SKC Annual Report 2007

Swedish Centre for Nuclear Technology

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Contents

Summary of 2007

Two PhD projects sponsored by SKC were finished in 2007. One project reached its licentiate thesis presentation status. Eleven PhD projects have been in progress during 2007 with funding from SKC. Five PhD theses were competing for the Sigvard Eklund prize to the best thesis of the year. Dr. Carl Sunde won the prize for his work on process status evaluations using noise measurements in BWRs and PWRs.

A large number of students have followed the courses relevant to the nuclear power industry. An international masters programme in nuclear energy engineering was started at KTH with 14 students attending. More than 30 students have attended nuclear engineering related courses at KTH and almost 20 at Chalmers. At Uppsala several nuclear technology related courses are part of the ordinary masters in engineering education with a large number of students participating. Furthermore, Uppsala university carries out graduate and post-graduate courses in nuclear engineering and radiation protection according to an agreement with KSU (Kärnkraftsäkerhet och Utbildning). Nine master theses were competing for the Sigvard Eklund price. Andreas Oskarsson won the prize for his work on starting a BWR with a core having a positive isothermal reactivity coefficient.

SKC finished its second six year period in 2007. Towards the end of the year, the financing partners agreed to continue supporting the centre for another six year period. SKC will now be organised as a centre at KTH.

SKC Sponsors in 2007

SKC has been sponsored by the following organisations during the six year period ending in 2007:

- Barsebäck Kraft AB
- Forsmark Kraftgrupp AB
- OKG AB
- Ringhals AB
- Swedish Nuclear Power Inspectorate (SKI)
- Westinghouse Electric Sweden AB

Total support from these organisations has been more than 90 million Swedish kronor for the six year period.

2007 was the last year in a 6-year agreement

SKC financing organisations have contributed 96 million kr during the last six years

Svenskt Kärntekniskt Centrum – SKC – finished its sixth and last year under the contract between the funding organizations and the universities KTH, Chalmers and Uppsala. The funding organizations have contributed 16 million Swedish kr each year to senior positions at the universities and to research projects. Totally 11 PhD students have been sponsored by SKC during the six year period.

The funding organizations have been:

- Barsebäck Kraft AB
- Forsmarks Kraftgrupp AB
- OKG AB
- Statens Kärnkraftinspektion
- Ringhals AB
- Westinghouse Electric Sweden AB

During the second half of 2007, the financing organizations agreed upon to continue the SKC activities, with the exception of Barsebäck Kraft AB who decided to leave and Ringhals AB took on the continued financing responsibility. Therefore, the organizational pre-requisites were developed for a new six year period and the parties determined to fund the activities with 17 million Swedish kr per year. This is further described on page 26.

The SKC Board has included:

- Bertil Dihné, Chairman, Vattenfall Bränsle (replaced in the summer of 2007 by Lennart Billfalk, Vattenfall)
- Eva Telg, Barsebäck Kraft AB
- Lars Berglund, Forsmarks Kraftgrupp AB
- Håkan Talts, OKG AB
- Leif Johansson, Ringhals AB
- Gustaf Löwenhielm, SKI
- Nils-Olov Jonsson, Westinghouse (replaced in the summer of 2007 by Stig Andersson)
- Tomas Lefvert, Vattenfall, director of SKC

Druing the six year period SKC has provided financial support to senior positions at KTH, Chalmers and Uppsala as outlined in the contracts with these organizations. Research projects have been performed on the basis of proposals to the SKC Board, who has decided which projects best suit the goals and purposes of the organization.

KTH – Royal Institute of Technology

Overview of Activities in 2007

At KTH, research and education within the field of nuclear energy engineering is performed at the following divisions:

- ✓ Reactor physics (department of physics)
- ✓ Reactor technology (department of physics)
- ✓ Nuclear power safety (department of physics)
- ✓ Nuclear chemistry (department of chemistry)

These four divisions are jointly members of CEKERT (Centre for nuclear energy technology) at KTH, together with representatives of SKI, Westinghouse atom and Forsmark kraft AB. Within CEKERT, the internal distribution of funds obtained from SKC is agreed. Further, CEKERT functions as a think-tank and centre of coordination for joint actions, such as the KTH masters programme in nuclear energy engineering.

In late 2007, 24 senior scientists and 26 PhD students were employed by the CEKERT divisions. Roughly 2/3 of the effort is focused on SKC related projects (i.e. R&D on the existing Swedish light water reactor park), the remainder being issues related to spent fuel, such as repository performance and transmutation.

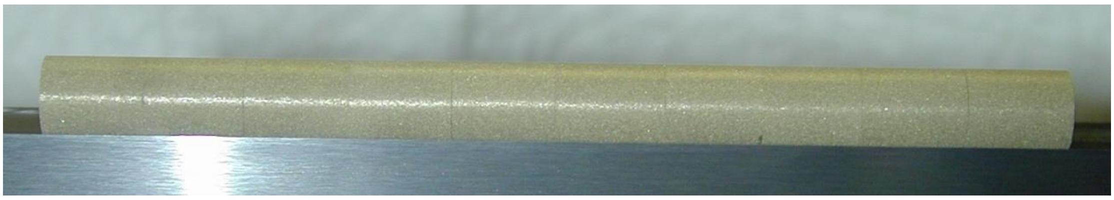
Three PhD thesis and five licentiate thesis were defended during 2007:

1) Daniel Westlen: Why faster is better - on minor actinide transmutation in hard neutron spectra (PhD in reactor physics).
2) Krzysztof Karkoszka: Mechanistic modelling of water vapour condensation in presence of noncondensable gases (PhD in reactor technology)
3) Fredrik Nielsen: The steady state approach – a model describing the dynamics of spent nuclear fuel dissolution in ground water (PhD in nuclear chemistry)
4) Carl-Magnus Persson: Reactivity assessment in sub-critical systems (Lic. Eng. in reactor physics)
5) Jan Dufek: Advanced Monte Carlo methods in reactor physics: eigenvalue and steady state problems (Lic. Eng. in reactor physics)
6) Roberta Concilio Hansson: An experimental study on the dynamics of melt-water micro interactions in a vapour explosion (Lic. Eng. in nuclear power safety)
7) Chi Thanh Tran: Development, validation and application of an effective convectivity model for simulation of melt pool heat transfer in a light water reactor lower head (Lic. Eng. in nuclear power safety).
8) Sandra García: The impact of groundwater chemistry on the stability of bentonite colloids (Lic. Eng. in nuclear chemistry).

Further, Andreas Oskarsson was awarded with the Sigward Eklund prize for best diploma work in nuclear energy engineering for a study of kinetics during a transient with positive isotermal coefficient.

A major event occurring in 2007 was the start of the international masters programme in nuclear energy engineering, with 14 students from the following countries: Estonia, Spain, Italy, Bulgaria, Mongolia, China, Pakistan, India, Sudan and the United States. Four out of these students are female. KTH students can select individual courses from the masters programme and a total of 33 students attended the reactor physics course in 2007.

On the research side, the CONFIRM irradiation of two (Pu,Zr)N fuel pins started in November 2007 in HFR (Petten). The CONFIRM project is coordinated by reactor physics at KTH, also providing an optimised process for fabrication of the fuel pins as a result of thermo-chemical modelling. The figure shows the CONFIRM fuel pellets as fabricated by PSI in Switzerland.



(Pu,Zr)N fuel pellets now under irradiation in HFR, fabricated by PSI according to a recipy developed at KTH.

Six SKC funded projects have been in progress during 2007. A summary description is provided for five of these projects in the following pages.

Study of post-dryout heat transfer and internal structure of annular and mist two-phase flows in annuli with spacers

Research leader: Associate Professor Henryk Anglart

PhD student: Ionut Anghel, Division of Nuclear Reactor Technology, KTH, Stockholm

Introduction

In the nuclear reactors one of the important influence to the operation, either economical, either safety is to understand the impact of the spacers on the heat transfer both in pre-CHF and post-CHF. The main function of the spacers is to maintain the fuel rod equidistant in the fuel rod assembly. The second function also very important is to enhance heat transfer in order to reduce the probability of dryout occurrence. Recently, the spacer design has become a very sensitive process in optimization of the geometry being a trade-off between pressure losses and improved dryout performance. Needless to say that understanding of the hydro-dynamic and thermal process may be helpful in the spacers design technology as well in the evaluation of the thermal margins which is very important parameter in the reactor operation.

The present project contains a detailed experimental program which has the main objective to improve the predictive capabilities of the spacers in pre- and post-dryout heat transfer. As a result a detailed experimental database will be obtained which is necessary to develop and validate a new mechanistic model for post-dryout heat transfer.

Objective of the project

Latest experimental study of post-dryout heat transfer in an annulus 10x22.1x3650 has shown a considerable influence of the spacers on the heat transfer intensity. The measurements reveal that dryout patches can be effectively quenched just downstream of spacers and that the measured wall temperature is much lower than that predicted from currently available correlations and/or models [1-6]. Figure 1 shows the measured heat surface superheat distribution on inner wall for low mass flux of cooling water. The temperature is reduced downstream of the spacer and the temperature reduction is easily seen.

In order to have a detailed view of the spacer influence the following issues will be addressed: influence of spacers shape and blockage ratio on heat transfer, level and influence of vapour superheat on heat transfer, exact position of quenching/rewetting fronts, influence of spacer in drop dynamics.

The post dryout measurement will be performed in two side heated annulus using various type of spacers. The main flow parameters such as liquid film flow rate, drop size, drop velocity, drop distribution and vapor temperature will be measured for various distances from a spacer and under different flow conditions. An original technique will be used to measure the wall temperature.

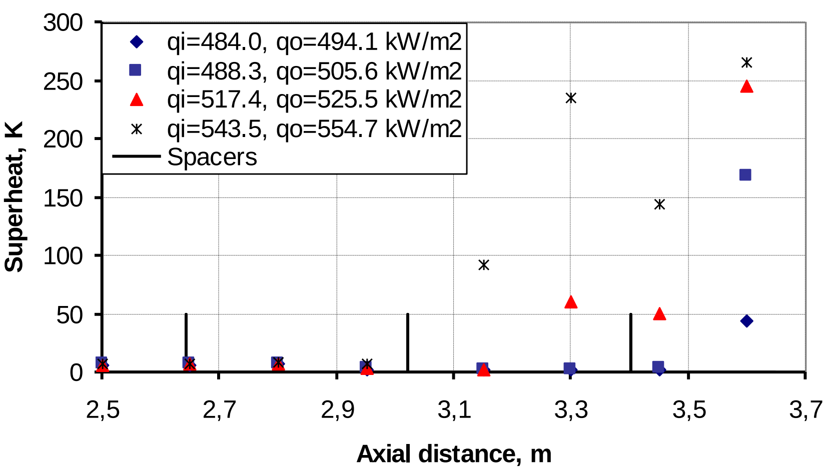


Fig. 1. Measured superheat of rod surface. Mass flux G = 500 kgm^{-2}s^{-1}, inlet subcooling ΔT_{sub} = 10 K, pressure p = 7 MPa, from [1].

Experimental matrix set-up

The experimental matrix includes measurements of wall temperature distributions for single and two phase flow and for both convective boiling and post-dryout heat transfer. Various mass fluxes, inlet subcooling values and system pressures will be studied.

During year 2007, the main effort was focused in rebuilding the KTH 250 bars loops. Since experiments will be carry out at very high temperatures (over 850 K) and high pressure (70-150 bars), parts of the loop have been replace. The new turbine flow-meter, new control valve and a new safety valve have been installed. The cooling side from the condenser has been completely replaced and two new control magnetic valves have been installed.

In order to be able to run with the desired inlet temperature conditions at the test section an electrical control PPR3-150 for the existing preheater has been mounted. The electrical control system for the entire loop has been renewed. A special attention has been payed to the operational safety, and for this purpose a control room has been built. The main focus is currently on the design of the test section and the device for measuring temperature. In parallel a Labview computer program is updated.

References

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[2] Groeneveld, D.C., Leung, L.K.H, Zhang, J., Cheng S.C., Vasic, A., 1999. Effect of appendages on film-boiling heat transfer in tubes, Proc. 9th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, San Francisco, USA.

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[6] Adamsson, C., Anglart, H., 2006, “Film Flow Measurements in High Pressure Diabatic Annular Flow with Various Axial Power Distributions”, Nuclear Engineering and Design, vol. 236, pp. 2485-2493.

Development of a Method for the Treatment of Two-Phase Flow Patterns in Nuclear Reactor Thermal Hydraulic CFD-Based Analysis

Scientist: Viet-Anh Phung

Research Leaders: Professor Nam Dinh, Dr. Pavel Kudinov

Division of Nuclear Power Safety, KTH, Stockholm

Background

Reactor thermal-hydraulics system computer codes based on two-fluid model, such as RELAP5 and TRACE, play an important role in assessing safety analysis for nuclear systems. These codes provide economical calculation tool in term of computational time, while giving reasonably good results for system steady-state and transient conditions. However, for closure of the two-fluid model equations the codes employ flow regime based empirical correlations. Moreover, the two-fluid model uses time and volume-averaged parameters of flows. The neglect of physical effects while using empirical correlations together with volume averaging results in ill-posed character of the model. Thus, there is a concern that the codes will fail in calculating complex transient state such as strongly oscillating two-phase flows, which exhibits rapid transitions between bubbly, slug and annular flow regimes. In addition to the system codes, Computational Fluid Dynamics (CFD) codes have been already used in new reactor system design. Their application is expected to be significant in future reactor system safety analysis as computing power is being improved rapidly. However, 3-dimensional two-phase flow simulations using CFD for LWR thermal-hydraulics and safety analysis are still rare and facing many challenges. A vital challenge to CFD codes for predicting two-phase flow accurately and reliably is to develop methods to overcome the weakness of averaged model and to introduce information about flow patterns and flow regime history into consideration.

Goal of the project

The first objective of the project is to analyze the performance of existing system codes in predicting two-phase flow in BWR instability transients, review and analyze modeling methods for description and prediction of two-phase flow regimes in unsteady flows. The second objective is to formulate and develop an effective treatment of two-phase flow regimes in 3D-CFD calculations. Information about the flow pattern mechanism, flow pattern history and other flow parameters will be utilized in flow regime recognition method for flow regime treatment purpose. Such numerical model will be applied to the calculation of the flow regime transitions such as bubbly-to-slug or slug-to-annular. The third goal is to validate the developed method on two-phase flow regime transition in a system code using selected two-phase flow experiments related to reactor thermal hydraulics.

Organization

The work is performed by a PhD student Viet-Anh Phung under the direction of Professor Truc-Nam Dinh and Dr. Pavel Kudinov, with scientific advice of Dr. Tomasz Kozlowski. The members of the reference group are: Ninos Garis (SKI), Gustav Dominiqus (Forsmark), Claes Halldin (OKG), Henrik Nylén (Ringhals) and Anders Andrén (Westinghouse).

Methodology

Activities in year 2007 include the analysis of RELAP5 system code performance in predicting unsteady, oscillatory flow in BWR system. For this purpose, SIRIUS-N tests were simulated using RELAP5 and the calculation results were compared with experimental data. SIRIUS-N was a natural circulation test loop facility which was started at Central Research Institute of the Electric Power Institute (CRIEPI, Japan) in 1993. The main objective of the SIRIUS-N project was to investigate flashing-induced instability and Type-I instability in large-scale natural circulation BWRs.

Simulation of SIRIUS-N experiments was performed and analyzed for cases at low system pressure (0.2 MPa) and high system pressure (2 MPa), at both steady and unsteady states, concentrated on intermittent and sinusoidal oscillatory flows. RELAP5 modeling results obtained for the cases at low system pressure, low heat flux, high inlet subcooling were relatively close to the experimental values. Therefore, it appears that RELAP5 can predict intermittent oscillatory flow at this operating range. However, the oscillation amplitude and frequency of the flow velocity were overestimated, as shown on Figure 1. For the cases at high system pressure, higher heat flux and low subcooling, modeling results were significantly different from the experiment.

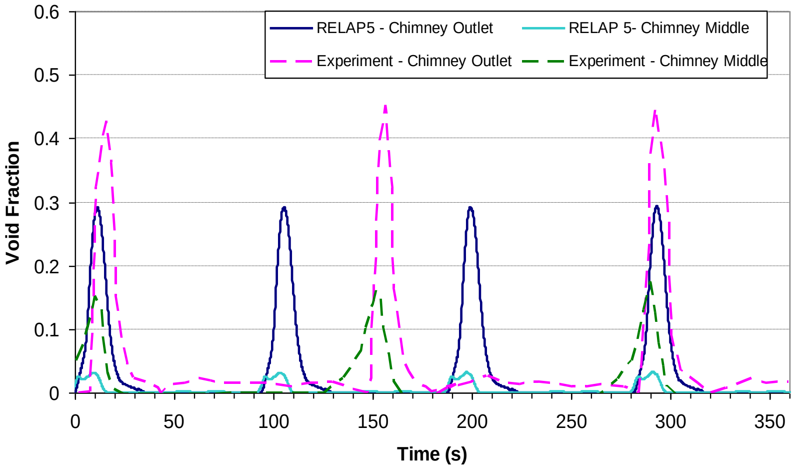


Figure 1: RELAP5 void fraction prediction for intermittent oscillation flow at low pressure.

The possible reasons for the discrepancy between simulation results and experiment are following. First, there is a deficiency of the code in predicting oscillatory flow with large velocity and void fraction amplitudes. In all oscillation cases modeled, transitions between bubbly and slug (and annular) flow regimes were observed where the code can fail to correctly predict such flow transition. Second, it is questionable that the code is capable of predicting operating points close to the stability boundary. And third, the geometrical uncertainty of the model itself, such as lack of the downcomer orifice data, and the utilization of equivalent downcomer, contributes to the result inaccuracy.

Development of a Multi-Scale Simulation Methodology for Nuclear Reactor Thermal Hydraulic and Safety Analysis

Scientist: Francesco Cadinu

Research Leaders: Professor Nam Dinh, Dr. Tomasz Kozlowski

Division of Nuclear Power Safety, KTH, Stockholm

Background

The thermal-hydraulics analysis of nuclear power plants has been traditionally carried out using so-called system thermal-hydraulic codes (STH). RELAP5, TRACE, CATHARE, ATHLET are notable members of this class.

The STH are based on a multi-fluid model of two-phase flow, whose closure is provided by flow regime maps and constitutive relations for fluid-fluid and fluid-wall mass/momentum/energy exchange. From the geometrical point of view, they employ a one-dimensional description of the plant, simplified as a series of control volumes. Even though this approach may seem overly simplified, system codes have enabled analysts to successfully perform simulations of complex transients of safety relevance. However, they cannot capture correctly the features of those transients where the multi-dimensionality of the flow plays a key role.

It was soon recognized, in the nuclear engineering community, that Computational Fluid Dynamics (CFD) could complement system codes in the toolbox of the safety analyst. Based on the solution of the Navier-Stokes equations, CFD has the capability to analyze multi-dimensional flows and, beyond the nuclear industry, it is considered a well-developed and reliable tool, especially for single-phase applications.

The most natural way to couple CFD and system codes is what we refer to as a "domain decomposition" approach. Namely, the computational domain is divided in a "CFD subdomain" and a "STH subdomain" where the corresponding solvers are used. Matching conditions, regarding the primitive variables or the fluxes, are imposed at the interface between different subdomains. We refer to this approach as "hard coupling". While this is very intuitive, there are fundamental issues such as the coupling instability between codes marching at different time steps and the difficulty of obtaining boundary conditions for the CFD subdomain from the 1D data provided by the system code. This suggests looking for alternative ways to couple CFD and system codes.

Goals of the project

Viewing the STH/CFD coupling as a multi-scale problem, our goal is to explore the possibility to introduce, in the analysis of nuclear power plants, the latest advances in the theory of multi-scale techniques for heterogeneous systems (such as E and Engquist's Heterogeneous Multiscale Method and Kevrekidis' Equation Free Method). At the same time we aim to create a theoretical basis and practical recommendations to guide the development of "hard coupling" algorithms when appropriate.

The ultimate goal is to develop a multi-scale computational platform for reactor safety and thermal-hydraulics analysis which makes an effective use of the available computational tools (system codes, CFD codes). The coupled code system and methodology developed will be used to perform plant simulations, to gain insights into various feedbacks and cross-scale interactions.

Organization

The work is performed by Ph.D. student Francesco Cadinu under the direction of Professor Truc-Nam Dinh and Dr. Tomasz Kozlowski, with scientific advice of Dr. Pavel Kudinov. The contact reference group consists of Wiktor Frid and Oddbjörn Sandervåg (SKI), Lilly Burel-Nilsson and Thomas Probert (OKG), Farid Alavyoon (Forsmark), Anders Andren (Westinghouse), Henrik Nylen (Vattenfall).

Methodology

The first part of the work has been devoted to a literature review on multi-scale methods with the goal of assessing their suitability for application to the CFD/STH coupling problem. Most multi-scale methods found in the literature are problem specific, so they cannot be easily extended. However, it was found that two popular frameworks used to study multi-scale systems, namely the Equation Free Method

(EFM) and the Heterogeneous Multiscale Method (HMM) are general enough to be considered for application to the CFD/STH coupling problem. In particular, it can be shown that it is possible to devise a coupling strategy between a system code and a CFD code which can be cast in a form very similar to the Equation Free Method. Such a strategy prescribes the use of a CFD code to calculate the coefficients (closures) needed by the system code. For the efficiency of the method, it is essential to enable the system code with the capability to decide when, during the transient, to invoke the CFD code. This procedure is called closure-on-demand.

The closure-on-demand strategy will be able to simulate phenomena where the overall behavior of the system is strongly influenced by local 3D effects due, for example, to the presence of spacer grids or other complicated geometrical features. This procedure is thought to be more robust than the domain decomposition approach with respect to the fact that accurate initial and boundary conditions for the CFD solver cannot be determined from the 1D data provided by the system code. It must be remarked, however, that the closure-on-demand strategy is not an appropriate simulation tool for those transient where the goal is the calculation of 3D flow and temperature patterns. This shortcoming is inherited from the philosophy underlying both the EFM and HMM. The objective of these methods, in fact, is the calculation of the macroscale evolution, and the microscale solution is used either to calculate the macroscale quantities of interest after an averaging procedure (EFM) or to provide closures or fluxes needed by the macroscale solver (HMM).

Publications

Francesco Cadinu, Tomasz Kozlowski, Truc-Nam Dinh, Relating System-to-CFD Coupled Code Analyses to Theoretical Framework of a Multiscale Method, International Congress on Advances in Nuclear Power Plants (ICAPP 2007), Nice, France, May 2007.

Measurements and Analysis of Dryout and Film Thickness in a Tube with Various Axial Power Distributions

Scientist: Carl Adamsson

Research Leaders: Associate Professor Henryk Anglart

Division of Nuclear Reactor Technology, KTH, Stockholm

Background

In high performance heat exchangers, such as nuclear reactors, the critical heat flux gives the most important design boundary. In a BWR nuclear reactor the critical heat flux occurs through the process of dryout, i.e the disappearance of liquid film from the fuel rod surface. At the transition to dryout the heat conduction between the fuel and the coolant is vastly reduced, leading to a sharp increase in fuel temperature and possible fuel damage. It is obvious that accurate methods to predict the dryout limit under various conditions are needed. Today the industry relies on empirical correlations, which require extensive full scale experiments. Moreover, since the correlations used today are not well-founded in physical reasoning they cannot be trusted if used outside the parameter range of the underlying experiment. In some cases this can be a severe limitation; e.g. there has recently been an increasing interest in the influence of the axial power distribution on the dryout power. For practical reasons it is only possible to perform experiments for a very limited set of power distributions. It is thus questionable if empirical correlations can be trusted to predict the quit large effect of the power distribution in an adequate way.

Most models developed are built on the assumption that the annular steam-water flow can be described as a balance between a gas-field, a liquid droplet-field and a liquid film-field. Dryout is then postulated to occur when the liquid film thickness becomes zero. To develop and validate mechanistic models however, experimental data on the film thickness and film flow rate are much more useful than data on only the dryout power itself. Such experiments have been performed by several researchers under various conditions, but most of this data do not focus on the axial power distribution.

Objectives and Methodology

During the first, experimental, phase of the project film flow measurements were carried out for three axial power profiles. The remaining part of the project will be focused on numerical simulations of two-phase annular flows in order to improve the understanding of mainly the drop deposition process.

For that purpose a code based on Lagrangian particle tracking is under development. This code will be used for detailed studies of drop motions in circular pipes, annuli and subchannels, see Figure 1.

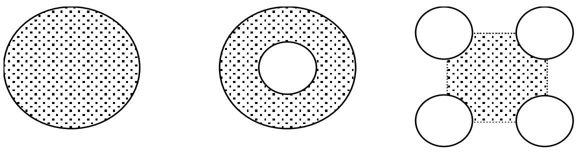


Figure 1 Three channel crossection to be investigated – pipe, annulus and subchannel

Results during 2007

The algorithm that has been developed during the project was presented at the NURETH-12 conference. Preliminary results for the three cross-section geometries shown in Figure 1 showed interesting trends but the absolute values of the predicted deposition rates sometimes deviated considerably from experimental data. The deviations were particularly large for high pressure steam-water flows of interest for BWRs. The work to find the cause of these deviations is ongoing.

As a comparison, a study with the Lagrangian algorithm available in the commercial code CFX-11 was performed. The results were presented at the Nuclear Energy for New Europe conference and showed similar deviations that had been observed with the in-house code.

Publications

1) Carl Adamsson and Henryk Anglart. Measurements of the Liquid Film Flow Rate in High Pressure Annular Flow with Various Axial Power Distributions. HEAT 2005, June 26-30 2005, Gdansk, Poland

2) Carl Adamsson and Henryk Anglart. Experimental Investigation of the Liquid Film for Annular Flow in Tube with Various Axial Power Distributions. NURETH 11, Avignon, France, October 2-6, 2005.

3) Carl Adamsson and Henryk Anglart. Film Flow Measurements for High Pressure Diabatic Annular Flow in Tubes with Various Axial Power Distributions. Nuclear Engineering and Design 236(23), p. *2485-2493*

4) Carl Adamsson and Henryk Anglart, An Investigation of Cross-Section Geomtry Effects on the Deposition Rate in Annular Two-Phase Flows with a Lagrangian Model, NURETH-12, Pittsburgh, Pennsylvania, USA, Sep 30 – Oct 4, 2007

5 Carl Adamsson and Henryk Anglart, A Sensitivity Study of Lagrangian Models for Application to Annular Two-Phase Flows, International Conference Nuclear Energy for New Europe, Portorož, Slovenia, Sept. 10-13, 2007

First-principles study of self diffusion parameters in a bcc metal; the case of Nb

Scientists: Christina Lagerstedt, Nils Sandberg, Division of Reactor Physics KTH
Research Leaders: Associate Professor Janne Wallenius, KTH

Introduction and Objective

In general, body centered cubic (bcc) materials are much more irradiation resistant compared to face centered cubic (fcc) materials. Thus, ferritic-martensitic (F–M) steels are superior to austenitic steels in this respect. In particular, some Fe-Cr based F–M steels show a much better resistance to swelling compared to austenitic steels. The fundamental reason for this is however not completely understood. Some rather advanced models exist to explain this in terms of interstitial mobility, interstitial clustering and binding to solutes.

A predictive model of swelling in a specific alloy should be applicable to other types of alloys as well. An interesting system in this respect is V-5%Fe. It has a bcc structure, but in contrast to Fe-Cr alloys it is known to resist irradiation very poorly. In fact, this system has the world record in swelling; 160% at 50 displacements per atom (dpa) [1]. It is obvious, therefore, that not only the crystal structure determines a materials response to irradiation, but also other factors such as the defect-defect and defect-solute interactions.

In the current work, we have used first-principles electronic structure methods in order to study point-defect properties in Nb. The idea is to use Nb as a bench-mark system for first-principles calculations of defect properties in a bcc metal. Nb then has the benefit of being non-magnetic, which removes some complications when experimental data are interpreted. There are also a fairly large ammount of data on vacancy-impurity interactions and impurity diffusion rates in Nb, which are in principle also attainable to first-principles calculations. We have focused on applying electronic structure methods in order to calculate i) the vacancy concentration and ii) the self-diffusion rate in pure Nb. The latter can be directly compared with experimental data. This comparison gives information on what accuracy that can be expected from similar calculations targeting defect-solute interactions, e.g., in V or Fe based alloys. Such numbers, in turn, are needed in quantitative and predictive models of swelling in bcc alloys.

Organisation

The project was initially carried out by Christina Lagerstedt (PhD student) and Nils Sandberg (researcher), and was later taken over by Nils Sandberg.

Method

It is possible to study self diffusion, impurity diffusion, and even diffusion in concentrated alloys by first-principles methods. In the case of self diffusion the calculational procedure is particularly simple, and the comparison with experiments is relatively straight-forward. The self diffusion rate is written as the product of the vacancy concentration c(T), and v(T), the jump-rate of neighboring atoms into the vacant site, and a trivial geometrical factor f; D(T) = fc(T)v(T). Both vacancy formation and vacancy mediated atomic jumps are thermally activated processes, and therefore experiments are available only at elevated temperatures, typically above half the melting temperature.

It is straight-forward to calculate the activation energy of vacancy formation, Ev, and the activation energy, Em, of vacancy-assisted migration jumps. They are obtained essentially from the total energy of the static structures representing the vacancy and the migration jump, respectively. There are also two corresponding prefactors, exp(Sv/kB) and exp(Sm/kB), that may be calculated from the phonon-frequencies of the two structures. Computationally, these factors are much more demanding to extract. However, it can be done, and the benefit of doing that is that one can predict absolute diffusion rates, and not only relative ones. If calculated activetion energies are to be critically compared with experimental information, it is better to calculate the prefactors, rather than to “guess” them, which is normally done.

Calculated Diffusion Parameters

Table I shows vacancy formation and migration parameters calculated by two different methods. First, electronic structure calculations based on density functional theory (DFT) were carried out, using the simulation package VASP. Second, an empirical model potential was used. In the DFT calculations, Hv was calculated using simulation cells of different size (54 and 128 atoms) and the convergence with respect to various numerical approximations was carefully tested. Hm was obtained in a similar calculation, and we also tested and ruled out other migration mechanisms (e.g., second nearest neighbor jumps).

The only parameter that we were unable to calculate is the prefactor for migration jumps. A possible explanation is the use of a 54 atom cell in this calculation. The transition state leads to more strain in the simulation cell compared to the formation of a vacancy, and it may be more appropriate to use a larger simulation cell. However, this leads to much more demanding calculations. A second factor is that Nb is known to have long-range force-interactions in a Born-von-Karman fit to measured phonon dispersion data. Phonon-dispersion curves and densities of states (DOS) are a by-product in our phonon calculations, and are compared with measured ones in Fig. 1. It is seen that a 128 atom cell is required to reproduce the main features of the experimental DOS, although a spurious peak at 4 THz is still present. Again, this indicates that it would be preferable to use a larger simulation cell in diffusion calculations on Nb. The calculated vacancy formation and migration parameters, with Sm taken from model potential calculations, can be put together to obtain the predicted self diffusion rate. It is shown in Fig. 2, along with available experimental tracer diffusion data. The agreement between theory and experiments, both in terms of the self-diffusion activation enthalpy (given by the slope in Fig. 2) and in terms of absolute self-diffusion rates, is surprisingly good, considering that the present calculations are to a large degree free of experimental input.

Conclusions

In the present project, an attempt was made to calculate self-diffusion parameters in Nb directly from first principles electronic structure methods. The computational scheme is very demanding in terms of cpu-power, but the benefit of using this approach is that defect and defect interaction parameters that are very hard or impossible to obtain by other means can be calculated. Such parameters are fundamental in quantitative models of irradiation swelling in metals. We use pure Nb as a bench-mark system, and find that the calculated pre-factor for atom migration does not agree with experiments. By taking that factor from a model calculation, we show that the self-diffusion rate, which depends on the vacancy concentration times the vacancy mediated atomic jump-rate, can be calculated in good agreement with experiments.

TABLE I: Calculated and experimental self-diffusion parameters.
The sums HD = Hv + Hm and SD0 = Sv + Sm are given.

	VASP	*Model potential*	*Experiments*
Hv	2.85 (eV)	2.50 (eV)	2.6 ± 0.3 (eV)
Sv	3.10 (kB)	2.75 (kB)	-
HD	3.77 (eV)	3.31 (eV)	4.1 (eV)
SD	-	5.0 (kB)	5.2 (kB)

References

[1] K. Fukumoto, A. Kimura, and H. Matsui, J. of Nuclear Materials 258-263, 1431 (1998).

[2] F. Guthoff et al., J. Phys.: Condens. Matter 6, 6211 (1994).

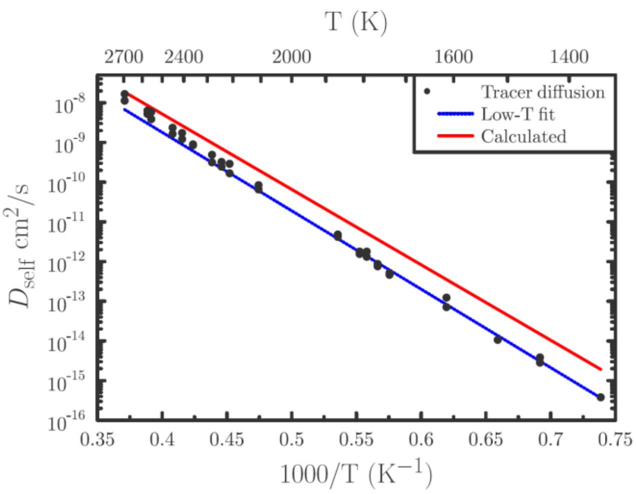


FIG. 1: Calculated phonon density of states. Experimental data from Ref. 2.

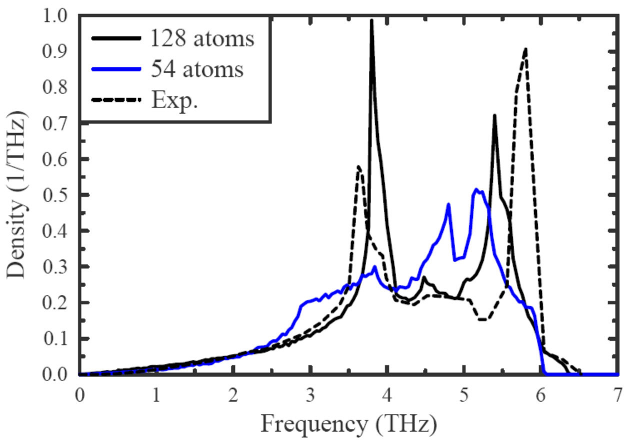
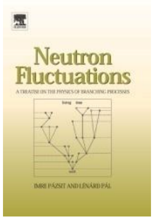


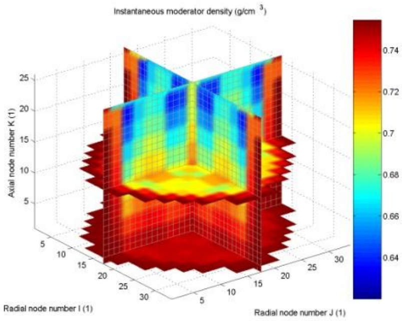
FIG. 2: Calculated and experimental self-diffusion rates in Nb.

Chalmers University of Technology

Overview of Activities in 2007

Research and education in nuclear engineering is pursued at the Departments of Nuclear Engineering (Applied Physics) and Nuclear Chemistry (Chemical and Biological Engineering) in Chalmers. The research is pursued separately, but as from the academic year 2007/08, the specialized nuclear engineering course is given jointly by the two groups

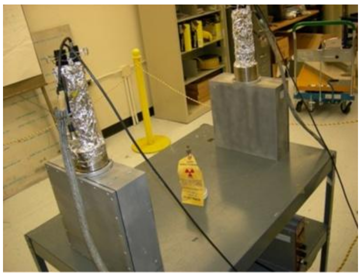




Nuclear Engineering

Research in:

- reactor physics, dynamics and noise diagnostics; deterministic and stochastic transport; nuclear safeguards; random aspects of advanced reactors;
- coupled core physics – thermal hydraulics: method development, application to safety analysis of power uprates; full static and dynamic modelling of all Swedish reactor units; competence centre for SKI; BWR instability research, NORTHNET projects;
- nuclear measurement methods for material science, positron annihilation techniques;
- thorium fuel cycle; Gen-IV reactors, in particular gas-cooled fast reactors and molten salt reactors.



Facilities, tools and other data:

Access to all major system codes for neutronic and thermal hydraulic calculations.
A pulsed beam for variable energy slow positrons.
A portable 14 MeV pulsed neutron generator.
5 PhD students (3 with SKC support, 1 jointly with Nucl. Chemistry).

Highlights of the year:

Carl Sunde won Sigvard Eklund's prize for best PhD thesis.
Book on Neutron Fluctuations by I Pázsit and L. Pál at Elsevier.

Nuclear Chemistry

Research in:

- actinide science; nuclear waste repository investigations
- nuclear reactor chemistry including accidents
- separation and transmutation; nuclear fuel investigations

Facilities and other data:
Laboratories for low activity α, β, γ experiments and activity measurements; hot cell laboratory for γ activity.
9 PhD students (2 with SKC support, 1 jointly with Nucl. Engng)

Courses

Package of full nuclear engineering courses was given the first time in the fall semester of 2007. The package is a specialisation within the master course of Applied Physics, but can be selected also in other master courses. The package structure is:

Nuclear Engineering I	7.5 pts	Reading period 1
Nuclear Engineering II	7.5 pts	Reading period 1
Nuclear Engineering III	7.5 pts	Reading period 2
Nuclear Engineering IV	7.5 pts	Reading period 2

Together with other specialized courses of Nuclear Chemistry, the total number of undergraduate students taking courses in nuclear engineering is 18.

Nucl. Engng diploma work (master thesis): 30 pts.
Reading periods 3 and 4 (7 students)

Higher level graduate courses given:

Advanced reactor theory,	15 pts;
Radiation detection and measurements,	7,5pts;
Actinide chemistry,	15pts;
Solvent extraction,	6pts.

Uncertainty and sensitivity analysis applied to the simulation of the Swedish Boiling Water Reactors

Research leaders: Professor Christian Ekberg and Associate Professor Christophe Demazière

Research scientist: Augusto Hernández-Solís

Background

In earlier days, the modelling of nuclear reactors, both for static and transient calculations, was very often performed via very conservative tools. Such analyses were rather crude and only worked analytically for a number of simple cases. This conservatism was, among others, the result of limited computer power, which prevented using sophisticated models, especially on the thermal-hydraulic side. With the recent increase of cheap CPU power, advanced modelling methods are now in reach. The actual trend worldwide is to develop and use so-called Best-Estimate (BE) methods for nuclear reactor simulations. These BE methods are based on coupled (or sometimes integrated) neutronic/thermal-hydraulic calculations, where the interplay between the neutron kinetics and the thermal-hydraulics can be properly accounted for. This coupling thus makes it necessary to have detailed modelling tools on both the neutronic and the thermal-hydraulic sides. Although this coupling allows significantly improving the accuracy of the calculations, a full evaluation of the uncertainties associated to these BE methods is highly beneficial, in order to assess the reliability, the robustness and the fidelity of the simulations. The main advantage of uncertainty evaluation is to decrease even further the conservatism of the safety analyses, which can lead to a decrease of the safety margins and thus to a maximisation of the reactor output/utilization.

Goals of the project

Developing an uncertainty analysis methodology is highly beneficial for many different reasons:

- For licensing and safety purposes: if a BE approach is used in connection with an uncertainty evaluation, a relaxation of the licensing rules is possible, leading to less conservative safety margins, and a maximization of the reactor output/utilization. This is of particular interest for the extensive program of power uprates in Sweden.
- For identifying deficient models: the sensitivity and uncertainty analysis can provide some guidance about the correlations and the code models that would lead to a significant increase of the accuracy of the calculations if these correlations and models were to be improved.

The goal of the present project is thus to develop a tool for uncertainty and sensitivity analysis applied to nuclear reactor simulations. This project exclusively focuses on the case of the Swedish BWRs. The simulation tool is based on the POLCA-T code. In this framework, a close collaboration with the POLCA-T code developers (Westinghouse Electric Sweden AB) is planned within this project. If successful, the last past of the project will be devoted to a generalization of the methodology to other types of reactors/codes.

Organization

The work is performed by PhD student Augusto Hernández-Solís under the supervision of Professor Christian Ekberg and Associate Professor Christophe Demazière. Dr. Paolo Vinai and Arvid Ödegaard - Jensen also support Augusto Hernández-Solís on some aspects of the project. The members of the reference group are: Oddbjörn Sandervåg, SKI, Henrik Nylén, Ringhals, Pär Lansåker, Forsmark, Christer Netterbrant, OKG, and Ulf Bredolt, Westinghouse.

Methodology

There are existing methods for quantifying and calculating the effect of data uncertainties in the field of nuclear reactor calculations, e.g. the CIAU [Code (with the capability of) Internal Assessment of Uncertainty] methodology from Pisa [1], [2] and the CSAU (Code Scaling, Applicability, and Uncertainty) technique developed by the USNRC [3]. In this project, another approach for uncertainty and sensitivity analysis than the ones used in the CIAU and CSAU methods is proposed, based on the experience of the Nuclear Chemistry group at Chalmers [4]. The usual definition of sensitivity and uncertainty analysis is that the sensitivity analysis answers the question of how important an input parameter is and the uncertainty analysis gives the uncertainty on the output resulting from the uncertainties in the input parameters. In some cases when many input parameters are required, it may be possible to perform a screening calculation first to limit the number of parameters to those that have some effect within the selected uncertainty range. In addition to the rather straightforward data uncertainties, there are also conceptual uncertainties, e.g. which method and which program are used to calculate a certain event. The evaluation of these uncertainties is not clear but still plays an important role in the resulting accuracy of the calculations. As a starting point, the Latin Hypercube Sampling (LHS) method, which the Nuclear Chemistry group at Chalmers has some experience with, will be investigated. The general idea with the LHS method is that given every input data with uncertainty and distribution, several samples are taken and the results recorded. From there on, simple statistics may be used for evaluation. The LHS method is an effective sampling method that allows a minimization of the computer time required for the sampling. The present aim for the PhD project is to construct a program package that controls the real simulation code, makes the sampling, and performs an uncertainty and a sensitivity analysis. Although the LHS method is not new, its applications to the nuclear engineering area were rather limited in the past, focusing mostly on the evaluation of the uncertainties associated to the peak cladding temperatures, and to the Departure from Nucleate Boiling Ratio (DNBR). Furthermore, the LHS method has the advantage of requiring much less computer runs compared to the Response Surface Methodology used in the CSAU method. Nevertheless, it may well be that during the start up of the project a more effective method for either the uncertainty or sensitivity analysis is selected and in such a case this will be used instead.

Augusto Hernández-Solís started his employment on October 1st, 2007. Since then, Augusto Hernández-Solís mostly attended courses in Nuclear Engineering and started looking at the literature published in the field of uncertainty and sensitivity analysis.

References

1) F. D'Auria, and W. Giannotti (2000), "Development of a code with the capability of internal assessment of uncertainty." Nuclear Technology, 131 (2), pp. 159-196.

2) A. Petruzzi, F. D'Auria, and W. Giannotti (2005), "Methodology of internal assessment of uncertainty and extension to neutron kinetics/thermal-hydraulics coupled codes." Nuclear Science and Engineering, 149 (2), pp. 211-236.

3) B. E. Boyack, I. Catton, R. B. Duffey, P. Griffith, K. R. Katsma, G. S. Lellouche, S. Levy, U. S. Rohatgi, G. E. Wilson, W. Wulff, and N. Zuber (1990), "Quantifying the reactor safety margins part I: An overview of the Code Scaling, Applicability and Uncertainty evaluation methodology." Nuclear Engineering and Design, 119 (1), pp. 1-15.

4) C. Ekberg, A. Ödegaard-Jensen, and G. Meinrath (2003), "LJUNGSKILE 1.0: A computer program for investigation of uncertainties in chemical speciation." SKI report 2003:03.

Development of an integrated neutronic/thermal-hydraulic tool for noise analysis

Research leader: Associate Professor Christophe Demazière

Research scientist: Viktor Larsson

Background

The neutron noise, i.e. the difference between the time-dependent neutron flux and its time-averaged value, assuming that all the processes are stationary and ergodic in time, allows determining many interesting features of a reactor. The neutron noise can be used either for diagnostic purposes, when an abnormal situation is suspected, or for estimating a dynamical core parameter, whereas the reactor is at steady-state conditions. Noise diagnostics has the obvious advantage that it can be used on-line without disturbing reactor operation. Such a monitoring technique received further attention in the past few years due to the extensive program of power uprates worldwide. Some of main issues/ concerns related to the operation of the plants at the uprated power level are the reduction of the safety margins, such as the margins to instability for BWRs, and increased vibrations (flow induced vibrations). When analyzing neutron noise measurements, the knowledge of the so-called reactor transfer function is of prime importance. This transfer function gives the space-dependent response of the reactor to perturbations that might be localized or spatially-distributed. As a matter of fact, most of the diagnostic tasks require the prior determination of the reactor transfer function, since the original perturbation has to be estimated from the detector reading (unfolding task).

Goals of the project

The Department of Nuclear Engineering, Chalmers University of Technology, developed in the past a tool, usually referred to as a "neutron noise simulator", allowing determining the reactor transfer function [1]. This simulator is able to calculate the response of a nuclear core to perturbations expressed as fluctuations of the macroscopic nuclear cross-sections or of the possible external neutron source, assuming that the operating conditions of the reactor are stationary. The noise simulator was successfully benchmarked against analytical or semi-analytical solutions and was already used in many diagnostic tasks (see [2] for an overview of some of those). This preliminary version of the tool was demonstrated to work properly and to give new physical insights for the interpretation of noise measurements. Nevertheless, the existing tool has some shortcomings, such as its inability to model closed-loop reactor transfer functions. The goal of the present PhD project is to further develop this tool to bring it to a level of development/sophistication/reliability similar with coupled time-dependent codes. The PhD project is thus aiming at developing a full-core integrated neutronic/ thermal-hydraulic tool for noise analysis. This requires extensive work both on the neutronic side (use of nodal methods) and on the thermal-hydraulic side (development of thermal-hydraulic models). The main advantage of the new tool would be that the neutronics is based on the calculation of the actual Green's function of the reactor, and that all the time-dependent equations describing the fluctuating quantities are Fourier-transformed. The applications of this tool would be numerous for noise analysis. Due to the coupling to any code system, this tool could be easily applied to any of the Swedish nuclear power plants.

Organization

The work is performed by PhD student Viktor Larsson under the supervision of Associate Professor Christophe Demazière. Prof. Imre Pázsit and Dr. József Bánáti are also supporting Viktor Larsson on some aspects of the project. The members of the reference group are: Ninos Garis, SKI, Tell Andersson, Ringhals, Farid Alavyoon, Forsmark, Christer Netterbrant, OKG, and Camilla Rotander, Westinghouse.

Methodology

In 2007, several semi-analytical 1-dimensional models were developed in 2-group theory in order to determine the spatial dependence of the neutron noise throughout the reactor. First, a 1-region reactor model was developed in both diffusion theory and P_1 theory. The purpose of this model was twofold: first to get familiar with noise theory and to get some physical intuition about the results to be expected, second to determine whether P_1 theory would give significantly advantageous results for the calculation of the neutron noise compared to diffusion theory. This preliminary study demonstrated that the only case where significant results can be expected are when steep gradients of the static neutron flux exist. Based on these observations, a 2-region reactor model was developed, where the regions represent the active fuel and the reflector, respectively. As an illustrative example, the neutron noise induced by a local noise source is given in Figs. 1 and 2, for both a central perturbation and a non-central perturbation, respectively. In these figures, the neutron noise calculated using a semi-analytical solution and a fully numerical solution based on finite differences are also represented. The full results of such calculations and the resulting conclusions will be presented at the PHYSOR 2008 conference [3]. In parallel with these investigations, nodal methods are being studied in order to determine the most suitable one to be applied to the calculation of the neutron noise.

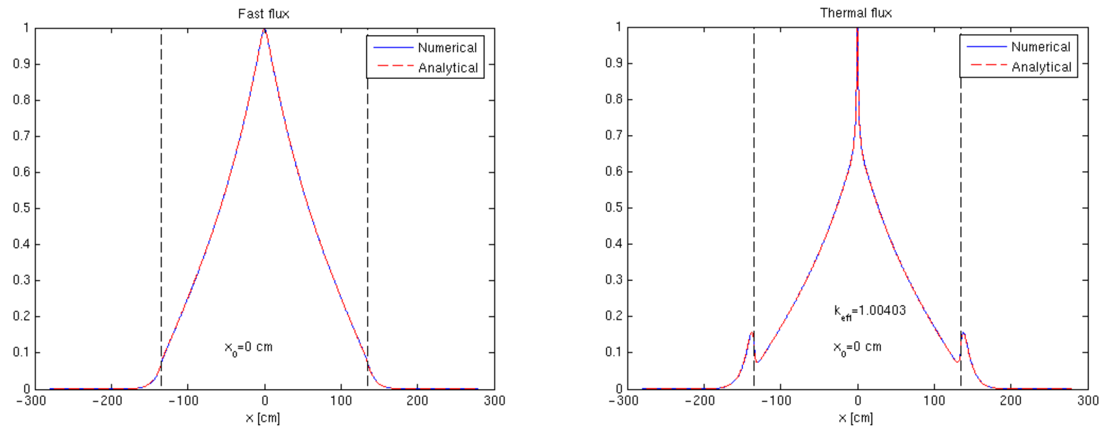


Figure 1. Amplitude of the neutron noise induced by a central perturbation (in the fast and thermal groups, on the left and right hand-side, respectively).

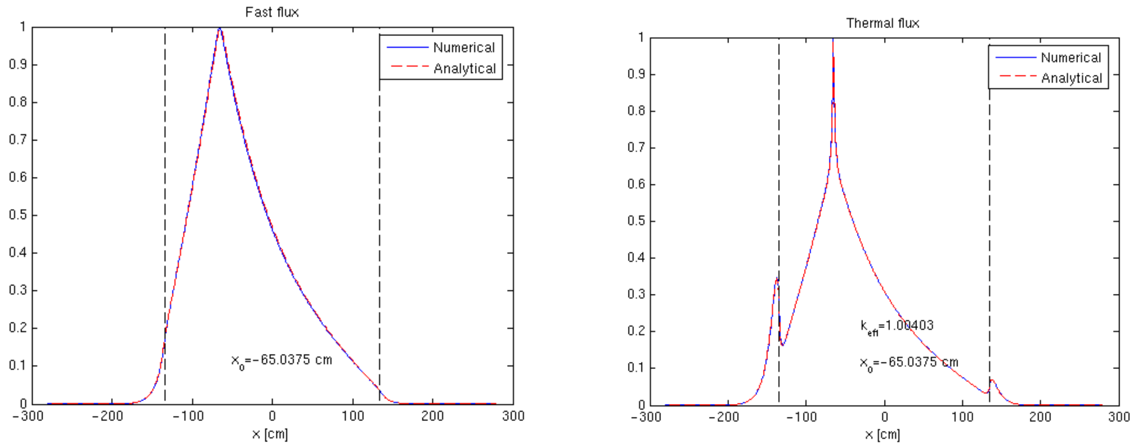


Figure 2. Amplitude of the neutron noise induced by a non-central perturbation (in the fast and thermal groups, on the left and right hand-sides, respectively).

Publications

1) C. Demazière, "Development of a 2-D 2-group neutron noise simulator," Annals of Nuclear Energy, 31, pp. 647-680 (2004).

2) C. Demazière and I. Pázsit, "Numerical tools applied to power reactor noise analysis," Accepted for publication in Progress in Nuclear Energy (2008).

3) V. Larsson and C. Demazière, "Semi-analytical calculations of the neutron noise in 2-group theory for 1-D homogeneous systems", submitted to PHYSOR 2008, International Conference on Reactor Physics, Nuclear Power: A Sustainable Resource, Casino-Kursaal Conference Center, Interlaken, Switzerland, September 14-19, 2008.

Advanced analysis methods for non-stationary processes in reactor cores

Research Leaders: Professor Imre Pázsit, Department of Nuclear Engineering, Chalmers University of Technology, Göteborg and Docent Ninos Garis, Swedish Nuclear Power Inspectorate, Stockholm.

Scientist: Carl Sunde, Department of Nuclear Engineering, Chalmers University of Technology, Göteborg.

Background

Diagnostics of reactor cores with noise methods is usually performed with FFT based spectral analysis, such as auto and cross spectral power densities and coherence/phase between variables. Such an analysis assumes that the system behaviour is stationary during the measurement period. In other words the status of the system is assumed to be unchanged over several tens of thousands of the periods of the characteristic frequencies of the system.

Nevertheless, the behaviour of complex systems such as a reactor core is often non-stationary, i.e. the state of the system often changes during a much shorter period. Non-stationary processes and transients are in fact quite common in reactor systems, such as the occurrence and development of local and global core instabilities in BWRs, the short-term changes of vibration properties (core-barrel, fuel assembly etc) in PWRs, and the various phenomena in two-phase flow (vortex shedding, slug flow etc). Apart from temporal transients, spatial transients or non-stationarities may also occur, such as in represented by the spatial structure of two-phase flow.

Goals

Powerful mathematical methods have been developed lately and applied for the analysis of such non-stationary processes, out of which wavelet analysis appears to be one of the most promising. The first purpose of the project was to introduce the use of known analysis methods of non-stationary processes, and primarily wavelet analysis, into the noise diagnostic work of our Department and to explore their possibilities for diagnosing non-stationary processes. A second goal was to develop wavelet-based methods further for tackling new problems which are of practical interest in our R&D work. Both a more effective early detection and quantification of beginning anomalies as well as elaborating of new, wavelet-based methods of reliable parameter estimation under non-stationary circumstances were addressed. The test of the methods was performed through both simulated signals as well as measurements taken in Swedish power plants.

Organisation

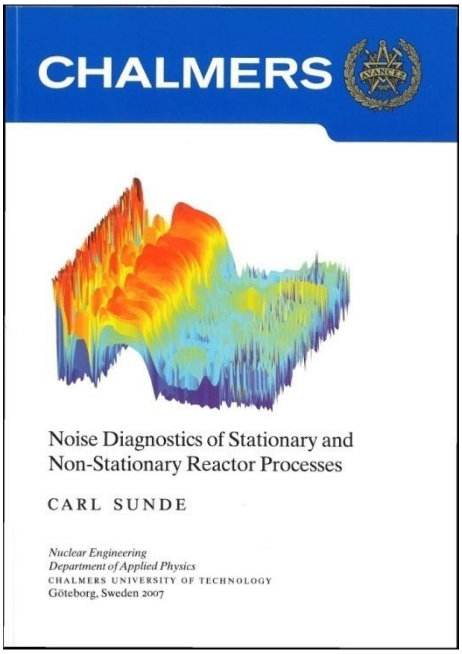
The reactor diagnostic group is headed by Prof. Imre Pázsit, who was also the leader of this SKC-project. The procject has been going on since July 2002. At the beginning of the project Dr. Vasily Arzhanov, now at KTH, introduced Kalle to noise theory and analytical models. Doc. Christophe Demaziere, university lecturer, was also supporting Carl Sunde during several stages of the project.

The members of the reference group were: Pär Lansåker Forsmark, Henrik Eisenberg OKG, Henrik Nylen Ringhals, Ninos Garis SKI, Johan Larsson Ringhals, and Camilla Rotander, Westinghouse. Doc. Ninos Garis served as deputy adviser to Kalle. The last and final reference group meeting was held in December 2006.

Methodology

The methodology is similar to traditional noise analysis work, which consists of both evaluation of measurements, and elaborating models of the reactor and its processes to expedite the interpretation of the measurement analysis. Hence both theoretical model development and analysis of measurements is involved. In the analysis part, in contrast to the FFT tool, used in the traditional methods, continuous (CWT) or discrete fast wavelet transform (DWT) is used.

Activity in 2007



All technical work which was included into the thesis was completed in 2006 and already published or accepted for publication. The activity in 2007 was devoted to the writing up of the thesis, organising and executing the PhD exam. In addition several further piece of work was performed by Kalle wich was not included into the thesis, such as evaluating measurement from Ringhals-1, as well as preparations for a future coming article on the accuracy of the source modulation method.

The PhD exam was made on 27 April 2007 and Kalle has defended his thesis entitled “Noise diagnostics of stationary and non-stationary reactor processes”. External examiner was Prof. Y. Yamane, Nagoya University. Members of the committee were Prof. Mats Viberg, Signals and Systems, Chalmers, Assoc. Prof. Tatiana Tambouratzis, University of Piraeus, Greece, and Dr. Joakim Karlsson, Studsvik AB.

The thesis is based on five papers published in international journals, and one conference proceedings.

Summary of the results

The type of problems investigated and corresponding results/conclusions can be summarized as follows.

<u>1. Identification of two-phase flow with wavelet preprocessing and neural networks.</u> It was found that artificial neural networks, ANN, can be trained to identify two-phase flow patterns from neutron radiography images and optical images of the flow by visible light. The wavelet preprocessing did not increase the success ratio of the identification, i.e. did not increase the discriminative power of the selected features, but made the training stage significantly faster. Hence the experience was positive.

<u>2. Detector string impacting.</u> This was presumable the most successful application of wavelets in reactor diagnostics. Both discrete and continuous wavelet transform methods were used with success. Especially the wavelet coherence, based on the continuous transform, appeared to be an objective method for detecting impacting of detector tubes from the detector signals, which can be applied without either need of input from experts or need for calibration and special tailoring to the concrete reactor unit in question. The basis of the method is empirical in its character, but the procedure is based on an algorithm that can be run by non-experts. Several measurements from Ringhals were evaluated, but we are still waiting for the results of the visual inspections made during the following refuelling.

<u>3. Investigation of the use of wavelet techniques for BWR stability analysis.</u> In this study several applications of wavelets were tested. One was to improve the quality of the detector ACF (autocorrelation) signals, i.e. eliminate trends and parasitic noise, before applying the classical methods of determining the decay ratio (DR) from the ACFs. Although the wavelet filtering indeed improved the visual quality of the ACFs, the DR determined from the filtered signals was not always better than the value obtained without wavelet filtering. Another application was to use continuous wavelet transform methods for direct calculation of the DR from the raw signals. This method worked quite satisfactorily, and could be used to determine the larger DR (i.e. that of the more unstable component) in case of dual oscillations. If the frequency of the simultaneous oscillations differ even only with a slight value, i.e. a few percent, the method is capable to identify both decay ratios.

<u>4. Core barrel vibrations, and the break-frequency method vs point-kinetics.</u> These parts of the project concern traditional, stationary reactor diagnostic methods, and will not be reviewed here. They were executed partly due to their own interest for the research of the Department, and partly to give Kalle an opportunity to deepen his knowledge in reactor physics and core dynamics as well as core calculations. These studies were also published internationally and well received.

Carl Sunde received two awards at international conferences for the work included into his thesis, and he became the winner of 2007 years’ Sigvard Eklunds prize for best PhD thesis.

Ruthenium chemistry in the reactor containment during severe accidents in nuclear power plant

Research leaders: Prof. Christian Ekberg[1] and Dr. Henrik Glänneskog[2]

Research scientist: Ph.D. Student Joachim Holm[1], Department of Nuclear Engineering, Chalmers University of Technology, Gothenburg

[1] Nuclear Chemistry, Chalmers University of Technology, Gothenburgh

[2] Ringhals, Väröbacka

Background

Ruthenium is semi-volatile fission product, which is formed during normal operations in nuclear power plant. During a "normal" severe accident, with core melt and so on, ruthenium doesn't constitute a threat to environment, because of its non-volatile properties. However, during a severe accident with an air-ingress into the reactor vessel ruthenium can be oxidized and adopt a volatile form like ruthenium tetroxide (RuO_4). This specie can be released from the nuclear fuel and the reactor vessel and be transported to reactor containment. It is therefore really important investigate if a containment has the ability to reduce and prevent the source term of ruthenium, in a case of a severe accident.

Goals:

The goals of the project are to investigate both the distribution of RuO_4 between an aqueous phase and gaseous phase and also study the reactions between RuO_4(g) and some metals surfaces that are in a Boiling Water Reactor (BWR) containment. The metals are copper, zinc and aluminium. The atmosphere and the conditions in the experiments are similar to the conditions in a containment of BWR. The temperature interval in the experiments is 20-50°C.

Organisation:

The work is performed by Ph.D. student Joachim Holm under the supervision of Prof. Christian Ekberg and Dr. Henrik Glänneskog.

Methodology:

The experiments are performed in the experimental set-up seen in figure 1.

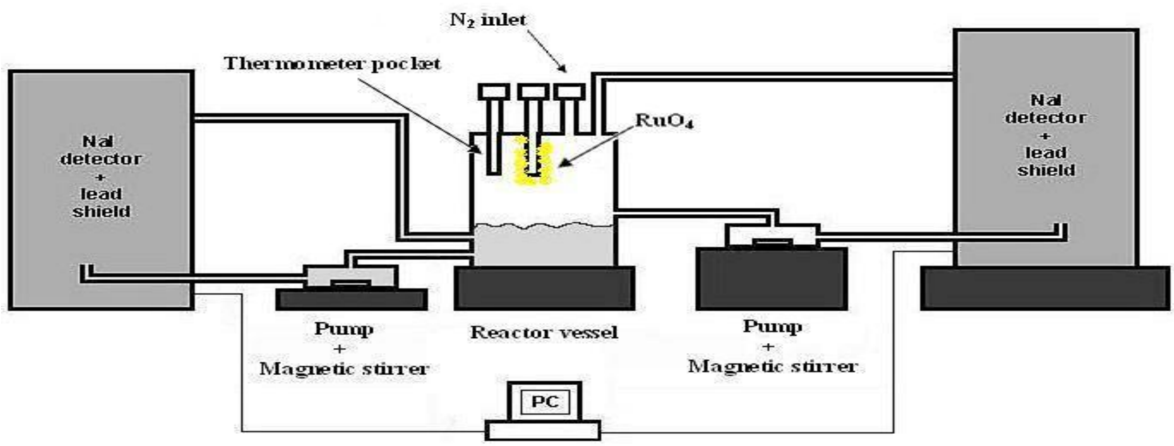


Figure 1: The experimental set-up, used for the RuO_4 distribution experiments.

The experimental set-up consists of a reactor vessel connected to two loops, one loop for the water phase and one loop for the gas phase. The two loops pass by two new 2in. × 2in. NaI(Tl)-detectors, which are connected to a computer. The detectors are used for detection of the 496 keV γ-rays emitted in the decay of ^{103}Ru. The temperature is generated by heating band twirled around the pipes and the vessel.

Every experiment is started by filling the aqueous loop and half of the reactor vessel with distilled water. The circulation is then started in the two phases, the flow rates are about 5 $cm^3 \cdot s^{-1}$ and 2 $cm^3 \cdot s^{-1}$ in the aqueous and the gaseous phase, respectively. The system is then flushed with nitrogen gas to secure that only a few percent of oxygen is presented. Gaseous ruthenium tetroxide spiked with an arbitrary amount of ^{103}Ru is introduced into system by letting $^{103}RuO_4$(cr) vaporize in the reactor vessel. The distribution of $^{103}RuO_4$ between the two phases is measured by the two NaI-detectors.

The interactions between the three metals and gaseous ruthenium tetroxide are investigated by, letting RuO_4(cr) sublimate into a gaseous form in a reaction bottle with the metal samples present. The temperature for the experiments is room temperature. After the experiments are the metals sample investigated by the surface analysis methods ESCA and SEM-EDX.

Results:

From the performed distribution experiments following results have been received.

- RuO_4(g) was almost immediately distributed in aqueous phase after the introduction of RuO_4 in the gaseous phase in the system, in the temperature interval 20-50°C.
- There was a significant deposition/adsorption of ruthenium on the glass surfaces in the experimental set-up. The speciation of deposited ruthenium was analyzed by SEM-EDX and ESCA and was determined to hydrated RuO_2.

The conclusion from these experiments is that the probability of a release of RuO_4 from the containment to the environment seems to be extremely small.

The experiments with interactions of RuO_4 and the metals are not finished yet. But some preliminary results have been received.

Significant deposition of RuO_4 on the metals, especially on zinc.

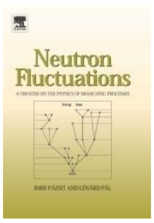
Publications:

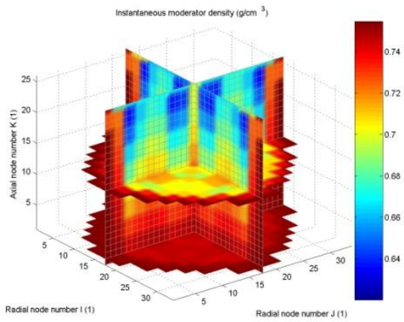
1) J. Holm, C. Ekberg and H. Glänneskog, Reactions of RuO_4 under severe nuclear reactor accident conditions, Submitted to Radiochimica Acta.

Chalmers University of Technology

Overview of Activities in 2007

Research and education in nuclear engineering is pursued at the Departments of Nuclear Engineering (Applied Physics) and Nuclear Chemistry (Chemical and Biological Engineering) in Chalmers. The research is pursued separately, but as from the academic year 2007/08, the specialized nuclear engineering course is given jointly by the two groups

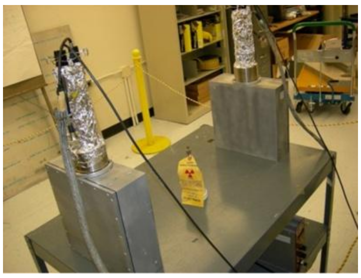




Nuclear Engineering

Research in:

- reactor physics, dynamics and noise diagnostics; deterministic and stochastic transport; nuclear safeguards; random aspects of advanced reactors;
- coupled core physics – thermal hydraulics: method development, application to safety analysis of power uprates; full static and dynamic modelling of all Swedish reactor units; competence centre for SKI; BWR instability research, NORTHNET projects;
- nuclear measurement methods for material science, positron annihilation techniques;
- thorium fuel cycle; Gen-IV reactors, in particular gas-cooled fast reactors and molten salt reactors.



Facilities, tools and other data:

Access to all major system codes for neutronic and thermal hydraulic calculations.
A pulsed beam for variable energy slow positrons.
A portable 14 MeV pulsed neutron generator.
5 PhD students (3 with SKC support, 1 jointly with Nucl. Chemistry).

Highlights of the year:

Carl Sunde won Sigvard Eklund's prize for best PhD thesis.
Book on Neutron Fluctuations by I Pázsit and L. Pál at Elsevier.

Nuclear Chemistry

Research in:

- actinide science; nuclear waste repository investigations
- nuclear reactor chemistry including accidents
- separation and transmutation; nuclear fuel investigations

Facilities and other data:
Laboratories for low activity α, β, γ experiments and activity measurements; hot cell laboratory for γ activity.
9 PhD students (2 with SKC support, 1 jointly with Nucl. Engng)

Courses

Package of full nuclear engineering courses was given the first time in the fall semester of 2007. The package is a specialisation within the master course of Applied Physics, but can be selected also in other master courses. The package structure is:

Nuclear Engineering I	7.5 pts	Reading period 1
Nuclear Engineering II	7.5 pts	Reading period 1
Nuclear Engineering III	7.5 pts	Reading period 2
Nuclear Engineering IV	7.5 pts	Reading period 2

Together with other specialized courses of Nuclear Chemistry, the total number of undergraduate students taking courses in nuclear engineering is 18.

Nucl. Engng diploma work (master thesis): 30 pts.
Reading periods 3 and 4 (7 students)

Higher level graduate courses given:

Advanced reactor theory,	15 pts;
Radiation detection and measurements,	7,5pts;
Actinide chemistry,	15pts;
Solvent extraction,	6pts.

Uncertainty and sensitivity analysis applied to the simulation of the Swedish Boiling Water Reactors

Research leaders: Professor Christian Ekberg and Associate Professor Christophe Demazière

Research scientist: Augusto Hernández-Solís

Background

In earlier days, the modelling of nuclear reactors, both for static and transient calculations, was very often performed via very conservative tools. Such analyses were rather crude and only worked analytically for a number of simple cases. This conservatism was, among others, the result of limited computer power, which prevented using sophisticated models, especially on the thermal-hydraulic side. With the recent increase of cheap CPU power, advanced modelling methods are now in reach. The actual trend worldwide is to develop and use so-called Best-Estimate (BE) methods for nuclear reactor simulations. These BE methods are based on coupled (or sometimes integrated) neutronic/thermal-hydraulic calculations, where the interplay between the neutron kinetics and the thermal-hydraulics can be properly accounted for. This coupling thus makes it necessary to have detailed modelling tools on both the neutronic and the thermal-hydraulic sides. Although this coupling allows significantly improving the accuracy of the calculations, a full evaluation of the uncertainties associated to these BE methods is highly beneficial, in order to assess the reliability, the robustness and the fidelity of the simulations. The main advantage of uncertainty evaluation is to decrease even further the conservatism of the safety analyses, which can lead to a decrease of the safety margins and thus to a maximisation of the reactor output/utilization.

Goals of the project

Developing an uncertainty analysis methodology is highly beneficial for many different reasons:

- For licensing and safety purposes: if a BE approach is used in connection with an uncertainty evaluation, a relaxation of the licensing rules is possible, leading to less conservative safety margins, and a maximization of the reactor output/utilization. This is of particular interest for the extensive program of power uprates in Sweden.
- For identifying deficient models: the sensitivity and uncertainty analysis can provide some guidance about the correlations and the code models that would lead to a significant increase of the accuracy of the calculations if these correlations and models were to be improved.

The goal of the present project is thus to develop a tool for uncertainty and sensitivity analysis applied to nuclear reactor simulations. This project exclusively focuses on the case of the Swedish BWRs. The simulation tool is based on the POLCA-T code. In this framework, a close collaboration with the POLCA-T code developers (Westinghouse Electric Sweden AB) is planned within this project. If successful, the last past of the project will be devoted to a generalization of the methodology to other types of reactors/codes.

Organization

The work is performed by PhD student Augusto Hernández-Solís under the supervision of Professor Christian Ekberg and Associate Professor Christophe Demazière. Dr. Paolo Vinai and Arvid Ödegaard - Jensen also support Augusto Hernández-Solís on some aspects of the project. The members of the reference group are: Oddbjörn Sandervåg, SKI, Henrik Nylén, Ringhals, Pär Lansåker, Forsmark, Christer Netterbrant, OKG, and Ulf Bredolt, Westinghouse.

Methodology

There are existing methods for quantifying and calculating the effect of data uncertainties in the field of nuclear reactor calculations, e.g. the CIAU [Code (with the capability of) Internal Assessment of Uncertainty] methodology from Pisa [1], [2] and the CSAU (Code Scaling, Applicability, and Uncertainty) technique developed by the USNRC [3]. In this project, another approach for uncertainty and sensitivity analysis than the ones used in the CIAU and CSAU methods is proposed, based on the experience of the Nuclear Chemistry group at Chalmers [4]. The usual definition of sensitivity and uncertainty analysis is that the sensitivity analysis answers the question of how important an input parameter is and the uncertainty analysis gives the uncertainty on the output resulting from the uncertainties in the input parameters. In some cases when many input parameters are required, it may be possible to perform a screening calculation first to limit the number of parameters to those that have some effect within the selected uncertainty range. In addition to the rather straightforward data uncertainties, there are also conceptual uncertainties, e.g. which method and which program are used to calculate a certain event. The evaluation of these uncertainties is not clear but still plays an important role in the resulting accuracy of the calculations. As a starting point, the Latin Hypercube Sampling (LHS) method, which the Nuclear Chemistry group at Chalmers has some experience with, will be investigated. The general idea with the LHS method is that given every input data with uncertainty and distribution, several samples are taken and the results recorded. From there on, simple statistics may be used for evaluation. The LHS method is an effective sampling method that allows a minimization of the computer time required for the sampling. The present aim for the PhD project is to construct a program package that controls the real simulation code, makes the sampling, and performs an uncertainty and a sensitivity analysis. Although the LHS method is not new, its applications to the nuclear engineering area were rather limited in the past, focusing mostly on the evaluation of the uncertainties associated to the peak cladding temperatures, and to the Departure from Nucleate Boiling Ratio (DNBR). Furthermore, the LHS method has the advantage of requiring much less computer runs compared to the Response Surface Methodology used in the CSAU method. Nevertheless, it may well be that during the start up of the project a more effective method for either the uncertainty or sensitivity analysis is selected and in such a case this will be used instead.

Augusto Hernández-Solís started his employment on October 1st, 2007. Since then, Augusto Hernández-Solís mostly attended courses in Nuclear Engineering and started looking at the literature published in the field of uncertainty and sensitivity analysis.

References

1) F. D'Auria, and W. Giannotti (2000), "Development of a code with the capability of internal assessment of uncertainty." Nuclear Technology, 131 (2), pp. 159-196.
2) A. Petruzzi, F. D'Auria, and W. Giannotti (2005), "Methodology of internal assessment of uncertainty and extension to neutron kinetics/thermal-hydraulics coupled codes." Nuclear Science and Engineering, 149 (2), pp. 211-236.
3) B. E. Boyack, I. Catton, R. B. Duffey, P. Griffith, K. R. Katsma, G. S. Lellouche, S. Levy, U. S. Rohatgi, G. E. Wilson, W. Wulff, and N. Zuber (1990), "Quantifying the reactor safety margins part I: An overview of the Code Scaling, Applicability and Uncertainty evaluation methodology." Nuclear Engineering and Design, 119 (1), pp. 1-15.
4) C. Ekberg, A. Ödegaard-Jensen, and G. Meinrath (2003), "LJUNGSKILE 1.0: A computer program for investigation of uncertainties in chemical speciation." SKI report 2003:03.

Development of an integrated neutronic/thermal-hydraulic tool for noise analysis

Research leader: Associate Professor Christophe Demazière

Research scientist: Viktor Larsson

Background

The neutron noise, i.e. the difference between the time-dependent neutron flux and its time-averaged value, assuming that all the processes are stationary and ergodic in time, allows determining many interesting features of a reactor. The neutron noise can be used either for diagnostic purposes, when an abnormal situation is suspected, or for estimating a dynamical core parameter, whereas the reactor is at steady-state conditions. Noise diagnostics has the obvious advantage that it can be used on-line without disturbing reactor operation. Such a monitoring technique received further attention in the past few years due to the extensive program of power uprates worldwide. Some of main issues/ concerns related to the operation of the plants at the uprated power level are the reduction of the safety margins, such as the margins to instability for BWRs, and increased vibrations (flow induced vibrations). When analyzing neutron noise measurements, the knowledge of the so-called reactor transfer function is of prime importance. This transfer function gives the space-dependent response of the reactor to perturbations that might be localized or spatially-distributed. As a matter of fact, most of the diagnostic tasks require the prior determination of the reactor transfer function, since the original perturbation has to be estimated from the detector reading (unfolding task).

Goals of the project

The Department of Nuclear Engineering, Chalmers University of Technology, developed in the past a tool, usually referred to as a “neutron noise simulator”, allowing determining the reactor transfer function [1]. This simulator is able to calculate the response of a nuclear core to perturbations expressed as fluctuations of the macroscopic nuclear cross-sections or of the possible external neutron source, assuming that the operating conditions of the reactor are stationary. The noise simulator was successfully benchmarked against analytical or semi-analytical solutions and was already used in many diagnostic tasks (see [2] for an overview of some of those). This preliminary version of the tool was demonstrated to work properly and to give new physical insights for the interpretation of noise measurements. Nevertheless, the existing tool has some shortcomings, such as its inability to model closed-loop reactor transfer functions. The goal of the present PhD project is to further develop this tool to bring it to a level of development/sophistication/reliability similar with coupled time-dependent codes. The PhD project is thus aiming at developing a full-core integrated neutronic/ thermal-hydraulic tool for noise analysis. This requires extensive work both on the neutronic side (use of nodal methods) and on the thermal-hydraulic side (development of thermal-hydraulic models). The main advantage of the new tool would be that the neutronics is based on the calculation of the actual Green’s function of the reactor, and that all the time-dependent equations describing the fluctuating quantities are Fourier-transformed. The applications of this tool would be numerous for noise analysis. Due to the coupling to any code system, this tool could be easily applied to any of the Swedish nuclear power plants.

Organization

The work is performed by PhD student Viktor Larsson under the supervision of Associate Professor Christophe Demazière. Prof. Imre Pázsit and Dr. József Bánáti are also supporting Viktor Larsson on some aspects of the project. The members of the reference group are: Ninos Garis, SKI, Tell Andersson, Ringhals, Farid Alavyoon, Forsmark, Christer Netterbrant, OKG, and Camilla Rotander, Westinghouse.

Methodology

In 2007, several semi-analytical 1-dimensional models were developed in 2-group theory in order to determine the spatial dependence of the neutron noise throughout the reactor. First, a 1-region reactor model was developed in both diffusion theory and P_1 theory. The purpose of this model was twofold: first to get familiar with noise theory and to get some physical intuition about the results to be expected, second to determine whether P_1 theory would give significantly advantageous results for the calculation of the neutron noise compared to diffusion theory. This preliminary study demonstrated that the only case where significant results can be expected are when steep gradients of the static neutron flux exist. Based on these observations, a 2-region reactor model was developed, where the regions represent the active fuel and the reflector, respectively. As an illustrative example, the neutron noise induced by a local noise source is given in Figs. 1 and 2, for both a central perturbation and a non-central perturbation, respectively. In these figures, the neutron noise calculated using a semi-analytical solution and a fully numerical solution based on finite differences are also represented. The full results of such calculations and the resulting conclusions will be presented at the PHYSOR 2008 conference [3]. In parallel with these investigations, nodal methods are being studied in order to determine the most suitable one to be applied to the calculation of the neutron noise.

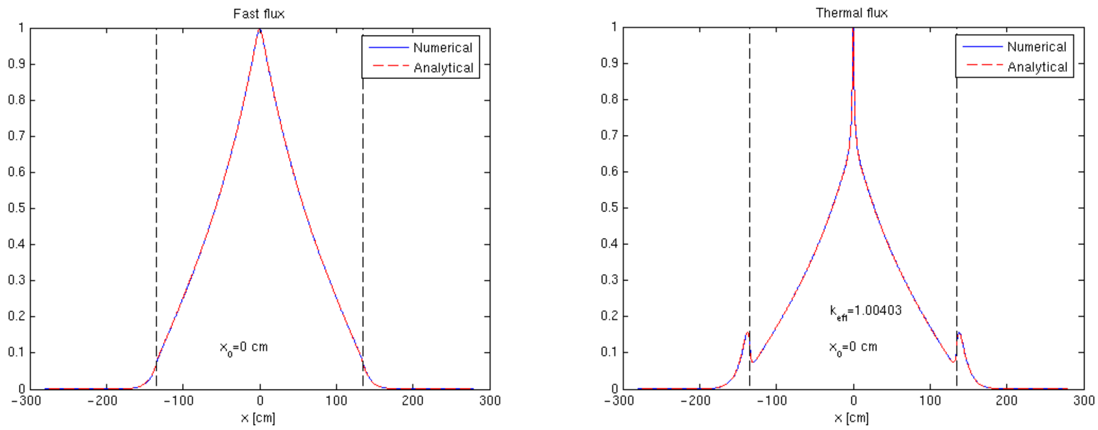


Figure 1. Amplitude of the neutron noise induced by a central perturbation (in the fast and thermal groups, on the left and right hand-side, respectively).

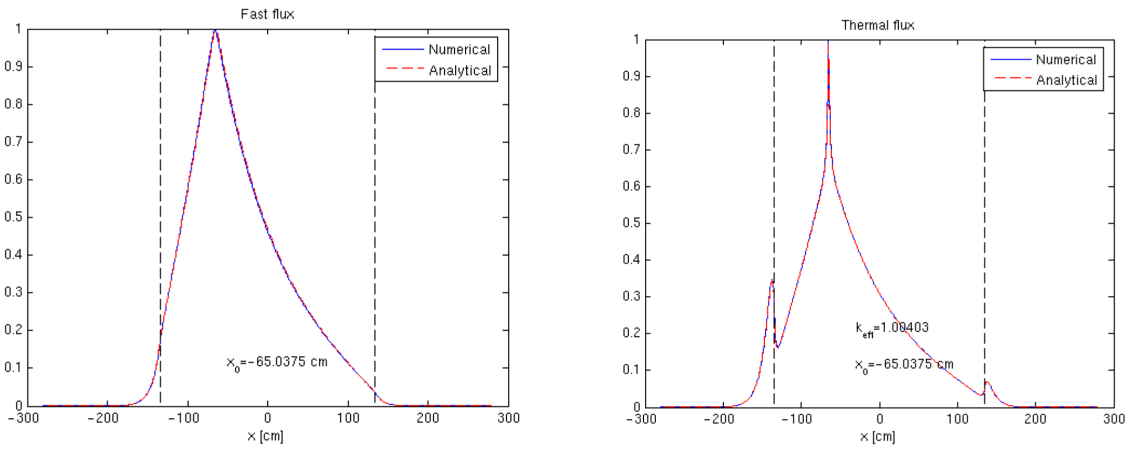


Figure 2. Amplitude of the neutron noise induced by a non-central perturbation (in the fast and thermal groups, on the left and right hand-sides, respectively).

Publications

1) C. Demazière, "Development of a 2-D 2-group neutron noise simulator," Annals of Nuclear Energy, 31, pp. 647-680 (2004).

2) C. Demazière and I. Pázsit, "Numerical tools applied to power reactor noise analysis," Accepted for publication in Progress in Nuclear Energy (2008).

3) V. Larsson and C. Demazière, "Semi-analytical calculations of the neutron noise in 2-group theory for 1-D homogeneous systems", submitted to PHYSOR 2008, International Conference on Reactor Physics, Nuclear Power: A Sustainable Resource, Casino-Kursaal Conference Center, Interlaken, Switzerland, September 14-19, 2008.

Advanced analysis methods for non-stationary processes in reactor cores

Research Leaders: Professor Imre Pázsit, Department of Nuclear Engineering, Chalmers University of Technology, Göteborg and Docent Ninos Garis, Swedish Nuclear Power Inspectorate, Stockholm.

Scientist: Carl Sunde, Department of Nuclear Engineering, Chalmers University of Technology, Göteborg.

Background

Diagnostics of reactor cores with noise methods is usually performed with FFT based spectral analysis, such as auto and cross spectral power densities and coherence/phase between variables. Such an analysis assumes that the system behaviour is stationary during the measurement period. In other words the status of the system is assumed to be unchanged over several tens of thousands of the periods of the characteristic frequencies of the system.

Nevertheless, the behaviour of complex systems such as a reactor core is often non-stationary, i.e. the state of the system often changes during a much shorter period. Non-stationary processes and transients are in fact quite common in reactor systems, such as the occurrence and development of local and global core instabilities in BWRs, the short-term changes of vibration properties (core-barrel, fuel assembly etc) in PWRs, and the various phenomena in two-phase flow (vortex shedding, slug flow etc). Apart from temporal transients, spatial transients or non-stationarities may also occur, such as in represented by the spatial structure of two-phase flow.

Goals

Powerful mathematical methods have been developed lately and applied for the analysis of such non-stationary processes, out of which wavelet analysis appears to be one of the most promising. The first purpose of the project was to introduce the use of known analysis methods of non-stationary processes, and primarily wavelet analysis, into the noise diagnostic work of our Department and to explore their possibilities for diagnosing non-stationary processes. A second goal was to develop wavelet-based methods further for tackling new problems which are of practical interest in our R&D work. Both a more effective early detection and quantification of beginning anomalies as well as elaborating of new, wavelet-based methods of reliable parameter estimation under non-stationary circumstances were addressed. The test of the methods was performed through both simulated signals as well as measurements taken in Swedish power plants.

Organisation

The reactor diagnostic group is headed by Prof. Imre Pázsit, who was also the leader of this SKC-project. The procject has been going on since July 2002. At the beginning of the project Dr. Vasily Arzhanov, now at KTH, introduced Kalle to noise theory and analytical models. Doc. Christophe Demaziere, university lecturer, was also supporting Carl Sunde during several stages of the project.

The members of the reference group were: Pär Lansåker Forsmark, Henrik Eisenberg OKG, Henrik Nylen Ringhals, Ninos Garis SKI, Johan Larsson Ringhals, and Camilla Rotander, Westinghouse. Doc. Ninos Garis served as deputy adviser to Kalle. The last and final reference group meeting was held in December 2006.

Methodology

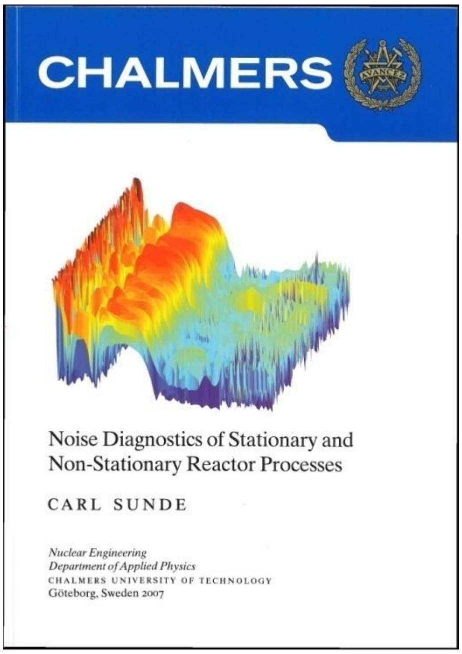
The methodology is similar to traditional noise analysis work, which consists of both evaluation of measurements, and elaborating models of the reactor and its processes to expedite the interpretation of the measurement analysis. Hence both theoretical model development and analysis of measurements is involved. In the analysis part, in contrast to the FFT tool, used in the traditional methods, continuous (CWT) or discrete fast wavelet transform (DWT) is used.

Activity in 2007

All technical work which was included into the thesis was completed in 2006 and already published or accepted for publication. The activity in 2007 was devoted to the writing up of the thesis, organising and executing the PhD exam. In addition several further piece of work was performed by Kalle wich was not included into the thesis, such as evaluating measurement from Ringhals-1, as well as preparations for a future coming article on the accuracy of the source modulation method.

The PhD exam was made on 27 April 2007 and Kalle has defended his thesis entitled “Noise diagnostics of stationary and non-stationary reactor processes”. External examiner was Prof. Y. Yamane, Nagoya University. Members of the committee were Prof. Mats Viberg, Signals and Systems, Chalmers, Assoc. Prof. Tatiana Tambouratzis, University of Piraeus, Greece, and Dr. Joakim Karlsson, Studsvik AB.

The thesis is based on five papers published in international journals, and one conference proceedings.



Summary of the results

The type of problems investigated and corresponding results/conclusions can be summarized as follows.

1. Identification of two-phase flow with wavelet preprocessing and neural networks. It was found that artificial neural networks, ANN, can be trained to identify two-phase flow patterns from neutron radiography images and optical images of the flow by visible light. The wavelet preprocessing did not increase the success ratio of the identification, i.e. did not increase the discriminative power of the selected features, but made the training stage significantly faster. Hence the experience was positive.

2. Detector string impacting. This was presumable the most successful application of wavelets in reactor diagnostics. Both discrete and continuous wavelet transform methods were used with success. Especially the wavelet coherence, based on the continuous transform, appeared to be an objective method for detecting impacting of detector tubes from the detector signals, which can be applied without either need of input from experts or need for calibration and special tailoring to the concrete reactor unit in question. The basis of the method is empirical in its character, but the procedure is based on an algorithm that can be run by non-experts. Several measurements from Ringhals were evaluated, but we are still waiting for the results of the visual inspections made during the following refuelling.

3. Investigation of the use of wavelet techniques for BWR stability analysis. In this study several applications of wavelets were tested. One was to improve the quality of the detector ACF (autocorrelation) signals, i.e. eliminate trends and parasitic noise, before applying the classical methods of determining the decay ratio (DR) from the ACFs. Although the wavelet filtering indeed improved the visual quality of the ACFs, the DR determined from the filtered signals was not always better than the value obtained without wavelet filtering. Another application was to use continuous wavelet transform methods for direct calculation of the DR from the raw signals. This method worked quite satisfactorily, and could be used to determine the larger DR (i.e. that of the more unstable component) in case of dual oscillations. If the frequency of the simultaneous oscillations differ even only with a slight value, i.e. a few percent, the method is capable to identify both decay ratios.

4. Core barrel vibrations, and the break-frequency method vs point-kinetics. These parts of the project concern traditional, stationary reactor diagnostic methods, and will not be reviewed here. They were executed partly due to their own interest for the research of the Department, and partly to give Kalle an opportunity to deepen his knowledge in reactor physics and core dynamics as well as core calculations. These studies were also published internationally and well received.

Carl Sunde received two awards at international conferences for the work included into his thesis, and he became the winner of 2007 years’ Sigvard Eklunds prize for best PhD thesis.

Ruthenium chemistry in the reactor containment during severe accidents in nuclear power plant

Research leaders: Prof. Christian Ekberg[1] and Dr. Henrik Glänneskog[2]

Research scientist: Ph.D. Student Joachim Holm[1], Department of Nuclear Engineering, Chalmers University of Technology, Gothenburg

[1] Nuclear Chemistry, Chalmers University of Technology, Gothenburgh

[2] Ringhals, Väröbacka

Background

Ruthenium is semi-volatile fission product, which is formed during normal operations in nuclear power plant. During a "normal" severe accident, with core melt and so on, ruthenium doesn't constitute a threat to environment, because of its non-volatile properties. However, during a severe accident with an air-ingress into the reactor vessel ruthenium can be oxidized and adopt a volatile form like ruthenium tetroxide (RuO_4). This specie can be released from the nuclear fuel and the reactor vessel and be transported to reactor containment. It is therefore really important investigate if a containment has the ability to reduce and prevent the source term of ruthenium, in a case of a severe accident.

Goals:

The goals of the project are to investigate both the distribution of RuO_4 between an aqueous phase and gaseous phase and also study the reactions between $RuO_4(g)$ and some metals surfaces that are in a Boiling Water Reactor (BWR) containment. The metals are copper, zinc and aluminium. The atmosphere and the conditions in the experiments are similar to the conditions in a containment of BWR. The temperature interval in the experiments is 20-50°C.

Organisation:

The work is performed by Ph.D. student Joachim Holm under the supervision of Prof. Christian Ekberg and Dr. Henrik Glänneskog.

Methodology:

The experiments are performed in the experimental set-up seen in figure 1.

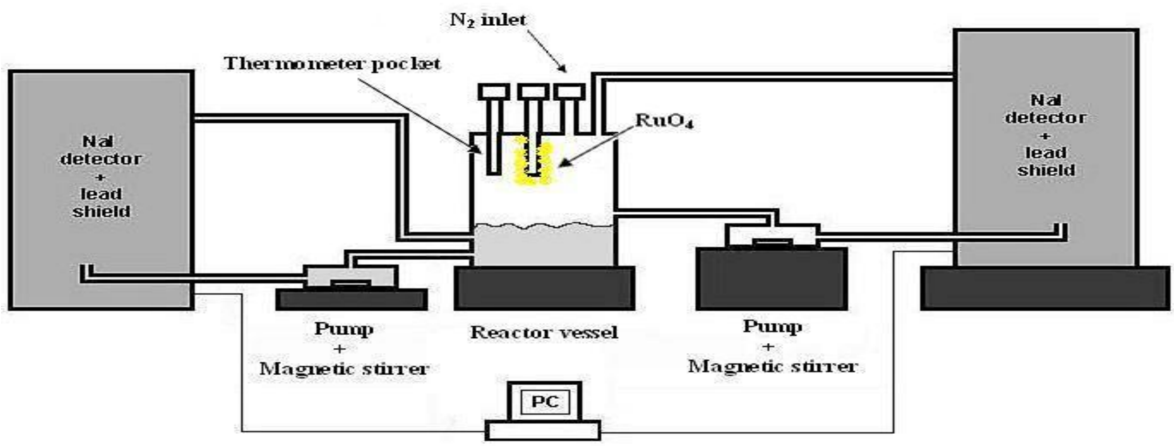


Figure 1: The experimental set-up, used for the RuO_4 distribution experiments.

The experimental set-up consists of a reactor vessel connected to two loops, one loop for the water phase and one loop for the gas phase. The two loops pass by two new 2in. × 2in. NaI(Tl)-detectors, which are connected to a computer. The detectors are used for detection of the 496 keV γ-rays emitted in the decay of ^{103}Ru. The temperature is generated by heating band twirled around the pipes and the vessel.

Every experiment is started by filling the aqueous loop and half of the reactor vessel with distilled water. The circulation is then started in the two phases, the flow rates are about 5 $cm^3 \cdot s^{-1}$ and 2 $cm^3 \cdot s^{-1}$ in the aqueous and the gaseous phase, respectively. The system is then flushed with nitrogen gas to secure that only a few percent of oxygen is presented. Gaseous ruthenium tetroxide spiked with an arbitrary amount of ^{103}Ru is introduced into system by letting $^{103}RuO_4$(cr) vaporize in the reactor vessel. The distribution of $^{103}RuO_4$ between the two phases is measured by the two NaI-detectors.

The interactions between the three metals and gaseous ruthenium tetroxide are investigated by, letting RuO_4(cr) sublimate into a gaseous form in a reaction bottle with the metal samples present. The temperature for the experiments is room temperature. After the experiments are the metals sample investigated by the surface analysis methods ESCA and SEM-EDX.

Results:

From the performed distribution experiments following results have been received.

- RuO_4(g) was almost immediately distributed in aqueous phase after the introduction of RuO_4 in the gaseous phase in the system, in the temperature interval 20-50°C.
- There was a significant deposition/adsorption of ruthenium on the glass surfaces in the experimental set-up. The speciation of deposited ruthenium was analyzed by SEM-EDX and ESCA and was determined to hydrated RuO_2.

The conclusion from these experiments is that the probability of a release of RuO_4 from the containment to the environment seems to be extremely small.

The experiments with interactions of RuO_4 and the metals are not finished yet. But some preliminary results have been received.

Significant deposition of RuO_4 on the metals, especially on zinc.

Publications:

1) J. Holm, C. Ekberg and H. Glänneskog, Reactions of RuO_4 under severe nuclear reactor accident conditions, Submitted to Radiochimica Acta.

Uppsala University

Division of Applied Nuclear Physics

Education in the field of energy systems and technology is conducted through a number of courses within the various civil engineering programs. In addition, education is performed on commission within the framework of an agreement with the Nuclear Safety and Training Centre (KSU), a subsidiary of Vattenfall AB.

The research with relevance for the nuclear power industry is performed within two research groups:

The Fuel Diagnostics and Safeguards Group.

Research in the technical aspects of international safeguards, diagnostics of irradiated nuclear fuel and development of advanced measuring techniques for encapsulation and final storage of spent nuclear fuel.

The Neutron Reaction Group.

Research and development of advanced measuring and analysis tools for measurements of nuclear cross sections with relevance for applications within transmutation of nuclear waste and dosimetry. Together with the Fuel Diagnostics and Safeguards Group, this group develops new equipment and methods for fission reactor diagnostics as well as technologies for future reactor concepts.

New employees of the nuclear industry attending the course H1 in Uppsala (spring 2008).

Education

The following courses are given within the civil engineering programmes and within the collaboration with KSU.

Intro STS	1.5 pts
Elektroteknik	4.5 pts
Energifysik I	6 pts
Energifysik II	6 pts
Energifysik projekt	7.5 pts
Energisystemfysik	7.5 pts
Risker i komplexa system	7.5 pts
Kärnkraft ToS	7.5 pts
Joniserande strålning och detektorer	7.5 pts
Kärnkraftteknik H1	12 pts
Strålningsfysik FS1	6 pts
Termohydraulik	1.5 pts
Tillämpad reaktorfysik	7.5 pts
Kärnämneskontroll och Icke-spridning av kärnvapen	7.5 pts
Fortbildning för KSU:s personal	1.5 pts

Theses presented during 2007

The following theses, with relevance for the nuclear industry, have been presented during 2007:

- Otasowie Osifo: "Automatic Gamma-Scanning System for Measurement of Residual Heat in Spent Nuclear Fuel" (Ph. Lic.)
- Tobias Lundqvist Saleh: "Tomographic techniques for safeguards measurements of nuclear fuel assemblies" (Ph. Lic.)
- Karen Sihm Kvenangen: "Alternative Measuring Approaches in Gamma Scanning on Spent Nuclear Fuel" (M.Sc.)
- Charlotte Lager: "Investigation of the Spectroscopic Performance of a Cadmium Telluride Detector Using Pulse Shape Correction and Digital Techniques" (M.Sc.)
- Anna Kindberg: "Emergy Evaluation of a Swedish Nuclear Power Plant" (M.Sc.)

More information is available at:
www.fysast.uu.se

Tomographic verification of the integrity of nuclear fuel assemblies

Research leader: PhD Staffan Jacobsson Svärd
Scientist: PhD student Tobias Lundqvist Saleh
Department of Neutron Research, Uppsala University

Background

A tomographic measuring technique for irradiated nuclear fuel assemblies was developed at Uppsala University in an earlier project supported by SKC [1]. Two applications of the technique were demonstrated; (1) highly accurate determination of the thermal pin-power distribution, and (2) verification of the integrity of spent nuclear fuel.

The current project is focused on the latter application, in particular in connection to the future final repository that is planned for spent nuclear fuel from Swedish nuclear power plants. For this type of "difficult-to-access storage", the International Atomic Energy Agency (IAEA) sets safeguards criteria to ensure non-proliferation of nuclear materials. One such criterion is that the integrity of the fuel has to be verified.

Currently, there are no measuring techniques for verifying the integrity on the 100 % level, i.e. for ensuring that all declared fuel rods are present in a fuel assembly. However, tomographic techniques have been identified as the strongest candidates for this type of verification [2].

Goals of the project

The objective of the current projects is to further develop the tomographic measurement technique from [1] and to adapt it especially for integrity verification of spent nuclear fuel, with focus on encapsulation and final reposition. The goal is to present a fast and highly confident measuring technique that is capable of detecting partial defect on the single rod level.

Organization

The work is performed by PhD student Tobias Lundqvist Saleh under the supervision of PhD Staffan Jacobsson Svärd at the Department of Neutron Research at Uppsala University. The results are presented at international conferences and published in refereed journals. The reference group consists of Kåre Axell (SKI), Björn Bjurman (Forsmark), Henrik Nylén (Ringhals), Mats Thunman (Westinghouse) and Göran Wiksell (OKG).

Methodology

The tomographic technique will be developed to enable a safeguards inspection procedure based on the following methodology:

1) Tomographic measurements of the gamma-ray flux distribution around a nuclear fuel assembly.
2) Tomographic reconstruction using fast, analytic algorithms that enable on-line inspection.
3) Image analysis of the images resulting from the on-line reconstructions in order to identify and count the number of fuel rods in the assembly.

In addition, techniques to determine whether there is a replacement material in the position of a missing fuel rod will be investigated. In this context, the algebraic reconstruction algorithms from [1] may be used in off-line analyses to enhance precision and confidence.

The techniques are developed based on theoretical considerations and computer simulations. Verification of the theoretical framework will be performed using a laboratory measuring device including a mock-up of a fuel assembly.

Results

The project has just passed half-time, and methodology and algorithms have been developed according to the plans accounted for above. The methodology has been presented in a refereed paper [3] and the reconstruction algorithms developed have been presented in a manuscript that will be submitted to *Nuclear Instruments and Methods A* [4].

The results of a tomographic reconstruction of data from an authentic fuel assembly, recorded in 2002 at the Forsmark 2 NPP, are presented in Figure 1. Also shown in the figure are reconstructed pixel activities in a central row of fuel rods and the background level deduced by means of image analysis. The results demonstrate the technique's potential for verifying the presence of all fuel rods.

In addition, the construction of a proposed measuring device has been presented in a paper [5], which has been submitted to *Nuclear Science and Engineering*. The work was based on experiences gained when using a test platform (see ref. [1]). The suggested device was designed to ascertain more efficient shielding to background radiation and it has advantageous properties regarding transports and decontamination.

Furthermore, a study of the plutonium composition in spent nuclear fuel and its suitability in nuclear weapons has been performed, which was presented at the ESARDA conference in May 2007 [6]. The work concludes that plutonium from fuel with low burnup would be the most attractive target for weapons production. Thus, methods for determining fuel burnup are highly relevant and tomographic verification of integrity particularly interesting for assemblies with low burnup.

Papers [3-6] serve as the basis of the Licentiate Thesis [7] presented in October 2007.

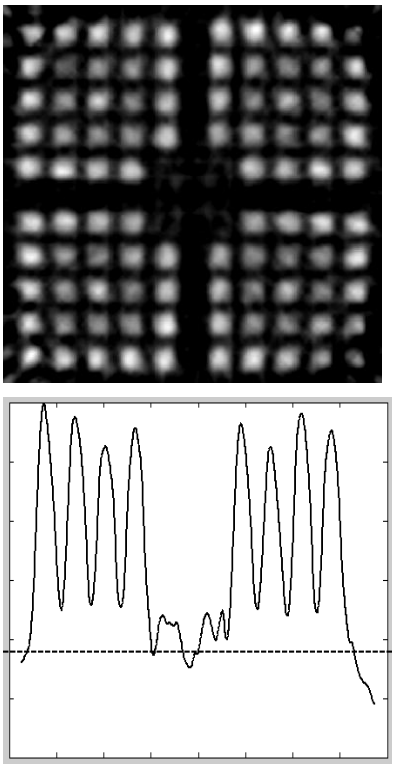


Figure 1: Image of the ^{140}Ba distribution (top) reconstructed using a newly developed algorithm [4]. The data were collected in measurements of a SVEA-96 assembly at the Forsmark 2 NPP. Reconstructed pixel activities in row 5 are also illustrated (below) together with the background level in the image, deduced using image analysis.

References

1) S. Jacobsson Svärd, A Tomographic Measurement Technique for Irradiated Nuclear Fuel Assemblies, Comprehensive Summaries of Uppsala Dissertations from the Faculty of Science and Technology 967, ISSN 1104-232X, ISBN 91-554-5944-7, Uppsala (2004).

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3) T. Lundqvist, S. Jacobsson Svärd, A. Håkansson, SPECT imaging as a tool to prevent proliferation of nuclear weapons, Proceedings of Imaging 2006, Stockholm, Sweden, June 27-30, 2006, Nuclear Instruments and Methods A, vol. 580, issue 2, pp. 843-847 (2007).

4) T. Lundqvist Saleh, S. Jacobsson Svärd, A. Håkansson, Reconstruction algorithms for SPECT on irradiated nuclear fuel assemblies and their applicability, in manuscript, to be submitted to Nuclear Instruments and Methods A.

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6) T. Lundqvist, Safeguards aspects of recycling plutonium into MOX fuel in light water reactors, 28th Annual Symposium on Safeguards and Nuclear Materials Management, ESARDA, Aix en Provence, France, May 22-25, 2007.

7) Tobias Lundqvist Saleh, Tomographic Techniques for Safeguards Measurements of Nuclear Fuel Assemblies, Licentiate Thesis, UU-NF 07#14, ISSN 1401-6269 (2007). Available at http://www.inf.uu.se/Reports/UUNF07-14.pdf

SKC Financials in 2007

The following table summarises the SKC financials for 2007

Received from Financing parties		16 000 000 kr
Saved from previous years		2 509 401 kr
Costs for post-graduate school	693 365 kr	
Paid to KTH	3 150 000 kr	
Paid to Chalmers	2 000 000 kr	
Paid to Uppsala University	1 000 000 kr	
Research Projects	5 190 880 kr	
Other costs	1 676 355 kr	
Sum of costs in 2007		13 710 600 kr
Transfer to new SKC		4 798 801 kr

The contributions from the financing organisations are split as follows:

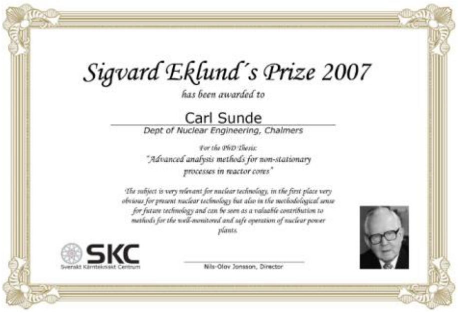
SKI	33%
Westinghouse	20%
Forsmark	17%
Ringhals	13%
OKG	13%
Barsebäck	4%

The Activity Plan for 2008-2010 includes a suggestion that 3 075 000 kr out of the total savings of 4 798 801 kr should be allocated mainly to new research projects in 2008.

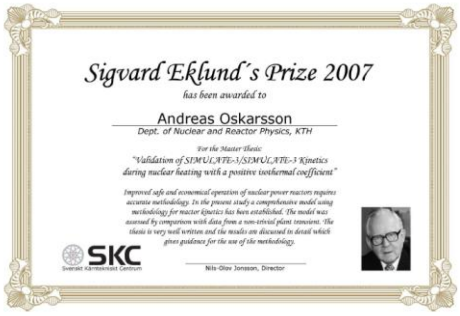
Winners of Sigvard Eklund's Price in 2007

Carl Sunde, Chalmers, was awarded the price for the best PhD thesis, which has the title "Noise Diagnostics of Stationary and Non-Stationary Reactor Processes".

Andreas Oskarsson, KTH, was awarded the price for the best master thesis, which has the title "Validation of SIMULATE-3/SIMULATE-3 Kinetics during nuclear heating with a positive isothermal coefficient".



The picture below is from the price award ceremony. The price winners (holding their diploma) are Dr. Carl Sunde to the left and MSc Andreas Oskarsson to the right. Members of the SKC Board are also in the picture.



SKC – Partners, Tasks and Goals

By Nils-Olov Jonsson,
Director of SKC

SKC - Swedish Center for Nuclear Technology or Svenskt Kärntekniskt Centrum in Swedish - has been active since 1992 in providing support to education and research within the nuclear power area. From the first of January 2008 the SKC partners have entered a new six year period of support to KTH, Chalmers and Uppsala University for senior positions at these universities and for research projects.

The partners are:

- Statens Kärnkraftinspektion (SKI, the Swedish Nuclear Power Inspectorate)
- Forsmark Kraftgrupp AB
- Ringhals AB
- OKG AB
- Westinghouse Electric Sweden AB

and the three universities:

- Kungliga Tekniska Högskolan (KTH)
- Chalmers Tekniska Högskola AB
- Uppsala Universitet

SKC is active within three research programs:

1) Nuclear Power Plant Technology and Safety
2) Reactor Physics and Nuclear Power Plant Thermal Hydraulics
3) Materials and Chemistry

An education program is also supported by financial contributions to senior positions at the universities.

Within the research programs the focus is on the areas of primary interest to the SKC partners, as shown in the following list:

- Thermal-Hydraulics
- Core Physics
- Core and Plant Dynamics
- Chemistry
- Material physics and engineering
- Safety & Severe Accidents
- Reactor Diagnostics
- Detectors and measurement
- Safeguards
- Fuel Technology

SKC shall provide long-term support to securing knowledge and competence development at an academic level for the Swedish nuclear technology programs. This shall be a basis for providing resources to the Swedish nuclear industry and its regulators. It means that SKC will contribute to a safe, effective and thus reliable nuclear energy production, which is an important part of the Swedish energy supply.

SKC has five top level goals for reaching its vision:

1. Increase the interest among students to enter nuclear technology education.
2. Make sure that the needs of the SKC financing parties to recruit qualified personnel with a nuclear technology education are met. To meet this goal, the universities will offer relevant basic education, execute research projects and support continued education of engineers already active in the nuclear technology area.
3. Offer attractive education in the nuclear technology area.
4. Maintain strong and internationally acknowledged research groups within areas which are vital for and unique to the nuclear technology area.
5. Create organizations and skills at the universities such that research can be performed on account of the financers of the SKC also outside the boundaries of the SKC agreement.

Formally, SKC is organized as a center within the School of Science at KTH.

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