SKC Annual Report 2008

Swedish Centre for Nuclear Technology

March 2009

SKC 09-01

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Contents

1 Summary of 2008

1 SKC Sponsors in 2008

2 SKC-Partners Tasks and Goals

3-4 SKC-a look into the future

5 2008 was the first year in a 6-year agreement

6-14 KTH - Overview of activities in 2008

Research Projects at KTH

15-25 Chalmers – Overview of activities in 2008

Research Projects at Chalmers

26-29 Uppsala – Overview of activities in 2008

Research Projects at Uppsala

30 SKC Financials

30 Winners of the Sigvard Eklund's Price in 2008

Summary of 2008

Twelve PhD projects have been in progress during 2008 with funding from SKC. The Sigvard Eklund prize to the best thesis of the year was awarded to Dr. Olivia Roth for her work on the chemistry of radiation-induced dissolution of spent nuclear fuel. Andreas Carlson won the prize for the best masters´ thesis for his work in the area of multi-phase flow in micro-channels.

A large number of students have followed 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 about 160 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).

SKC Sponsors in 2008

SKC has been sponsored by the following organisations during 2008:

- Forsmark Kraftgrupp AB
- OKG AB
- Ringhals AB
- Swedish Radiation Safety Authority (SSM) from July 1st, 2008 former Swedish Power Inspectorate (SKI)
- Westinghouse Electric Sweden AB

Total support from these organisations was 17 million Swedish kronor during 2008.

SKC-Partners, Tasks and Goals

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:

- Swedish Radiation Safety Authority (SSM, Strålsäkerhetsmyndigheten)
- 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 Sciences at KTH.

For further information see:
www.swedishnuclear.eu

SKC – a look into the future

A message from the director

Since February 1, 2009, I am the new director of the Swedish Nuclear Technology Centre. It is a challenging task ahead to lead SKC in the times of nuclear renaissance. The reason for establishing SKC in the mid-90s was to remedy the difficult situation concerning academic competence in nuclear power relevant education and research. With the present comeback internationally of nuclear power, it is evident that SKC has fulfilled an important mission. The resources allocated during the last decade have made it possible today to embrace the new situation much faster than otherwise would have been possible.

Although industry certainly would like more people present right now when the activities at the plants are the most intensive since the plants were constructed, it is widely recognized that without the financial support during the harsh years, there would be essentially no people present at academia today, making the comeback even more difficult. We have a situation now where a generation change has already taken place at Swedish academia. At all the three universities supported by SKC, most research leaders are below the age of 50. Thus, we have good conditions for successful growth.

The renewed interest in nuclear power has prompted a new strategy for SKC. We can foresee that the needs in industry for qualified staff will increase significantly. First, the industry is presently in a retirement phase; a large fraction of the staff will retire within the coming decade. This will obviously motivate large recruitments. Until now, recruitment needs have been met by employing fresh engineers, with little or no nuclear engineering in their curriculum. This in turn has necessitated educational efforts for this staff, managed by KSU.

Presently, the three SKC partner universities are developing deeper educational programmes in nuclear engineering. KTH has a masters program running on its second season, Chalmers starts a new program in the autumn 2009, and Uppsala is launching a nuclear specialization within the existing energy systems program, also in autumn 2009, the latter in collaboration with KSU. Moreover, a national program on bachelor’s level nuclear engineering is planned to start autumn 2010 at Uppsala University, also that in collaboration with KSU.

In parallel, the first signs of a changed attitude from the government towards nuclear power research have been seen. Since 1980, there has been essentially no funding available for novel nuclear power research in Sweden. The late SKI (now SSM) has funded research activities with the aim to support the present fleet in general, and its safety in particular, whereas Vetenskapsrådet (VR), the largest public general research funding organization has in reality been forbidden to support nuclear engineering, although auxiliary research in, e.g., nuclear physics has been funded, however modestly.

In the end of 2008, the government announced an instruction to VR to support research on Gen-IV reactors, and an indicative amount of up to 15 MSEK per year was given. This is the first occasion in a generation when few experts will be shared among operation, modernization and lifetime extension projects, power upgrades as well as public funding has been announced for truly novel nuclear power research.

I have taken the initiative to gather the three SKC universities to a single, joint project, GENIUS (Generation-IV Industry-University consortium Sweden), where we request the total available frame. The project is industry-supported, in the sense that various industry research activities on Gen-IV is launched in parallel, and significant synergy between the academic and industry activities are foreseen.

If this application is approved, it opens the door for continued expansion. There are several other funding bodies (SSF, Vinnova, etc.) that have no formal instructions against supporting nuclear power research and development, but up to now they have provided no funding. Previous attempts to get funding there have in fact been turned down with the motivation that public funding should not be used for nuclear power research. With an approved VR application, such “alibis” against financial support to nuclear power research vanish.

To be able to efficiently utilize the presently favourable situation, SKC has recently decided to streamline its internal administration, especially related to the base support. With less time and other resources spent on the distribution of the base funding to the universities, increased attention can be paid to applications to the agencies mentioned above for increased funding to expand the total volume of activities.

Another change is that SKC will become more visible, to its partners as well as to society at large. Starting already in the autumn 2009, we will have an annual meeting that will circulate among the parties, with Ringhals as first hub. We expect a broad participation from both universities and industry. Especially important is that this will provide a showcase for the PhD students, where they can get feedback from industry on their work, and industry can get a preview of the finest of the SKC products: PhDs to employ.

We are living in exciting times. New-build projects are no longer just dreams, they are already reality. In a long term, I believe this is very positive. This boosts confidence into the entire industry, and I think it is a pre-requisite for safe, reliable and efficient operation also of the existing fleet. In a short term, we can foresee a strained situation when few experts will be shared among operations, lifetime extension projects, power upgrades as well as construction and commissioning of new nuclear power. SKC is ready to give its contribution to successful completion of our joint mission!

Jan Blomgren
SKC director

2008 was the first year in a new 6-year agreement

SKC financing organizations entered a new six year period assuring 102 million Swedish kronor to the universities

Svenskt Kärntekniskt Centrum – SKC – finished its sixth and last year 2007-12-31 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:

- Lennart Billfalk, Chairman, Vattenfall
- Lars Berglund, Forsmarks Kraftgrupp AB
- Magnus Antonsson, OKG AB
- Leif Johansson, Ringhals AB
- Björn Sjöström, Ringhals AB – replacing Leif Johansson at the end of 2008
- Gustaf Löwenhielm, SKI/SSM
- Stig Andersson, Westinghouse
- Gustav Amberg, KTH
- Irene Kolare, Uppsala University
- Imre Pazsit, Chalmers
- Nils-Olov Jonsson, Vattenfall, director of SKC

During 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, which has decided what projects best suit the goals and purposes of the organization.

Jan Blomgren, Vattenfall, replaces Nils-Olov Jonsson as the director of SKC from 1st February 2009.

KTH – Royal Institute of Technology

Overview of Activities in 2008

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

- Reactor physics (dept. of physics)
- Reactor technology (dept. of physics)
- Nuclear power safety (dept. of physics)
- Nuclear chemistry (dept. of chemistry)

These four divisions are jointly members of CEKERT (Centre for nuclear energy technology) at KTH, together with representatives of SKI, Westinghouse Electric 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 2008, 20 senior scientists and 28 PhD students were employed by the CEKERT divisions. Roughly 2/3 of the effort is focused on R&D related to the existing Swedish light water reactor park, the remainder being issues related to spent fuel, such as repository performance and transmutation.

In April 2008, Janne Wallenius was promoted to full professor in reactor physics.

One PhD thesis was defended and one licentiate thesis was presented during 2008:

• Olivia Roth: "Redox chemistry in radiation induced dissolution of spent nuclear fuels - from elementary reactions to predictive modelling", PhD thesis in nuclear chemistry.

• Sara Nilsson: "Influence of metallic fission products and self irradiation on the rate of spent nuclear fuel-matrix dissolution", Lic. Eng. thesis in nuclear chemistry.

The international masters programme in nuclear energy engineering enrolled seven students in the class of 2008. In addition, four students from KTH are following the nuclear energy engineering direction of competence in the engineering physics programme. All 14 students from the class of 2007 were transferred to the second year of the programme, and will make their masters theses during spring 2009.

1050 ECTS were awarded in masters programme courses taught by the CEKERT divisions, corresponding to 17.5 full student years. In addition, eleven masters theses (330 ECTS) were completed during 2008.

In June 2008 the CONFIRM irradiation of two (Pu,Zr)N fuel pins was successfully completed in HFR (Petten). The CONFIRM project was coordinated by reactor physics at KTH. Figure 1 shows a neutron radiography of one CONFIRM fuel pin after completed irradiation. Destructive post irradiation examination of the fuel will be made by NRG and PSI in 2009.

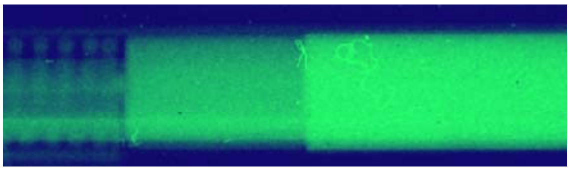

Figure 1: Neutron radiography of one (Pu,Zr)N fuel pin irradiated in HFR, fabricated by PSI according to a recipe developed at KTH.

Five SKC funded PhD projects have been in progress during 2008. A summary description for four of these projects is provided in the following pages.

More information is available at:
www.cekert.kth.se

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

For safe and economical operation of nuclear reactors it is important that the mechanisms which govern heat transfer and fluid flow in fuel assemblies are well understood. The objective of this project is to investigate the influence of spacers on post-dryout heat transfer. The main function of spacers is to keep fuel rods equidistant in the lattice of the fuel rod assembly. The second - equally important - function is to enhance heat transfer and to reduce the probability of the dryout occurrence. The spacer design is a very complex process of optimization of the over-all fuel bundle performance, being a trade-off between pressure losses and improved dryout margins. Needless to say that understanding of the underlying fluid-flow and heat transfer phenomena is an important prerequisite in a successful spacer design. The present project contains a detailed experimental program, which has the main objective to improve the predictive capabilities of the spacer behavior in pre- and post-dryout heat transfer conditions. 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.

Objectives of the project

Recent experimental study of post-dryout heat transfer in an annulus 10x22.1x3650 mm has shown a considerable influence of the spacers on the heat transfer intensity, [1]. 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]. In order to have a detailed view of the spacer influence, the following issues will be addressed: (a) the influence of spacers shape and blockage ratio on heat transfer, (b) the level and the influence of vapor superheat on heat transfer, (c) exact position of quenching/rewetting fronts, and (d) the influence of spacers on drop dynamics. The post-dryout measurement will be performed in two-side heated annulus and using various types of spacers. In addition to the classical procedure [1], the following experimental procedures are foreseen: the *constant-quality-line* method and the *constant-local-quality* method. The constant-quality-line method is based on an approach, in which the parameters (i.e. heat flux, pressure, inlet subcooled and mass flux) are changed in such a way that the quality distribution in the whole test section remains constant.

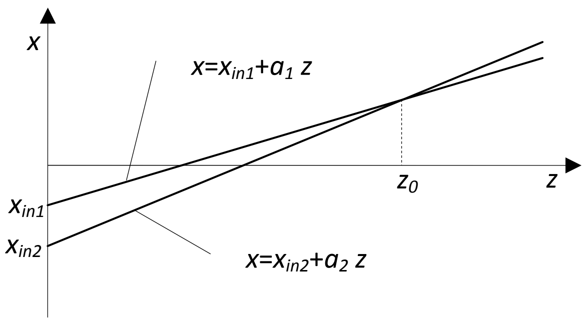

Figure 1: Illustration of the constant-local-quality method

$$x(z) = x_{in} + a\left(G, i_{fg}, q_i'', q_o''\right) \cdot z \qquad (1)$$

where x_{in} is the inlet subcooling, G is the mass flux, i_{fg} is the latent heat, q_i'' is the heat flux on the inner wall surface, q_o'' is the heat flux on the outer wall

The constant-local-quality method is based on an approach, in which the parameters are changed in such a way that the quality at a given selected point in the test section remains constant. The method is illustrated in the figure 1.

Experimental matrix set-up

The experimental matrix includes measurements of wall temperature distributions for single and two phase flow for both convective boiling and post-dryout heat transfer. The results will include as well pressure drop measurements for single phase in the test section in order to obtain the friction coefficient relationship and an expression for local pressure losses for spacers. Three possible situations are investigated: occurrence of the dryout on the inner rod, occurrence of the dryout on the outer tube and occurrence of the dryout on tube and rod simultaneously. Various mass fluxes, inlet sub-cooling values and system pressures will be studied.

During year 2008, the main effort was devoted to assemble the test section and set the instrumentation. The heated annulus (12.7x24.3x3650) mm is manufactured from Inconnel 600. The design pressure and temperature is 18.3 MPa and 700 ° C, respectively. The temperature of the annulus walls are recorded with 80 thermocouples, 40 located axially inside of the inner rod and 40 located as well axially outside of the outer tube. The other 8 thermocouples were distributed azimutally - 4 before and 4 after the last spacer- on the outer tube to capture the azimuthal distribution of the temperatures on the tube outer surface. In order to control conditions during the loop operation, temperatures at 6 locations must be measured on the continuous basis: inlet and outlet test section temperatures necessary for the heat balance calculation; coolant water temperature of the loop before the flow measurement system necessary to calculate viscosity, specific volume and the mass flux; water temperature after preheater to refine inlet conditions; water temperature before the pump to avoid cavitations; coolant temperature for the pump from secondary circuit. A calibration of flow meter and the heat balance for a single tube was performed. Currently the heat balance for annulus is performed and the test section is prepared for.

References

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

Research Leader: Associate Professor Henryk Anglart
PhD student: Carl Adamsson, 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 with focus on the influence of the power distribution.

Results during 2008

The project changed focus from Largrangian methods to more conventional film-flow analysis in sub-channels. Such models rely on empirical correlations for the deposition and entrainment rates but handle the power distribution in an essentially mechanistic manner. The MEFISTO code that has recently been developed by Westinghouse will be used for this purpose. Dryout measurements in full scale fuel assemblies with various power distributions from Westinghouse's FRIGG loop and the international OECD/NRC BFBT benchmark will be used.

Detailed study of simulated film-flow distributions can help to explain how the power distribution influences the dryout power in a rod bundle. If weaknesses in current film-flow models can be identified this will aid future research efforts in further improving these models.

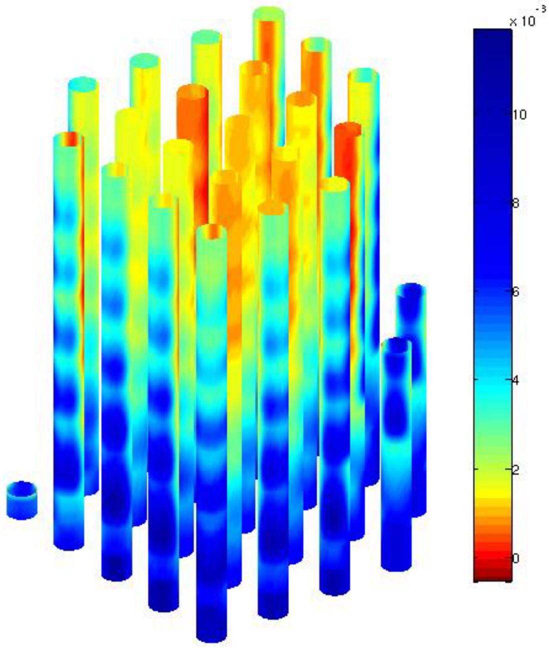

Figure 1: Example of simulated film-flow distribution in rod bundle

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

Research Leaders: Professor Tomas Lefvert and Dr. Pavel Kudinov
Scientist: Viet-Anh Phung, Division of Nuclear Power Safety, KTH, Stockholm

Background

Reactor thermal-hydraulic system computer code such as RELAP5 and TRAC play an important role in assessing safety analysis for nuclear plants, designing thermal-hydraulic experimental facilities, research reactors and commercial nuclear reactors. Main advantage of these codes is that they provide economical calculation tools in terms of computational time while giving reasonably good results for system steady-state and transients. For closure, the codes, however, employ correlations with empirical coefficients from different scale separate effect experiments. In addition, to simulate two-phase system behavior, a two-fluid model with time- and volume-averaged parameters of flows is used. The neglect of physical effects 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 system behavior such as strongly oscillating two-phase flows with rapid transitions between bubbly, slug and annular 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.

Goals of the project

Because the correct prediction of multiphase flow is important for designing and especially for safety analysis of BWR plants, a treatment of two-phase flow pattern in nuclear reactor thermal-hydraulic system code is necessary.

First, the work in this project will focus on investigating the capability of the system code to predict two-phase oscillatory flows. A number of experimental facilities with relevant data will be modeled using the system code.

Second, those oscillatory flows will be simulated and analyzed using three-dimensional Computational Fluid Dynamics (CFD) codes. Taking the advantage that CFD codes can calculate more precisely two-phase flows than system code does, result from CFD calculation will be used as another “experiment” in evaluating the system code.

Finally, based on understanding of sensitive parameters of the system code and operating region of thermal-hydraulic systems which strongly affect correct simulation result, a method for the treatment of two-phase flow pattern will be derived. The method will be developed and implemented into the system code for a better two-phase system simulation.

Organization

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

Methodology

In year 2008, sensitivity analysis was used for RELAP5 validation process for two-phase oscillatory flow in CIRCUS-I experiment. CIRCUS-I was a test loop, located at Delft University of Technology (the Netherlands), which was used to investigate two-phase flow instability at low pressure in natural circulation Boiling Water Reactor systems. The sensitivity analysis has quantified possible error range of experimental data, and identified the most influential boundary parameter affecting the calculation results.

The result showed that RELAP5 can predict well two-phase oscillatory flows having low frequency and low amplitude. For high frequency and high amplitude oscillation, it was found that RELAP5 modeling result has very non-linear behavior with changes in system pressure. As shown in figure 1, inlet flow rate significantly changed when slightly increasing system pressure from 1.1 to 1.13 bar.

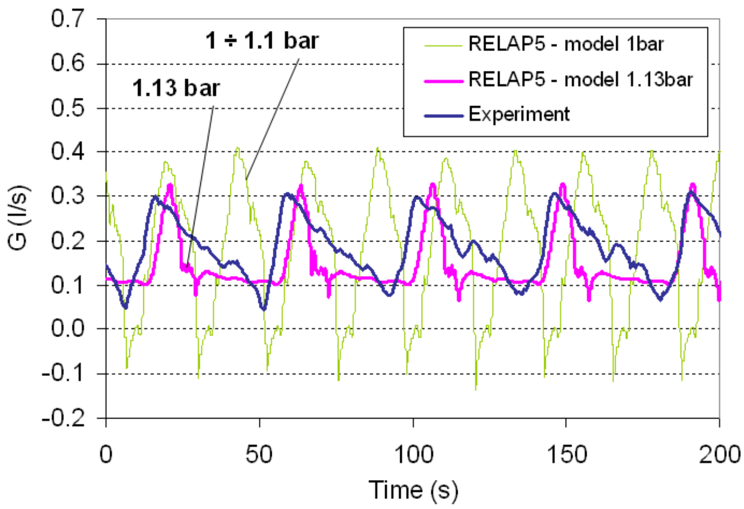

Figure 1: Inlet flow rate shows very non-linear behavior with changes in system pressure for the case having high frequency and high amplitude oscillation.

The work has been tackling with new and generic challenges for the modern code validation activities when there is a necessity of reusing previous valuable and expensive experimental data for code validation. A procedure was developed to utilize incomplete and uncertain experimental data for code validation purposes.

Modifications for future CIRCUS-IV measurement system and experimental procedure were suggested to TUDelft. The purpose of modifications is to reduced experimental uncertainty in the future and to make the experimental data more valuable for system code validation related to two-phase flow patterns.

Publications

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

Research Leader: Professor Nam Dinh
Scientist: Francesco Cadinu, 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 (STH) codes. RELAP5, TRACE, CATHARE, ATHLET are notable members of this class.

STH codes 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 into a “CFD subdomain” and a “STH subdomain” where the corresponding solvers are used. Matching conditions on the primitive variables or the fluxes, are imposed at the interface between different subdomains. While this is very intuitive, there are fundamental issues such as possible coupling instabilities 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; generally speaking, this is possible only if more information on the physics at the interface between different subdomains is available: for example if such interface is located in a region of fully developed flow.

However, the calculation of multidimensional temperature and velocity profiles, which requires a “domain decomposition” approach, is not the only instance where coupling between CFD and STH codes is needed.

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. 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.

This strategy will be applied to those phenomena that cannot be properly simulated by a STH code because of the lack of appropriate closures. 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.

The issues of the closure-on-demand procedure are explored considering, as a sample problem, the flow in an axisymmetrical sudden expansion subject to time-dependent inlet and outlet pressure boundary conditions.

Despite its simplicity, this is an example of a phenomenon which cannot be adequately simulated by a STH code. The missing closure is, in this case, the loss coefficient across the expansion, which exhibits a strong dependence on time, as it can be shown by a full transient CFD simulation.

Within the closure-on-demand framework, a combined CFD/STH analysis enables the STH code to predict correctly the system behavior at a much lower computational cost than the one required by a CFD simulation of the entire transient.

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.

Francesco Cadinu, Tomasz Kozlowski, Pavel Kudinov, *Study of Algorithmic Requirements for a System-to-CFD Couling Strategy*, XCFD4NRS, Grenoble, 10-12 September, 2008.

Francesco Cadinu, Tomasz Kozlowski, Pavel Kudinov, *A Closure-On-Demand" Approach to the Coupling of CFD and System Thermal-Hydraulic Codes*, NUTHOS-7, Seoul, October 5-9, 2008.

Chalmers University of Technology

Overview of Activities in 2008

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.

A main event of the year was a contract with E.ON and a related decision by Chalmers to start an international master's course in Nuclear Engineering in 2009.

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). 1 PhD exam and 1 licentiate during 2008.

Highlights of the year:

J. Rydberg received the Hansen award for outstanding contributions within nuclear engineering and fluid extraction.

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 modeling of all Swedish reactor units; competence centre for SSM; BWR instability research;
- nuclear measurement methods for material science, positron annihilation techniques;
- thorium fuel cycle; Gen-IV reactors, in particular 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.
- 7 PhD students (5 with SKC support, 1 jointly with Nucl. Chemistry).

Highlights of the year:

IAEA workshop at the ICTP, Trieste, on Neutron Fluctuations – main organizers and lecturers.

Courses

Package of full nuclear engineering courses was given the first time in the fall semester of 2007. The package is a specialization 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

A total of 12 students took the course package in nuclear engineering, and the number of students taking other specialized courses at Nuclear Chemistry was 26.

Nucl. Engng diploma work (master thesis): 30 pts Reading periods 3 and 4 (14 students)

Higher level graduate courses given: Actinide chemistry, 15 pts; Solvent extraction, 6 pts.

More information is available at:

www.nephy.chalmers.se

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 modeling 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 modeling 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 modeling 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

In 2008, a literature survey was performed in the fields of uncertainty and sensitivity analysis, focusing on three different approaches that have been applied to nuclear reactor calculations: the Uncertainty Methodology Based on Accuracy Extrapolation (UMAE), deterministic and statistical approaches, respectively. The UMAE was born in the nuclear safety analysis field and was developed by the University of Pisa, and constitutes the backbone of the CIAU methodology (Code (with the capability of) Internal Assessment of Uncertainty). Regarding the other approaches, the methodologies based on deterministic concepts first compute the local sensitivities by perturbing the physical models. Such sensitivities are thereafter used for carrying out uncertainty analysis. For instance, the Adjoint Sensitivity Analysis Procedure (ASAP) is one of the most time efficient deterministic methods which has been traditionally used in neutronic applications that make use of adjoint flux calculations [1]. By comparison to the deterministic approaches, statistical methods are relatively easy to develop and start with one of the several sensitivity analysis techniques that exist for this scope (i.e. scatter plots, regression analysis, etc.), removing the majority of unimportant parameters in order to perform an efficient uncertainty analysis. Sampling (or Monte Carlo) based methods are the most popular statistical procedures which have a number of desirable properties such as stratification over the range of each uncertain variable. Such methods are e.g. the Latin Hypercube Sampling (LHS), the Importance sampling, etc., which proved to have less variability and also to be more time efficient than random sampling since statistics converge faster to a fixed significance level. Both LHS and random sampling have been applied to BWRs thermal hydraulic calculations, the latest being the basis of uncertainty analysis through the use of tolerance limits developed by GRS, and also the basis of the CSAU method. Nevertheless, applications to neutronic calculations have been rather limited in the past by sampling based methods or even not considered by them when studying the effects of uncertainties in the basic nuclear data [2] (e.g. microscopic cross-sections, branching ratios, etc.) and therefore, further studies on the inclusion of such sources of uncertainty may constitute an important aim of the PhD project.

In parallel with the above literature survey, the POLCA-T code was used to model the BWR Full-Size Fine-Mesh Bundle Test (BFBT). The analysis of these test data is part of a NEA/NRC project that Chalmers joined. Chalmers has also joined the OECD/NEA expert group on Uncertainty Analysis in Modeling (UAM). Chalmers' goals for the BFBT analysis are threefold: to gain experience with POLCA-T, to develop a model of a Separate Effect Test on which different uncertainty/sensitivity methodologies can be tested, and to evaluate many of the code features in predicting steady-state and transient void fractions, pressure drops and critical powers under a wide range of system conditions. As can be appreciated in figure 1, the qualitative behaviour of the code is very good in transient analysis for both one dimensional void fraction and critical power predictions. The complete benchmark was submitted to the 2009 American Nuclear Society annual meeting [3].

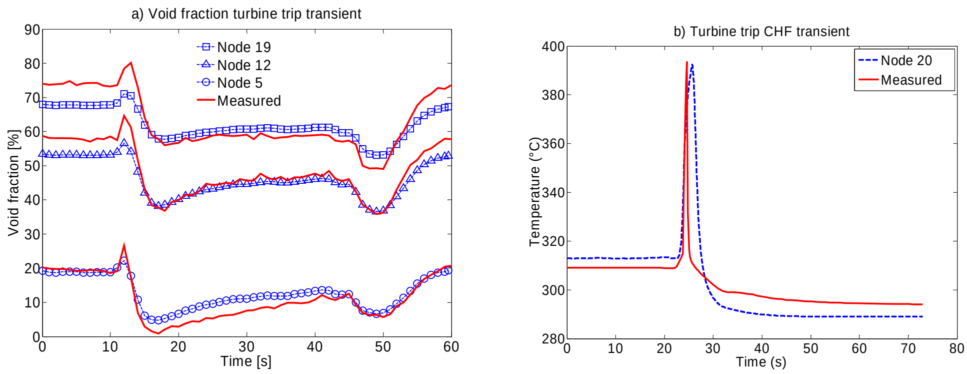

Figure 1: Void fraction and critical power turbine trip transient benchmark with POLCA-T

References

1) D. G. Cacuci. (1988), “The Forward and the Adjoint method of Sensitivity Analysis, Ch. 3, Uncertainty Analysis.” Ed. Y. Ronen, *CRC Press*, Florida, USA.

2) R. Macian, M. A. Zimmerman, R. Chawla (2007), “Statistical Uncertainty Analysis Applied to Fuel Depletion Calculations.” *Nuclear Science and* Technology, **44** (6), pp. 875-885.

3) A. Hernandez-Solis, P. Vinai, U. Bredolt. (2009), “An Assessment Study of the POLCA-T Code Based on NUPEC Data.” Submitted to *ANS 2009 Annual Meeting Transactions*.

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 the determination of the reactor transfer function [0]. 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. 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, Henrik Nylén, Ringhals, Farid Alavyoon, Forsmark, Christer Netterbrant, OKG, and Camilla Rotander, Westinghouse.

Methodology

In 2008, the semi-analytical 1-dimensional models used for determining the spatial dependence of the neutron noise were finalized and presented at PHYSOR in Interlaken. This study demonstrated that there were only negligible benefits in using higher order theories (such as P_1) for performing noise calculations on Swedish power plants. An article [0] will soon be submitted to Annals of Nuclear Energy, containing further description on the theories and modelling done. The article includes the 2-region 1-dimensional reactor using P_1 and diffusion theories, both for a LWR and a HWR. The calculations were performed numerically, after first being verified against semi-analytical calculations. The advantage of the numerical calculations is that the full reactor transfer function is obtained with one single inverse matrix for all possible types and locations of point-like perturbations corresponding to fast and/or thermal noise sources. In the case of the semi-analytical calculations, the reactor transfer function obtained will only be valid for a perturbation in one of the groups and at a certain spatial point. Thus, using the numerical way for solving the problem is much more efficient when comparing the two theories.

The next step of the project was also initiated, although not with immediate success, namely to use nodal methods for calculating the neutron noise. The nodal method chosen was the Analytical Nodal Method (ANM) which is a simple but yet powerful method. The programming language used for this was FORTRAN since the need for large matrices requires heavy computations. In addition, most codes related to the nuclear industry use FORTRAN. Furthermore, there are also extensive library sets with mathematical functions available in FORTRAN, for example LAPACK/ARPACK. These libraries are free of charge. They have been tested for many years and are thus reliable.

The ANM is being implemented for the full static neutronic 3-dimensional calculations. Thereafter, going to the noise calculations will be a relatively easy task as all equations are very similar to the static ones. The biggest differences are the inhomogeneous nature of the noise equations, for which complex arithmetic is also required.

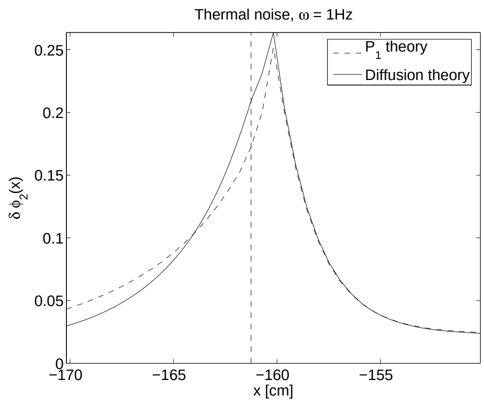

Figure 1: Amplitude of the thermal noise for a noise source located close to the core/reflector boundary as a result of a removal cross section perturbation with a frequency of 1 Hz.

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,” *Progress in Nuclear Energy*, **51**, pp. 67-81 (2008).

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

4) V. Larsson and C. Demazière, “Comparison of 2-group P1 and diffusion theories using neutron noise for 1-D 2-region systems”, to be submitted to *Annals of Nuclear Energy*.

Reactor diagnostics with advanced signal analysis (READS)

Research leader: Professor Imre Pázsit
Research scientist: Victor Dykin

Background

This project is a continuation of a previous PhD project, pursued by Carl Sunde whose project finished in April 2007. The goal is the development of new and more effective methods for the diagnostics of the reactor core and the primary circuit. The work consists of two ingredients. One is the development of models of the perturbations and the core transfer properties that are more advanced than the ones in use, by physical modeling and a qualitative and quantitative study of the properties of the system response to various perturbations. The second is the elaboration of powerful inversion methods, by which the searched diagnostic parameters can be unfolded from the measured noise, assuming that the relationship between the measured noise and the inducing perturbation has a functional form described by the theory.

It is in this step where the new advanced signal analysis methods come into play. These can take into account that the behaviour of the system is often non-stationary and/or non-linear, by replacing the FFT based spectral methods with wavelet analysis, and also invoking fractal and bifurcation analysis in the diagnostic step. The non-linearity has to be taken into account partly at the model construction stage, and partly at the inversion stage. In the latter case the non-linearity, and possible redundance in the measured data, can be handled by the use of artificial neural networks. There are in addition several other promising non-parametric methods emerging in the field which open new possibilities for extending the power of diagnostic methods.

Goals

The goals of the project is to give contributions for method development both regarding advancement of modeling the system and the various normal and abnormal regimes, and to apply them to solve relevant diagnostic problems in collaboration with the power plants. Phenomena which need further development of models and methods include BWR instability, vibrations of internals and the core-barrel in PWRs, and diagnostics of two-phase flow regimes and determination of two-phase flow parameters in BWRs. Regarding the new signal analysis methods, we plan to investigate several new methods that were reported to be very suitable for various diagnostic applications. These include fuzzy logics, fuzzy inference systems, principal component analysis etc. whose suitability for diagnostic applications will be explored. The test of the methods will be performed on both simulated signals as well as measurements taken in Swedish power plants.

Organisation

The reactor diagnostic group is headed by Prof. Imre Pázsit, who is also the leader of this SKC-project. The project started during the summer of 2008, so it is at the beginning only. Assoc. Prof. Christophe Demaziere, senior lecturer, acts as a deputy adviser. Other PhD students at the department, and some of our foreign collaborating partners, primarily Assoc. Prof. Tatiana Tambouratzis, also support the project.

The organisation of a reference group for the project is underway.

Methodology

The methodology should be clear from the background and goals described above. In brief, like traditional noise analysis, it 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, new signal processing and inversion methods will be tested and evaluated.

Activity in 2008

Since the student, Victor Dykin from Kaliningrad, did not have nuclear engineering subjects in his undergraduate education, in the fall period he took the minor program in Nuclear Engineering (I – IV) given by our Department and Nuclear Chemistry jointly. He also attended the IAEA workshop on Neutron Fluctuations, Reactor Noise, and their Applications in Nuclear Reactors", held at the International Centre for Theoretical Physics, Trieste, 22—26 September 2008, with our Department as the main organiser.

Victor Dykin has also started working on various diagnostic projects. First he got acquainted with the application of traditional FFT-based spectral and correlation analysis methods of evaluating measurements. Then he performed a study whose purpose what the analysis of BWR stability in a model driven by a driving force with a non-white power spectrum.

Summary of the Results

A model of BWR instability was investigated. The model is based on a previous work [1] which assumes the core being a second order system perturbation with a white noise driving source. Actually the existence of such a model lies behind of the concept of the decay ratio (DR) as a stability indicator, as calculated from the autocorrelation function (ACF) of detector signals. This model was extended to the case when the driving force is not a white noise, i.e. it has a coloured spectrum. This spectrum was taken from the reactivity effect of simple propagating perturbations, as used in traditional models of BWR in-core noise. Despite that this case is significantly more complicated than the case of white noise driving source, the analysis still can be performed analytically. One can then investigate the magnitude of the error in the estimation of the DR response of the core due to the fact that the evaluation is based on the white noise driving source. It was shown that for the case of the propagating perturbations serving as the driving source, the error in the estimate is small. However, one can construct certain types of coloured spectra, where the effect of the driving force spectrum on the results is significant. Among others, with fitting the sink structure of the perturbation to the resonance of the system response, a large deviation from the ACF of a second order system was found, showing large similarities to measured spectra. This shows that the model treated has a certain relevance for the interpretation of stability measurements.

These results are illustrated in the Figure below. The figure on the left shows the power spectrum of the system transfer function (which is equal to the system response with a white driving force), the power spectrum of the driving source showing the known sink-structure of the power spectrum of a propagating perturbation, and finally the product, which is equal to the autospectrum of the system response to non-white perturbation. The right figure shows the ACF calculated from the APSD of the system response, showing a clear deviation from the response of a second order system. A determination of the DR by a curve fitting to the "measured" ACF gives, on the other hand, still a reasonable estimate of the true DR, if the parts of the ACF for small time delay values are omitted.

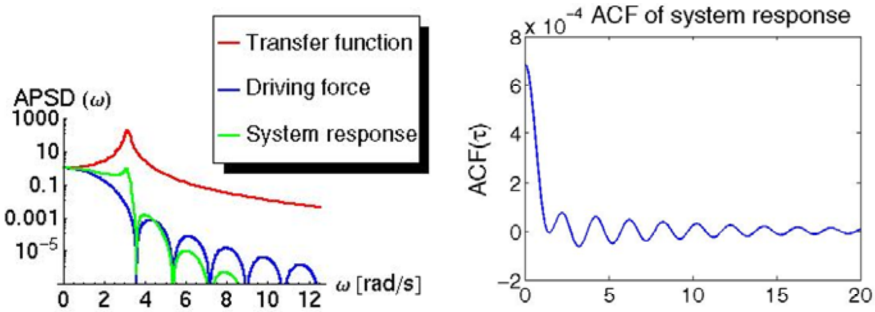

Figure 1: Left: APSD of the system transfer, the driving force and the resulting system response. Right: the ACF of the system response to the non-white driving source, showing a significant deviation from a traditional second order system.

References

1) I. Pázsit, "Determination of reactor stability in case of dual oscillations". *Ann. nucl. Energy* **22**, 377 - 387 (1995)

Neutron fluctuations in zero power systems and power reactors

Research Leader: Professor Imre Pázsit
Scientist: PhD Student Anders Jonsson

Background

Neutron fluctuations in multiplying systems can be divided into two classes which differ from each other what regards their origin, mathematical treatment and domain of dominance. One is the fluctuations in zero power systems with a constant material composition, where the reason of the non-trivial (non-Poisson) character of the neutron statistics is due to the branching (fission) process. Such processes dominate at low reactor power and are described by the master equation technique. The other area is high power systems, where the origin of the fluctuations is due to the temporal changes of reactor material (boiling, vibrations etc). Such neutron fluctuations dominate in high power systems, and are described by the Langevin equation technique.

Since the theory of zero power noise for reactivity determination was elaborated long ago, during a long period no new work was performed in the area. However, due to recent developments it became interesting again. Partly, measurement of reactivity during start-up has got increased attention and new methods were suggested without a full theory behind (e.g. the Cf-252 method). Further, new systems such as accelerator driven systems (ADS) raised new challenging problems in the area. The zero power noise methods proved to be applicable even in material control and accounting in nuclear safeguards. The intermediate case of medium power systems, where zero reactor noise and power reactor noise co-exist, became also a subject of interest. Last, but not least, several of the planned future systems, such as some of the Gen-IV reactors, raise challenging questions of stochastic nature (random core composition etc). These require advanced stochastic modeling, both for a deeper understanding of the physics of the systems and for studying their noise diagnostics. This is also planned in the project.

Goals

Research is planned be conducted in the above described areas. There are many open questions in the applicability of the zero power noise methods, i.e. the master equation technique, for systems with temporally varying composition. Several of these, such as the use of the zero-power reactor methods for reactivity measurement in zero power systems, will be investigated in the project. Stochastic problems, kinetics and dynamics in systems that arise in the development of new reactor concepts, such as those containing a random distribution of fissile material as well as non-stationary fuel, will also be investigated.

Organisation

The research in neutron fluctuations and stochastic theory is led by Prof. Imre Pázsit, who is also the leader of this SKC-project. The project started during the summer of 2008, so it is at the beginning only. A senior PhD student, Andreas Enqvist, working with zero power noise methods in nuclear safeguards, is supporting the project.

The organisation of a reference group for the project is underway.

Methodology

As mentioned, the methodology of zero power noise is based on the master equation, or Chapman-Kolmogorov equation. Forward and backward forms of these equations will be formulated and solved, as well as their validity will be investigated. Special emphasis is on the use of the results for novel reactivity measurement techniques. For the case of systems in temporally and/or spatially varying random character, the Langevin equation technique will also be used.

Activity in 2008

Since the student, Anders Jonsson, did not have nuclear engineering subjects in his undergraduate education, in the fall period he took the minor program in Nuclear Engineering (I – IV) given by our Department and Nuclear Chemistry jointly. He also attended the IAEA workshop on “Neutron Fluctuations, Reactor Noise, and their Applications in Nuclear Reactors”, held at the International Centre for Theoretical Physics, Trieste, 22—26 September 2008, with our Department as the main organiser.

A smaller project was also executed which had two parts. One was the investigation how the traditional Feynman-alpha formula changes in the case when instead of a capture detector, a fission chamber is used, hence giving rise to fission neutrons, extending the subcritical chains in which detection is performed. The other part was the investigation of the quantitative error of the backward equation solution when applied to time-varying system. For such systems an assumption made in the traditional cases (which is necessary to achieve closure) is not valid, hence the results are in error. The goal was to quantify the magnitude of this error.

Summary of the Results

Regarding the Feynman-alpha formula with the fission chamber, it turned out that the time-dependence of the formula is the same as for the case of the capture detector, except that the presence of the detectors slightly influences the criticality of the system in opposite ways. Also, even for the same total reactivity (alpha value), the asymptotic amplitude (for long measuring times) of the variance to mean is higher for the fission detector. In the course of the work the variance to the mean (Feynman-alpha) formula for one single initial particle was also calculated, for methodological reasons. There an interesting qualitative difference was found between the capture detector and the fission detector, namely that the relative variance of the detector counts, starting from its initial value of unity, starts decreasing (becomes sub-Poisson) for a short time, before raising back too above unity and getting to the same asymptotic track as the one with a fission detector, which does not have a local minimum for short measurement times. This is illustrated in the Figure below. This explains the mechanism behind the fact that the asymptotic variance to mean is higher for the fission detector. Namely, an early detection by a capture detector kills all possible progeny of the initiating neutron, leading hence to a lower than unity relative variance, whereas a detection with a fission detector does not terminate the chain.

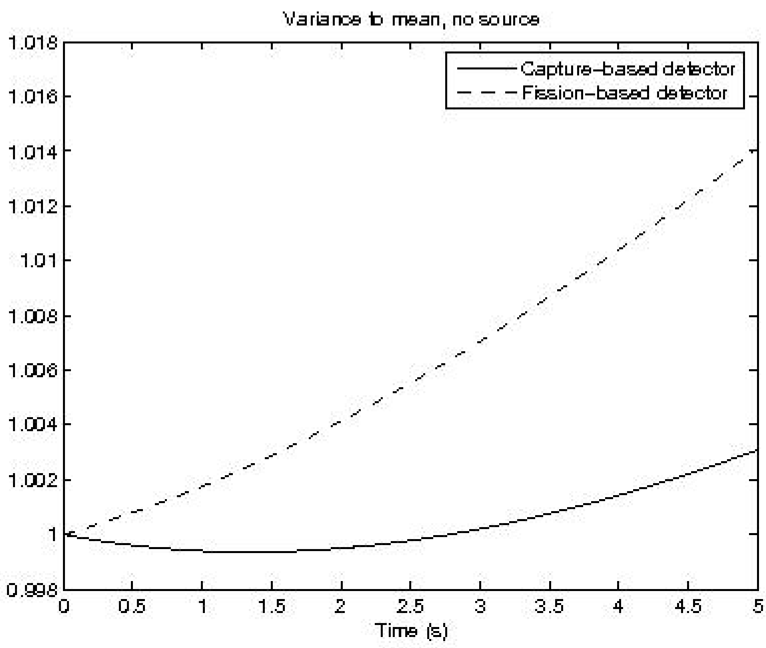

Figure 1: Dependence of the relative variance (Feynman-alpha formula) of the detector counts due to one single initiating neutron, for a capture and a fission detector in a supercritical system. It is seen that for short measurements times, the relative variance has a local minimum in the case of a capture detector.

Ruthenium chemistry in the reactor containment during a severe accident

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, Gothenburg
[2] Ringhals, Väröbacka

Background

Ruthenium is a semi-volatile fission product, which is formed during normal operations in a nuclear power plant. Formed ruthenium metal in the core is slight volatile and consequently not released from the vessel. However, during a severe accident with air-ingress into the reactor vessel, volatile ruthenium oxides can be released and constitute a threat to environment. The only stable gaseous ruthenium oxide below 1000°C is ruthenium tetroxide, and thus the only gaseous ruthenium specie that can reach the containment and be released to the environment. Thus it is important to investigate the behaviour of RuO_4 in a reactor containment under severe accident conditions, among other things interactions with different surfaces and the water phases.

Goals:

To investigate the speciation of ruthenium deposits on different metals, like aluminium, cupper and zinc, present in a boiling water reactor containment. We wanted also to investigate the atmosphere's influence on the speciation of the ruthenium deposits.

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.

The set-up is a simple glass bottle, with sample holders in glass for placing of the metals. An experiment was started by placing one or three metals, depending on which experiments, on the sample holders. The atmosphere in system was nitrogen (humid or dry) or air. RuO4(cr) attached to the surface of a sample vial in glass was introduced through an opening in the lid of the glass bottle. The ruthenium tetroxide crystals were let to sublimate and vaporize, so clean gaseous RuO4 interacted with the metals. The temperature in the set-up was room temperature, ~25°C.

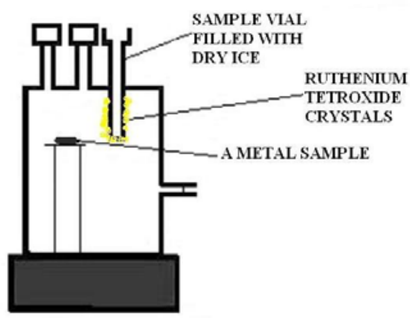

Figure 1: The experimental set-up, used for the RuO_4 (g) – metal interactions

The ruthenium deposited metals were analyzed with different speciation methods, like ESCA (Electron Spectroscopy for Chemical Analysis), SEM (Scanning Electron Microscopy) and XRD (X-Ray Diffraction). The ruthenium quantity on the metals was determined by dissolution of the deposits with a 0.2 M NaOH solution with 2 g/l K2S2O8. The solutions were analyzed with ICP-MS (Inductively Coupled Plasma-Mass Spectroscopy).

Results

An ESCA spectrum from one of the analyzed ruthenium deposited metals can be seen in figure 2. The binding energy of the surface electrons can be related to which element and its speciation appearing on the surface. From the positions of the main ruthenium peaks in the spectra at 281.8 eV and 285.9 eV, the conclusion can be made that the ruthenium deposit consists of $RuO_2 \cdot xH_2O$. The remainder peaks in the spectrum is carbon contamination peaks (284.5 eV and 288.7 eV) and the peaks at 283 eV and 287 eV are formed due to an unscreened final state of RuO_2, which is extensively described by Rochefort et al. [1]. As can be seen in table 1, either the atmosphere or the kind of metal has no influence of the speciation of the ruthenium deposit.

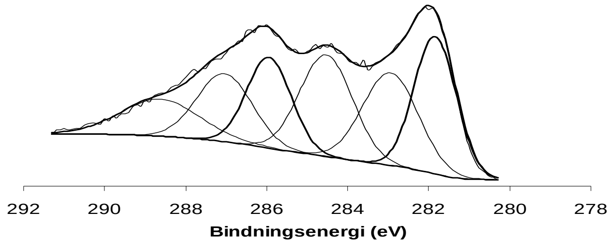

Figure 2: A magnification of an ESCA spectrum from the measurements.

Table 1: Binding energy (eV) of the main ruthenium peak Ru 3d5/2 for the experiment

Metal	Nitrogen	Air	Humid Nitrogen	Average	±
Al	281.7	281.5	281.6	281.6	0.1
Cu	281.8	281.9	282.0	281.9	0.1
Zn	282.0	282.1	281.8	282.0	0.2

Some of the RuO_4(g) interacted metal samples were heated in an oven to 300°C. The ESCA analyses of the heated samples showed that RuO_2 was presented on the metal surfaces. The XRD analyses showed also that RuO_2 was formed when RuO_4 was stained over the different metal surfaces.

Pictures (Figure 3) from the SEM analyses illustrated that the ruthenium deposition was more significant at the copper and zinc surface compared to aluminium surfaces. This was also confirmed by the ICP-MS measurements of dissolved ruthenium deposits.

The results from this work together with previous works show that release of ruthenium from a BWR during a severe accident should not constitute a large threat for the environment.

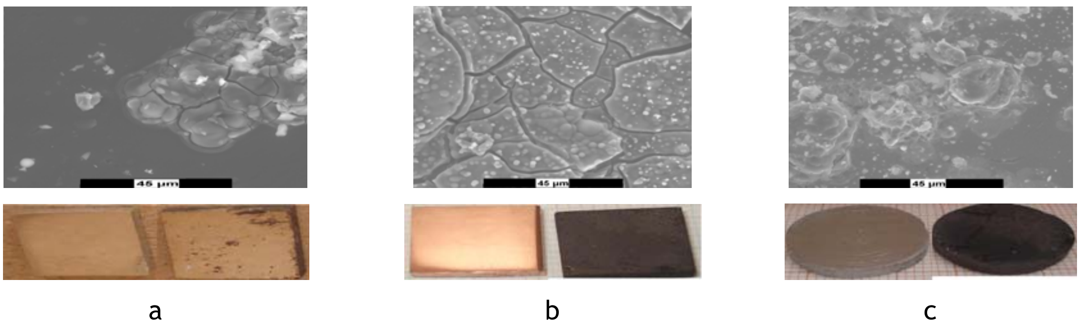

Figure 3: SEM pictures (above) and ordinary photographs (below) of the metals; a) aluminium b) copper and c) zinc. The ordinary photographs show to the left an untreated metal sample and to the right a RuO_4(g) stained sample.

Publications:

1) J. Holm, C. Ekberg and H. Glänneskog, Deposition of RuO4 on various surfaces in a nuclear reactor containment, submitted for reviewing in Journal of Nuclear Materials.

Reference

2) D. Rochefort, P. Dabo, D. Guay, P.M.A. Sherwood, Electrochim. Acta. 48 (2003) 4245-4252.

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 master engineering programs and "högskoleingenjörsprogrammen". In addition, education is performed on commission within the framework of an agreement with the Nuclear Safety and Training Centre (KSU).

Previously, in the Division of Applied Nuclear Physics, the research with relevance for the nuclear power industry was performed within two research groups. These two research groups have now merged together in order to gain administrative and operative advantages. It is believed that this merging will augment efficient use of available resources.

In addition, there is an on-going process that aims at merging the Dept. of Physics and Astronomy with the Dept. of Physics and Materials into one department. The resulting department is expected to be a strong actor on the nuclear technology scene as it will cover a broad range of research activities of interest for the nuclear industry.

While awaiting for such a department there are well-evolved discussions between the Division of Applied Nuclear Physics with the material research and chemistry in Uppsala in order to create a "nuclear platform" at Ångström laboratory in order to enhance the ability to serve a broad set of research tasks and coordinate these tasks.

At the end of 2008, two researchers were employed: Henrik Sjöstrand was employed after his PhD exam. Henrik will initially join several projects as well as conducting undergraduate teaching. Cecilia Gustafsson will start her work 1st of March 2009 after several years' employment at Forsmark as a core physicist. Cecilia's focus areas will be nuclear data and teaching.

Research

Research is being performed in the following areas:

- The technical aspects of international safeguards.

- Development of advanced measuring techniques based on detection of ionising radiation. These methods are intended for validation of core simulators and determination of fuel parameters of relevance for encapsulation and final storage of spent nuclear fuel.

- 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. One aim of this research is the development of new equipment and methods for fission reactor diagnostics as well as technologies for future reactor concepts.

- Theoretical studies of the neutronics in sub-critical cores and development of a high-intensity neutron source for use in such cores. This work has also implication for some concepts within GenIV.

Education

The nuclear renaissance has spawned a huge increase in student interest for courses within the field of nuclear engineering, approximately 160 students took part in master engineering. Consequently, in addition to the existing courses new courses are being developed within the master engineering programmes as well as in collaboration with KSU.

The number of course weeks for industry participants has approximately doubled to 30 weeks since the previous year.

The following courses have been given within the framework of KSU. A significant part of the teaching was conducted at OKG and Ringhals power plants.

Kärnkraftteknik H1	12 pts
Tillämpad reaktorfysik	7.5 pts
Fördjupad strålningsutbildning FS1,	2 weeks
Fördjupad strålningsutbildning FS2,	1 week
Termohydraulik,	1 week
Aktivitetsmätning med germaniumdetektor,	1 week
Värme- och strömningslära	1 week
Fortbildning av KSU-personal	1 week

Participants of one of the H1 courses together with some teachers.

During 2008, discussions were initiated between KSU and Uppsala University with the aim of introducing a third year, devoted to nuclear engineering, on the “Högskoleingenjörsprogram”. The idea behind this concept is to allow students from various engineering programmes, e.g. mechanical engineering and electrical engineering, all over Sweden to apply for this unique education.

The program is planned to cover all non-site specific issues and it is believed that such an education would 1) increase the volume of employable people to the nuclear industry and 2) decrease the industry’s total training cost. The latter is due to the fact that almost a year of training, that otherwise would charge each power plant individually, would be centrally funded. The discussions have been very fruitful and both parties are, in principle, positive to the idea.

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

Theses presented during 2008

The following theses, with relevance for the nuclear industry, have been presented during 2008:

Henrik Sjöstrand: “Neutron Spectroscopy - Instrumentation and Methods for Fusion Plasmas” (Ph.D.)

Angelica Öhrn: “Neutron scattering at 96 MeV” (Ph.D.)

John Loberg: “Neutron-detection based void monitoring in boiling water reactors” (Lic.)

Anni Fritzell, "System Aspects on Safeguards for the Back-End of the Swedish Nuclear Fuel Cycle" (Lic.)

Patrik Tegelberg: “Assessment of TRACE V5.0 using the NRC/General Electric level swell problem” (M.Sc.)

Robert Larsson: “Methods for determining nuclide specific levels of contamination on waste from nuclear facilities” (M.Sc.)

Lisa Bladh: “Thermal hydraulic modelling of Forsmark 1 NPP in TRACE – validation versus the 25th of July 2006 plant transient” (M.Sc.)

Tobias Winblad von Walter: “Comparison of nuclear fuels for LWR applications” (M.Sc.)

Marcus Molander, "Quantitative image analysis of Cherenkov light from nuclear fuel assemblies" (M.Sc)

Neutron-detection based void monitoring in boiling water reactors

Research leaders: PhD Michael Österlund[1], Adj. Prof. Klaes-Håkan Bejmer[2], Prof. Jan Blomgren[1]
Scientist: PhD student John Loberg[1]

[1]Division of applied nuclear physics, Uppsala University
[2] Vattenfall AB

Background

In Boiling Water Reactors (BWR) the uranium fuel is surrounded by water that is brought to boiling within the reactor core by the heat released due to fission. The void fraction, i.e., the volume part of the steam is an important parameter for BWR operation. Too high void fraction can lead to dry-out, which may cause severe fuel damages. The void fraction is also important for the moderation of neutrons and thus the power level of the reactor and optimized fuel utilization.

Presently there is no technique for measuring the void fraction or its distribution in the reactor core so the void fraction has to be calculated. Calculating local void fractions with high accuracy is very difficult which leads to some conservatism in the operation of a BWR. If the void fraction could be measured with better accuracy than the calculated void fraction, both safety and economical features could be improved.

Goals of the project

This project presents a new approach to in-core void fraction determination by monitoring of neutrons of different energies; as the void fraction determines the efficiency of the moderation, the relation between the thermal and fast neutron flux will reflect the moderation. Out of the ration between the simultaneously measured thermal and fast neutron flux, the void fraction can be determined. Also, because the thermal and fast neutron fluxes changes with void fraction changes in opposite direction, a ratio of the two signals enhances the change in moderation density. There is also a possibility to detect channel bow due to the different sensitivity of the thermal and fast neutron fluxes to local changes in moderation.

Organization

The work is performed by PhD student John Loberg under the supervision of Dr. Michael Österlund and Adj. Prof. Klaes-Håkan Bejmer. Prof. Jan Blomgren is involved as assistant supervisor for the project.

Methodology

A void prediction model will be developed that will be able to predict absolute void fractions out of the ratio of the thermal and fast neutron fluxes. The model will be based on 2D and 3D calculations using commercial codes such as CASMO and POLCA, which may be verified in experimental tests further on. Another model is also under consideration that will indicate the presence of channel bow in the core.

Results

The void correlation, i.e., the correlation between the void fraction and the ratio of the thermal and fast neutron fluxes has proven to be a strong linear correlation. The influence of normal reactor parameters such as burnable asborbers, burn up and control rods on the correlation is very small, except for channel bow that biases linear void correlation strongly, see Fig 1. This bias however pose a possibility to separate channel bowing from void fraction changes.

The results from the 3D calculations looks very promising. The void prediction model is able to predict the absolute void fraction in 792 positions in the core with an uncertainy of ±1.5%, when taking burnup and control rods into account. The model is however not able to account for channel bow, because the presence of channel bow often is unknown. Nevertheless, if channel bow is present it will be visible as a non-linear response in the void prediction model due to the fact that channel bow has a parabolic shape and gradually will decrease/increase the moderation close to the detector, while the void prediction model presumes a linear increase in moderation. This makes it possible to detect wheter channel bow is present or not, at least if it is bigger than 2 mm.

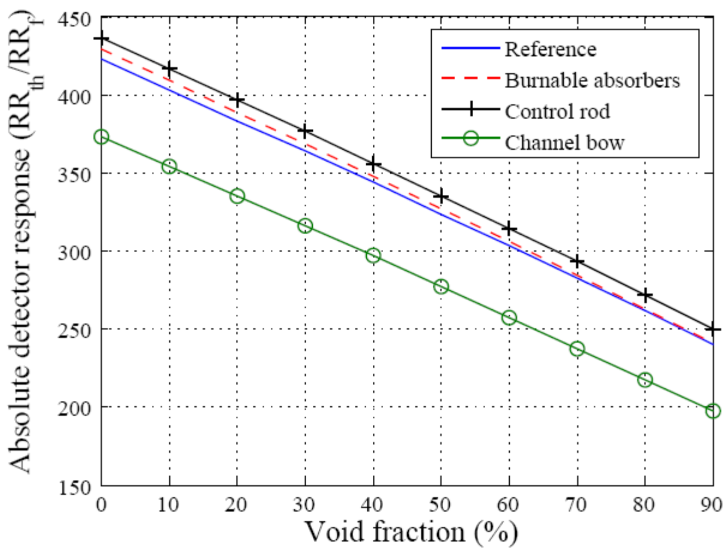

Figure 1: The ratio of the thermal and fast neutron flux plotted against the void fraction for different reactor scenarios. Channel bow, 4 mm in this calculation, affects the ratio significantly.

The results from the 3D calculations looks very promising. The void prediction model is able to predict the absolute void fraction in 792 positions in the core with an uncertainy of ±1.5%, when taking burnup and control rods into account. The model is however not able to account for channel bow, because the presence of channel bow often is unknown. Nevertheless, if channel bow is present it will be visible as a non-linear response in the void prediction model due to the fact that channel bow has a parabolic shape and gradually will decrease/increase the moderation close to the detector, while the void prediction model presumes a linear increase in moderation. This makes it possible to detect wheter channel bow is present or not, at least if it is bigger than 2 mm.

References

1) Neutron-detection based monitoring of void effects in boiling water reactors , konferensbidrag, International Symposium on Reactor Dosimetry, ISRD13, Holland, May 2008.

2) Neutron-detection based monitoring of void effects in boiling water reactors, konferensbidrag, PHYSOR08, Interlaaken, Schweiz, Sep 2008.

3) Neutron-detection based void monitoring in boiling water reactors, Licentiate thesis, Oktober, 2008.

SKC Financials in 2008

The following table summarises the SKC financials for 2008

Received from Financing parties		15 850 619 kr
Saved from previous year		5 406 969 kr
Sum of income 2008		**21 257 588 kr**
KTH	8 588 733 kr	
Chalmers	6 500 061 kr	
Uppsala University	3 550 124 kr	
SKC administration	1 724 994 kr	
Sum of costs in 2008	**20 363 912 kr**	
Transfer to 2009		**893 676 kr**

The contributions from the financing organizations are split as follows:

SKI/SSM	33,33%
Westinghouse	20%
Forsmark	14%
Ringhals	19,67%
OKG	14%

The unused funding from previous year will be used in 2009.

Winners of Sigvard Eklund's Price in 2008

Olivia Roth, KTH, was awarded the price for the best PhD thesis, which has the title ”Redox Chemistry in Radiation Induced Dissolution of Spent Nuclear Fuel – from elementary reactions to predictive modeling”. Her work is characterized by professor Mats Jonsson:

“The PhD-thesis by Olivia Roth covers a wide range of important aspects in the field of Spent Nuclear Fuel Dissolution. Novel mechanistic studies on the chemical processes involved in radiation induced oxidative dissolution of the UO2 matrix and the inhibition thereof are presented. The results of these studies are used in predictive modeling of spent nuclear fuel dissolution under deep repository conditions."

Andreas Carlson, KTH, was awarded the price for the best master thesis, which has the title ”Numerical Simulations of Slug Flow in a Micro Channel”. His work is characterized by the review committee:

“Cooling of a melted core includes multi-phase flow in micro-channels. The thesis demonstrates an exceptional breadth and depth in understanding such multi-phase flow phenomena resulting in improvements to state-of-the art codes, which are then applied to show unique physical results”

SKC AlbaNova / KTH

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