Deterministic Safety Analysis Group

Review for the period of 2008 – 2010

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**Background**

SSM has signed an agreement with KTH and Chalmers for a long-term commitment within the deterministic safety analysis (transient and severe accident analysis). The objective for this co-operation is to promote the national competence and to establish groups, whom can support SSM to perform safety analysis, reviews and inquiries, as well as participate in international projects and working groups within these areas.

According to the agreement, the analysis groups at KTH and Chalmers have worked with R&D within the following areas:

- plant analysis;
- evaluation and contribution to international projects;
- training and education of SSM’s personnel.

The first phase of the Technical Support Organization for the Deterministic Safety Analysis (TSO-DSA) agreement covered the period of 2008 – 2010. The objective of this document is to describe the work performed within the TSO-DSA and summarize the TSO-DSA achievements for the initial 3-year period.

In order to facilitate evaluation of the TSO-DSA achievements, the document summarizes following information:

- a short description of the work so far, including the results, and what the work has meant for KTH and the national competence;
- a list of the persons which have been involved, including their contribution;
- a list of publications (articles, conference papers, internal reports, etc);
- a list of reports which are aimed for publication within SSM report series.

1. **Selection of the simulation tools**

The U.S. NRC codes RELAP5/PARCS, TRACE/PARCS and MELCORE were studied to determine fusibility of performing safety analysis and transient calculation.

RELAP5 is a light water reactor (LWR) transient analysis code, originally developed at the Idaho National Engineering Laboratory (INEL) for the U.S. NRC. RELAP5 is a highly generic code that, in addition to calculating the behavior of a reactor coolant system during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and nonnuclear systems involving mixtures of steam, water, noncondensable and solute. Specific applications include simulations of transients in LWR systems such as loss of coolant, anticipated transients without scram (ATWS) and operational transients such as loss of feedwater, loss of offsite power, station blackout and turbine trip.

TRACE is a best-estimate transient analysis code based on consolidation of TRAC-P, TRAC-B, RELAP5 and RAMONA programs, capable of modeling PWR, BWR, and scaled thermal-hydraulics test facilities. It was developed by recoding the TRAC-P algorithms to take
advantage of the advanced features available in the Fortran 90 programming language, while conserving the computational models available in the original code. It is currently NRC’s primary best-estimate tool for predictions of postulated accidents in LWRs.

PARCS is a three-dimensional reactor core simulator developed at Purdue University. It solves the steady-state and time-dependent multi-group neutron diffusion and SP3 transport equations to predict the dynamic response of the reactor to reactivity perturbations such as control rod movements, boron concentration or changes in the temperature/fluid conditions in the reactor core. The code is applicable to both PWR and BWR cores loaded with either rectangular or hexagonal fuel assemblies. PARCS is coupled directly to the U.S. NRC thermal-hydraulics systems codes TRACE and RELAP5 using a message passing interface.

MELCOR is a U.S. NRC computer code for severe accident analysis of light water reactors (LWRs). It is viewed as a state-of-the-art (best-estimate) tool for severe accident analysis and source term calculation. The main objective of the MELCOR activities at KTH is to build expertise on using the code for safety analysis of Swedish NPPs, with the near-term goal of supporting SSM’s decision-making for power uprates of NPPs.

The main outcome was that it is possible to use above mentioned codes, however, comprehensive access to information from the power plant is necessary. RELAP5/PARCS and TRACE/PARCS were recommended for transient analysis and MELCOR was recommended for severe accident analysis.

However, creation of the cross-section library in the format accepted by PARCS code from CASMO4 output showed to be very time consuming due to lack of information necessary for the conversion of the cross-sections to format accepted by PARCS code. This obstacle was overcome by collaboration with Purdue University and PARCS development team (currently at University of Michigan).

2. Plant analysis activities

The plant analysis activities comprise of the power uprate and plant modernization support, and the analysis of past plant events.

The aim of the power uprate and plant modernization support is to perform, independent from the utility, transient safety analysis using coupled Thermal-Hydraulics and Neutron Kinetic (TH/NK) codes and severe accident analysis to support SSM’s review of the updated Safety Analysis Report (SAR) submitted by the utilities as part of the license application.

The objective of the analysis of past plant events is to support SSM’s review of plant events by using independent analysis with advanced simulation codes and to contribute to better understanding of Swedish reactors behavior and specific problems. Analysis of past plant events is also used to validate the tools (codes) and models (code input) used for safety analysis.

Tomasz Kozlowski, Sean Roshan and Andrej Kubarev are responsible for the safety analysis simulation work using the RELAP5/PARCS and TRACE/PARCS codes. Their main contributions are development and/or adaptation of the input decks for various NPPs, performing the simulation calculations and writing technical reports for the safety analysis.
Weimin Ma is responsible for the severe accident simulation work using the MELCOR code. His main contributions are development and/or adaptation of the input decks for various NPPs, performing the simulation calculations and writing technical reports for the safety analysis.

KTH was responsible for the transient analysis of Oskarshamn-3, Oskarshamn-2, Forsmark-3 and Ringhals-1. KTH was also responsible for severe accident analysis of Oskarshamn-3, Oskarshamn-2 and Ringhals-3.

### 2.1 Oskarshamn-3 activities

#### 2.1.1 Transient analysis at uprated power

The objective of this work was to provide SSM with support in evaluating O3 transients at uprated power level.

RELAP5 and PARCS model were developed, coupling between these codes was created and a steady-state at the plant operating conditions was achieved. The model was validated against O3 2002 turbine trip event.

A great effort has been put into developing and maintaining the inputs necessary for this task, the following tasks were a part of that effort:

- study of the spatial coupling between the neutronic and thermal-hydraulic codes;
- study of the number of thermal-hydraulics channels required for each transient;
- study of the effect of K-loss on the core stability;

The validated code package was used for transient safety analysis. Analysis was performed of following transients at current and uprated power conditions:

- turbine trip;
- CR drop;
- FW loss of preheaters;
- steam line break.

The work with the developed code package continues in order to have an up to date and accurate tool for future needs of SSM. The model can be modified to correspond to the diverse needs that might arise in the future.

The work has been carried out by Sean Roshan, Andrej Kubarev and Tomasz Kozlowski.

#### 2.1.2 Power uprate start-up test analyses

The objective of this work was to support SSM in evaluation of startup tests at uprated power level. The models developed during O3 power uprate project are being used for O3 pre- and post-uprate tests calculations.
The power uprate startup testing test plans have been reviewed and a number of tests have been selected in collaboration with SSM. After adopting the models to the current plant data, these tests were simulated blindly. KTH performed pre-analysis of the following power uprate startup tests:

- Test 604, stability measurement
- Test 663, loss of one feedwater pump
- Test 680, SCRAM from full power

KTH will perform post-analysis of the power uprate startup tests when the results become available. The post test simulation which, will be performed after the actual tests have been performed will provide us with valuable information about the plant vs. code status.

The work has been carried out by Andrej Kubarev and Sean Roshan.

2.1.3 Severe Accident analysis

The objective of this work was to perform MELCOR analysis for severe accident scenarios of the Oskarshamn-3 BWR related to power uprate. The results from the independent analysis can be used to compare with the results of MAAP code analysis submitted by NPPs, so as to reduce the uncertainties in quantification of severe accident risk. The most dangerous scenarios (accidents such as station blackout and station blackout plus loss of coolant) were chosen in the simulations using MELCOR 1.8.5. Generally speaking, the power uprate do not explicitly increase the severe accident risk, in terms of source term release, and maximum of pressure and thermal loads. Supplementary studies using of MELCOR higher versions (1.8.6 or 2.1) are under way. The new versions of MELCOR are supposed to increase the simulation fidelity of the in-vessel melt behavior and progression.

The MELCOR 1.8.5 code was also used to investigate the direct containment heating (DCH) of Oskarshamn-3 BWR. For DCH estimation, the MELCOR simulation shows that in a conservative case (without actuation of ADS; 35% of the ejected melt is dispersed in the atmosphere of the lower drywell and the upper drywell), the pressure peak can reach 5 bar at vessel failure, which triggers the containment venting but does not fail the containment, and the peak temperature in the containment at vessel failure is around 550°C which just stands a few seconds.

The work has been carried out by Weimin Ma.

2.1.4 Past event analysis

On June 7, 2002, Oskarshamn-3 suffered from a SCRAM due to SS12 (high power in the reactor pressure vessel), SS5 (high pressure in the reactor pressure vessel) and IM2 (high water level in the reactor pressure vessel). This happened during a test of the safety equipments of the unit. Failure in some systems caused a turbine trip (TS) while the bypass valves did not manage to take over the plants pressure control in time. This was most probably due to leakage of hydraulic oil and loosing the pressure required for regulating valve 421VB4.
The event was simulated in order to validate the code package and O3 model developed for O3 transient analysis and power uprate. The result achieved showed good agreement with the recorded data during the event.

The work has been carried out by Sean Roshan.

### 2.2 Oskarshamn-2 activities

#### 2.2.1 Transient analyses

Oskarshamn-2 will shortly undergo an extended power uprate to increase the plants power to 135%. The objective of this task was to support SSM in review of the O2 safety analysis for operation at the uprated power and evaluating O2 transients at uprated power level. The main question to address was the core cooling during a large break LOCA in the bottom of the pressure vessel (at the recirculation pump line) at uprated power.

SSM has required best-estimate capabilities to respond to the forthcoming needs related to review of the new plant conditions. The work consists of development and validation / verification of coupled code input files using PARCS and TRACE codes. This package will later be used for safety analysis of the power plant. The project includes:

- Development of PARCS input file for Oskarshamn-2
- Development of TRACE input file for Oskarshamn-2
- Coupling between PARCS and TRACE O2 model
- Validation / verification of the model

KTH, in collaboration with University of Michigan and Pennsylvania State University, prepared steady-state TRACE/PARCS model, and the model was upgraded by incorporating the following changes to transient conditions:

- 4 independent steam lines, steam valves and turbine connections;
- 4 steam bypass lines;
- 4 independent feedwater lines, feedwater valves and feedwater control logic;
- 4 independent recirculation loops;
- new ADS with 4 different set of valves;
- new auxiliary feedwater system and ECCS;
- new vessel nodalization;
- modified control system.

An analysis of the core cooling during large rupture in the recirculation pump line (LB-LOCA) will be performed later in 2010. The purpose of the study will be to answer questions raised about the ability of the low pressure core cooling spray system to cool the core sufficiently due to existence of counter-current flow phenomena at the core upper plate. This phenomena and its effect on core cooling will be investigated. A hypothesis is that an important part of the core cooling happens via internal recirculation, and such recirculation will be studied, as well.

The work has been carried out by Sean Roshan and Tomasz Kozlowski.
2.2.2 Severe Accident analysis

The objective of this task was to support SSM in review of the severe accident analysis of O2 related to power uprate. This task included development of MELCOR model (input decks) for O2, validation of the model (comparisons of code-to-code calculations) and analysis of severe accident scenarios of risk importance. The work was postponed due to other tasks, and the input deck will be developed in the end of this year.

The work will be carried out by Weimin Ma.

2.2.3 Analysis of past events

KTH, in collaboration with University of Michigan and Pennsylvania State University, prepared steady-state TRACE/PARCS model and started modeling of feedwater transient that occurred on February 25, 1999. This is a very challenging problem for which TRACE/PARCS and most of the other coupled codes have not been validated yet.

The modeling of the event was successful and the analysis is ongoing. The event was proposed and was accepted as international benchmark by OECD/NEA.

The work has been carried out by Tomasz Kozlowski and Sean Roshan.

2.3 Forsmark-3 activities

2.3.1 Transient analyses

The objective of this task was to support SSM in review the F3 safety analysis for operation at the uprated power and evaluating F3 transients at uprated power level.

Due to similarities between O3 and F3, O3 model has been used as the starting point for the F3 power uprate project. Even though these plants started as sister plants, the modifications and changes made during the years have made them different from each other. Therefore, the O3 RELAP5 model has gone through a comprehensive comparison with the F3 available data. O3 RELAP5 model was improved to fit actual state of F3 and cover the differences between F3 and O3. The necessary changes to model were made in the input resulting in an independent, self-sufficient F3 RELAP5 model.

F3 power plant has faced many problems in the recent years resulting in power uprate delays, and the F3 uprate project has taken a dormant state with no more work performed on the neutronic part of the code package, e.g., PARCS.

Consequently, the analysis of F3 transients has not yet started. Following transients are considered to evaluate when the project resumes: the turbine trip, reactivity initiated accident (CR drop), feedwater transients and LOCA (steam line break) at uprated power conditions.

In addition, following past plant events are planned for future analysis:
load rejection, turbine trip, or similar pressure event;
unauthorized insertion of the CR (pancake core event) from June 15 1994, since there are no LPRM measurements, only global parameters can be compared with a simulation, and code-to-code comparison will be performed for local parameters.

The work has been carried out by Joanna Peltonen.

### 2.4 Ringhals-1 activities

#### 2.4.1 Analysis of past events

Nuclear Engineering students at KTH and other universities have shown a great interest to join NPS team for their MSc thesis work. Although these students need to use codes and inputs during the course of their studies, for proprietary reasons they can’t use inputs developed for safety assessment of Swedish power plant. This need has prompted KTH to develop a coupled code model for Ringhals-1. Since the plant design has been published as an international benchmark, it is considered open source information. The model is also used by students for educational purposes.

The objective of this task was to create and validate R1 model for educational purposes and future support of SSM needs. KTH, in collaboration with University of Michigan, prepared steady-state TRACE/PARCS model and performed validation using Cycle 14, 15, 16, 17 stability measurements.

The primary contributions to the TSO-DSA were:

- validation of TRACE/PARCS for BWR stability prediction;
- development of time-domain stability analysis methodology;
- development of S/U methodology for BWR stability prediction.

The TRACE/PARCS model has been frequently used at NPS for the educational purposes; however, they could at anytime be used in support of SSM needs.

The work has been carried out by Ivan Gajev and Tomasz Kozlowski.

### 2.5 Ringhals-3 activities

#### 2.5.1 Severe Accident analysis

The objective of this work was to perform MELCOR analysis for severe accident scenarios of the Ringhals-3 PWR related to power uprate. The results from the independent analysis can be used to compare with the results of MAAP code analysis submitted by NPPs, so as to reduce the uncertainties in quantification of severe accident risk. The most dangerous scenarios (accidents such as station blackout and station blackout plus loss of coolant) were chosen in the simulations using MELCOR 1.8.5. Generally speaking, the calculations show that the power uprate do not explicitly increase the severe accident risk, in terms of source term release, and maximum of pressure and thermal loads. More studies using newer
MELCOR versions (1.8.6 or 2.1) are under way, which are supposed to increase the simulation fidelity of the in-vessel melt behavior and progression.

The MELCOR 1.8.5 code was also used to investigate the ECCS efficacy of Ringhals-3 PWR. The calculated results showed that

- 20 kg/sec water injection at 20°C to the secondary side of a steam generator from the beginning is sufficient to establish natural circulation in the primary system, enabling the effective removal of decay heat in the core;
- after half of the core height was uncovered the injection into a primary loop (the cold leg) was not sufficient to mitigate the accident, although the coolant injection to a primary loop significantly delays vessel failure.

The work has been carried out by Weimin Ma.

### 3. Support to SSM in addressing emerging safety question

The TSO-DSA agreement allowed KTH-NPS to provide support to SSM in addressing safety questions, by directly responding to SSM requests for analysis. The most significant safety questions addressed are described below.

#### 3.1 Analysis of Forsmark-1 station blackout event

The work started in response to the Forsmark-1 2006 event with loss of the offsite electric power. The RELAP5 and MELCOR codes were used to simulate the event, to determine its severity.

The simulation results have a good agreement with the plant recording data during the event. This analysis did not only demonstrate the capabilities of the RELAP5 and MELCOR for thermal-hydraulic analysis of a BWR, but also answered several “what-if” questions related to the safety margin of the event.

The analysis confirms that failure of only two auxiliary diesel generators as occurred in the Forsmark-1 event does not lead to core uncoverly and hence could not cause any core damage. The analysis predicts that with all four auxiliary diesel generators failed as in a postulated common cause failure, it would take an hour for the fuel cladding temperature to reach the critical level. Although the threat of core damage was not imminent in the actual event, the operator’s success in recovering the failing auxiliary diesel generators conclusively prevented the core from being uncovered, even if a common cause failure had disabled all four auxiliary diesel generators. In addition, the successful recovery of the auxiliary power had an important effect on decision making during the Emergency Operation Procedure (EOP). Namely, it recovered the control room’s monitoring systems which were partially lost during the event’s initial phase. The failure of the monitoring system may have a serious impact on the plant operator’s ability, both technically and psychologically, to manage the plant in such emergency.

The work has been carried out by Sean Roshan and Weimin Ma.
3.2 Effect of CR velocity on fuel temperature for turbine trip transient

The peak cladding temperature (PCT) has been different in two consecutive analyses performed by OKG due to change in safety analysis methodology. NPS was asked to analyze the two cases to support SSM in the review of the result of analysis performed by OKG. The two major questions addressed were:

- How would the change of the cycle point from BOC to EOC affect the peak cladding temperature?
- How would the change of the CR insertion rate affect the peak cladding temperature?

It was found that the effect on PCT is negligible, as long as other safety systems work.

The work has been carried out by Sean Roshan.

3.3 Evaluation of CRGT cooling in BWR severe accident

The MELCOR code was used to assess the effectiveness of Control Rod Guide Tube (CRGT) cooling as a severe accident management measure for BWRs. The first-cut results show that the nominal flow rate (10.5 kg/sec) of CRGT cooling is sufficient to maintain the integrity of the vessel in a Swedish designed BWR of 3900 MWth, if the water injection is activated no later than 1 hour after scram. For late recovery of the CRGT cooling (later than 1 hour after scram), a higher flow rate rather than the nominal is needed to contain the melt in the vessel. For instance, if water injection through CRGTs is activated after 2 hours following scram, much higher flowrate (~40 kg/sec) is required for in-vessel melt retention (IVR).

The work has been carried out by Weimin Ma.

3.4 Influence of steam separator pressure drop on BWR stability

Replacement of BWR steam separator causes change in the total pressure drop. RELAP5/PARCS model was create and used to investigate influence of steam separator pressure drop on BWR stability.

The effect was evaluated at various power/flow conditions. The main result of the study was to confirm that decrease of steam separator pressure drop increases core stability. However, the magnitude of the effect is very uncertain, and probably small for steam separator. This is consistent with results reported in literature, that BWR stability increases when two-phase pressure drop decreases.

The work has been carried out by Joanna Peltonen.
3.5 Oskarshamn-3 operation with partially inserted CR (Issue 86-14)

Due to the cracking problems in control rods discovered in Oskarshamn-3, SSM was informed by OKG that the O3 will be operating on a 86-14 strategy: all of the control rods would be inserted into the core by 14% to avoid the thermal fatigue on the most exposed part of the rod. OKG had submitted a number of calculations supporting the feasibility of the case in order to receive approval for this condition. NPS has performed numerous analyses to corroborate or contradict this and the results have been presented to SSM.

Dynamic of the core was analyzed for cycle effect (Cycle 24 vs. Cycle 31), CR position effect (100% vs. 86%) and burnup effect (BOC vs. EOC). No significant effect was found between the Cycle 24 core with 100% CR out and the Cycle 31 with 86% CR out. The differences were clearly due to small difference in the Total Power Coefficient (TPC).

The main conclusion of the study was that the operation of the new core loading and the 86-14 Control Rod pattern has only marginal effect on the reactor dynamic behavior.

The work has been carried out by Sean Roshan and Andrej Kubarev.

3.6 CCFL in the external pump reactors during a LB-LOCA

SSM has requested a study of the Counter Current Flow Limiting (CCFL) phenomena during a LB-LOCA in the external pump reactors. TRACE/PARCS was selected as a suitable tool for this purpose. This study aims to show if the CCFL limits will be exceeded in the event of a double-edge guillotine break in one of the main circulation pumps’ pipe. The study includes:

- development of a TRACE input file for LOCA (complete);
- adaptation of PARCS input file for LOCA (complete);
- coupling between TRACE and PARCS input files (complete);
- study of LOCA transient with different pipe break area and comparing the results with the FSAR (on-going);
- study of different CCFL models and their effect on the outcome (future work);
- recommendations to the SSM based on the study results (future work).

The work has been carried out by Sean Roshan.

4. Evaluation and contribution to relevant international projects

SSM is participating in many international reactor safety projects within thermal-hydraulics and severe accidents area. KTH had contributed analysis of the OECD and other relevant international reactor safety research problems. Validation of the codes on experiments gives comprehensive knowledge about codes’ behavior. Contribution to the thermal-hydraulic reactor safety international project and code validation activities relevant to the power uprate needs are summarized in this section.
4.1 U.S. NRC CAMP contribution

The analysis groups at KTH and Chalmers have taken the responsibility to fulfill CAMP (Code Assessment and Maintenance Program) agreement. The analysis groups and SSM select appropriate plant events and experiments for CAMP contribution, and submit them to U.S. NRC for approval.

KTH has prepared a draft report on Marviken Critical Flow experiment. Currently the report is under internal review, to be submitted to the U.S. NRC by the end of 2010. The next planned contributions are Marviken Level Swell experiments and NUPEC BWR void fraction and critical power experiments.

The work on Marviken Critical Flow and Level Swell experiments has been carried out by Lukasz Sokolowski. The work on NUPEC BWR void fraction and critical power experiments has been carried out by Francesco Cadinu, Joanna Peltonen and Tomasz Kozlowski.

4.2 U.S. NRC CSARP contribution

KTH is also a member of CSARP (Cooperative Severe Accident Research Program). KTH contribution is to present the outcomes of APRI research and MELCOR activity. In return, KTH has full access to all documents of CSARP and MCAP meetings, and MELCOR user workshops. KTH also receives all versions of MELCOR code and support to use them.

KTH is actively participating in MELCOR Code User Workshop and MELCOR Code Assessment Program (MCAP) meeting hosted by the code developer (Sandia National Laboratories) and the sponsor (USNRC). So far KTH have been following the progress of the code development and applications, with all access to all versions of the code. The highlights and major results of KTH MELCOR activities were also well received and appreciated by colleagues at European MELCOR User Group (EMUG) meeting.

This collaboration is very beneficial for the TSO-DSA work, as it allows access to MELCOR code, user workshop and developer support.

The work has been carried out by Weimin Ma.

4.3 OECD ATLAS thermal-hydraulics reactor safety projects

The objective of the project was modeling of Direct Vessel Injection (DVI) Line Break test in ATLAS facility. The new phenomenon addressed by this test is 3D flow distribution in downcomer during Direct Vessel Injection (DVI). In particular, the thermal-hydraulics phenomena occurring in the upper annulus down-comer region between the DVI nozzle and the cold leg nozzles are expected to be complicated due to the countercurrent flow of the upward break flow and the downward safety injection flow. Therefore, the relevant models need to be incorporated into the safety analysis codes in order to predict these thermal hydraulic phenomena correctly.
KTH has created TRACE model of the ATLAS facility and participated in the International Standard Problem No. 50 (ISP-50). Modeling of the experiment was successful and blind calculation results have been submitted at the beginning of January 2010. The KTH TRACE calculation predicted the core power, primary system pressure, and accumulated mass of break flow with an acceptable accuracy. However, prediction performance for the core wall temperatures, secondary system pressure, collapsed water levels in intermediate loop, and primary flow rates were comparatively poor.

Further analysis of the experiment is ongoing, and additional model improvements were identified:

- choking on breaks and orifices;
- safety valve on SG;
- pressurizer heater behavior during the transient;
- pressure distribution in the primary loop;
- containment simulator is modeled by a break component

The model improvements will be made later in 2010, the results will be re-submitted and the conclusions about TRACE predictive capabilities will be made.

The work has been carried out by Erdenechimeg Suvdantsetseg and Tomasz Kozlowski.

4.4 OECD BFBT thermal-hydraulics benchmark

The OECD/BFBT (BWR Full-size Fine-mesh Bundle Tests) benchmark contains the highest quality void and critical power measurements available publically. The objective of this project was to validate RELAP5 and TRACE for BWR void fraction and critical power prediction.

KTH prepared RELAP5 and TRACE model and completed all calculations. Both codes performed well for void fraction prediction, but deficiencies were found with critical power predictions. The results have been communicated to SSM and U.S. NRC. The final report will be proposed to the U.S. NRC as CAMP contribution.

The work has been carried out by Francesco Cadinu, Joanna Peltonen and Tomasz Kozlowski.

4.5 FIX-II 31% break LOCA Test Number 3025 (ISP-15)

BWR power uprate presents significant challenges to the thermal-hydraulic codes. The most significant need is the validation for RELAP5 and TRACE for integral BWR transient behavior. FIX-II integral test facility was selected to perform validation of TRACE for BWR LOCA and other transients. The aim of the FIX-II experiments was to increase the knowledge and improve the data basis for the calculations of certain accident transients for Swedish BWRs. The experiments included the following cases:
• static dryout experiments;
• LOCA experiments with simulated breaks on a main recirculation line of an external pump type reactor;
• pump trip experiments for internal pump type reactors.

In case of LOCA experiments, the facility is a scaled down model of reactor with external recirculation pumps. The scaling was based on Oskarshamn-2 reactor and gives volume ratio of 1:777.

TRACE model of FIX-II facility has been completed and analysis of Test 3025 (ISP-15) is on-going. The FIX-II Test 3025 simulates an intermediate size split break in one of the four main recirculation lines. The break area was 31 per cent of the scaled down pipe area of the reactor. The initial power of the 36-rod bundle was 3.38 MW, corresponding to the hot channel power of the reactor. Phenomena included in the test were:

• collapsed level behaviour in downcomer;
• core thermal hydraulics;
• void collapse and temperature distribution during pressurization;
• critical power ratio.

The work is not yet complete and the experiment analysis is on-going. The work is being carried out by Łukasz Sokolowski and Tomasz Kozlowski.

4.6 Marviken critical flow experiments

The objective of this task was to validate RELAP5 and TRACE critical flow model with Marviken critical flow experiments (CFT 1-27). The experiments provided essential data for code validation of critical flow model.

KTH prepared RELAP5 and TRACE model and completed all calculations. Calculated mass flow rates are in good agreement with the experimental data.

The final report is under internal review and will be submitted to SSM and U.S. NRC as NUREG/IA.

The work has been carried out by Łukasz Sokolowski.

4.7 Marviken level swell experiments

The objective of this task was to validate RELAP5 and TRACE level-swell model with Marviken level-swell experiment (based on Jet Impingement Test series). JIT experiments provided essential knowledge about the properties of critical flow and differential pressures inside the pressure vessel and are useful in evaluation of LOCA.

KTH prepared RELAP5 and TRACE model and completed calculations for level-swell experiments (JIT test 5 and 11). Calculated pressures in the steam dome and mass flow rates
are in good agreement with the experimental data. Calculated differential pressures are in good agreement only within one phase region.

The final report has not yet been completed. The report will be submitted to SSM and U.S. NRC as NUREG/IA.

The work has been carried out by Lukasz Sokolowski.

5. Additional analysis tools development

In order to facilitate variety of safety analysis needs, several codes were modified and new codes were developed.

5.1 PARCS code modifications

PARCS code was modified to add important features necessary to BWR analysis:

- point kinetic reactivity calculation to simulate the reactivity measurements by detectors;
- option of opposite motion for dropped and scrammed rods;
- control rods acceleration motion option (for hydraulic engine).

The work has been carried out by Andrej Kubarev.

5.2 RELAP5 code modifications

The following issues were encountered, which limited the codes’ capability for modeling of Swedish BWRs:

- memory limit on the number of components
- memory limit on the length of the input

Both issues were submitted to U.S. NRC and were resolved promptly.

5.3 TRACE code modifications

An issue was encountered, where a small change in the vessel nodalization significantly affects the results. This limits the user’s confidence in the codes’ modeling capabilities.

The issue will be studied with additional nodalization studies, and if confirmed, will be submitted to U.S. NRC as a code deficiency.
5.4 **MELCOR code modifications**

The following bugs have been reported to the developers:

- The value of plotting parameter COR-EMWR-TOT in MELCOR 1.8.5 is incorrect.
- The control parameter COR-MLTFR.n.m.k is not usable in CF.
- The control parameter COR-ABRCH is not physical after it is larger than the vessel cross-section area.
- The MELCOR 1.8.6 could not run to completion if the eutectics model is activated on COR00006. The suggestion to the problem is to run the problem by deactivating the eutectics model. It is said the code developers will to fix this model in MELCOR 2.1.

5.5 **Core behavior visualization software**

A tool was developed to visualize and compare radial, axial power and control rods behavior during a transient. The visualization software consists of two parts:

- Fortran program for reading PARCS output files and generates appropriate MATLAB data input;
- MATLAB program for generating the figures and movies.

The work has been carried out by Andrej Kubarev.

5.6 **Detect and suppress methodology for Swedish designed BWR**

The concept of instability suppression system and its performance in a BWR plant was presented in the PHYSOR-2008 conference.

Suppression system was designed to provide a compensation for reactivity deviations in the reactor core. In the present approach, the entire core is hypothetical divided into 10 Local Automatic Control (LAC) zones. Each LAC zone contains 2 LAC rods and 2 Local Emergency Protection (LEP) rods. Each zone contains radial and axial neutron flux detectors – LPRM (divided into several independent groups), which operate the relevant control rods, i.e. the rods in a certain local sub-region are designed to move in response to signals reflecting the misbalance between the core operational set point and the local power level (neutron flux).

The signals from the detectors are processed by external equipment: they are processed and discrepancies against the local operational set point are evaluated. In order to compensate for the discrepancies, one can initiate motions of the appropriate LAC and LEP rods to bring the monitored system into balance. Each zone has at least 4 (up to 6) detector groups. In order to activate the suppression system, the relevant rods are required exceeded pre-set actuation signal in at least three detector groups. This was done to minimize the probability of spurious activation due to possible LPRM malfunction.
The major conclusion has been that the proposed instability suppression system can be effective in suppressing instability for the frequency range expected in BWR (less than 1 Hz). For higher frequencies the system can activate only reactor scram function.

The work has been carried out by Andrej Kubarev.

5.7 Best-estimate plus uncertainty methods development

Usage of the best-estimate codes as tools for safety analysis has brought enormous advantages to the safety assessment arena. However, these codes require a solid understanding of the uncertainty involved and methods to deal with that. This project started with identification of all possible sources of uncertainty in the work ahead and classification of those.

Three different methods were reviewed for uncertainty quantification:

- propagation of input errors (e.g. GRS method);
- propagation of output errors (e.g. D’Aura method);
- adjoint sensitivity method (e.g. Cacuci method).

Propagation of Input Errors (PIE) method was found to be very practical, flexible and has showed potential to investigate problems in the plant and guide improvements of code models.

In order to implement results of the study, a program called ASSUME (Automatic Software for Sensitivity and Uncertainty Methods) was developed to service the needs of uncertainty calculations. It was implemented for coupled codes RELAP5/PARCS and TRACE/PARCS and applied to several problems (BFBT benchmark and R1 stability benchmark).

KTH plans to continue S/U methodology development and implement it in the ASSUME program. Current needs are:

- extend to different sources of uncertainty;
- extend to other codes (e.g. MELCOR);
- validate for a range of problems.

It is expected that in the future the methodology will be routinely used for all TSO-DSA activities. The goal is to apply the methodology to all power uprate analysis.

The work has been carried out by Sean Roshan and Ivan Gajev. The ASSUME software has been developed by Ivan Gajev.

6. Training and education of SSM's personnel

KTH has organized several educational courses, workshops and seminars. SSM personnel was invited and participated in all of them.
• Computational Nuclear Power Safety (CNPS) seminar – organized every 6 months since April 2008

The purpose of the CNPS seminar is to present a scope and philosophy of CNPS research work at KTH-NPS and establish a discussion between CNPS developers and end-users on priorities in CNPS research directions. The objective of the seminar is to keep Steering Committee and Reference Group members of relevant projects (SSM, SKC, NORTHNET, NKS, EU) informed about progress of their projects, and provide a regular feedback to CNPS scientists at KTH-NPS from end users.

The first 5 seminars were held on:

- CNPS-1 2008-04-10
- CNPS-2 2008-10-31
- CNPS-3 2009-04-24
- CNPS-4 2009-12-04
- CNPS-5 2010-05-17

• RELAP5 intro course – August 2008

Introductory RELAP5 course was organized in August 2008. The training objectives were:

- learn RELAP5 control-volume modeling approach;
- learn how to read and understand RELAP5 input;
- learn how to run RELAP5 model;
- learn how to check calculation results;
- learn how to make model modifications and how to fix problems.

• Seminar and Training on Scaling, Uncertainty and 3D Coupled Code Calculations in Nuclear Technology (SUNCOP), October 12 – October 30, 2009.

The objective of the seminar is to transfer to the participants competence and experience in uncertainty methodologies and 3D coupled code calculations from activities carried out over the last two decades by a group of experts from different organizations (universities, research labs, industry) through participation in benchmarks, International Standard Problems and international cooperation. The seminar is subdivided into three parts:

- the concepts and theory of the proposed methodologies;
- methodology training and application;
- user qualification.

• CAMP 2010 Spring meeting, June 9 – 11, 2010

CAMP (Code Maintenance and Assessment Program) is a cooperation program between U.S. NRC and other country safety authority that allows foreign countries to use U.S. NRC best-estimate thermal-hydraulic codes (RELAP5, TRACE and PARCS).
CAMP meeting is a professional user group meeting on thermal-hydraulic plant safety analysis using U.S. NRC best-estimate codes.

7. TSO-DSA contribution to MSc education

The TSO-DSA funding has allowed KTH to contribute to MSc education in area of reactor safety and thermal-hydraulics. Following MSc theses have been completed in area of reactor safety and thermal-hydraulics with direct support from TSO-DSA staff.

- Christoffer Wigert, 2009

- Ivan Gajev, 2009
  “Sensitivity and Uncertainty of Ringhals-1 TRACE/PARCS Stability Prediction”

- Marco Gaboardi, 2009
  “Sensitivity of Void Fraction Prediction to Physical and Numerical Models in System Codes”

- Elsa de Alfonso, 2009
  “Qualification of TRACE/PARCS Spatial Coupling for Ringhals-1 Stability Prediction”

- Lukasz Sokolowski, 2009
  “Sensitivity of Critical Flow Prediction to Physical and Numerical Models in System Codes”

- Erdenechimeg Suvdantsetseg, 2010
  “Analysis of Direct Vessel Injection Line Break LOCA in the ATLAS Facility”

- Daniel Heredia, 2010
  “Sensitivity of Departure from Nucleate Boiling Prediction to Physics and Numerical Models in System Codes”

8. Planned future activities

This Chapter outlines the activities planned for the years 2011 – 2013.

8.1 Plant analysis activities

8.1.1 Oskarshamn-3 activities

- Power uprate start-up test analyses

  KTH performed pre-analysis of the following power uprate startup tests:
- Test 604, stability measurement
- Test 663, loss of one feedwater pump
- Test 680, SCRAM from full power

KTH will perform post-analysis of the power uprate startup tests when the results become available. The post test simulation which, will be performed after the actual tests have been performed will provide us with valuable information about the plant vs. code status.

TSO-DSA group at KTH will continue to support the SSM during the remaining of project PULS and will keep the O3 inputs up to date in order to be able to aid SSM if any needs occur.

- Severe Accident analysis

Severe accident analysis will be done using the latest MELCOR release (versions 1.8.6 and 2.1) for O3 related to power uprate and other safety issues (e.g., DCH, CRGT cooling as a potential SAM measure, ECCS efficiency/reflooding, in-vessel melt progression). The uncertainty analysis using the MELCOR code will be gradually introduced in the future work, to increase the credibility of the simulation results. The future work may also include the utilization of graphic tool of SNAP for MELCOR modeling development and animation of the simulation results by GRS analysis simulator ATLAS.

8.1.2 Oskarshamn-2 activities

- Power uprate transient analysis

An analysis of the core cooling during large rupture in the recirculation pump line (LB-LOCA) is on-going and will continue in 2011.

A model for safety analysis of the plant is ready and will undergo validation against plant data. This validated model will be used to support SSM in review the O2 safety analysis for operation at the uprated power and evaluating O2 transients at uprated power level achieved during Project PLEX. The analysis will continue in 2011, 2012 and possibly 2013, depending on the time schedule of Project PLEX.

The analysis also requires code validation for CCFL (counter current flow limiting) condition and vessel internal recirculation. The code validation for these phenomena will occur in 2011 and 2012.

- Severe Accident analysis

The development of MELCOR modeling for O2 is ongoing. MELCOR analysis will be performed for severe accident scenarios related to power uprate, after completion and testing of the input decks. The uncertainty analysis using the MELCOR code will be gradually introduced in the future work, to increase the credibility of the simulation results.
8.1.3 Forsmark-3 activities

- Transient analyses

KTH will support SSM in reviewing the F3 safety analysis for operation at the uprated power and evaluating F3 transients at uprated power level. The following transients are considered to evaluate:

  - the turbine trip;
  - reactivity initiated accident (CR drop);
  - feedwater transients;
  - LOCA (steam line break) at uprated power conditions.

- Past event analysis

The following past plant events are planned for analysis for 2011 and 2012:

  - load rejection, turbine trip, or similar pressure event;
  - unauthorized insertion of the CR (pancake core event) from June 15 1994, since there are no LPRM measurements, only global parameters can be compared with a simulation, and code-to-code comparison will be performed for local parameters.

- Severe Accident analysis

MELCOR code modeling (input decks) can be developed for F3 during 2011-2012 and MELCOR analysis can be performed for severe accident scenarios related to power uprate. This proposal needs to be discussed with SSM.

8.1.4 Ringhals-1 activities

- BWR stability analysis methodology and S/U methodology development

The TRACE/PARCS model developed at KTH-NPS can be used at anytime in support of SSM needs.

The currently on-going activities are planned to continue in the future:

  - validation of TRACE/PARCS for BWR stability prediction;
  - development of time-domain stability analysis methodology;
  - development of S/U methodology for BWR stability prediction.

8.1.5 Ringhals-3 activities

- Severe Accident analysis

Severe accident analysis will be done using the latest MELCOR release (versions 1.8.6 and 2.1) for R3 related to power uprate and other safety issues (e.g., H2 risk, DCH, reflooding of damaged core/ECCS efficiency). The uncertainty analysis using the MELCOR code will be gradually introduced in the future work to increase the
credibility of the simulation results. The future work may also include the utilization of graphic tool of SNAP for MELCOR modeling development and animation of the simulation results by GRS analysis simulator ATLAS.

8.2 Support to SSM in addressing emerging safety question

KTH-NPS is ready to support SSM in addressing and resolving emerging safety questions. The particular issues pending for period 2011 – 2013 are:

- Analysis of the O3 unsignaled TSxD event from February 12, 2010.
- Analysis of O2 March 12, 2008 FW transient, trip on low level.
- Analysis of F3 June 15, 1994 unauthorized insertion of the CR (pancake core event).
- Validation of CCFL in the external pump reactors during a LB-LOCA.
- Validation internal vessel re-circulation in the external pump reactors during a LB-LOCA.
- Validation of transient level prediction in the vessel after power uprate, in order to determine if the water level should be increased at uprated conditions.
- In addition to the R3, O3 and O2 reactors, MELCOR modeling (input decks) can be developed for other Swedish reactors (e.g., F3) during 2011-2013. The specific needs should be discussed with SSM.

The priorities of safety analysis (TRACE/PARCS, RELAP5/PARCS) and severe accident (MELCOR) analysis activities can be changed upon the request of SSM.

8.3 Evaluation and contribution to relevant international projects

8.3.1 U.S. NRC CAMP contribution

KTH will continue to fulfill CAMP commitments. The contributions planned for 2011 – 2013 are:

- Marviken Level Swell experiments
- NUPEC BWR void fraction and critical power experiments
- NUPEC PWR void fraction and DNB experiments
- GÖTA spray cooling and CCFL experiments
- FIX-II LOCA and pump trip experiments
- O2 Feb. 25, 1999 feedwater event
- Other plant transients used for model validation.

8.3.2 U.S. NRC CSARP contribution

KTH-NPS will continue to participate and contribute to the meetings of MELCOR Code User Workshop, MELCOR Code Assessment Program (MCAP) and European MELCOR User Group (EMUG).
8.3.3 GÖTA and SVEA spray cooling

The objective of this project is to validate RELAP5 and TRACE for CCFL condition at the core upper plate. These experiments are important for code validation in support of BWR LB-LOCA.

The experiment description and experimental data have been received from SSM, but the work was delayed to too lack of man power. It is expected that the project will commence at the end of 2010 and continue in 2011.

8.3.4 OECD PSBT thermal-hydraulics benchmark

The OECD/PSBT (PWR Sub-Channel and Bundle Tests) benchmark contains the highest quality void and DNB measurements available publically. It is analogous to BFBT benchmark for PWR. The objective of this project was to validate RELAP5 and TRACE for PWR subcooled boiling and DNB prediction.

KTH will prepare TRACE model of PSBT facility and perform the benchmark. The work is scheduled for 2011 – 2012.

8.3.5 FIX-II transient analysis

The objective of this project is to validate TRACE for integral BWR transient behavior. FIX-II integral test facility was selected to perform validation of TRACE for BWR LOCA and other transients.

TRACE model of FIX-II facility has been completed and analysis of Test 3025 (ISP-15) is on-going. The project is performed to support LB-LOCA study for O2.

Following additional experiments are scheduled for 2011 – 2013:

- static dryout experiments;
- LOCA experiments with simulated breaks on a main recirculation line of an external pump type reactor;
- pump trip experiments for internal pump type reactors.

These experiments are important for code validation in support of BWR LOCA, FW transient and other BWR transients.

8.4 Additional analysis tools development

The most important analysis tools and methods development needs are described below.

8.4.1 Sensitivity/uncertainty methodology

Based on the previously developed ASSUME (Automatic Software for Sensitivity and Uncertainty Method) program, KTH-NPS will continue development of this S/U methodology. It is expected that in the future the methodology will be routinely used for all TSO-DSA activities.
The 2011 – 2013 activities are:

- Continue S/U methodology development, implement it in ASSUME program.
- Continue development of ASSUME program
  - extend it to other sources of uncertainty;
  - extend it to other codes (PARCS, MELCOR);
  - validate it for a range of problems.
- Apply the methodology to O2 LBLOCA and power uprate analysis.
- Release official version of ASSUME that could be used by other Swedish partners.

The long-term goal of this project is to apply the S/U methodology to all deterministic analysis at KTH-NPS.

8.4.2 BWR stability analysis methodology

KTH proposes to continue development of time-domain stability analysis methodology. The methodology will be applied to relevant plant events.

The 2011 – 2013 activities are:

- continue validation of RELAP5 and TRACE for stability prediction;
- validate RELAP5/PARCS or TRACE/PARCS with R1 stability measurements
- validate RELAP5/PARCS or TRACE/PARCS with O2 1999 event.

8.5 Training and education of SSM’s personnel

The following training and education of SSM’s personnel is planned for 2011 – 2013:

- Short courses on RELAP5, TRACE, PARCS, MELCOR can be organized upon request from SSM;
- OECD Uncertainty Analysis Methodology (UAM) Workshop is planned at KTH for April 2011;
- Computational Nuclear Power Safety (CNPS) seminars should continue at 6-8 month intervals. This seminar is supported jointly by SSM, SKC, NORTHNET, NKS and EU projects.

8.6 TSO-DSA contribution to MSc education

KTH-NPS will continue to contribute to MSc education in the area of reactor safety and thermal-hydraulics, which are subjects relevant to TSO-DSA topics. Based on the past experience, we expect to educate 2-5 MSc students per year. The students will receive direct support from TSO-DSA staff.
9. Publication list

9.1 Journal publications


9.2 Conference proceedings


9.3 Technical reports


9.4 NUREG/IA reports


9.5 Other publications


Weimin Ma, Chi-Thanh Tran, On the Effectiveness of CRGT cooling as a Severe Accident Management Measure for BWRs, Proceedings of OECD/NEA Workshop on Implementation of Severe Accident Management Measures, Villigen, Switzerland, October 26-28, 2009.

Weimin Ma, MELCOR activities at KTH, The 2nd Meeting of the European MELCOR User Group, Prague, Czech Republic, March 1 - 2, 2010.

Weimin Ma, Severe accident analysis for Swedish NPPs using MELCOR 1.8.5 and MELCOR 1.8.6, MELCOR Code Assessment Program (MCAP) Technical Review Meeting, Residence Inn Bethesda, Bethesda, Maryland, September 16-17, 2010.