distribution and fraction of agglomeration [18,19]. In these works phenomena relevant to the debris bed formation and coolability were also addressed experimentally and analytically.

A steam explosion is a fast fuel-coolant interaction that might occur as molten core material is released into the flooded lower drywell. Steam explosions have three distinct stages: premixing, triggering and propagation. In the premixing stage, the molten corium is fragmented into the coolant due to thermohydraulic forces. A large portion of the corium forms molten drops suspended in the coolant by vapour film. A triggering pulse locally collapses this instable liquid-vapour-liquid system. If the mixture properties are favourable, the trigger propagates in the mixture collapsing all the melt drops so that the thermal energy of the melt is almost instantly transferred to the coolant causing instantaneous high-pressure increase.

The effect of an ex-vessel steam explosion was analysed at VTT via computational models using the MC3D code, a multiphase CFD (computational fluid dynamics) code for fuel coolant interactions [20]. The focus of the analysis lies on the dynamic loads on the cavity wall imposed by the explosion. Simulations were made to analyse the effect of different triggering times on a standard case with central break at the lower head of the reactor vessel. The results showed that as long as the mixture properties allow the trigger to propagate then the resulting explosion behaves in a similar way. Different side breaks scenarios were also tested but here the mixture did not trigger. The sensitivity analysis was done for melt temperature, coolant subcooling, cavity water level and melt drop size. The results show that the parameter with the strongest effect is the drop size formed in the premixing phase, which is largely tied to the physical properties of the melt.

In an earlier sensitivity analysis [21], the three input parameters, melt temperature, triggering time and ambient pressure, were varied to see how changes in these affect the results. Among them, changing the triggering time had the largest impact on the steam explosion occurrence and strength. The melt temperature is not a limiting factor in this case. The ambient temperature had no effect on the explosion strength but instead on the probability of the explosion to occur.

A study performed at KTH deals with the premixing and explosion phase calculations of a Nordic BWR flooded drywell cavity, using MC3D [22]. The main goal of the study was the assessment of pressure build-up in the cavity and the impact loading on the sidewalls. The study included a sensitivity analysis of the parameters in modelling of fuel coolant interactions, to reduce uncertainty in assessment of steam explosion energetics. Moreover, the effects

of water pool depth and jet velocity on the energetics of steam explosion were investigated. In an earlier study [23], it was shown that the amount of liquid melt droplets in the water is maximum even before the jet reaches the bottom of the pool. In the explosion phase, maximum pressure is attained at the bottom and the maximum impulse on the wall is at the bottom of the wall. The pressure attained and impulses on the wall are higher for bigger jet diameters, bigger melt droplets and higher subcoolings.

An approach for analysis of steam explosion in Nordic BWR and sensitivity of steam explosion impulse to the uncertain modelling and scenario parameters has been developed using the TEXAS-V code [17]. The Texas-V is a code for analysis of gas, liquid and fuel particles. It comprises two modules: one for calculation of pre-mixing and another one for calculation of steam explosion. The first results indicated that the most influential parameters are water level and water temperature [24]. Extensive simulations using the TEXAS-V revealed that explosion impulse is a chaotic function of the triggering time [17]. The obtained database of impulse and pressure is used in the development of a computationally efficient tool for assessment of ex-vessel steam explosion risk in Nordic BWRs. This modelling approach is further developed in the ongoing SPARC project (see below).

3.2. The effect of aerosols and air radiolysis products on the transport of ruthenium (ATR)

The behaviour of the fission product ruthenium in a model primary circuit during a severe accident has been studied by VTT and Chalmers University of Technology in Sweden (CTH) in the ATR project [25]. The radiotoxicity of ruthenium oxides is similar to that of iodine in the short term and similar to that of caesium in the long term. All experiments were conducted using VTT's Ru transport facility. The results show that the impact of additional NO₂ gas feed (75 ppm volume) to the flow of ruthenium oxides (in humid air) was significant both on the transport of ruthenium through the facility and on the speciation of the transported ruthenium. Transport of gaseous RuO₄ was increased significantly, whereas at the same time the amount of aerosols was decreased.

The following year the effect of additional air radiolysis products N₂O, NO₂, HNO₃ [26] and CsI aerosol [27] on the transport of gaseous and particulate ruthenium species through a model primary circuit was studied. In the experiments, nitrogen oxides as well as nitric acid originating from air radiolysis had a significant effect on the ruthenium chemistry in the model primary circuit. The obtained results indicate a strong effect of air radiolysis products on the quantity of transported ruthenium and its partition to gaseous and aerosol compounds.

3.3. Scenarios and phenomena affecting risk of containment failure and release characteristics (SPARC)

The SPARC project is a collaboration between KTH, LRC and VTT on scenarios and phenomena affecting risk of containment failure and release characteristics. The work is co-supported by the APRI-MSWI program, co-funded by APRI and ENSI (Swiss Federal Nuclear Safety Inspectorate), CASA (Comprehensive Analysis of Severe Accidents) and PRAMEA (Probabilistic Risk Assessment Method Development and Applications) projects in the frame of the SAFIR2018 program. The work is co-funded also by VTT and the Nuclear Waste Management Fund and from LRC's own research funds. The project involves a number of young scientists in both Sweden and Finland.

The project is motivated by an apparently high sensitivity of effectiveness of severe accident management (SAM) strategy in Nordic type BWR to the uncertainties in physical phenomena (deterministic) and accident scenarios (stochastic). The ROAAM+

(Risk-Oriented Accident Analysis Methodology) approach is employed in order to address both epistemic (systematic) and aleatory (statistical) sources of uncertainty in a consistent manner. In the first report [28], the state of the art review of Integrated Deterministic Probabilistic Safety Analyses (IDPSA) is presented. The ROAAM+ framework addressing all stages of the accident progression from initial plant damage states, through core degradation and vessel failure, melt ejection mode to ex-vessel melt-coolant interactions and debris coolability, is discussed in detail along with implementation details of the ROAAM + framework itself. Main findings of the analysis of effectiveness of SAM strategy in Nordic BWRs using ROAAM + framework and main results are presented using failure domain maps, which indicate conditions at which the mitigation strategy of the SAM fails. The outcome of this ongoing project will allow the end users to enhance understanding, completeness and consistency of safety analysis dealing with risk analysis.

4. Risk analysis and probabilistic methods

4.1. Addressing off-site consequence criteria using level 3 PSA (L3PSA)

Level 3 Probabilistic Safety Analysis (Level 3 PSA) provides a tool to assess the risks to the society posed by a nuclear plant, and could be integral in making objective decisions related to the off-site risks of nuclear facilities. Results from the site components and human factors (Level 1 PSA) and the severe accident and radioactive source term analysis (Level 2 PSA) are incorporated with meteorological data, radionuclide release data, population and agricultural data to estimate the risks to the public in terms of health, environmental and economic effects. The interest in this topic has risen within the Nordic region, and around the world, mainly as a consequence of the Fukushima Daiichi accident.

The goal of the L3PSA project has been to develop Nordic understanding of the potential for Level 3 PSA to determine the influences and impacts of off-site consequences, the effectiveness of off-site emergency response, and the potential contributions of improved upstream Level 1 and Level 2 PSAs. The project has included four of the leading Nuclear Risk Analysis consultancies in the Nordic countries (Lloyd's Register Consulting, ÅF, Risk Pilot, and VTT), and also Vattenfall in the final year of the project. The group has conducted a survey to develop a baseline for the state of knowledge, the opinions, and interests concerning Level 3 PSA. They also performed a study of potential risk metrics, e.g. the probability per reactor year versus consequences such as health effects, environmental or economic impact, a summary of regulations and standards, as well as one Swedish and one Finnish pilot study. The outcome from these activities provided input into the document Level 3 PSA Guidance for Nordic Conditions with user recommendations that were presented at a seminar in 2017 and that is included in the L3PSA final report [29].

4.2. Modelling as a tool to augment ground motion data in regions of diffuse seismicity (ADdGROUND)

After the Fukushima accident, seismic safety of nuclear power plants and other nuclear installations has become an increasingly important topic also in regions with low seismic activity, including the Nordic nuclear sites. The technical aim of the ADdGROUND project was to refresh existing and to build new capabilities in earthquake source modelling for ground motion simulations in the context of stable continental regions, specifically the Fennoscandian shield. Another aim was to establish and maintain a network of experts focused on diffuse seismicity areas of the Nordic countries and further enhance the cooperation between VTT and

Uppsala University (UU) in the area of earthquake source modelling. The project also involved University of Helsinki, Institute of Seismology (SEI), ÅF-Consult (ÅF), Aalto University (AAU) and Geological Survey of Denmark and Greenland (GEUS).

The focus in the first report lies on mid-magnitude earthquakes located near the plant [30]. This region is called the “near-field” and is known for its particularities when compared to “far-field”. For example, significant duration of the ground motions is shorter, corresponding to shear wave and surface wave arrivals; there are distinctive high velocity peaks in the ground motions and vertical shaking components may exceed horizontal components. These particularities are known to have design consequences, but are often overlooked by engineering codes. In Fennoscandia, near-field observations of larger magnitude ($M > 3$) earthquakes are missing. Some of the potential design consequences of near-source earthquakes to nuclear installations are highlighted in the report. The consensus seems to be that the destructive potential of these types of earthquakes is generally low. However, they can produce surprisingly larger acceleration values in the range of high frequencies, and can generate high strain rates in the loaded structures and components. In nuclear installations, with stiff components, the effect of high frequency shaking should be carefully considered.

The moment magnitude of the simulated earthquakes is $M_w = 5.5$, which corresponds approximately to the magnitudes of the largest historic events in the region. The modelling techniques are described in the second report [31], in which the modelling outcomes are compared with ground motion prediction equations (GMPEs) developed for stable continental regions. The results of the project are of great significance for the seismic assessment of structures, systems and components in nuclear facilities.

5. Organisational issues and safety culture

5.1. Safety culture assurance and improvement methods in complex projects (SC_AIM)

A good safety culture is an essential ingredient for ensuring safety in the nuclear industry. Current safety culture and safety management models and practices are largely focused on single organisations, which are stable and relatively homogeneous, and it is far from clear how to apply them in the dynamically changing project networks.

The project SC_AIM is aiming at identifying and specifying methods to improve, facilitate and assure safety culture in complex projects [32]. In this collaboration between VTT and KTH, with involvement from industry partners such as Fortum, Fennovoima, OKG and Forsmarks Kraftgrupp, an initial literature review concerning project environments revealed a multitude of project-specific challenges and boundary conditions in the domains of time, team, task and context that can potentially influence safety culture assurance and improvement. Temporary organisational forms are inherently time-delimited, which means that they have defined starting and ending points. The tasks are often specific, finite, often unique and complex. This leads to conceiving time in a linear manner and dividing the task into a number of consecutive phases. Special team and group interdependencies also characterize temporary organisations, where one challenge for the team is to manage a diverse set of skills and knowledge without prior collective working experience.

Three empirical case studies in Nordic nuclear industry organisations were conducted. In the first case study, which focused on the use of an in-house self-assessment team called “safety culture ambassadors” to promote and facilitate safety culture improvements. It was found that this method can influence safety culture through multiple mechanisms and that the flexibility of this

method can potentially rectify some of the challenges posed by project environment, or even benefit from them. Another case study focused on a safety-oriented project management seminar and showed the potential of this method in influencing safety culture through providing a forum for dialogue between different stakeholders. Finally, information exchange with experts provided additional insight into the current challenges and opportunities of safety culture work in projects. As a result of the theoretical and empirical work, a preliminary framework for evaluating the applicability of safety culture assurance and improvement methods has been developed.

5.2. Learning from successes in nuclear power plant operation (LESUN)

Learning from experience is essential to achieve safe and efficient operations at nuclear power plants. In the nuclear industry, licensees are required to collect lessons from unwanted events in order to prevent the recurrence of similar events. The LESUN project is a collaboration between IFE, VTT and Ringhals AB (RAB) that aims at improving nuclear safety by enhancing organisational learning from successful actions and decisions instead of from failures [33]. The project wanted to develop an operating experience method for capturing, analysing and communicating lessons learned based on successes.

From a literature review and two case studies in nuclear power plants, the project found that success is a complex and multidimensional concept that can take many forms. Three broad categories of success were identified: normal performances, extraordinary performances and recoveries. It was also observed that success can have properties such as time and situation-dependence and that it relates to the objective or subjective expectations of multiple stakeholders. The project formulated a preliminary framework for capturing successes that could be useful for identifying successful situations for learning purposes. It was also found that successes are often less salient and less likely to trigger intentional learning processes than failures. Regardless, the empirical studies indicate that there was a clear interest in successes at the power plants: existing methods, albeit not very refined, were already in place that could be utilized to learn from successes more systematically. Further developing these activities is also important in order to avoid unwanted side-products of learning from success such as organisational drift or complacency. In addition, because lessons learned from success are often tacit, exploring the possibilities of developing learning that relates to tacit knowledge may be useful. Operating experience activities have a central role in facilitating the development of these learning activities.

6. Decommissioning

6.1. Challenges and opportunities for improving nordic nuclear decommissioning (NORDEC)

Approaching large-scale nuclear decommissioning projects in the Nordic countries make it important for both regulators and operators to build new capabilities for handling up-coming challenges. Sweden and Finland both have a mixed legacy of nuclear sites, including plants and research reactors in different stages of operation or decommissioning, whereas in Denmark some decommissioning projects have been completed for research reactors and others are well on the way to completion. In Norway, while no immediate decommissioning activities are foreseen, the existing decommissioning plans and regulations can be improved by means of the information and lessons learned from the other

Nordic countries.

NKS sponsored a decommissioning seminar at IFE in Halden, Norway in 2013 [34]. The conference aimed to bring together operators, regulators, scientists, consultants and contractors from the Nordic countries (and invitees from other countries) to exchange information and views on a number of the topics related to decommissioning, release of materials and management of waste. A number of presentations about new knowledge and lessons learned were given in addition to round table discussions where issues and experiences of decommissioning from the Nordic countries were discussed.

The NORDEC project, which started in 2017, is conducting a study on how decommissioning is regulated, planned and performed in the Nordic countries, in order to identify where the main challenges lie, collect best practices and share experiences between the Nordic participants. A wide span of stakeholders are involved in the project; The Norwegian Radiation Protection Authority (NRPA), Swedish Radiation Safety Authority (SSM), Danish Health Authority (SIS), Finnish Radiation and Nuclear Safety Authority (STUK), the energy companies Fortum and Vattenfall, the consulting firm ÅF of Sweden and VTT Technical Research Centre of Finland will participate in the project, which is led by the Institute For Energy Technology (IFE) in Norway. The project involves collecting experiences from completed and ongoing decommissioning-related activities in Sweden, Finland, Denmark and Norway. The experiences' evaluation aims to identify possible improvements in processes, methods and tools. The project will foster collaboration among Nordic stakeholders through sharing of challenges and best practices.

7. Plant life management and extension

7.1. Barsebäck as research and development platform, extraction and analysis of reactor pressure vessel material (BREDA-RPV)

Irradiation induced ageing of the weld material of the reactor pressure vessel (RPV) is a limiting factor from a long term operation perspective. The closed Barsebäck 2 reactor gives an opportunity to harvest samples from the RPV, which was manufactured and welded with the same technique and high amounts of nickel and manganese as many Nordic RPVs. A test program to analyse the as-aged material properties has been prepared within BREDA-RPV [35]. The work is ongoing and has been supported from both SSM and SKC (Svenskt Kärntekniskt Centrum) in Sweden and by the Finnish nuclear safety program, the SAFIR-program.

VTT, CTH and KTH have collaboratively prepared an extraction outline to give the basis for further discussions with the Swedish utilities regarding the materials extraction scheme and proposed amounts of materials and positions in the RPV. The work at CTH focused on base-line high resolution atom probe tomography (APT), testing on un-irradiated material as well as sample materials irradiated in a test reactor. VTT has performed a base-line testing utilizing miniature fracture toughness testing samples of un-irradiated RPV material obtained from the original tests of the RPV of Barsebäck 2. The actual retrieval of materials from Barsebäck, is foreseen to occur in 2018 and -19.

By this study, it is possible to verify the validity of the surveillance specimen used for the monitoring of the RPV status, e.g. by investigating whether the heat treatment of the surveillance samples gives representative values as compared to the RPV itself. It is possible also to study the irradiation induced embrittlement of the RPV by analyzing the degradation gradient in the RPV wall. A possibility to acquire for example three or four trepan samples from locations at different axial positions would make it possible to study the metallurgic variability and different ageing phenomena from thermal and radiation induced degradation: The core region

has substantial neutron flux, while the RPV top lid has a substantial thermal component while the neutron flux is orders of magnitude lower. The upcoming study will be carried out at KTH, in strong collaboration with CTH on micro-structural analysis, and VTT for mechanical testing.

8. Summary and conclusions

NKS-R funds research activities with particular relevance for the Nordic development of research within reactor safety. The grants are used to co-fund activities and add important value by facilitating the coordination of existing collaborations. The NKS-R covers programs within the areas of 'severe accidents', 'thermal hydraulics', 'risk analysis and probabilistic methods', 'organisational issues and safety culture', 'decommissioning' and 'plant life management and extension', 'reactor physics' and 'automation and control room'. The activities involve experimental and analytical studies, workshops, case studies and seminars, and bring together technical research organisations, universities, authorities, industries and consultant companies from the Nordic countries. There is also a particular focus on facilitating the competence building and development among young scientists.

As exemplified in this report, the NKS-R program provides support with high additionality particularly in smaller R&D projects and pilot projects, and offers the opportunity for network building and valuable collaborations between experts at the leading Nordic research facilities within reactor safety.

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