NKS-R STATUS REPORT

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Table of contents

Appendices

A) "STATUS REPORT OF NKS-SPARC PROJECT - Scenarios and Phenomena Affecting Risk of Containment Failure and Release Characteristics", May 2017 – Full report May 15, 2017
B) "Status report for Project: WRANC", May 2017 – Full report

1 Overall status summary

This report provides a short overview of the current status of the NKS-R programme.

Since the last NKS board meeting in January, nine final reports from five of the eight NKS-R activities from CfP 2016 have been published on the NKS website.

Nine out of eleven contracts have been signed for the seven activities from CfP 2017. The process for the remaining two contracts from COPSAR and SPARC is expected to be completed in June.

All activities from CfP 2015 and earlier are completed. The publication issue related to the ATR-2015 activity that appeared in the end of 2016 has been resolved. The partners of ATR-2015 will resubmit a revised version of their final report with a link to published material.

1.1 Published NKS reports

The following reports have been published within the NKS reports series since the last board meeting in January. All the reported work was performed in 2016.

Report nr	Project	Title	Published
NKS-381	SC_AIM	*Safety Culture Assurance and Improvement Methods in Complex Projects – Intermediate Report from the NKS-R SC_AIM*	25 Jan 2017
NKS-382	COPSAR	*Sparger Tests in PPOOLEX on the Behaviour of Thermocline*	17 Mar 2017
NKS-383	COPSAR	*Mixing Tests with an RHR Nozzle in PPOOLEX*	17 Mar 2017
NKS-384	COPSAR	*Preliminary Spray Tests in PPOOLEX*	17 Mar 2017
NKS-385	BREDA	*Barsebäck as Research and Development Platform, Extraction and Analysis of Reactor Pressure Vessel Material*	17 Mar 2017
NKS-386	L3PSA	*Addressing off-site consequence criteria using Level 3 PSA*	27 Mar 2017
NKS-387	HYBRID	*Development of a hybrid neutron transport solver in 2 energy groups*	25 Apr 2017
NKS-388	HYBRID	*Data and visualization solutions for HYBRID core simulation method*	27 Apr 2017
NKS-389	COPSAR	*Simulation of PPOOLEX stratification and mixing experiment SPA-T1*	27 Apr 2017

1.2 Seminars and publications

Project	Seminar date
L3PSA	Final seminar, L3PSA 2016, was held on 14th of February 2017.
NORDEC	Workshop on challenges and opportunities for improving Nordic nuclear decommissioning, 20-21 November 2017, Halden, Norway
SC_AIM	Three researcher workshops on the following topics: a) *How to build an adaptive safety culture in the nuclear industry?* (28-29 March, 2017) b) *Safety culture improvement and assurance methods* (3-4 May, 2017) c) *Safety culture methods and their underlying assumptions in the context of safety paradigms* (21-22 June, 2017)
FIREBAN	Workshop for PRA Integration in Q4 2018.

Project	Publications
ATR-2015	Kärkelä, T.; Kajan, I.; Tapper, U.; Auvinen, A.; Ekberg, C. *Progress in Nuclear Energy*, Vol. 99, **2017**, 38–48.
SPARC	1. Dmitry Grishchenko, Simone Basso, Pavel Kudinov, "Development of a surrogate model for analysis of ex-vessel steam explosion in Nordic type BWRs," Nuclear Engineering and Design, Volume 310, 15 December 2016, Pages 311-327, 2016. 2. Basso S., Konovalenko A., and Kudinov P., "Preliminary Probabilistic Risk Analysis of Debris Bed Coolability for Nordic BWRs Under Severe Accident Conditions," Nuclear Engineering and Design, Submitted 2017. 3. Galushin S. and Kudinov P., "Analysis of Core Degradation and Relocation Phenomena and Scenarios in a Nordic-type BWR," Nuclear Engineering and Design, Volume 310, 15 December 2016, Pages 125–141, 2016. 4. Kudinov P., Grishchenko D., Konovalenko A., Karbojian A. "Premixing and Steam Explosion Phenomena in the Tests with Stratified Melt-Coolant Configuration and Binary Oxdic Melt Simulant Materials," Nuclear Engineering and Design, Volume 314, Pages 1-338 (1 April 2017). 5. L. Manickam, P. Kudinov, W.M. Ma, S. Bechta and D. Gishchenko, "On the influence of subcooling and melt jet parameters on debris formation," Nuclear Engineering and Design 309: 265-276, 2016. PhD Dissertations: • Viet-Anh Phung, "Input Calibration, Code Validation and Surrogate Model Development for Analysis of Two-phase Circulation Instability and Core Relocation Phenomena," KTH, March, 2017. • Simone Basso, "Particulate Debris Spreading and Coolability," KTH, April, 2017.

1.3 Young scientist travel support

Two requests have been received from Chalmers students presenting at the 18th biennial meeting on Reactor Physics in the Nordic Countries (RPNC-2017), hosted by DTU Nutech, Risø on May 8-9, 2017.

- Klas Jareteg (PhD student): *"Fine-mesh multiphysics of LWRs: two-phase flow challenges and opportunities"*
- Huaiqian Yi (MSc student): *"Sensitivity analysis in reactor noise simulations"*

The total request is ca 8000 DKK.

2 Summary and status for activities initiated in 2016

Eight activities were initiated in 2016. Three of the activities were continuing activities and five were new. Nine final reports from five of the eight NKS-R activities from CfP 2016 have been published on the NKS website. Four activities are completed and one activity has submitted two out of three reports. A draft report was received from one activity. From two activities no reports have been received.

An overview of the status of the 2016 NKS-R activities is presented below in Table 1.

Table 1. NKS-R 2016 activities

Activity	Title	First invoice	Second invoice	Report number	Status
ADdGround	*Modelling as a tool to augment ground motion data in regions of diffuse seismicity*	3/6	-	-	a)
BREDA-RPV	*Barsebäck RPV trepan*	x	x	NKS-385	Done
COPSAR	*Containment Pressure Suppression Systems Analysis for Boiling Water Reactors*	x	2/3	NKS-382 NKS-383 NKS-384 NKS-389	b)
FIREBAN	*Determination of fire barriers's reliability for fire risk assessment in NPP*	2/5	-	-	c)
HYBRID	*Development of hybrid neutron transport methods and data visualization tools*	1/2	-	NKS-387 NKS-388	Done
L3PSA	*Addressing off-site consequence criteria using Level 3 PSA*	x	3/5	NKS-386	Done
SC_AIM	*Safety culture assurance and improvement methods in complex projects*	x	1/2	NKS-381	Done
SPARC	*Scenarios and Phenomena Affecting Risk of Containment Failure and Release Characteristics*	x	-	-	d)

a) **ADdGround –** Delayed reporting announced on Jan 26. New request for update sent on April 18.
b) **COPSAR –** Two partners out of three are done. Final report from the third partner will be delivered in June.
c) **FIREBAN –** Draft report received on May 16.
d) **SPARC –** Final report will be delivered in mid-June.

2.1 ADdGround

Modelling as a tool to augment ground motion data in regions of diffuse seismicity

Research Area: *Risk Analysis*

The technical aim of the ADdGROUND project is to build new capabilities in earthquake source modelling for ground motion simulations. The scarcity of empirical observations of near-field ground motions from large magnitude earthquakes in Fennoscandia has been an impediment for deeper understanding of the possible earthquake loading scenarios on nuclear installations, even if empirical data has been exhaustively analysed. With recent advances in computational methods, the opportunity exists for numerical models to give realistic estimates of earthquake loads. In addition to the technical outcome, the ADdGROUND project also aims 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 in the area of earthquake source modelling. The project outcomes will support STUK and SSM, providing background information for the safety assessments of nuclear plants, but are also relevant for nuclear repositories.

Activity leader: Ludovic Fülöp, VTT Technical Research Centre of Finland.
Funded organizations: VTT, SEI, ÅFC, AAU, GEUS, UU

Funding: 500 kDKK

Status: Delayed reporting announced on Jan 26. New request for update sent on April 18.

2.2 BREDA-RPV

Barsebäck RPV trepan

Research Area: *Plant life management and extension*

Studies of mechanical and microstructural properties of Irradiated Low Alloy Steel trepan samples from the Reactor pressure vessel wall of Barsebäck 2 (BREDA-RPV).

Irradiation induced ageing of the reactor pressure vessel (RPV) steel is closely monitored in specified ageing management programs called surveillance programs. These consist of a number of capsules positioned inside the RPV to allow for accelerated irradiation of the RPV material to predict the evolution of the mechanical properties of the material as a function of neutron dose. The closed Barsebäck 2 reactor gives an opportunity to harvest samples from the aged reactor pressure vessel (RPV).

Activity leader: Pål Efsing, Royal Institute of Technology (KTH)

Funded organizations: KTH, VTT, Chalmers

Funding: 400 kDKK

Status: *Completed*

2.2.1 Final report NKS-385

Report Number: NKS-385 (ISBN 978-87-7893-471-0)

***Report Title**: Barsebäck as Research and Development Platform, Extraction and Analysis of Reactor Pressure Vessel Material*

Abstract:
As part of the NKS-R program, VTT, Chalmers and KTH has performed a baseline study to prepare for a test program to analyze the as aged material properties of the retired reactor pressure vessel, RPV, from Barsebäck unit 2. The project started at July 1st, 2016. The initial activities focused on mapping of possibilities for future work between VTT, Chalmers and KTH, liason activities with Vattenfall to discuss extraction of the test material from the Barsebäck plant and collection of material for the base line testing. The group has 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 Chalmers University 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. In addition to this some samples of thermally aged material was included to visualize the features that develops during both types of ageing. 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. The material harvesting is outside the scope of the research oriented program that was supported in 2016. The work has been supported from both SSM and SKC in Sweden and by the Finnish nuclear safety program, the SAFIR-program. The main outcome so far apart from the actual data that has been produced and the proposed cutting scheme for materials retrieval, is the fact that the work enhances the collaboration in this technology driven area between two Swedish technical universities KTH and CTH and Aalto University in Finland, and the Finnish research institute VTT. In addition to this, it is functioning as a facilitator for contacts between the research driven academic world, safety and operability driven Finnish and Swedish nuclear operating companies and the Finnish and Swedish nuclear safety authorities.

Keywords: Low alloy steel, irradiation effects, fracture toughness, ductile to brittle transition temperature, constraint effects, high resolution microscopy

2.3 COPSAR

Containment Pressure Suppression Systems Analysis for Boiling Water Reactors

Research Area: Thermal Hydraulics

Thermal hydraulics experiments on the behaviour of a safety relief sparger (SRV) and a containment spray system are carried out at the PPOOLEX facility at Lappeenranta University of Technology (LUT). The effectiveness of mixing a thermally stratified water pool due to injection through a sparger is studied. Modelling work is done at VTT Technical Research Centre of Finland Ltd (VTT) and at Kungliga Tekniska Högskolan (KTH).

Activity leader: Markku Puustinen, Lappeenranta University of Technology (LUT)

Funded organizations: LUT, VTT, KTH

Funding: 500 kDKK

Status: Two partners out of three are done. Final report from the third partner will be delivered in June.

2.3.1 Final report NKS-382

***Report Number**:* NKS-382 (ISBN 978-87-7893-468-0)

***Report Title**: Sparger Tests in PPOOLEX on the Behaviour of Thermocline*

***Abstract**:*
This report summarizes the results of the two sparger pipe tests (SPA-T8R and SPA-T9) carried out in the PPOOLEX facility at LUT in 2016. Steam was blown through the vertical DN65 sparger type blowdown pipe to the condensation pool filled with sub-cooled water. Two different flow conditions were tested. Flow was either through all the 32 injection holes at the sparger head or just through eight holes in the bottom row. The main objective of the tests was to obtain data for the development of the EMS and EHS models to be implemented in GOTHIC code by KTH. KTH plans to extend the models to cover also situations where steam injection into the pool is via a sparger pipe. The test parameters were selected by KTH on the basis of pre-test simulations and analysis of the results of the earlier sparger tests in PPOOLEX. Particularly the behaviour of the thermocline between the cold and warm water volumes was of interest. For this purpose also PIV measurements were tried during the tests. In SPA-T8R, where flow was via 32 injection holes, the thermocline seemed to be around the elevation of 670 mm at the end of the stratification phase just as predicted by the pre-test simulations. The thermocline moved downwards as the erosion process progressed. The prevailing mixing mechanism during the final mixing phase was also erosion rather than internal circulation. In SPA-T9, where flow was via eight injection holes, the thermocline was at first at a higher elevation than in SPA-T8R. It then started to shift downwards as the flow rate was increased in small steps. Complete mixing of the pool was achieved with the steam mass flow rate of 85 g/s. Erosion was again the prevailing mechanism in the mixing process. The few sequences with recognized flow patterns from the PIV measurements indicate that some kind of swirls could exist at the elevation of the thermocline. The flow direction just under the thermocline can also be opposite to that just above the thermocline. The somewhat chaotic nature of the investigated phenomenon creates problems when measuring with a slow-speed PIV system and therefore definitive conclusions on the detailed behaviour of the thermocline can't be made. These tests in PPOOLEX verified that mixing of a thermally stratified water pool can happen through an erosion process instead of internal circulation if suitable flow conditions prevail.

Keywords: condensation pool, sparger, thermocline, mixing

2.3.2 Final report NKS-382

Report Number: NKS-383 (ISBN 978-87-7893-469-7)

Report Title: *Mixing Tests with an RHR Nozzle in PPOOLEX*

Abstract:
This report summarizes the results of the RHR nozzle tests carried out in the PPOOLEX facility at LUT in 2016. The test facility is a closed stainless steel vessel divided into two compartments, drywell and wetwell. For the RHR nozzle tests the PPOOLEX facility was equipped with a model of an RHR nozzle and an associated water injection line. The main objective of the tests was to obtain additional data for the development of the EMS and EHS models to be implemented in GOTHIC code by KTH. Mixing of a thermally stratified pool with the help of water injection through an RHR nozzle was of special interest. Particularly the effects of nozzle orientation, ΔT in the pool, injection water temperature and injection water mass flow rate were studied. In the tests there were two stratification phases and two mixing phases. During the stratification phases two regions with clearly different water temperatures and a narrow thermocline region between them developed in the pool. When the target temperature difference between the bottom and the top layer of the pool had been reached the mixing process was initiated by starting water injection into the pool through the RHR nozzle. With the vertical orientation of the RHR nozzle mixing was otherwise successful but incomplete above the nozzle elevation. This was the case with both of the used water injection flow rates, 0.5 kg/s and 0.3 kg/s. Compete mixing was achieved with the horizontal orientation of the RHR nozzle by using a large injection flow rate (1.0-1.05 kg/s). The pool mixed in about 4000 seconds. With a 0.3 kg/s injection flow rate the water volume above the thermocline started to cool down as soon as the mixing phase started whereas below the thermocline the mixing process proceeded very slowly and only a small fraction of the bottom volume mixed completely before the test was terminated because the wetwell became full of water. These tests in PPOOLEX verified that orientation of an RHR nozzle plays an important role in the success of the mixing process of a thermally stratified pool. The nozzle injection flow rate, injection water temperature and ΔT in the pool have an effect on the mixing process but it is not as dominant as the nozzle orientation.

Keywords: condensation pool, RHR nozzle, mixing

2.3.3 Final report NKS-384

Report Number: NKS-384 (ISBN 978-87-7893-470-3)

Report Title: *Preliminary Spray Tests in PPOOLEX*

Abstract:
This report summarizes the results of the preliminary spray tests carried out in the PPOOLEX facility at LUT. The test facility is a closed stainless steel vessel divided into two compartments, drywell and wetwell. For the spray tests the facility was equipped with a model of a spray injection system with four nozzles. The main objective of the tests was to study interplay between suppression pool behaviour and the spray system operation. Particularly we were interested to find out if mixing of a thermally stratified pool with the help of spray injection from above is possible. An additional goal was to obtain data for improving simulation models related to spray operation in CFD and system codes as well as contribute to the development of the EMS and EHS models for sprays to be implemented in the GOTHIC code by KTH. In the first two tests the initial stratified situation was created by injecting first warm and then cold water from the tap into the wetwell. In the third test the stratified situation was created with the help of small steam injection through the model of the sparger pipe in PPOOLEX by starting from a cold state. In all three tests, the spray injection flow rate was the maximum available from the water supply system of the laboratory i.e.

about 128 l/min. When divided to the four spray nozzles it gives 32 l/min per nozzle. In the first two tests, mixing of the topmost layers of the pool was achieved easily. The initial temperature difference between the bottom and surface was 28 °C and 33 °, respectively. It can be speculated that the whole water volume could have been mixed if the tests had been continued for a longer period of time. In the third test, complete mixing of the initial 60 °C temperature difference between the pool bottom and the surface layer was achieved in about 4200 seconds as a result of internal circulation in the pool induced by the density difference between the cold spray water and warm pool water. The pool water level rose by 2 meters during the spray operation. These preliminary spray tests in PPOOLEX indicate that it might be possible to mix a stratified pool with the help of spray injection from above. If spray injection was continued long enough internal circulation developed and finally mixed the pool.

Keywords: condensation pool, spray, mixing

2.3.4 Final report NKS-389

Report Number: NKS-389 (ISBN 978-87-7893-475-8)

Report Title: *Simulation of PPOOLEX stratification and mixing experiment SPA-T1*

Abstract:
Thermal stratification of the pressure suppression pool of the PPOOLEX facility has been studied at Lappeenranta University of Technology in experiments, where steam was injected into water pool through a sparger. In the stratification phase of the experiment SPA-T1, steam was injected into the pool at a small mass flow rate of 30 g/s for time 13 650 s. Then the mass flow rate was increased to 123 g/s in order to mix the pool. In the present report, CFD calculation of the experiment SPA-T1 is presented. The stratification phase and the mixing phase of the experiment were calculated by using the ANSYS Fluent 16.2 CFD code. Single-phase calculation was performed, where the mass, momentum and enthalpy sources of the injected steam were added in front of the sparger holes. Comparison of the CFD calculation to the measurements shows that the simulation predicts the temperature trends over time rather well. However, during the long stratification phase the calculated mixing between the lower part and the upper part is too strong. This might be corrected by adding grid resolution in the density and velocity gradient layer near the injection. Due to the excessive mixing during the stratification phase the predicted thermal transient in the mixing phase is somewhat milder than in the experiments.

Keywords: BWR, pressure suppression pool, condensation pool, stratification, mixing, CFD, computational fluid dynamics

2.4 FIREBAN

Determination of fire barriers's reliability for fire risk assessment in NPP

Research Area: Risk Analysis

The scope of the project is to investigate and assess the reliability of fire barriers in NPP during realistic fire scenarios to support the plant-scale risk assessment. The objective is to establish data and methods to determine the conditional probabilities for failure of fire barrier. Statistics, literature review, calculation and specific unique designed fire tests are used as methods.

Activity leader: Patrick van Hees, Lund University

Funded organizations: LU, VTT, AAU, DBI, RAB

Funding: 450 kDKK

Status: Draft report received on May 16.

2.5 HYBRID

Development of hybrid neutron transport methods and data visualization tools

Research Area: Reactor physics

The modelling of neutron transport typically relies on two rather opposite approaches: the probabilistic approach, and the deterministic approach. The probabilistic approach or Monte Carlo approach relies on tracking the individual lives of neutrons, and requires a large computing power for nuclear reactors. The deterministic approach, on the other hand, is based upon fast running algorithms, that solve the problem at hand in only an approximate manner. The purpose of HYBRID is to combine both approaches in order to obtain fast running methods (thanks to the deterministic route) and accurate results (thanks to the probabilistic route).

Activity leader: Christophe Demazière, Chalmers University of Technology

Funded organizations: Chalmers, IFE

Funding: 500 kDKK

Status: *Completed*

2.5.1 Final report NKS-387

Report Number: NKS-387 (ISBN 978-87-7893-473-4)

Report Title: *Development of a hybrid neutron transport solver in 2 energy groups*

Abstract:
This project investigates the feasibility of performing reactor physics calculations for nuclear cores using a hybrid neutron transport methodology, by combining deterministic and probabilistic modelling techniques. In the presented implementation, a deterministic response matrix method was developed in Matlab. The necessary probabilities appearing in the response matrix method were estimated in advance using a probabilistic solver – the Monte Carlo code Serpent2. Ultimately, the hybrid framework will combine the advantages of the deterministic approach (fast running calculations) with the ones of the probabilistic approach (high flexibility in modelling any geometry and high accuracy). In the response matrix method, two grids are used: one fine grid for estimating the scalar neutron flux and a coarse grid for computing the neutron currents on this grid. Because of the large efforts developing a new computational framework represents and because such a developmental work is error-prone, this first phase of the project implemented and tested the hybrid framework on a system as simple as possible: a two-dimensional representation of a simplified BWR fuel assembly. Such a choice was governed by the necessity to lower the computational time and to have a tractable system during the developmental phase of the framework. The development of the hybrid route was demonstrated to be feasible, after some modifications of the Serpent2 code. Although promising, the solution computed by the framework was demonstrated to be not fully realistic. Additional investigations are necessary to identify the root cause of the observed deviations from the expected physical behaviour of the system.

Keywords: nuclear reactor calculations, neutron transport, deterministic methods, probabilistic methods, hybrid methods

2.5.2 Final report NKS-388

Report Number: NKS-388 (ISBN 978-87-7893-474-1)

Report Title: *Data and visualization solutions for HYBRID core simulation method*

Abstract:
The modelling of neutron transport typically relies on two rather opposite approaches: the probabilistic approach, and the deterministic approach. The purpose of the present project is to combine both approaches in order to obtain fast running methods (thanks to the deterministic route) and accurate results (thanks to the probabilistic route). This so-called hybrid method will result in larger amounts of high-fidelity data than previous solutions to this problem. Viewing, comparing and storing this data should utilize the latest in data handling technology, covering input generation, data storage and output visualization. This report summarizes work performed so far in analysing the data aspects of this problem. This data system will not only be required to interface correctly with the proposed HYBRID method but will also have to interact with the envisaged user organization. At this stage of the project, the organizations are research institutes and universities. In the future, they may be reactor operators, fuel vendors or even reactor construction companies. Even further in the future spent fuel disposal companies may require some parts of the data

solution. Considering these users we have proposed a list of requirements related to quality assurance, continuous development and aging management. This report makes a start at describing the data problem. Data types, uses and possible database configurations are discussed. Finally, some examples of different data structures are given and possible consequences investigated. The next project phase will focus on constructing and testing different data solutions and showing possible visualizations.

Keywords: Neutron Transport, Database, SQL, NoSQL, Big Data

2.6 L3PSA

Addressing off-site consequence criteria using Level 3 PSA

Research Area: Risk analysis and probabilistic methods

Level 3 PSA provides a tool to assess the risks to society posed by a nuclear plant, and could be useful in making objective decisions related to the off-site risks of nuclear facilities. The intention of this study was to further Nordic understanding of the potential of 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 PSA.

Activity leader: Andrew Wallin-Caldwell Lloyd's Register Consulting

Funded organizations: LRC, Risk Pilot, ÅF, Vattenfall AB, VTT

Funding: 140 kDKK

Status: *Completed*

2.6.1 Final report NKS-386

Report Number: NKS-386 (ISBN 978-87-7893-472-7)

Report Title: *Addressing off-site consequence criteria using Level 3 PSA*

Abstract:
The goal of this project is to further 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. This report summarizes the developments from four years of work, but focuses on the finalization of a Nordic Level 3 PSA Guidance Document, which has been worked upon mainly during calendar years 2015 and 2016. Other activities that has been conducted, and provided valuable input to the Guidance Document, are an Industrial Survey, a study of potential Risk Metrics, a summary of Regulations & Standards, and two Pilot Studies (one Swedish and one Finnish). The main objective of the pilot studies was to gain practical experience that, together with insights from the other tasks included in the project, could be transferred to recommendations into a final guidance document. During the project, targeted discussions between consultancies, utilities, regulators, and insurance companies on the subject of Level 3 PSA have taken place and at the end of each years working period a seminar has been arranged. The working group has also been

engaged in international activities surrounding Level 3 PSA, i.e. the development of the IAEA Level 3 PSA TECDOC and the ANS/ASME Level 3 PSA Standard through the 2016 continuation of the project.

Keywords: PSA, PRA, Level 3 PSA, Probabilistic Consequence Analysis

2.7 SC_AIM

Safety culture assurance and improvement methods in complex projects

Research Area: *Organisational issues and safety culture*

Networks of companies typically carry out major projects in the nuclear industry. Current safety culture and safety management models and practices are largely focused on single organisations and it is far from clear how to apply them in the dynamically changing project networks. Traditional cultural approaches emphasize that it takes time and certain amount of continuity to create a culture, both of which are in short supply in projects with short time frames, diversity in both personnel and companies involved, and often a high personnel turnover.

Several issues remain unanswered, e.g., what should a safety culture improvement or assurance program be like in an “organization”, which is actually a dynamic network of actors from different companies? How to utilize the concept of safety culture in network and project settings?

Activity leader: Elina Pietikäinen, VTT Technical Research Centre of Finland

Funded organizations: VTT, Vattenfall AB

Funding: 410 kDKK

Status: *Completed*

2.7.1 Final report NKS-381

Report Number: NKS-381 (ISBN 978-87-7893-467-3)

Report Title: *Safety Culture Assurance and Improvement Methods in Complex Projects – Intermediate Report from the NKS-R SC_AIM*

Abstract:
A good safety culture is an essential ingredient for ensuring safety in the nuclear industry. The predominant approaches for safety culture are based on the assumption of stable and relatively homogeneous organizations, which often does not apply to contemporary project-oriented and turbulent environments. This study aims to identify and specify safety culture assurance and improvement methods for project environments. A variety of approaches and practical methods for safety culture improvement was identified in the literature. Based on their apparent objectives, the methods were classified into the following groups: organizational structures, direct behavioural modification, interaction and communication, commitment and participation, training, promotion and selection. The literature review did not reveal methods intended specifically for project environments or guidelines for tailoring the existing ones to suit project environment. Further

review of the literature 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. Three empirical case studies in Nordic nuclear industry organizations were conducted. In the first case study, which focused on the use of safety culture ambassador group, 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 was developed.

Keywords: Safety culture, project management, organizational change

2.8 SPARC

Scenarios and Phenomena Affecting Risk of Containment Failure and Release Characteristics

Research Area: *Severe Accidents*

A robust severe accident management strategy is paramount for minimizing the environmental impact in the case of a severe accident involving melting of a reactor core. Both physical phenomena (deterministic) and accident scenarios (stochastic) are sources of uncertainties in the assessment of effectiveness of the accident mitigation. Adequate approaches are necessary in order to address both deterministic (epistemic) and stochastic (aleatory) sources of uncertainty in a consistent manner.

Activity leader: Pavel Kudinov, Royal Institute of Technology (KTH)

Funded organizations: KTH, LRC, VTT

Funding: 600 kDKK

Status: Final report will be delivered in mid-June, see Appendix A for more details.

3 Summary and status for activities initiated in 2017

Seven activities were approved funding in CfP 2017. Five of these are continuing activities and two are new (NORDEC and WRANC). Nine out of eleven contracts have been signed. Contracts remain to be signed for one partner in COPSAR and one partner in SPARC. The process for these contracts is expected to be completed in June.

An overview of the 2016 NKS-R activities is presented below in Table 2.

A request for status updates of ongoing activities were sent to the Activity Leaders on April 25. The status of all activities are summarized in the sections below.

Additional details for two activities are available in Appendix A (SPARC) and B (WRANC).

Table 2. NKS-R 2017 activities

Activity	Partners	Contract sent	Contract received	First invoice	Funding
COPSAR	LUT	2017-01-26	2017-03-02	-	493
	KTH	2017-01-26	Expected May 31	-	
	VTT	2017-01-26	2017-04-06	-	
FIREBAN	LU + 4 partners	2017-01-27	2017-05-03	-	393
HYBRID	CTH + 1 partner	2017-01-30	2017-03-03	1(2)	493
NORDEC	IFE + 8 partners	2017-01-26	2017-03-08	3(9)	524
SC_AIM	VTT + 1 partner	2017-01-26	2017-03-16	1(2)	279
SPARC	KTH	2017-02-08	Expected May 31	-	524
	VTT	2017-02-08	2017-05-29	-	
	LRC	2017-02-08	2017-05-30	-	
WRANC	Inspecta + 2 partners	2017-01-31	2017-03-09	1(3)	393

3.1 COPSAR

Containment Pressure Suppression Systems Analysis for Boiling Water Reactors

Summary

Thermal hydraulics experiments on the behaviour of a safety relief sparger (SRV) and a containment spray system are carried out at the PPOOLEX facility at Lappeenranta University of Technology (LUT). The effectiveness of mixing a thermally stratified water pool due to injection through a sparger is studied. Modelling work is done at VTT Technical Research Centre of Finland Ltd (VTT) and at Kungliga Tekniska Högskolan (KTH).

Summary of the experimental work at LUT:

Efficiency of mixing a thermally stratified pool with the help of steam injection through a safety relief valve (SRV) sparger pipe or water injection through a residual heat removal (RHR) nozzle has been studied in tests carried out with the PPOOLEX facility in 2016. In 2017, the SRV sparger will be moved to the center of the pool and the submergence will be reduced from 1.8 to 1.5 m. This will allow developing a thicker stratified layer at the bottom and will contribute to the Effective Heat Source (EHS) and Effective Momentum Source (EMS) models based on the

Richardson scaling. A small-scale separate effect facility, where it is possible to measure directly the effective momentum induced by a steam injection through a single hole, will be designed and constructed. Tests with the facility will help to map the effective momentum of many condensation regimes. Closures for the EMS model development for spargers by KTH will be provided.

Wet well spray tests for studying the interplay between the suppression pool behavior and the spray system will continue. Mixing of a thermally stratified pool as a result of spray injection from above will be of interest. With the help of pre-test simulations done at VTT and KTH a representative test matrix, the initial thermal hydraulic state of the facility and the correct spray injection rate to be used can be determined.

Summary of the modelling work at VTT:

Pre-calculations will be performed with ANSYS Fluent for the small-scale separate effect facility, where steam will be injected through a single hole into water pool. The Euler-Euler method of Fluent with condensation model will be used. The effective momentum and heat sources generated by the steam injection into the water pool will be studied and later compared to the experimental results.

A spray experiment performed at PPOOLEX will be calculated with ANSYS Fluent. The water pool will be modelled with the Euler-Euler model of Fluent, where droplets will be described with the Discrete Particle Model (DPM). The effect of the spray droplets on the stratified pool will be calculated. The results will be compared to PPOOLEX experiment.

Summary of the modelling work at KTH (contract expected by May 31):

KTH will perform pre-test analysis and simulations for selection of operational regimes and test procedures, and post-test analysis and validation with EHS/EMS models implemented in GOTHIC against PPOOLEX tests. Further development of the EHS/EMS models for spargers and RHR nozzles will be pursued to simulate dynamics of the pool mixing and stratification. The models will be validated against respective separate effect tests.

Activity leader: Markku Puustinen, Lappeenranta University of Technology (LUT)

Funded organizations: LUT, VTT, KTH

Funding: 493 kDKK

Milestones

Lappeenranta University of Technology:

Milestones in October 2017:

1. The SRV sparger has been moved to the centre position in PPOOLEX.
2. A small scale separate effect test facility for direct measurement of momentum has been designed and constructed.
3. Wet well spray tests in PPOOLEX have been carried out.

Deliverables of LUT in 2017:

1. A SRV sparger test with the sparger in the centre of the pool.
2. A small scale separate effect facility for the direct measurement of effective momentum
3. Tests in the small scale separate effect facility with different condensation regimes
4. Wet well spray injection tests in PPOOLEX
5. Delivery of relevant experiment data to the simulation partners

VTT Technical Research Centre of Finland Ltd:

Milestones to be achieved by October 2017 are:

- Pre-calculation of small-scale separate effect test on condensation has been performed.
- The first CFD calculations of the spray effect on the stratification in the wet well have been performed.

Deliverables of the Contractor in 2017 are:

- Report on the CFD calculations of the condensation in the small-scale separate effect facility.
- Report on the CFD calculations of the spray effect on the stratification in the wet well.

Kungliga Tekniska Högskolan:

The milestones below are the expected deliverables based on the status update on May 15 (contract expected by May 31):

Deliverable 1: Pre-test analysis for selection of operational regimes and test procedures
Deliverable 2: Development of the EHS/EMS models
Deliverable 3: Post-test analysis and validation using GOTHIC and Fluent codes
Deliverable 4: Reporting

Status (May 15, 2017)

Work progressing according to plan. The update below was received on May 15.

Work at LUT, Markku Puustinen, Jani Laine, Antti Räsänen, Eetu Kotro, Lauri Pyy

<u>*Deliverable 1*</u>*:* A SRV sparger test with the sparger in the centre of the pool and with reduced submergence
The sparger test series in PPOOLEX will be completed in 2017 with the sparger first moved to an alternative position, center of the pool, and the submergence reduced from 1.8 to 1.5 m. Modifications needed to the PPOOLEX test facility and its instrumentation will be implemented as soon as KTH delivers a detailed suggestion. The test parameters will be decided together with KTH. The test will be carried out after the summer.

<u>*Deliverable 2*</u>*:* Construction of a small scale separate effect facility for the direct measurement of effective momentum
A small pool with transparent walls and a sparger pipe having a single injection hole will be designed and constructed. An updated design of the test facility will be received from KTH on week 20. The test facility will be constructed during the summer.

<u>*Deliverable 3*</u>*:* Tests in the small scale separate effect facility with different condensation regimes
Effective momentum will be evaluated with the help of a direct force measurement. High speed cameras will be used for recording condensation regimes and collapsing bubbles and high frequency pressure measurements for obtaining the detachment and collapse frequencies of the bubbles. The tests will help to map the effective momentum of many condensation regimes and hopefully will provide closures for the EMS model development for spargers by KTH. The tests will start after the summer when the construction of the test facility has been finished.

<u>*Deliverable 4*</u>*:* Wet well spray injection tests in PPOOLEX
Mixing of a thermally stratified pool with the help of spray injection from above will be studied. Test matrix is being developed on the basis of the preliminary spray tests in January 2017.

Deliverable 5: Delivery of relevant experiment data to the simulation partners
No activity.

<u>*Work at VTT*</u>*, Timo Pättikangas and Ville Hovi*

<u>*Deliverable 1*</u>*:* Report on the CFD calculations of the condensation in the small-scale separate effect facility
Pre-calculations are performed with ANSYS Fluent for the small-scale separate effect facility, where steam will be injected through a single orifice into water pool. The calculated condensation rate and penetration of the vapor jet into the pool is calculated and later compared to the experimental data.
New condensation models of ANSYS Fluent 18.0 have been tested for the pre-calculations of the small-scale separate effect facility. The Lee model and the Thermal Phase Change model of Fluent have been tested. The results have not so far been very promising. Therefore, the work will be continued with the condensation model implemented at VTT with User-Defined Functions of Fluent. Design of the test facility has been delayed at LUT, which affects the pre-calculations and achieving the Milestone in October.

<u>*Deliverable 2*</u>*:* Report on the CFD calculations of the spray effect on the stratification in the wet well
A spray experiment performed at PPOOLEX is calculated with ANSYS Fluent. The water pool is modelled with the Euler-Euler model of Fluent, where droplets are described with the Discrete

Particle Model (DPM). The effect of the spray droplets on the stratified pool is calculated. The possible deterioration of the thermal stratification of the pool is studied. The results will be compared to PPOOLEX experiment.
Preliminary wet well spray test SPR-T3 performed with the PPOOLEX facility is calculated. In the experiment, spray was injected from four nozzles into thermally stratified water pool. Simulation of mixing of the pool caused by the spray injection has been started.

Work at Royal Institute of Technology (KTH), Ignacio Gallego-Marcos, Walter Villanueva and Pavel Kudinov

Deliverable 1: Pre-test analysis for selection of operational regimes and test procedures
Design and test procedure of the separate effect facility has been proposed to LUT. The goal of this experiments is to measure the effective momentum induced by a steam injection in the oscillatory bubble and stable jet regimes. Sensitivity studies will be done on the steam mass flux, pool temperature, injection hole diameter, number of holes, and geometry of the hole (chamfer).

Deliverable 2: Development of the EHS/EMS models
EHS/EMS models for blowdown pipes have been extended to enable prediction of the pool behaviour during a prototypic LOCA transient using GOTHIC. The model estimates the condensation regime based on a new regime map, the effective momentum induced by chugging using new correlations, and allows computing non-uniform heat fluxes along the blowdown pipe walls. The model has been validated against the PPOOLEX MIX-04 experiment.

EHS/EMS models for spargers are under development in Fluent. The results show a large sensitivity on the flow structure and the effective momentum. Thus, the separate effect facility is needed to measure the effective momentum and reduce the uncertainty of the simulations.

Deliverable 3: Post-test analysis and validation using GOTHIC and Fluent codes
An in-depth analysis of the PPOOLEX and PANDA experiments with spargers has been performed. The results showed similar phenomena in PPOOLEX and PANDA in terms of steam jet condensation, and pool behaviour of stratification, erosion, and mixing. A correlation has been proposed to model the erosion velocity of the stratified layer as a function of the pool geometry and steam injection conditions. All of these observations have been used for the development of the EHS/EMS models for spargers.

Deliverable 4: Reporting
NKS report will be delivered in June.

3.2 FIREBAN

Determination of fire barriers's reliability for fire risk assessment in NPP

Summary
The scope of the project is to investigate and assess the reliability of fire barriers in NPP during realistic fire scenarios to support the plant-scale risk assessment. The objective is to establish data and methods to determine the conditional probabilities for failure of fire barrier. Statistics, literature review, calculation and specific unique designed fire tests will be used as methods. The next steps in the process are the final definition of criteria for reliability and also further calculation supported by fire tests.

Activity leader: Patrick van Hees, Lund University

Funded organizations: LU, VTT, AAU, DBI, RAB

Funding: 393 kDKK

Milestones

Tasks, milestones and deliverables until 2017-12-31

Risk-based acceptance criteria (MS1, included in D1)
Current state of the art for determination of reliability of fire barriers (MS2), First year report (D1)
Risk besign design curves for nuclear facilities (MS3)
Second Year report (D2)

Status (May 16, 2017)
Work progressing according to plan. The update below was received on May 16.

The project has now delivered it first year report. This includes an extra paper on sensitivity analysis which was produced in the end of 2016 but published early 2017. The report includes three publications of which one is a peer review paper.

The project group needed to discuss first the reduction of the budget as it had implications on the fire test program. This caused some delay in the activities and in the first year reporting. We have now signed the second year contract and it is foreseen that this will be solved in the second part of 2017. During June one of the reports in the first year report will be presented at the IAFSS conference in Lund as a poster.

The project group will meet during the IAFSS conference to discuss the progress and the test programme. No real technical problems are foreseen.

Apart from dissemination at the IAFSS conference (the largest conference on fire safety science in the world) also a workshop will be organised by VTT. Participants of the workshop are mainly from the NPP companies, Finnish authorities, and research organizations. One objective of the workshop is to create a roadmap for the Finnish Fire PRA development work.

3.3 HYBRID

Development of hybrid neutron transport methods and data visualization tools

Summary

The modelling of neutron transport typically relies on two rather opposite approaches: the probabilistic approach, and the deterministic approach. The probabilistic approach or Monte Carlo approach relies on tracking the individual lives of neutrons, and requires a large computing power for nuclear reactors. The deterministic approach, on the other hand, is based upon fast running algorithms, that solve the problem at hand in only an approximate manner.

The purpose of the present project is to combine both approaches in order to obtain fast running methods (thanks to the deterministic route) and accurate results (thanks to the probabilistic route). The so-called response matrix method was the method investigated in the first phase of the project undertaken in 2016 with NKS support. This method was originally derived in the early seventies in a pure deterministic sense. In the proposed project, the computation of the collision probabilities required for applying the method is carried out using a probabilistic solver instead.

The level of details of the simulations, and the approach allowing a direct computation of whole core problems produces a large-scale data set. There is however, a need to support rapid awareness of the complex 4D (3D + time) data-set for end users. This problem can be divided into;

a) Which data are necessary for situational awareness (power, flux, etc.)?
b) How should these data be visualized for rapid visual perception?
c) How can the visualization principles be implemented in a software application?

The outcome is enhanced visualization tools. This requires the construction of an adequate data management system with visualization capabilities. In sum, the technology is supporting the efficient development of reactor core simulations, useable first for research purposes by Chalmers, and later by commercial companies.

In 2017, the project will involve 2 MSc students under the supervision of senior scientists, and make use of the complementary expertise from Chalmers University of Technology (deterministic neutron transport), the Technical Research Centre of Finland - VTT (probabilistic neutron transport), and the Institute for Energy Technology - IFE (visualization tools).

Activity leader: Christophe Demazière, Chalmers University of Technology

Funded organizations: Chalmers, IFE

Funding: 493 kDKK

Milestones

Tasks	*Milestones*	*Deliverables*
Consolidation of the hybrid probabilistic/deterministic framework based on the work performed in 2016 for two-dimensional systems.	Assessment of the reliability of the hybrid solution as compared to the reference solution. Assessment of the influence of the boundary conditions applied in the coarse mesh modelling and in the corresponding computation of the probabilities. Estimation of the computational cost of the hybrid framework in relation to the cost of the full Monte-Carlo solution.	Short progress report (2-3 pages maximum) summarizing the work performed and the achieved milestones to be ready by October 31, 2017
Development of a benchmark between the developed method and a reference solution obtained entirely from Monte Carlo on a two-dimensional test case of small/medium size.		
Extension of the hybrid probabilistic/deterministic framework to three dimensions.	Assessment of the reliability of the hybrid solution as compared to the reference solution. Assessment of the influence of the boundary conditions applied in the coarse mesh modelling and in the corresponding computation of the probabilities. Estimation of the computational cost of the hybrid framework in relation to the cost of the full Monte-Carlo solution.	Short progress report (2-3 pages maximum) summarizing the work performed and the achieved milestones to be ready by December 31, 2017
Development of a benchmark between the developed method and a reference solution obtained entirely from Monte Carlo on a three-dimensional test case of small/medium size.		

Identify the criteria for selecting which data is suitable for visualisation in 4D (3D + time). Identifying the research-oriented principles for how to visualize the data-set.	Criteria for selecting variables suitable for visualization and examples of their use. A set of design-principles describing how to visualize the data set.	Short progress report (2-3 pages maximum) summarizing the work performed and the achieved milestones to be ready by September 30, 2017
Continue to develop problem analysis methodology. Apply this methodology to the current data structures proposed by Chalmers for the HYBRID project. Suggest a data-base architecture and demonstrate with practical examples. Demonstrate alternative methods of accessing the data as an aid to development. E.g connection of data base to Python or Matlab.	Updated report on data-architecture assessment methodology. Description and demonstration of a data-base and processing tools for HYBRID data showing a link to a general processing tool such as Python or Matlab.	Short progress report (2-3 pages maximum) summarizing the work performed and the achieved milestones to be ready by September 30, 2017
		Final report to be ready by January 31, 2018

Status (April 26, 2017)
No major deviations have been identified. The update below was received on April 26, 2017.

In summary, we are creating the conditions to have staff allocated to the project, but the actual work will not start before the summer.

For the data visualization part, IFE has started the process of finding an MSc student. There will be a presentation of the project at the Østfold University College next week. The MSc project will be starting after the summer vacation. Preliminary work by other IFE researchers are scheduled to start in the summer.

For the neutron hybrid solver, Chalmers is currently preparing the advertisement of a PhD position (the position will be mainly financed by a grant from the European Union). Nevertheless, we plan to have this PhD student working on the HYBRID project as well, as part of his/her "institutionstjäntgöring". This will hopefully create the conditions for continuity in the HYBRID project. The PhD position will start from September 1st, 2017.

3.4 NORDEC

Challenges and opportunities for improving Nordic nuclear decommissioning

Summary
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.

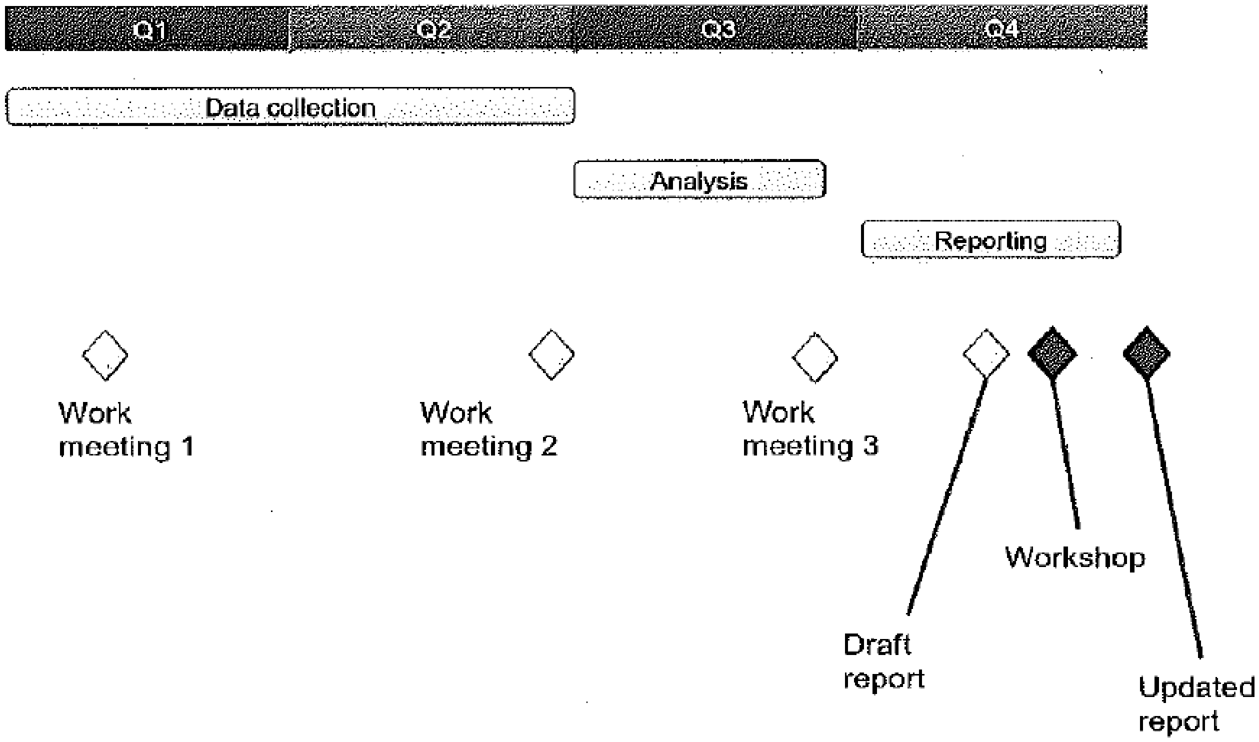
This project will conduct a study on how decommissioning is regulated, planned and performed in the Nordic countries, identify where the main challenges lie, collect best practices and share experiences between the Nordic participants. The contributions for this project will come from regulators, operators and contractors, thus having a wide span of stakeholder involvement. 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, VTT Technical Research Center of Finland, and Institute For Energy Technology (IFE) in Norway are participating in the project. The project will involve 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.

Activity leader: István Szőke, Institute for Energy Technology

Funded organizations: IFE, NRPA, SSM, STUK, SIS, VTT, Vattenfall AB, Fortum, ÅF

Funding: 524 kDKK

Milestones



As shown in the figure above, the main phases for all work packages will be:

- Data collection: Semi-structured interviews with key actors in Nordic decommissioning.
- Analysis: Work meetings, capability maturity analysis, and identification of key challenges and opportunities.
- Reporting: Draft report, workshop and final report.

Main milestones for all the activities will be three work meetings, a draft report, workshop and final report, as shown in the figure.

Work packages

Activity 1: Decommissioning of Nordic legacy sites
Data collection to identify main challenges and best practices for planning of decommissioning of legacy sites in Nordic countries

- Main challenges identified
- Interactions between regulatory body, licensees and contractors regarding decommissioning approaches and practices
- Lessons learned in a Nordic setting

Activity 2: Large scale decommissioning in a Nordic setting
Data collection to identify main challenges and best practices for planning of decommissioning of commercial reactors. Use Nordic experiences as well as the NorDec participants' collected knowledge of international lessons learned.

- Interactions between regulatory body, licensees and contractors regarding decommissioning approaches and practices
- Foreseen challenges and needs for future research and technology development
- Experiences with international decommissioning that may be transferred to Nordic projects

Activity 3: Nordic collaboration arena

Comparisons based on an analysis of the insights from Nordic decommissioning regulators, utilities, contractors and research organizations. Specific and common issues and practices among and within Nordic countries from a regulatory, licensee, contractor and research organization perspective.

- Common interests among Nordic countries for collaborative developments for solving issues and share experience and results, including adapting international lessons learned to a Nordic setting
- Ways for shortening the gap between different stakeholders in each country by collaboration around identified issues
- Issues and practices related to decommissioning strategy, e.g. immediate and deferred dismantling. Which considerations, such as radiation protection optimization, co-implementation of the decommissioning with other nuclear facilities, or the availability of disposal facilities, support the chosen strategy?

Status (May 15, 2017)
Work progressing according to updated plan. The update below was received on May 15.

Based on the current work progress there are no issues foreseen that might cause major deviations to the deliverables promised or the proposed work strategy. However, due to delayed decision from the NKS, we condensed the work schedule initially proposed. Due to the shorter schedule, we decided to merge Work meeting 2 and 3 (see schedule in the proposal) into one work meeting followed by a Workshop as planned.

Current progress

Work meetings and workshop:

- *Collaboration between participants of the project will mainly be realised through work meetings and a workshop. Work meeting 1 has been performed using video/phone conferencing. Individual phone / email communication has been conducted with participants that were not available for the meeting.*
- *Work meeting 2 is scheduled to be performed on Tuesday 13th June close to the Gardermoen airport.*
- *The Workshop is scheduled to be held in Halden, Mon 20th - Tue 21st November.*

Data collection:

- *On Work meeting 1 (video/phone meeting) main focus areas for this project have been discussed. During and after the meeting, suggestions for topics of interest have been received from the participants.*
- *Preliminary work for performing a literature review has been performed to identify a literature base for the topics of this project. One important piece of input will be the results*

of the group discussions during an international decommissioning workshop held in February by the implementation team of this project.
- *A questionnaire has been developed and sent out to the project team to be completed and forwarded to other nominees.*
- *An interview guide has been developed.*

Next steps
- *Individual interviews (in person or through video/phone) will be scheduled with people nominated by the project participants.*
- *Results of questionnaires and interview will be analysed.*
- *A second work meeting will be performed to discuss preliminary findings with the project team.*
- *Further analyses of questionnaire and interview results, cross referenced with findings of the literature study, will be discussed on a Workshop.*
- *A final report will be produced as described in the project proposal.*

3.5 SC_AIM

Safety culture assurance and improvement methods in complex projects

Summary

Despite a long research tradition, empirical studies of culture improvement in the safety field are scarce, especially in comparison to the amount of research on identifying the elements of safety culture or evaluation of safety culture. Safety culture and safety management models and practices have largely focused on single organisations, mainly in the operational phase. It is far from clear how to apply them in the dynamically changing project settings or other transitional phases such as commissioning or decommissioning. The methods that are effective in a project environment may differ from "traditional" methods of safety culture improvement as promoted by e.g. IAEA and WANO. The question is, what should a safety culture improvement and assurance program be like in an "organization" which is in a dynamic state of transition and may involve actors from different companies?

A basic premise of the project is that so far there has been a lot of attention on how to diagnose and evaluate safety culture, but actually not so much on how to improve the safety culture. A second premise is that improvement of safety culture in projects sets some unique requirements due to e.g. multiple organizations interacting, diverse background of personnel, schedules and contract issues etc. The same methods that have been applied in operating power plants may not work. Further, the long supply chains and the licensee's responsibility to oversee the safety culture of the entire network put more demands on safety culture assurance methods.

The project is planned as a two years' effort (2016-2017) between partners in two Nordic countries: VTT and Tmi Teemu Reiman (Finland) and Royal Institute of Technology, KTH (Sweden). The project has two aims:

1. To identify and specify methods to improve and facilitate safety culture in complex projects
2. To identify and specify methods to assure safety culture in complex projects

In the year 2017, we will carry out a follow-up study on the implementation progress of Safety Culture Ambassadors Group. This work provides valuable insight regarding good practices and

other experiences of implementing a safety culture improvement method in a growing organization, which is at design phase of the NPP life cycle. The information exchange with other power companies will continue, which full provide further information about safety culture improvement in various organizational contexts. The information exchange partnership is also an opportunity for the power companies to gain information from researchers. Furthermore, three researcher workshops will be held on the topics of safety culture improvement methods, assurance methods and an integrative workshop on the topic of building an adaptive safety culture in the nuclear industry. The findings from these workshops will result in three scientific publications. In addition, new methods for safety culture improvement or assurance will be developed and piloted based on needs identified in collaboration with the case organizations. Finally, the overall project findings from the two-year's effort will be documented in NKS final report.

TASKS IN 2017

1) Follow-up study at Fennovoima on the implementation progress of Safety Culture Ambassadors Group
2) Information exchange with additional organizations (incl. Forsmark, Fortum and OKG)
3) Three researcher workshops on the following topics:
    a. Safety culture improvement and assurance methods
    b. Safety culture methods and their underlying assumptions in the context of safety paradigms
    c. How to build an adaptive safety culture in dynamic organizational environments
4) Three scientific publications based on the findings from tasks 1-3
5) Development of new methods based on identified needs, potentially useful methods, and existing methods in workshops with the researchers and the case organizations
6) Pilot test of the selected new methods in the selected case organizations
7) Final report and dissemination of results
8) Administration

Activity leader: Kaupo Viitanen, VTT Technical Research Centre of Finland

Funded organizations: VTT, KTH

Funding: 279 kDKK

Milestones

MILESTONES AND DELIVERABLES IN 2017

- Researcher workshop (How to build an adaptive safety culture in dynamic organizational environments) 28.3. – 29.3.2017
- Conference paper: "Building an "adaptive safety culture" in a nuclear construction project – insights to safety practitioners" 30.4.2017
- Researcher workshop (Safety culture improvement and assurance methods) 3.5.2017 – 4.5.2017
- Researcher workshop (Safety culture methods and their underlying assumptions in the context of safety paradigms) 21.6. – 22.6.2017
- Main case study completed (Task 1) 30.6.2017
- Workshop paper and presentation: "Towards actionable safety science" [working title] 30.6.2017
- Scientific publication "Improving safety culture – what do we really know?" [working title] 31.12.2017
- Final report 31.12.2017

Status (May 15, 2017)
Work progressing according to updated plan. Note that there is a minor change in relation to original plans: the Fennovoima case study follow-up will be conducted during the autumn (instead of spring as was originally planned). The update below was provided on May 15.

<u>Project group:</u> Kaupo Viitanen (VTT, coordinator), Carl Rollenhagen (KTH/Vattenfall), Nadezhda Gotcheva (VTT)

<u>Description:</u> The SC AIM project aims to increase understanding on how to improve nuclear safety culture in complex project settings (e.g. in the presence of multiple organizations interacting, diverse background of personnel, etc.). The practical goals of the projects are to identify and specify methods to improve and facilitate safety culture in complex projects and to identify and specify methods to assure safety culture in complex projects.

<u>Overall evaluation and status:</u> Overall, the project is progressing according to plan. The only deviation from the original plans is the scheduling of Fennovoima case study follow-up (implementation of the Safety Culture Ambassadors Group), which was originally planned to be conducted in spring but will be postponed to autumn. The reason for the delay is that significant developments have not been achieved at the case study organization, rendering the follow-up study less relevant at this point. It was agreed with the Fennovoima representatives that follow-up will instead be carried out during this autumn.

To date, the project has held researcher workshops on the topics of adaptive safety culture and safety culture improvement and assurance methods. Researcher workshops will continue during the spring. A two-day researcher workshop is being arranged and will be held in Stockholm with all the participating research organizations (VTT, Tmi Teemu Reiman, KTH/Vattenfall) to enable Nordic collaboration.

The project has produced a finalized conference paper on the topic of adaptive safety culture. The paper discusses safety management from the perspective of complex adaptive systems and relates this thinking to safety culture improvement tools with the purpose of increasing understanding of how safety culture can be developed in dynamic environments such as projects. Another paper on the topic of actionable safety science is currently being written. This paper examines what needs to be taken into account when attempting to utilize insights from safety science in the practical work of safety practitioners.

Information exchange workshops with the NPPs are planned to be carried out during the autumn. Arrangements are being made for holding a workshop with OKG and Fennovoima in Stockholm in October (specific date not yet set) on the topic of Safety Culture Ambassadors. Opportunities for organizing workshops or online sessions with other NPPs are being discussed.

Milestones and deliverables for 2016		
Milestone / deliverable	*Planned completion date*	*Status*
Researcher workshop (How to build an adaptive safety culture in dynamic organizational environments)	*28.3. – 29.3.2017*	*Completed.*
Conference paper: "Building an "adaptive safety culture" in a nuclear construction project – insights to safety practitioners"	*30.4.2017*	*The conference paper was completed and submitted to Resilience Engineering Symposium. A poster presentation will be held at the conference in Liège, Belgium in 26th-29th June.*
Researcher workshop (Safety culture improvement and assurance methods)	*3.5.2017– 4.5.2017*	*The first part of this workshop was completed. Another researcher workshop on this topic will be held on 5.-6.6. in Stockholm. The arrangements and preliminary agenda for this workshop have been made.*
Researcher workshop (Safety culture methods and their underlying assumptions in the context of safety paradigms)	*21.6. – 22.6.2017*	*Arrangements have been made.*
Main case study completed (Task 1)	~~*30.6.2017*~~ *31.12.2017*	*Follow-up study at Fennovoima on the implementation progress of Safety Culture Ambassadors Group will be delayed and will be completed by the end of the year.*
Workshop paper and presentation: "Towards actionable safety science"	*30.6.2017*	*Abstract has been completed and submitted to WOS2017 conference.*
Scientific publication "Improving safety culture – what do we really know?" [working title]	*31.12.2017*	*The content has been discussed and preliminarily planned in researcher workshops*
Final report	*31.12.2017*	*The preliminary structure of the final report has been prepared*

Detailed status report by tasks		
Task description	*Estimated completion*	*Progress by 16.05.2017*
1) Follow-up study at Fennovoima on the implementation progress of Safety Culture Ambassadors Group	*5 %*	• *Will be delayed to autumn. Follow-up interviews are planned to be conducted.*
2) Information exchange with additional organizations (incl. Forsmark, Fortum and OKG)	*10 %*	• *A workshop with OKG and Fennovoima facilitated by VTT has been planned for the autumn. The date has not yet been fixed.*
3) Three researcher workshops on the following topics: *a) How to build an adaptive safety culture in dynamic organizational environments* *b) Safety culture improvement and assurance methods (partly in Stockholm)* *c) Safety culture methods and their underlying assumptions in the context of safety paradigms (in Paris together with J.-C. Le Coze and a group of invited young generation safety scientists)*	*50 %*	• *Researcher workshop a completed* • *Research workshop b partially completed*
4) Three scientific publications based on the findings from tasks 1-3 *a) "Towards actionable safety science" [working title]* *b) "Building an 'adaptive safety culture' in a nuclear construction project – insights to safety practitioners"* *c) "Improving safety culture – what do we really know?" [working title]*	*50 %*	• *Deliverable a abstract completed and submitted to WOS2017, and full paper is being written* • *Deliverable b completed and submitted to Resilience Engineering Symposium*
5) Development of new methods based on identified needs, potentially useful methods, and existing methods in workshops with the researchers and the case organizations	*10 %*	• *Preliminary ideas have been discussed with case study organizations*
6) Pilot test of the selected new methods in the selected case organizations	*10 %*	• *Preliminary ideas have been discussed with case study organizations*
7) Final report and dissemination of results	*5 %*	• *Preliminary structure of the final report has been prepared*
8) Administration	*33 %*	• *Ongoing*

3.6 SPARC

Scenarios and Phenomena Affecting Risk of Containment Failure and Release Characteristics

Summary
A robust severe accident management strategy is paramount for minimizing the environmental impact in the case of a severe accident involving melting of a reactor core. Both physical phenomena (deterministic) and accident scenarios (stochastic) are sources of uncertainties in the assessment of effectiveness of the accident mitigation. Adequate approaches are necessary in order to address both deterministic (epistemic) and stochastic (aleatory) sources of uncertainty in a consistent manner.

The goal of the project is to develop approaches and data for addressing the effects of scenarios and phenomena on the risk of containment failure and characteristics of release in case of a severe accident. There are 4 work packages that provide tightly coupled with each other activities.

WP1: *Development and application of risk oriented accident analysis framework (ROAAM+) for prediction of conditional containment failure probability for a Nordic type BWR.* (KTH)

The main tasks are:
1.1 Core degradation and relocation to the lower head (using MELCOR code). Obtained results will be compared with VTT analysis for Station Blackout (SBO) with delayed power recovery and other scenarios of risk importance.
1.2 In-vessel debris coolability (using DECOSIM code).
1.3 Debris remelting, melt pool formation and vessel failure (using PECM model).
1.4 Experiments on multi-component debris remelting will be carried out to understand basic physical phenomena.
1.5 Melt release and vessel ablation model and experiments for validation of the model.
1.6 Ex-vessel debris bed formation, agglomeration, spreading and coolability (using DECOIM and Agglomeration models). With focus on the mechanisms of the debris spreading that can help to reach a coolable state. Particulate Debris Spreading (PDS) experiments on debris spreading in the pool and after settlement will be carried out using different particles. Collaboration with VTT will be established for validation of the models.
1.7 Steam explosion (using TEXAS code) will be carried out with quantification of different sources of uncertainty.
2. Development of computationally efficient (surrogate) models for approximation of the full model response parameters.
3. Coupling of the surrogate models into ROAAM+ framework.
4. Connection of the framework to PSA-L1 and different plant damage states will be carried out in collaboration with LRC and VTT.
5. Development and implementation of the methods for quantification of uncertainty, identification of failure domains and prediction of the conditional failure probabilities using ROAAM+ framework will be carried out.
6. Development of data clustering techniques for coupling of ROAAM+ frameworks with PSA-L2, source term prediction tools and PSA-L3 will be done in collaboration with LRC and VTT.

WP2: *Development of the methods for coupling of Integrated Deterministic Probabilistic Safety Analysis tools such as ROAAM+ developed by KTH with PSA in general and PSA-L2 in particular.* (LRC)

The main tasks are:
1. Development of IDPSA generated data processing techniques for informing PSA about importance of (i) timing of events and (ii) epistemic uncertainty.
2. Different approaches will be considered in collaboration with KTH and VTT to addressing of dynamic events and physical phenomena in (i) cut sets; (ii) success and failure paths; (iii) connections to PSA-L3.
3. Cross code comparison for modelling of key phenomena of different accident progression scenarios (in collaboration with WP1 and WP3).

WP3: *Deterministic modelling of core degradation, melt relocation, vessel failure, debris spreading and coolability and threats for the containment integrity.* (VTT)

The main tasks are:
1. Development and verification of modelling approaches to core degradation, melt relocation and vessel failure. Comparison of MELCOR and ASTEC results for SBO with delayed power recovery and other scenarios of risk importance in collaboration with KTH.
2. Implementation and validation of debris bed spreading models (e.g. Lagrangian particle tracking model in CFD) against PDS-P data in collaboration with KTH.
3. Analytical investigation of the effect of debris bed multidimensionality on coolability (using the CFD approach developed at VTT and the MEWA code). This consist of refining the temperature-based coolability criteria for heap-like debris beds, which is a main unresolved question in the coolability of realistic debris beds. Collaboration with KTH on comparison of results obtained with DECOSIM code analysis.
4. MELCOR analyses of hydrogen explosions in order to address the interactions between deterministic phenomena, stochastic events and operator actions (in collaboration with WP1 and WP4).
5. MC3D analysis on the effect of vessel breaking mode to dynamic pressure loads on cavity wall induced by steam explosion.
6. Consideration of the implications of the analysis results for source term characteristics in collaboration with KTH, LRC and WP4.

WP4: *Level 2 PSA modelling of phenomena and factors affecting containment failure probability and release characteristics. The input is from KTH, LRC and VTT analysis in WP1, WP2 and WP3.* (VTT)

The main tasks are:
1. PSA-L2 analysis with the focus on the factors affecting source term characteristics. The factors to be considered are: (i) plant damage states (from PSA level 1), (ii) plant design and (iii) accident progression phenomena.
2. Consideration of the factors affecting the probability and magnitude of relevant phenomena such as (i) hydrogen explosions (in collaboration with WP3), (ii) steam explosions (in collaboration with WP1); (iii) non-coolable debris bed formation and core-concrete interaction (in collaboration with WP1 and WP3).

Activity leader: Pavel Kudinov, Royal Institute of Technology.

Funding: 524 kDKK

Milestones

KTH work is focused in WP1 on Tasks 1, 2, 3 and 5 with the following goals:

1. Development and validation of detailed (full) deterministic models for analysis of severe accident phenomena in Nordic BWRs.
2. Development and application of computationally efficient surrogate models for uncertainty and risk analysis.
3. Collaboration with VTT and LRC on cross code comparison, code validation, and development of approaches to informing PSA with the ROAAM+ framework results.
4. Reporting of the results.

LRC work is focused in WP2 on Task 1 and 2 with the following goals:

1. Perform the integration case outlined during 2016 with a large scale PSA model.
2. The test case will identify the need for further necessary refinements of the method for including time into cut sets (dynamic approach).
3. Discuss the potential for the method (dynamic cut-set) to be used for other purposes than tested in the integration test.
4. Reporting of the results.

VTT work in WP3 will be focused on Task 3 and 5 with the following goals:

1. Further analyses of debris bed temperature in post-dryout conditions for developing temperature-based coolability criterion.
2. Analysis on the effect of vessel breaking mode to the steam explosion loads.
3. Comparison of obtained results with KTH and LRC data.
4. Reporting of the results.

VTT *in* WP4 will be focused on Task 1 and 2 with the following goals:

1. PSA-L2 analysis results addressing important factors for the release characteristics.
2. Consideration of the relevant phenomena, namely steam and hydrogen explosions.
3. Reporting of the results.

Status (May 16, 2017)

The partners have agreed upon the distribution of the funding offered by NKS. Contracts have been received from LRC and VTT. The contract from KTH is expected by May 31.

From the status report received on May 16, work is progressing according to updated plan from CfP 2017. There are no major deviations between plans and results so far. Five articles have been published in peer-review journals since last NKS Board meeting in January. Two PhD theses have been published.

More details about the status of SPARC can be found in the attached update in Appendix A.

3.7 WRANC

Warm Pre-Stressing – Validation of the relevance of the main mechanisms behind Warm Pre-Stressing in assessment of nuclear components

Summary

The embrittlement of the reactor pressure vessel (RPV) due to extended operation can lead to difficulties in demonstrating safe operation beyond 40 years when using traditional assessment methods. Therefore, utilizing the beneficial WPS (Warm Pre-Stressing) effect in assessments is important for continued operation beyond 40 years of the RPV. The practise of utilizing the beneficial WPS effect in RPV assessments have been adopted already in several European countries. However, there are still some uncertainties about the limitations of the engineering methods that are being used. These uncertainties need to be addressed to ensure safe utilization of the WPS effect.

The WPS effect is the increase of the apparent brittle fracture toughness for a ferritic component when pre-loaded at a temperature in the ductile upper shelf region and then cooled to the brittle lower shelf region of the material fracture toughness transition curve.

The WPS effect can be attributed to four main mechanisms. These mechanisms have different impact, depending on the pre-load level and load path. All the mechanisms are related to plastic straining at pre-load. The engineering methods used today do not consider constraint and do not take into account the different impact of the mechanisms in relation to different load paths.

There is a need to evaluate thoroughly the importance of the four main mechanisms behind WPS for realistic situations that could be encountered in a RPV. This in order to understand the limitations and possibilities in the engineering methods used to assess the magnitude of the WPS effect.

Within this research project (Inspecta Technology AB (Sweden), Royal Institute of Technology (Sweden), SINTEF (Norway) and Swedish Radiation Safety Authority (Sweden)), the main mechanisms behind WPS and their importance relating to RPV assessments will be validated using both experiments and numerical methods. This project will try to answer the question of which of these mechanisms, or combination of, is the governing mechanism in situations that closely resemble those that can arise in a RPV. This is important to be able to assess the reliability and limitations of the engineering methods that are employed today in assessing the magnitude of the WPS effect. The results will also be used to formulate guidelines in utilizing the WPS effect in RPV assessments.

Activity leader: Tobias Bolinder, Inspecta Technology AB

Funding: 393 kDKK

Milestones

Tasks and Deliverables:

Design of experimental program	2017-03-31
Execution of experimental program	2017-04-30
Numericall investigation	2017-05-31
Fractographical examination	2017-05-31
Formulate guidelines	2017-06-30
Final report	2017-09-30

Status (May 16, 2017)

Work progressing according to updated plan. The original experimental program has been expanded and revised. The update received on May 16 reveals that informative results were obtained from the first test sets. Additional funding from SSM enables the project to expand the experimental program with new test sets, see attached update in Appendix B for details.

Below is a summary of deliverables.

Completed tasks:

- *Design of experimental program*
- *Acquired material for testing (RPV steel 18MnD5) from EDF France.*
- *Numerical modelling*
- *Numerical investigation (Master thesis)*
- *Manufactured test specimens for experimental program*
- *Carried out 70 % of the complete experimental program*

Remaining tasks:

- *Complete the experimental program (will be completed before the mid of June)*
- *Carry out the fractographical examination (delayed, SINTEF have not yet received the test specimens this should not delay the final report)*
- *Analyse the results*
- *Write final report*

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Nordic nuclear safety research

4 Overview of all NKS-R activities in 2010-2016

It is seen from the table below that all activities started in 2015 and earlier have been finalised. ATR-2015 needs to submit a revised final report.

An activity is considered to be started at the January board meeting, and ended when the final report has been delivered.

Activity	NKS number	Started	Ended
Decom-sem	NKS_R_2010_83	01/2010	12/2010
DIGREL	NKS_R_2010_86	01/2010	12/2010
IACIP	NKS_R_2008_61	01/2010	12/2010
INCOSE	NKS_R_2009_75	01/2010	05/2011
MOSACA10	NKS_R_2008_69	01/2010	01/2011
NROI	NKS_R_2008_70	01/2010	04/2011
POOL VTT	NKS_R_2007_58	01/2010	05/2011
POOL KTH	NKS_R_2007_58	01/2010	06/2011
POOL LUT	NKS_R_2007_58	01/2010	03/2011
AIAS	NKS_R_2011_98	01/2011	12/2012
DIGREL	NKS_R_2010_86	01/2011	01/2012
ENPOOL	NKS_R_2011_90	01/2011	03/2012
ENPOOL	NKS_R_2011_90	01/2011	05/2012
ENPOOL	NKS_R_2011_90	01/2011	05/2012
MoReMO	NKS_R_2011_95	01/2011	02/2012
NOMAGE4	NKS_R_2008_63	01/2011	11/2011
POOLFIRE	NKS_R_2011_96	01/2011	02/2012
SADE	NKS_R_2011_97	01/2011	03/2012
RASTEP	NKS_R_2010_87	06/2011	09/2012
AIAS	NKS_R_2011_98	01/2012	06/2013
DECOSE	NKS_R_2012_100	01/2012	07/2013
DIGREL	NKS_R_2010_86	01/2012	02/2013
ENPOOL VTT	NKS_R_2011_90	01/2012	04/2013
ENPOOL LUT	NKS_R_2011_90	01/2012	03/2013
ENPOOL KTH	NKS_R_2011_90	01/2012	05/2013
MoReMO	NKS_R_2011_95	01/2012	03/2013
Nordic-Gen4	NKS_R_2012_103	01/2012	11/2012
POOLFIRE	NKS_R_2011_96	01/2012	02/2013
RASTEP	NKS_R_2010_87	01/2012	10/2013
SADE	NKS_R_2011_97	01/2012	03/2013
Decom-sem	NKS_R_2013_106	01/2013	02/2014

Activity	NKS number	Started	Ended	
DECOSE	NKS_R_2012_100	01/2013	10/2014	
DIGREL	NKS_R_2010_86	01/2013	03/2014	
DPSA	NKS_R_2013_107	01/2013	07/2014	
ENPOOL	NKS_R_2011_90	01/2013	10/2014	
Exam HRA	NKS_R_2013_110	01/2013	03/2014	
HUMAX	NKS_R_2013_108	01/2013	02/2014	
L3PSA	NKS_R_2013_109	01/2013	03/2014	
POOLFIRE	NKS_R_2011_96	01/2013	12/2014	
SADE	NKS_R_2011_97	01/2013	02/2014	
ATR	NKS_R_2014_111	01/2014	06/2015	
DECOSE	NKS_R_2012_100	01/2014	07/2015	
DIGREL	NKS_R_2010_86	01/2014	02/2015	
DPSA	NKS_R_2013_107	01/2014	08/2015	
ENPOOL	NKS_R_2011_90	01/2014	07/2015	
HUMAX	NKS_R_2013_108	01/2014	01/2015	
L3PSA	NKS_R_2013_109	01/2014	04/2015	
Nordic-Gen4	NKS_R_2012_103	01/2014	02/2015	
ProCom	NKS_R_2014_112	01/2014	03/2015	
ADdGROUND	NKS_R_2015_113	01/2015	04/2016	
ATR-2015	NKS_R_2014_111	01/2015	06/2016	revised report to be completed
COPSAR	NKS_R_2015_114	01/2015	08/2016	
DECOSE	NKS_R_2012_100	01/2015	10/2016	
L3PSA	NKS_R_2013_109	01/2015	11/2016	
LESUN	NKS_R_2015_115	01/2015	12/2015	
MODIG	NKS_R_2015_116	01/2015	03/2016	
PLANS	NKS_R_2015_117	01/2015	01/2016	
ADdGROUND	NKS_R_2015_113	01/2016	Not completed	
BREDA-RPV	NKS_R_2016_118	01/2016	03/2017	
COPSAR	NKS_R_2015_114	01/2016	Not completed	two out of three partners are done
FIREBAN	NKS_R_2016_119	01/2016	Not completed	draft report has been received
HYBRID	NKS_R_2016_120	01/2016	04/2017	
L3PSA	NKS_R_2013_109	01/2016	03/2017	
SC_AIM	NKS_R_2016_121	01/2016	01/2017	
SPARC	NKS_R_2016_122	01/2016	Not completed	

STATUS REPORT OF NKS-SPARC PROJECT
Scenarios and Phenomena Affecting Risk of Containment Failure and Release Characteristics
May 15, 2017

Work at Royal Institute of Technology (KTH), Division of Nuclear Power Safety NKS-SPARC and APRI-9

Pavel Kudinov, Galushin, Sergey, Dmitry Grishchenko, Sergey Yakush, Alexander Konovalenko, Simone Basso, Walter Villanueva.

WP1: Development and application of risk oriented accident analysis framework (ROAAM+) for prediction of conditional containment failure probability for a Nordic type BWR.

KTH work is focused in WP1 on Tasks 1, 2, 3 and 5 (see details below) with the following goals:

1. Development and validation of detailed (full) deterministic models for analysis of severe accident phenomena in Nordic BWRs.
2. Development and application of computationally efficient surrogate models for uncertainty and risk analysis.
3. Collaboration with VTT and LRC on cross code comparison, code validation, and development of approaches to informing PSA with the ROAAM+ framework results.
4. Reporting of the results.

1.1 Core degradation and relocation to the lower head (using MELCOR code). Obtained results will be compared with VTT analysis for Station Blackout (SBO) with delayed power recovery and other scenarios of risk importance.

MELCOR model of Nordic BWR has been used to evaluate the effect of severe accident scenario (timing of activation of safety systems) on the resultant properties of relocated debris in LP. Obtained Typical debris configurations are: small relocation; large relocation with significant debris oxidation; large relocation with smaller debris oxidation; transition regime. Major part of the core relocates to LP within ~30-60min after onset of core support plate failure. ECCS is effective in preventing massive core relocation only within relatively small time window after activation of ADS. Delay in activation of ADS can significantly delay massive core relocation to LP and results in greater extent of core materials oxidation. Debris composition (i.e. metallic/oxidic debris fraction) in different layers are highly influenced by severe accident scenario.

Sensitivity analysis has been performed to evaluate the effect of modelling options in MELCOR on the properties of relocated debris in LP. The most influential parameters for determining debris mass in LP and time of core support plate failure are: oxidized fuel rod collapse temperature and particulate debris porosity. Hydrogen generation and metallic debris fraction in the first axial level are mostly affected by: velocity of falling debris and particulate debris porosity. Non-linear interactions between physical models in MELCOR make results sensitive to selection of numerical parameters.

Data base of MELCOR solutions is being generated with new versions of MELCOR 2.1 and 2.2. Lower plenum nodalization has been refined to obtain properties of relocated debris in LP in the vicinity of vessel Lower Head. Noticeable differences have been found between predictions with 2.1 and 2.2, while a reasonable agreement was observed between 1.8.6 and 2.1 versions. Investigation of the reasons for the discrepancies is ongoing. Computationally efficient Core Relocation Surrogate Model will be used for prediction of the properties of relocated debris in LP that are necessary for the analysis of in-vessel debris coolability, debris remelting, melt pool formation and vessel failure in ROAAM+ framework.

1.2 In-vessel debris coolability (using DECOSIM code).

Coolability of a porous debris bed in the lower plenum of reactor pressure vessel is considered using DECOSIM code in the conditions of limited water supply, with initially dry and hot porous debris beds of different shapes (flat-top, heap), mass, and properties (e.g. particle size). It was shown that for larger particles, water penetration into the initially hot debris bed proceeds mainly along the vessel wall. As a result, temperature escalation and remelting occurs in the top part of debris bed. For smaller particles, hot zone can be in direct contact with the wall. Total evaporation of water occurs faster for larger particles due to different rates of water ingress. Temperatures of debris in the locations of the nozzle welds for penetrations are studied to clarify possible vessel failure modes for different debris bed configurations.

1.3 Debris remelting, melt pool formation and vessel failure (using PECM model).

An approach is under development for coupling of the PECM model with MELCOR data on the debris properties in the lower head for analysis of debris bed heatup, remelting, and melt pool formation is ongoing.

1.4 Experiments on multi-component debris remelting will be carried out to understand basic physical phenomena.

REMCOD (REmelting of MultiCOmponent Debris) with quartz walls was designed and manufactured. The feasibility of experimental approach i.e. no apparent wall effects and a possibility of visualization of debris remelting through a quartz glass in a 2D sliced test section is demonstrated in the commissioning tests. Exploratory tests of the REMCOD are ongoing with tin as melt simulant and steel and ceramic and glass particles as debris bed simulants. In each test up to 0.5 liters of superheated tin (~400 ºC) is poured into debris. The depth of penetration apparently depends on the debris: sizes and thermal properties (heat capacity and thermal conductivity). Several series of tests are planned to investigate (i) the effects of absolute temperature of debris and temperature profile; (ii) the effect of specific interfacial area (determined by the particle size); (iii) the effect of wettability of debris by liquid metal.

1.5 Melt release and vessel ablation model and experiments for validation of the model.

Melt release and vessel ablation model is currently under development. The ablation rate of initial breach is predicted given transient melt release properties. The preliminary analysis using the model has demonstrated that total melt mass defines final jet diameter. Melt release velocity affects jet diameter at the water surface and respective risks of containment failure due to steam explosion or debris bed agglomeration and non-coolability. Importance of different phenomena of debris remelting and melt relocation that may delay or limit melt release from the vessel are investigated parametrically.

1.6 Ex-vessel debris bed formation, agglomeration, spreading and coolability (using DECOSIM and Agglomeration models). The focus is on the mechanisms of the debris spreading that can help to reach a coolable state. Particulate Debris Spreading (PDS) experiments on debris spreading in the pool and after settlement will be carried out using different particles. Collaboration with VTT will be established for validation of the models.

Coolability of the debris depends on the bed shape. Therefore, particle spreading (i) after settlement on the debris bed; (ii) in the water pool above the bed affect coolability. Debris self-levelling model, based on PDS-C (Particulate Debris Spreading – Closures) experiments, was used to carry out sensitivity and uncertainty analysis for the efficacy of the particulate debris spreading in prototypic accident conditions. An artificial neural network was employed as a surrogate model (SM). It is demonstrated that conditional containment failure probability (CCFP) due to non-coolable debris bed can vary in wide ranges depending on the combinations of the randomly selected probability distributions for the input parameters. Sensitivity analysis identified: effective particle diameter and debris bed porosity as the largest contributors to the output uncertainty.

Investigation of the debris spreading driven by large turbulent flows in the pool (PSD-P) is ongoing in order to develop a database with wider ranges of pool configuration, particle properties and debris release conditions. The work on validation of the DECOSIM code against PDS-P experimental is ongoing. Predicted and experimental mass distributions of debris at the bottom of the pool (local values and mean spreading distance) were compared. A reasonable agreement is observed for steel and glass particles. A series of experiments on two-phase flows (no particles) with flow pattern identification has been carried out. Further validation of the code is ongoing. The study of the influence of debris agglomeration on coolability is ongoing using DECOSIM code with (i) impermeable "cake" on top of the bed and (ii) distributed fraction of agglomerates that reduce open porosity for coolant flow. It is shown that while a dry and hot zone almost certainly develops in the debris bed with a "cake", there exist conditions under which the dry zone temperature can stabilize at some level by steam cooling.

1.7 Steam explosion analyses (using TEXAS code) will be carried out with quantification of different sources of uncertainty.

The Full Model (FM) for analysis of steam explosion in Nordic BWR was implemented in TEXAS-V. A statistical characterization of the possible explosion energetics for a single melt release scenario was introduced, considering different possible timing of the triggering. An extended database of FM solutions has been generated and is used for the development of a computationally efficient Surrogate Model (SM) that predicts impulses corresponding to 50, 65, 78, 95, 99 and 100% percentiles of the cumulative distribution for the explosion impulse. The results of the failure domain analysis in ROAAM+ framework for Nordic BWR suggest that the impulse will be higher than ~6kPa s in most of the possible met release scenarios if jet diameters is larger than Ø26 cm (independently on the other parameters) or if jet diameters >Ø10 cm in case if melt release velocity is high (>7 m/s) and water pool depth is large (>9m). If jet diameters are limited to 30 cm the probability of exceeding 50 kPa s impulse is less than 10^{-3} for the most of the possible combinations of eth distributions of the uncertain parameters. Several critical modelling assumptions have been verified and demonstrated to be valid: fixed ratio between the jet radius and the mesh cell cross section, reduced free gas volume compared to containment volume.

Task 5: Development and implementation of the methods for quantification of uncertainty, identification of failure domains and prediction of the conditional failure probabilities using ROAAM+ framework.

A new approach was proposed for taking into account the uncertainty in approximation of the surrogate model of the full model solution. The approach is currently employed for analysis of the containment failure due to steam explosion.

Work on Tasks 4,6 is postponed due to reduction of the project budget:

Task 4: Connection of the framework to PSA-L1 and different plant damage states will be carried out in collaboration with LRC and VTT.

Task 6: Development of data clustering techniques for coupling of ROAAM+ frameworks with PSA-L2, source term prediction tools and PSA-L3 will be done in collaboration with LRC and VTT.

Work at LRC Loyd's Register Consulting – Energy AB
NKS-SPARC and LRC:

Yvonne Adolfsson, Ola Bäckström

LRC is responsible for WP2: Development of the methods for coupling of Integrated Deterministic Probabilistic Safety Analysis tools such as ROAAM+ developed by KTH with PSA in general and PSA-L2 in particular.

LRC work is focused in WP2 on Task 1 and 2 (see detailed status of each task below) with the following goals:

1. Perform the integration case outlined during 2016 with a large scale PSA model.
2. The test case will identify the need for further necessary refinements of the method for including time into cut sets (dynamic approach).
3. Discuss the potential for the method (dynamic cut-set) to be used for other purposes than tested in the integration test.
4. Reporting of the results.

Task 1: Development of IDPSA generated data processing techniques for informing PSA about importance of (i) timing of events and (ii) epistemic uncertainty.

Perform the integration case with a large scale PSA model

Risk Spectrum model of Nordic BWR has been used to evaluate the impact on the PSA results based on achieved results from MELCOR studies of severe accident scenario with the ROAAM+ framework which is described in WP1. Different methods to implement the results in PSA model has been studied during the implementation phase.

The test case will identify the need for further necessary refinements of the method for including time into cut sets (dynamic approach).

The results from the performed PSA study on the test case shows to some extent new results. Since the study is small these results indicates that there is a need for further refinement before, eventually, new assumptions are made related to the studied phenomena described in WP1 and how they in general are represented in a PSA model. The recommendations are at the moment formulated and reviewed.

Task 2: Different approaches will be considered in collaboration with KTH and VTT to addressing of dynamic events and physical phenomena in (i) cut sets; (ii) success and failure paths; (iii) connections to PSA-L3.

Studies have been made related to different possibilities to incorporate results from IDPSA analysis with the PSA model. Result from these studies have been presented at the SAFECOMP'2016 "*Effective Static and Dynamic Fault Tree Analysis*". The use of dynamic fault trees makes it possible to enrich the static analysis with a more precis modelling but they are at present only possible to use in small models. The paper presents so called SD fault trees. The purpose of these is to give the user a possibility to specify failure data as either traditionally, statical, or more dynamical. The applicability has also been studied on fault trees of nuclear power plants. Conclusions from studies with large PSA models has also been presented in "*Dynamic Features in Large PSA Studies*" at ESREL 2016.

Task 3: Cross code comparison for modelling of key phenomena of different accident progression scenarios (in collaboration with WP1 and WP3).

The comparison is on-going.

Work at VTT Technical Research Centre of Finland Ltd NKS-SPARC and SAFIR2018:

VTT is responsible for WP3 and WP4 of the SPARC project:

Anna Nieminen, Magnus Strandberg, Veikko Taivassalo

WP3: Deterministic modelling of core degradation, melt relocation, vessel failure, debris spreading and coolability and threats for the containment integrity. (VTT)

VTT work in WP3 will be focused on Task 3 and 5 (see detailed status of each task below) with the following goals:

1. Further analyses of debris bed temperature in post-dryout conditions for developing temperature-based coolability criterion.
2. Analysis on the effect of vessel breaking mode to the steam explosion loads.
3. Comparison of obtained results with KTH and LRC data.
4. Reporting of the results.

Work on Tasks 1-2 is postponed due to reduction of the project budget:

Task 1: Development and verification of modelling approaches to core degradation, melt relocation and vessel failure. Comparison of MELCOR and ASTEC results for SBO with delayed power recovery and other scenarios of risk importance in collaboration with KTH.

Task 2: Implementation and validation of debris bed spreading models (e.g. Lagrangian particle tracking model in CFD) against PDS-P data in collaboration with KTH.

Task 3: Analytical investigation of the effect of debris bed multidimensionality on coolability (using the CFD approach developed at VTT and the MEWA code). This consist of refining the temperature-based coolability criteria for heap-like debris beds, which is a main unresolved question in the coolability of realistic debris beds. Collaboration with KTH on comparison of results obtained with DECOSIM code analysis.

VTT's MEWA results on debris bed post-dryout temperature behaviour were compared to KTH's DECOSIM results and notable differences were found. Previously the effect of heat transfer models were studied and now the focus has been on the effect of friction models. Tung & Dhir models are considered being the most complete since they include also friction between liquid and gas. Modified Tung and Dhir model is considered most suitable for analysing small particle cases, but there are several versions of the model in different codes and code versions. The work on solving the reason for the differences between MEWA and DECOSIM results continue.

Task 4: MELCOR analyses of hydrogen explosions in order to address the interactions between deterministic phenomena, stochastic events and operator actions (in collaboration with WP1 and WP4).

The existing MELCOR input deck for Nordic BWR plant was converted from MELCOR 1.8.6 to MELCOR 2.1. The results of the new and old version were compared by analyzing a SBO accident scenario. New version produced lower corium temperature in the cavity, which caused differences e.g in timing of the RPV failure, containment pressure and concrete ablation. The risk of hydrogen fire in the reactor building was studied analyzing hydrogen concentrations in different volumes. The results for the SBO scenario showed such low concentrations that a hydrogen fire is considered very unlike. Also a SBO accident with a noninerted containment was analysed and this resulted in a hydrogen fire in the containment. However, these hydrogen fires did not cause explosions. The timing of the RPV failure was also very similar to the standard case result.

Task 5: MC3D analysis on the effect of vessel breaking mode to dynamic pressure loads on cavity wall induced by steam explosion..

Previously steam explosion loads in Nordic BWR geometry were assessed and sensitivity of the results to key input parameters was examined using MC3D code. First, the effect of triggering time was analysed. The results showed that as long as the mixture is triggerable the strength of the resulting explosion does not change notably. Sensitivity analysis results showed that the melt drop size that is dependent on the physical properties of the melt had the strongest effect on the explosion strength. Surprisingly, the melt temperature did not affect the explosion strength as long as the temperature was high enough to cause an explosion. Also different side breaks scenarios were tested in 3D but here the mixture did not trigger despite high explosivity value. This result is considered unphysical and several attempts were made to complete the analysis unsuccessfully. Now the effect of RPV breaking location on dynamic pressure load on cavity wall induced by a steam explosion has been analysed with MC3D performing a functioning 3D simulation. The results seem promising as mixture is now properly ignited and the results are much more in line with what could be expected based on the theory and previous 2D simulations..

Task 6: Consideration of the implications of the analysis results for source term characteristics in collaboration with KTH, LRC and WP4.

The work is postponed.

Ilkka Karanta, Tero Tyrvainen

WP4: Level 2 PSA modelling of phenomena and factors affecting containment failure probability and release characteristics.

***VTT** in* WP4 will be focused on Task 1 and 2 (see detailed status of each task below) with the following goals:

1. PSA-L2 analysis results addressing important factors for the release characteristics.
2. Consideration of the relevant phenomena, namely steam and hydrogen explosions.
3. Reporting of the results.

Task 1: PSA-L2 analysis with the focus on the factors affecting source term characteristics, i.e. release energy (temperature), altitude, and probability. The factors to be considered are: (i) plant damage states (from PSA level 1), (ii) plant design and (iii) accident progression phenomena.

Release height and temperature have been considered for different accident scenarios based on general knowledge, literature and discussions with deterministic safety analysis experts. Roughly speaking, there are three different cases with regard to the release height:

- The release height is the height of the place where the reactor building leaks after containment failure.
- The release height is the height of the stack if filtered venting is performed.
- An explosion throws the releases in the air above/surrounding the containment and reactor building.

In most cases, the location of the containment failure (which normally can be inferred from containment failure mode) is the basis for the analysis of the release height, but the reactor building also affects the height significantly. Therefore, in addition to the containment failure modes, the migration paths of the radionuclides in the reactor building need to be analysed to determine release height accurately. This is a challenge because safety analyses focus mostly on events occurring inside the containment.

Literature search gives very little about the release heights directly. Some papers where release heights for Fukushima accident were given were found. Concerning release altitude when the containment fails, a list of possible containment failure modes for generic BWR's and PWR's has been constructed. The list is based on research literature and international guidance (IAEA, Asampsa), and contains the failure modes, prerequisites of failure in a particular mode, and some major possible causes of a failure in a particular mode.

Also release energy has not received much attention in the scientific literature. The temperature of release from containment is in most cases close to 100°C, but the temperature of radionuclides can potentially change during their migration in reactor building. Building structures can cool down radionuclides, whereas fires and explosions either within or outside the containment can cause higher temperatures of the atmospheric release. There are fluid dynamics software that can be used to analyse radionuclide flows in reactor building and determine the release heights and temperatures.

An old BWR containment event tree model (see VTT-R-05974-13) has also been developed further by implementing uncertainty analysis for release probabilities, and adding release height and temperature variables. The plan is to continue the development of the model in the forthcoming years. A conference paper on the model was published in PSAM 13 proceedings:

Task 2: Consideration of the factors affecting the probability and magnitude of relevant phenomena such as (i) hydrogen explosions (in collaboration with WP3), (ii) steam explosions (in collaboration with WP1); (iii) non-coolable debris bed formation and core-concrete interaction (in collaboration with WP1 and WP3).

Hydrogen explosions in a BWR plant have been studied based on literature and discussion with deterministic safety analysis experts. Results from deterministic analyses were not available in 2016.

Typically in BWR plants, the containment is inert during operation which prevents the hydrogen explosions from occurring inside the containment with certainty. However, the containment is not inert during start-up and shut-down, and accidents occurring at those times can lead to hydrogen explosions. Also, it is possible that the inerting system fails. Hydrogen explosions are typically modelled in level 2 PSAs of BWRs in very simple ways with conservative probabilities.

In practice, there are three probabilities that need to be determined for a given accident scenario:

1. the probability that the containment is not inert

2. the probability that an explosion occurs if the containment is not inert

3. the probability that the containment is broken if an explosion occurs.

The probability that containment is not inert due to start-up, shut-down or refueling can be taken from level 1 results. In addition, reliability analysis can be performed for the inerting system to account for the possibility of inerting failure during normal operation. If the containment is not inert, the probability of an explosion can be analysed based on deterministic simulations that determine hydrogen and steam volumes in different accident scenarios. MELCOR software could be used, but start-up, shut-down and refueling require different models than normal operation. Deterministic analyses can also be used to estimate the strength of an explosion in order to estimate the probability of containment failure.

Hydrogen explosion can also occur outside the containment if hydrogen leaks from the containment to the reactor building. This kind of hydrogen explosions occurred at Fukushima causing significantly larger releases than what had occurred before that because the roofs were destroyed. Despite of their potentially significant impact on the releases, explosions outside containment have not usually been modelled in PSA. Taking ex-containment hydrogen explosions into account causes the need to redefine level 2 PSA, because traditionally level 2 has stopped at the loss of containment integrity whereas ex-containment hydrogen explosion is most likely a consequence of a containment leak. A more comprehensive definition of level 2 would encompass anything in accident progression inside the reactor building (naturally including the containment) that may affect release probability or its characteristics. PSA modelling will be considered in 2017 based on the results of WP3. In addition to events leading to loss of containment integrity, also events and factors leading to an ex-containment hydrogen explosion, and its consequences to release characteristics have to be taken into account; therefore PSA modelling is different from the inside containment case; Nevertheless, the approach of utilising deterministic analyses in the extended model can be quite similar.

Overall Project Summary

Comparison between plans and results with explanation of any deviations:
There are no major deviations between plans and results so far.

Expected submit date of the final report

- Expected date for submitting the reports for 2016 is mid of June 2017.

Any issues you would like the board to know

- No.

Relevant Publications

Defended PhD Dissertations:

1. Viet-Anh Phung, "Input Calibration, Code Validation and Surrogate Model Development for Analysis of Two-phase Circulation Instability and Core Relocation Phenomena," KTH, March, 2017.
2. Simone Basso, "Particulate Debris Spreading and Coolability," KTH, April, 2017.

Peer reviewed publications

1. Dmitry Grishchenko, Simone Basso, Pavel Kudinov, "Development of a surrogate model for analysis of ex-vessel steam explosion in Nordic type BWRs," Nuclear Engineering and Design, Volume 310, 15 December 2016, Pages 311-327, 2016.
2. Basso S., Konovalenko A., and Kudinov P., "Preliminary Probabilistic Risk Analysis of Debris Bed Coolability for Nordic BWRs Under Severe Accident Conditions," Nuclear Engineering and Design, Submitted 2017.
3. Galushin S. and Kudinov P., "Analysis of Core Degradation and Relocation Phenomena and Scenarios in a Nordic-type BWR," Nuclear Engineering and Design, Volume 310, 15 December 2016, Pages 125–141, 2016.
4. Kudinov P., Grishchenko D., Konovalenko A., Karbojian A. "Premixing and Steam Explosion Phenomena in the Tests with Stratified Melt-Coolant Configuration and Binary Oxdic Melt Simulant Materials," Nuclear Engineering and Design, Volume 314, Pages 1-338 (1 April 2017).
5. L. Manickam, P. Kudinov, W.M. Ma, S. Bechta and D. Gishchenko, "On the influence of subcooling and melt jet parameters on debris formation," Nuclear Engineering and Design 309: 265-276, 2016.
6. Phung, V.-A. Galushin, S. Raub, S. Goronovski, A., Villanueva, W., Kööp, K., Grishchenko, D., Kudinov, P., "Characteristics of debris in the lower head of a BWR in different severe accident scenarios," NED, Volume 305, 15, August 2016, pages 359-370, 2016.
7. Basso S., Konovalenko A., Yakush S. E. and Kudinov P., "The Effect of Self-Leveling on Debris Bed Coolability Under Severe Accident Conditions," Nuclear Engineering and Design, Volume 305, 246-259, 2016.
8. Basso S., Konovalenko A., Yakush S. E. and Kudinov P., "Effectiveness of the debris bed self-leveling under severe accident conditions," Annals of Nuclear Energy, Volume 95, September 2016, Pages 75-85, 2016.
9. Basso, S., Konovalenko, A., Kudinov, P. "Empirical Closures for Particulate Debris Bed Spreading Induced by Gas-Liquid Flow", Nuclear Engineering and Design, 297, 19-25, (2016).
10. Konovalenko A., Basso S., Kudinov P., Yakush S. E., "Experimental Investigation of Particulate Debris Spreading in a Pool", Nuclear Engineering and Design, Volume 297, pp208-219, 2016.
11. Y Butkova, H Hermanns, P Krcal, O Backstrom, W Wang, "*Dynamic Features in Large PSA Studies*" ESREL 2016.
12. Tyrväinen T, Silvonen T, Mätäsniemi T. Computing source terms with dynamic containment event trees. 13th International Conference on Probabilistic Safety Assessment and Management, PSAM 13, 2 - 7 October 2016, Seoul, Korea. International Association for Probabilistic Safety Assessment and Management, IAPSAM (2016).

Inspecta

Status report for Project: *WRANC*

Short summary:

The embrittlement of the RPV due to extended operation can lead to difficulties in demonstrating safe operation beyond 40 years when using traditional assessment methods. Therefore, utilizing the beneficial WPS (Warm Pre-Stressing) effect in assessments is an important possibility for demonstrating continued safe operation beyond 40 years of the RPV.
The WPS effect is the increase of the apparent brittle fracture toughness for a ferritic component when pre-loaded at a temperature in the ductile upper shelf region and then cooled to the brittle lower shelf region of the material fracture toughness transition curve. The WPS effect can be attributed to four main mechanisms. These mechanisms have different impact, depending on the pre-load level and load path. All the mechanisms are related to plastic straining at pre-load.
The project will contribute to answer which are the active main mechanisms behind the WPS effect for different situations that are realistic in a RPV. A thorough study of this has not, to our knowledge, been published before. This will lead to an understanding of the possibilities and limitations of the engineering methods for WPS. Hence, the project will clarify limitations for safe use of engineering methods for utilizing the WPS effect in RPV integrity assessments.
The approach taken to answer this is a combination of both numerical and verifying experimental work. Within the experimental program two of the mechanisms are isolated to evaluate their individual contribution to the WPS effect at different loading conditions. The remaining two mechanisms are studied using advanced numerical methods.
The experimental work will also lead to a deeper understanding of the origin of the initiation sites and the effect that WPS have on the initiation sites for brittle fracture.

Completed tasks:

- Design of experimental program
- Acquired material for testing (RPV steel 18MnD5) from EDF France.
- Numerical modelling
- Numerical investigation (Master thesis)
- Manufactured test specimens for experimental program
- Carried out 70 % of the complete experimental program

Remaining tasks:

- Complete the experimental program (will be completed before the mid of June)
- Carry out the fractographical examination (delayed, SINTEF have not yet received the test specimens this should not delay the final report)
- Analyse the results
- Write final report

Status of the project

A master thesis has been conducted within the project. The master thesis focused on two of the mechanisms in the warm pre-stressing phenomenon. By the use of a probabilistic model for evaluating cleavage fracture the two mechanisms have been evaluated and compared. The two mechanisms that were evaluated were the closing residual stress field around a macroscopic crack tip and the change of material properties during cooling.
The main results and conclusions from the master thesis will in short be summarised below.
The LCF (Load Cool Fracture) load cycle was the most beneficial load cycle. Thus a conclusion was that the mechanism referred to as ‘change of material properties during cooling’ is a more beneficial mechanism than the residual stress field.

As can be seen in Figure 1 the LTUCF (Load Transient Unload Cool Fracture) cycles showed similar results as the results obtained by the LUCF (Load Unload Cool Fracture) load cycles which suggests that path independence can be assumed from the end of preloading to the start of reloading under the assumption that the load does not increase during this phase.
Furthermore, as can be seen in Figure 1 below, the LPUCF (Load Partially Unload Cool Fracture) load cycles showed results similar to the results obtained from the LCF load cycles even though a large portion of the preload had been unloaded. This suggests that the mechanism called "change of material properties during cooling" is the dominating mechanism of the two investigated. All the results of the master thesis will be incorporated in the final report.

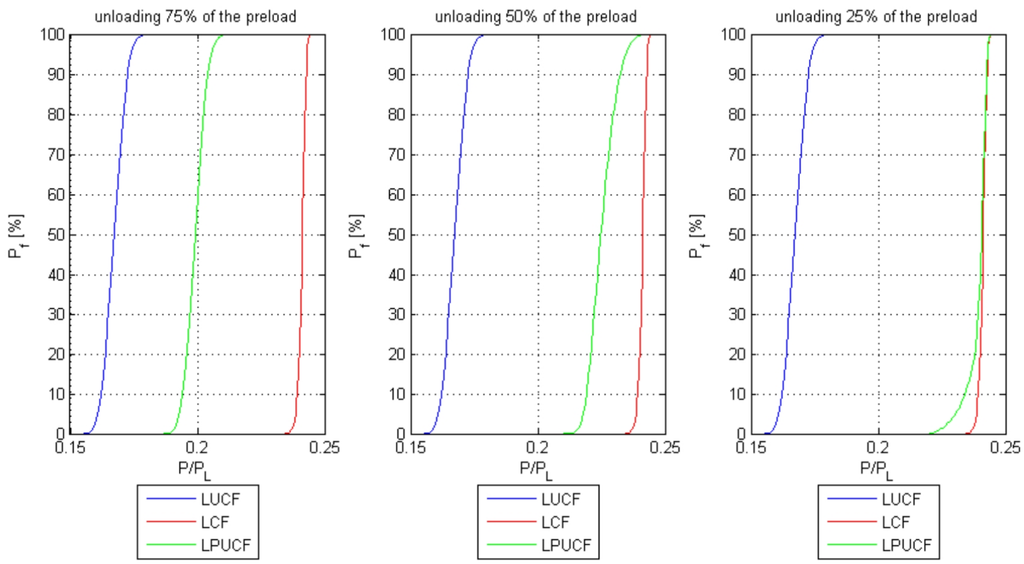

Figure 1. Comparison of LUCF, LCF and LPUCF load cycles with a pre-load level corresponding to load level C/D (J=105 kN/m).

The original experimental program have been expanded and revised. The motive for the changes arouse after the first Sets (Set 1, 2, 3, and 4) had been conducted at a pre-load level C/D. Very informative results were obtained from the first sets. The changes consists of adding three additional test sets with 7 specimens in each leading to 21 additional test specimens. The budget of the project has thus increased. Part of the funding for this increase of the project have been received from the Swedish Radiation Safety Authority.
Two additional sets Set 5 and 6 were added. Both sets have a LUCF load cycle and with sharp crack and EDM crack respectively. Set 5 and 6 are not heat treated after pre-loading. Set 5 and 6 were added as references cases with the full WPS effect. In addition Set 3 and 5 are pre-loaded to two different load levels (level A/B and C/D). This was earlier planned for set 3 and 4. Finally a Set 2* has been added where an EDM crack is machined with the same notch size as the notch size for Set 4 after pre-load. The revised experimental program is shown below in Figure 1.

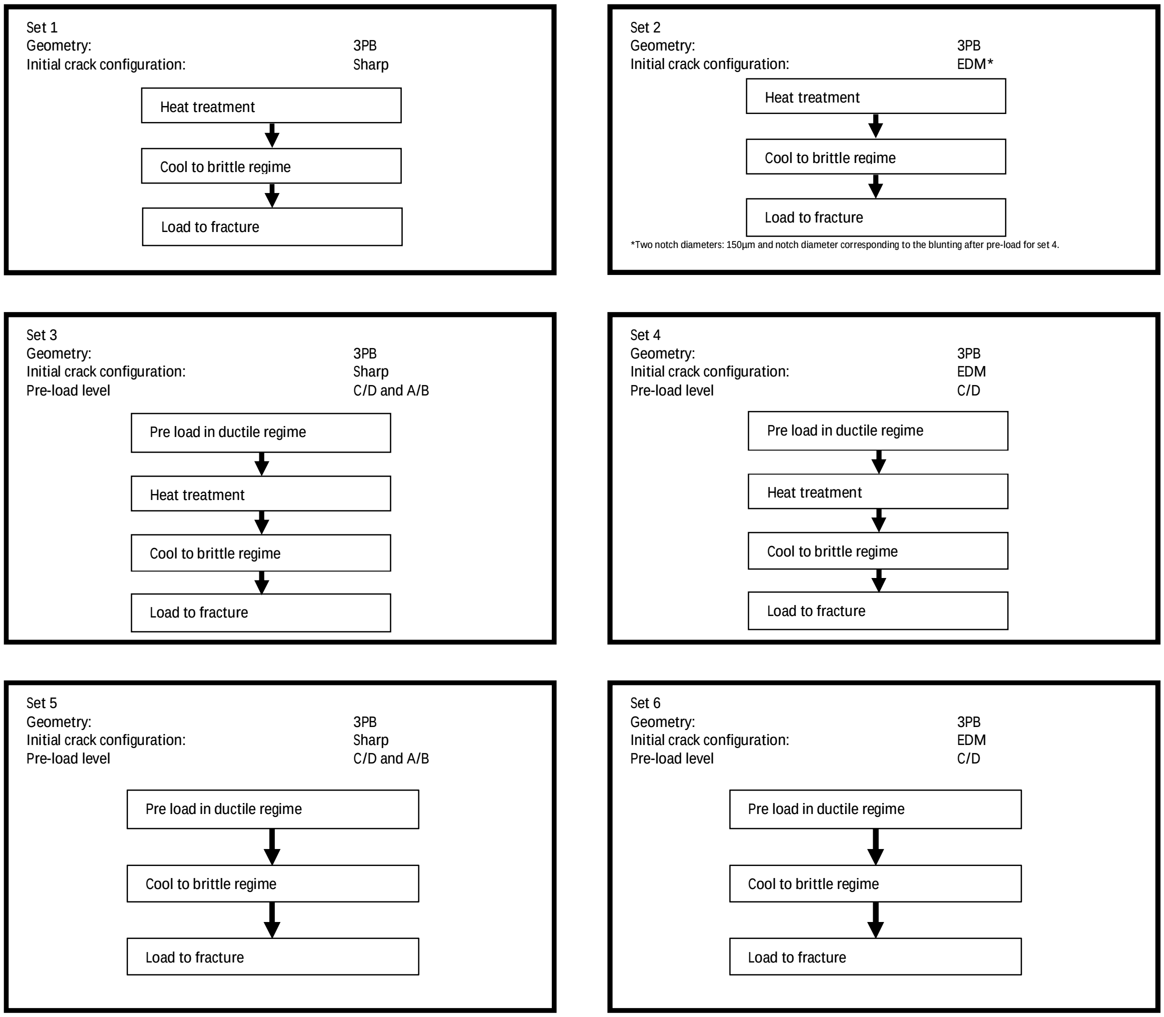

Figure 1. Revised experimental program a total of 63 test specimens.

Approximately 70 % of the test program have been completed. The results of the performed tests show some very interesting results. In Figure 2 results from Set 1, 2, 3 and 5 with a pre-load level corresponding to level C/D (K_I=155 MPam$^{1/2}$) are shown. As can be seen from the results a clear WPS effect is seen for Set 5 which is the set without any heat treatment after pre-load. This is as predicted. Active WPS mechanisms in Set 5 are compressive residual stress field, blunting of crack tip and inhibition of initiation sites. We can also see an effect on the fracture toughness for Set 2 compared with the reference case Set 1 where the mechanism is the artificially blunted crack tip due to EDM machining. Hence crack tip blunting can contribute to the WPS effect, it should be noted though that the size of the EDM crack for Set 2 is approximately 155 µm and this should be compared with the blunting due to pre-load which is approximately 72 µm at a pre-load level C/D (K_I=155 MPam$^{1/2}$). This is interesting due to the results from Set 3 which show a higher apparent fracture toughness even without a compressive residual stress field (heat treatment reduce the residual stresses to approximately 10%) and with a less blunted crack tip. This leads to the preliminary conclusion that the mechanism of inhibition of initiation sites is a mechanism that is active at least for high pre-load levels.

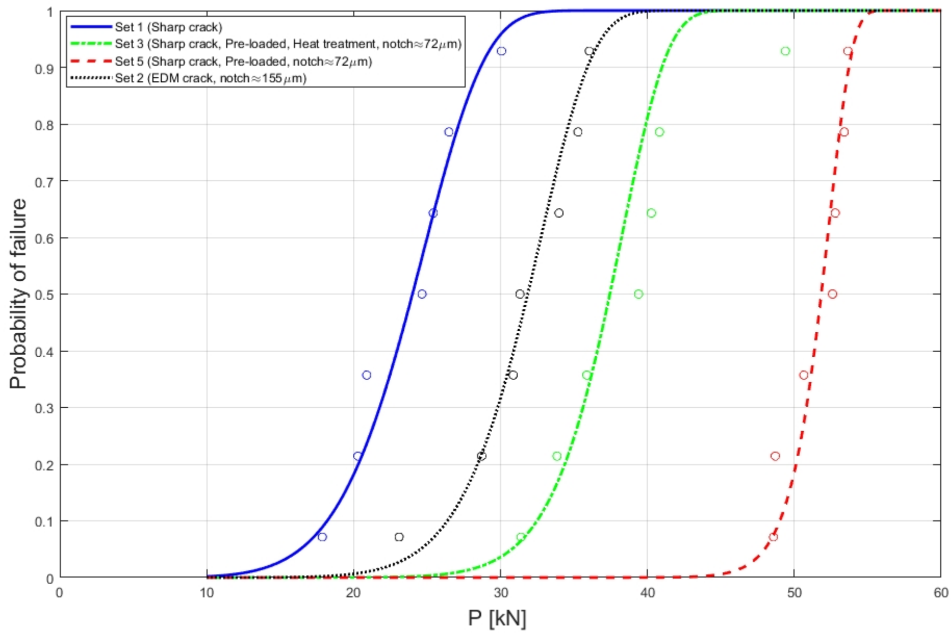

Figure 2. Preliminary experimental results from Set 1, 2, 3 and 5.

Remaining work

The experimental work will be completed with Set 6, Set 2* with pre-load level C/D and Set 3 and 5 with a pre-load level corresponding to load level A/B (K_I=70 MPam$^{1/2}$). This work is planned to be completed in the end of May or early June.

It remains to carry out the fractographical examination. This work has not yet started and is therefore delayed according to the original plan. The test specimens have not yet been sent to SINTEF. This should not necessarily influence the date for the delivery of the final report.

The results from the experimental program and fractographical examination will be studied and from this conclusions will be drawn. This work has already started by analysing the numerical and experimental results.

Write final report were the numerical, experimental and fractographical results are all presented and discussed to obtain a comprehensive understanding of mechanisms behind the WPS-effect.