10th International Topical Meeting on Nuclear Thermal Hydraulics, Operation and Safety

Okinawa Convention Center, Ginowan, Okinawa, JAPAN
Sunday, 14 December - Thursday, 18 December, 2014

NUTHOS-10
WELCOME TO NUTHOS-10

Foreword

It is a great pleasure to welcome and extend our appreciations to all the participants to the 10th International Topical Meeting on Nuclear Thermal Hydraulics, Operations and Safety (NUTHOS-10) in Okinawa, Japan.

I don’t have to write for the importance of nuclear energy which is clean, reliable and safe. It is clear that no other energy sources can replace the nuclear power base load. And we are well aware that the nuclear power is not without its challenges in the wake of the Fukushima Daiichi nuclear plant accident. The enhanced safety is emphasized where lessons learned from the accident should be implemented, while increasing cost must be overcome, and in the long range, perhaps most important to matured nuclear power countries, eventual drop-off of the current nuclear power generation from the grid should be compensated by new nuclear builds.

Enhanced safety and reducing cost are going together, which can be achieved through continued research and development efforts. NUTHOS keeps you abreast of the most updated information in the advancement of science and technology in nuclear thermal hydraulics, operations and safety, and provides you insights into the future.

The first NUTHOS was held in Taipei in 1984 followed by Tokyo (1986), Seoul (1988), Taipei (1994), Beijing (1997), Nara (2004), Seoul (2008), Shanghai (2010), and Kaohsiung (2012). The success of the previous conferences clearly indicates the high interest of the international nuclear community in the NUTHOS and has led to the establishment of NUTHOS as one of the major international conferences in the arena.

The NUTHOS-10 is sponsored by Atomic Energy Society of Japan, in cooperation with the International Atomic Energy Agency, and co-sponsored by American Nuclear Society Thermal Hydraulics Division among others.

Here, I would like to thank all the reviewers, Session Organizers, Session Chairs/Co-Chairs, for their time and efforts in maintaining the high quality of the NUTHOS program. Also my thanks go to members of Local Program Committee and Technical Program Committee for their exceptional devotion and self-sacrificing efforts to make this prestigious meeting successful.

Finally, I hope that all the participants will find the meeting very productive and the stay in Okinawa very enjoyable.

Hisashi Ninokata
NUTHOS-10 General Chair
Professor, Politecnico di Milano
Sponsors

Atomic Energy Society of Japan (AESJ)
[Thermal Hydraulics Division]

In cooperation with

IAEA
The International Atomic Energy Agency

Co-Sponsored by

American Nuclear Society
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Mexican Nuclear Society

Nuclear Society of Slovenia

Japan Atomic Industrial Forum, Inc.

Canadian Nuclear Society

Chinese Nuclear Society

Israel Nuclear Society

Korean Nuclear Society

Chung-Hwa Nuclear Society

Associação Brasileira de Energia Nuclear

Vietnam Nuclear Society

Indian Nuclear Society

The Federation of Electric Power Companies of Japan (FEPC)
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General Information

Registration
The NUTHOS-10 Registration Desk will be open at the venue of Welcome Reception on December 14 and at the entrance hall on the 1st basement of the Okinawa Convention Center from December 15 to 18 with the following opening hours:
Sunday, December 14, 16:00 - 18:00
Monday, December 15, 8:30 - 18:00
Tuesday, December 16, 9:00 - 18:00
Wednesday, December 17, 9:00 - 18:00
Thursday, December 18, 9:00 - 10:00
Participants who registered in advance should pick up the name badge, the receipt, and a bag with the program, USB proceedings, and Official Dinner ticket at the Registration Desk.

Important Information to Presenters and Chairs

For Presenters:
• You are requested to come to the room of the session at least 5 minutes before the session starts. The chairman will confirm all the presenters prior to starting the session.
• The presentation time is 20 minutes including Q&A (for example, 17 minutes of presentation and 3 minutes of discussion) basically for each paper. However, they depend on the total number of presentations and time length of your session.
• The room B5 in conference building B is open as a waiting room.

For Chairs:
• Please come to your session room 10 minutes before the session starts and confirm all the presenters prior to starting the session.
• The biography of each presenter is on the desk of session chair. Please confirm that the biography correspond to each presenter in the session. If the biography is not prepared, please ask the presenter to write a short note of biography.
• Selected papers from the NUTHOS-10 conference will be published in refereed archival journals as special NUTHOS-10 issues. Please recommend one or two good papers in your sessions and write the review form on the desk.

Internet connection
Free wifi service is available in the whole area of the Okinawa Convention Center.
Access point: Free_WiFi-Convention_Center, CVAP01, CVAP02,,,,,,,,,, CVAP22 (No password)
Events Highlights

Welcome Reception
All participants to NUTHOS-10 are invited to the Welcome Reception, which will be a cocktail party and is held in Okinawan Music&Dining "Tubaraama", December 14, from 18:00 to 20:00. Please note that “No” shuttle bus service from LAGUNA GARDEN HOTEL is available.

Opening and Plenary Session I and II
The Opening and Plenary Sessions will be held in the Room A1 on the 1st floor of the Okinawa Convention Center Conference Building A on Monday, December 15, from 9:00 to 11:35.

Plenary Session III
The Plenary Session III will be held in the Room A1 on the 1st floor of the Okinawa Convention Center Conference Building A on Tuesday, December 16, from 9:00 to 10:30.

Official Dinner
The Conference Official Dinner will be held at “Ryukyumura” in about 40 minutes drive from Okinawa Convention Center on Tuesday, December 16, from 19:00 to 21:00. Shuttle buses are arranged to transport participants between Okinawa Convention Center and Official Dinner site. The buses leave the conference site at 18:15, and after the dinner, they leave Official Dinner site at 21:10 – 21:20 to Naha hotels with stopping by LAGUNA GARDEN HOTEL. (Arrival time on Naha hotels: 22:10 – 22:30)

Closing
The Closing Session will be held in the Room A1 on the 1st floor of the Okinawa Convention Center Conference Building A on Thursday, December 18, from 11:20 to 12:00. Best paper award ceremony will be held in the closing session.
Technical Tour
We are planning the Technical Tour to visit Okinawa Institute of Science and Technology, OIST, and Sakiyama suyzo on Friday, December 19. The tour bus departs from Naha 8:00 with picking up the participants in each official hotels.
❖ Departure from hotel in Naha (8:00 - 8:30)
❖ Departure from Laguna Garden hotel near the conference venue (9:00 - 9:15)
❖ Tour of Okinawa Institute of Science and Technology (10:00 - 11:30)
❖ Transportation and lunch in bus (11:30 - 12:00)
❖ Tour of Sakiyama suyzo (12:00 - 13:00)
❖ Return to Laguna Garden hotel (13:45 - 14:00)
❖ Arrival in Naha Airport (15:00 - 15:15)
Return to hotels in Naha (15:30 - 16:00)
Conference Venue
Facility layout of Okinawa Convention Center

Address
Okinawa Convention Center, 4-3-1 Mashiki, Ginowan City, Okinawa, 901-2224, Japan
## Program Overview

**Room A1 (250)**  
**Room A2 (60)**  
**Room B1 (122)**  
**Room B2 (54)**  
**Room B3+B4 (60)**  
**Room B6+B7 (72)**

### 12/15 (Mon.)

<table>
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<tr>
<th>Time</th>
<th>Session</th>
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<tbody>
<tr>
<td>9:00-9:30</td>
<td>Opening remark</td>
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<tr>
<td>9:30-10:30</td>
<td>Plenary 1 (Prof. Dinh)</td>
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<tr>
<td>10:30-10:50</td>
<td>Coffee break</td>
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<tr>
<td>10:50-11:50</td>
<td>Plenary 2 (Prof. Yamana)</td>
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<tr>
<td>11:50-13:00</td>
<td>Lunch</td>
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<tr>
<td>13:00-15:00</td>
<td>Thermal Hydraulics and Safety of Water-Cooled Reactors 1</td>
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<td>(20min × 5+20min)</td>
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<tr>
<td>15:00-15:30</td>
<td>Coffee break</td>
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<tr>
<td>15:30-17:50</td>
<td>Thermal Hydraulics and Safety of Water-Cooled Reactors 3</td>
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<tr>
<td></td>
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<tr>
<td>19:00-19:30</td>
<td>Coffee break</td>
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<td>19:30-19:50</td>
<td>Lunch</td>
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### 12/16 (Tue.)

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<tr>
<td>9:00-10:30</td>
<td>Plenary 3 (Panel)</td>
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<td>10:30-11:00</td>
<td>Keynote Lecture 1</td>
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<td>Coffee break</td>
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<td>12:00-13:00</td>
<td>Lunch</td>
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<tr>
<td>13:00-15:00</td>
<td>Thermal Hydraulics and Safety of Water-Cooled Reactors 5</td>
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<td>Keynote Lecture 2</td>
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<td>10:00-10:30</td>
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<td>10:30-12:00</td>
<td>Thermal Hydraulics and Safety of Water-Cooled Reactors 8</td>
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<td>(20min × 4+10min)</td>
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<td>12:00-13:00</td>
<td>Lunch</td>
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<td>13:00-15:00</td>
<td>Thermal Hydraulics and Safety of Water-Cooled Reactors 9</td>
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<td>15:30-17:50</td>
<td>Thermal Hydraulics and Safety of Water-Cooled Reactors 10</td>
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### 12/18 (Thu.)

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<td>11:00-11:20</td>
<td>Coffee break</td>
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### Schedule Details

- **Opening remark**
- **Plenary 1** (Prof. Dinh)
- **Plenary 2** (Prof. Yamana)
- **Coffee break**
- **Lunch**
- **Keynote Lecture 1**
- **Keynote Lecture 2**
- **Keynote Lecture 3**
- **Coffee break**
- **Keynote Lecture 4**
- **Keynote Lecture 5**
- **Coffee break**
- **Keynote Lecture 6**
- **Keynote Lecture 7**
- **Keynote Lecture 8**
- **Coffee break**
- **Keynote Lecture 9**
- **Keynote Lecture 10**
- **Coffee break**
- **Coffee break**
- **Closing Remark**
Dec. 17 (Wed.), 15:30 – 17:50, Room B2
B-4-3. Verification and Validation of Numerical Codes 3
Chairs: Damar Wicaksono (École polytechnique fédérale de Lausanne, Switzerland), Aaron Simon Epiney (PSI, Switzerland)
NUTHOS 3D AGENT Methodology Validation for Prismatic High-Temperature Gas-Cooled Reactor
10–1070 Hermilo Hernandez, Sarah Obadina, Victor Bautista, Tatjana Jevremovic (The University of Utah, United States)
NUTHOS Uncertainty and Sensitivity Analysis of COBRA-TF for the Simulation of Selected OECD/NRC BFBT Void Experiments
10–1196 Simon Epiney, Omar Zerkak, Andreas Pautz (Paul Scherrer Institute, Switzerland)
NUTHOS Evaluation and Validation of Turbulent Models Used in Numerical Simulation of IRWST Experimental Model
10–1124 Gang Lu (North China Electric Power University, China), Zheng Du, Xiao Liang Fu (State Nuclear Power Software Development Center, China)
NUTHOS DEVELOPMENT AND VALIDATION OF COUPLED PARCS/RELAP5 MODEL FOR FORSMARK-2 NPP AT UPGRATED
10–1092 POWER Alexander Agung, József Bánáti (Chalmers University of Technology, Sweden)

Dec. 17 (Wed.), 09:00 – 10:00, Room B3+B4   KL-9. Keynote lecture 9
Chair: Marco Pellegrini (The Institute of Applied Energy, Japan)
NUTHOS SMR-IPWR concepts: thermal hydraulics R&Ds and other key aspects
10–KL09 Marco Enrico Ricotti (Politecnico di Milano, Italy)

Dec. 17 (Wed.), 10:30 – 12:00, Room B3+B4
A-6-1. Cross-Cutting Thermal-Hydraulics of Innovatie Nuclear Systems
Chairs: Vladimir Kriventsev (Karlsruhe Institute of Technology, Germany), Katrien Van Tichelen (Belgian Nuclear Research Centre, Belgium)
NUTHOS CFD analysis on Supercritical Pressure Water Heat Transfer in a 2x2 Rod Bundle
10–1041 Jinbiao Xiong, Xu Cheng (Shanghai Jiao Tong University, China)
NUTHOS Geometric Size Optimization and Behavior Analysis of a Dual-cooled Annular Fuel
10–1124 Wen Xi Tian, Sui Zheng Gao, Guang Hui Su, Wei Xu Zhang, Jun Mei Wu (Xi’an Jiaotong University, China)
NUTHOS Coupled Thermal-Hydraulic and Neutronic Simulations of Phenix Control Rod Withdrawal Tests with SIMMER-IV
10–1123 Vladimir Kriventsev, Fabrizio Gabrielli, Andrei Rineiski (Karlsruhe Institute of Technology, Germany)
NUTHOS FLUENT-based Neutronics and Thermal-Hydraulics Coupling Calculation for a Liquid-Fuel Molten Salt Reactor
10–1145 Chengdong Wang, Zuizheng Gao (Xi’an Jiaotong University, China), Zhi-gang Zhai (Xi’an Modnut Network Technologies Co. Ltd, China), Andrei Rineiski, Shisheng Wang (Karlsruhe Institute of Technology, Germany)

Dec. 17 (Wed.), 13:00 – 15:00, Room B3+B4
A-6-2. Cross-Cutting Thermal-Hydraulics of Liquid Metal and Fusion Systems
Chairs: Takashi Takata (Osaka University, Japan), Hongli Chen (University of Science and Technology of China, China)
NUTHOS Validation of a CFD code Star–CCM+ for liquid Lead–Bismuth Eutectic (LBE) thermal-hydraulics using TALL-3D experiment
10–1269 Marti Jeltsov, Kaspar Kööp, Walter Villanueva, Dmitry Grishchenko, Pavel Kudinov (Royal Institute of Technology, Sweden)
NUTHOS Multi-Scale Uncertainty and Sensitivity Analysis of the TALL-3D Experiment Using Thermal-Hydraulic Coupled Codes
10–1103 Clotaire Geffray, Rafael Macián-Juan (Technische Universität Muenchen, Germany), Angel Papukchiev (IRS Germany)
NUTHOS A Study of Different Approaches for Multi-Scale Sensitivity Analysis of the TALL-3D Experiment Using Thermal-Hydraulic Codes
10–1052 Computer Codes Clotaire Geffray, Rafael Macián-Juan (Technische Universität Muenchen, Germany)
NUTHOS Validation of the Validation of the Flow Distribution Characteristics of Water-Cooled Solid Breeder Blanket Module
10–1078 Mufei Wang, Lili Tong, Xuewu Cao (Shanghai Jiao Tong University, China)
NUTHOS Thermal-Hydraulic Design and Analysis of Helium Cooled Solid Breeder Blanket for Chinese Fusion Engineering Test Reactor
10–1221 Zhongliang Lv, Guangming Zhou, Qianwen Liu, Shuai Wang, Minyou Ye (University of Science and Technology of China, China)
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Dr. Martin Zimmerman, PSI, Swiss
Mr. John Kickhofel, ETH, Switzerland
Prof. Min Lee, NHTU, Taiwan
<table>
<thead>
<tr>
<th>Name</th>
<th>Institution/Location</th>
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<tbody>
<tr>
<td>Prof. Tsung-Kuang Yeh</td>
<td>NHTU, Taiwan</td>
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<td>Prof. C. K. Shih</td>
<td>NHTU, Taiwan</td>
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<td>Prof. Chin Pan</td>
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<td>Dr. S. Arndt</td>
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<td>Prof. D. Aumiller</td>
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Development and Validation of Coupled PARCS/RELAP5 Model for Forsmark NPP at Uprated Power

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ABSTRACT

This paper gives an account of the development and validation of an up-to-date coupled neutronic/thermal-hydraulic model for the Swedish Forsmark boiling water reactor. The model will be used for analyses of the consequences of the planned power uprate from 2928 MWth to 3253 MWth.

At first, the development of the PARCS and RELAP5 models are presented. On the neutronic side, cross-sections data was generated, allowing feeding PARCS with realistic data. This step was performed by converting the library data file from the power plant using the in-house cross-section interface code. The dependence of the material properties on history effects, burnup, and instantaneous conditions was accounted for, and the full heterogeneity of the core was thus taken into account. Each of the 676 fuel assemblies was modeled individually, while the 161 control rods were grouped into 6 different types. On the thermal-hydraulic side, the model consists of a model for the feedwater system, a model for the reactor vessel that include a model for the core channels, and a model for each of the four steam lines. The fuel assemblies were modeled as twelve flow channels in the core region. The coupling between the two codes is touched upon, with emphasis on the mapping between the hydrodynamic/heat structures and the neutronic nodes.

The validation efforts were focusing on benchmarking the code capabilities against measured plant data, both under steady-state and transient conditions. The PARCS standalone model was validated against traversing in-core probe (TIP) measurements, taken at different burnup level with operating power varies from 108% (nominal level) to 120% (uprated level). The coupled PARCS/RELAP5 model was validated against an operational transient. For this validation task, the transient chosen was a turbine trip test, which was performed on May 6, 2013.

Comparisons between calculated and measured parameters demonstrate that the coupled model was able to correctly represent the steady-state conditions of the plant. The validation of the coupled model against measured transient plant data was then performed. It has been demonstrated that the coupled model is able to capture the main features of the transient with a sufficient level of accuracy.

KEYWORDS

Model development and validation, transients, coupled neutronics/thermal-hydraulics, RELAP5, PARCS, turbine trip.
1. INTRODUCTION

Many power utilities worldwide have been implementing power uprates, i.e. increasing the power output of their reactors. Such uprates are economical ways of producing more electricity at a NPP and have attracted interest due to increased electricity prices. The uprates in BWRs (Boiling Water Reactors) can be achieved by increasing the core feed-water flow and steam flows [1, 2].

The Forsmark nuclear power plant is located on the eastern coast of Sweden and there are three BWRs that generate 20 – 25 TWh in a normal year. The commercial operation of the plants began in 1980s. During mid 1980s, uprates of around 8% were performed at each unit. A major uprate to 120% was announced in 2004 by Forsmark Kraftgrupp AB (FKA), but the plan was delayed and the upgrading work began in 2009 [3, 4]. In 2013 the Swedish Radiation Safety Authority (SSM) approved a set of operational tests to be performed at 120% uprated power.

In this context, it is then essential to identify the main consequences resulting from the increased power level of the reactor, and their impact on the safety of the plant. SSM then gave the task to Chalmers to perform independent safety analyses of the power uprate. For this purpose, a coupled neutron kinetics and thermal-hydraulics model was developed using PARCS [5] and RELAP5 [6]. A real operational transient was chosen as a basis for validation of the model.

This paper describes the development and validation of the coupled neutronic/thermal-hydraulic model for the Forsmark BWR unit and it is organized as follows. First, the development of the individual model is presented and the coupling method between the two codes is touched upon, with emphasis on the mapping between the hydrodynamic/heat structures and the neutronic nodes. Validation of the model is then described. The standalone neutronic model was validated against a traversing in-core probe (TIP) measured data and the coupled model was validated against a turbine trip transient test program. The results are discussed in the respective sections. Finally, conclusions are drawn and future activities are described.

2. DEVELOPMENT OF THE PARCS/RELAP5 MODEL

2.1. Features of the PARCS Model

The PARCS nodalization used for the Forsmark BWR core is such that each of the 676 fuel assemblies is modeled radially with one node and axially with 27 non-equidistant nodes. The first and the last axial levels are applied for the bottom and top reflectors outside the active core, respectively. The other axial levels are such that the different segments in which one fuel assembly consists of can be taken into account with little approximations. Besides, the radial reflector is reproduced by making use of dummy assemblies that surrounds the active core and that are associated to the proper radial reflector materials and conditions.

The core is loaded with 18 types of fuel assembly, each having different fuel segments. As one PMAXS cross-section file associates to each of the fuel segments, in total 52 cross-section sets are needed to fully describe the core. In this way the actual heterogeneity of the core is accounted for. Each node is represented by a set of core state variables (burnup, history variables, and instantaneous variables), which differ significantly throughout the core.
The spatial distribution of the exposure and of the history variables (e.g. moderator density) throughout the core is obtained from SIMULATE-3 at the core cycle exposure corresponding to each of the validation cases. These spatial distributions are then fed into PARCS. The PMAXS files are created by converting the CASMO library using an in-house interface code [7]. Fig. 1 describes the fuel loading configuration in the core.

![Fig. 1 Radial zoning of the core.](image)

![Fig. 2 Radial zoning of the control rods.](image)

(a) control rod banks  
(b) control rod types
In the Forsmark unit core, there are 161 cruciform control rods, and they are of 6 different types. Since PARCS cannot model explicitly cruciform control rods, each control rod is defined as a bunch of 4 control rods placed in those fuel assemblies that are near the specific cruciform control rod. Fig. 2 gives the radial position of the different control rod banks in the core. The 161 control rods are grouped into 68 banks as shown in Fig. 2(a) and the configuration of the corresponding types are shown in Fig. 2(b).

For simulating an uprated power condition, the initial core power level of 120% is defined in the PARCS input. As the original nominal power is 2711 MW, the average nuclear power per assembly value in the input card is 4.0104 MW/assembly. The calculation is performed as a criticality search with xenon and samarium options set to equilibrium during steady state. Decay heat calculation is included, while the pin power is not calculated. Zero incoming current is assumed as boundary condition for the neutron field, and a hybrid neutronic solver is used. Decupping of the control rods is not used in the model.

2.2. Features of the RELAP5 Model

The RELAP5 model consists of a model for the Feed-Water (FW) system, a model of the Reactor Vessel (RV) that includes a model for the core channels, and a model for each of the four Steam Lines (SLs). Fig. 3 presents the nodalization scheme of the system. The top figure shows the nodalization of RV and FW systems, while the bottom figure represents the SL systems. Only the SL number 1 and 3 are shown, while the other two SL are nodalized similarly although having different pipe lengths.

To represent the 676 fuel assemblies of the core, the core region is modeled by twelve parallel flow paths; each of those is described by a pipe and two heat structures that consist of 24 axial volumes. There is a single bypass channel which is connected to the lower head. The lower plenum and lower head are divided into four azimuthal sectors and three axial segments. The downcomer is divided into four azimuthal sectors with seven axial segments in the lower downcomer and three axial sectors in the upper downcomer. The flow is pumped from the downcomer to the lower plenum by the circulation pumps. Each reactor coolant pump is modeled explicitly with two pumps connected to the bottom of each of the downcomer azimuthal sectors. The pump speed is given in the input deck, and it determines the total recirculation mass flow rate in the RV. The upper plenum, standpipes, separators, dryers and the steam dome are modeled as normal volumes.

The model of the FW system is divided in four parts, so that the feed-water flow is specified for each of the four radial quadrants of the RV. For each quadrant, a time-dependent volume gives the condition of the FW fluid in terms of pressure and temperature, and a time-dependent junction provides the FW mass flow rate. The feed-water flow is injected into the lower downcomer of the RV.

Each of the eight steam line nozzles in the vessel is modeled and connected to the steam lines. The four steam lines are modeled individually in the same manner with no lumping, although their lengths are somewhat different. Each of the MSIVs and turbine control valves are modeled explicitly. Pressure boundaries are provided for the turbine and steam bypass inlets.
Fig. 3 Nodalization scheme in RELAP5.
(Top) RPV and FW models. (Bottom) FW model.
2.3. The Coupled PARCS/RELAP5 Model

As regards the coupling between the models discussed above, it is necessary to couple properly the RELAP5 nodes of the core region to the PARCS neutronic core mesh. Besides, factors calculated from the ratio between the thermal-hydraulic and the neutronics computational volumes must be provided in an external file named maptab, in such a manner that the transfer of information between the two codes (i.e., fuel temperature, moderator temperature and density from RELAP5 to PARCS; power density from PARCS to RELAP5) is consistently weighted. Two mappings are necessary to be set up: one mapping for the neutronic/hydrodynamic structures, and one mapping for the neutronic/heat structures. The first mapping allows specifying in the neutronic code which fuel temperature, moderator temperature/density from the thermal-hydraulic code needs to be associated with a specific neutronic node. The second mapping allows specifying in the thermal-hydraulic code which fission power needs to be associated with a specific thermal-hydraulic node. In both mappings, weighting factors have to be defined for each of the links between the neutronic and the thermal-hydraulic nodes.

Concerning the radial mapping between thermal-hydraulic and neutronic nodalization, one PARCS neutronic node is used for one fuel assembly, whereas one RELAP5 pipe includes more fuel assemblies. The PARCS neutronic nodes for the radial reflector are associated to the core by-pass channel. In the case of the axial direction, the 24 levels of the RELAP5 core channel are coupled properly to the 25 neutronic levels of the active region of those PARCS fuel assemblies that are contained in that specific RELAP5 channel. In addition, the first axial neutronic node of each fuel assembly (the bottom reflector) is linked to the guide tubes, and the last axial neutronic node (the top reflector) to the upper plenum. The axial neutronic nodes for the radial reflector are associated to the core by-pass channel.

3. VALIDATION OF THE MODEL

3.1. Validation against Traversing In-Core Probe Measurement

The PARCS model of the Forsmark BWR unit was validated against traversing in-core probe (TIP) measurements provided by FKA to Chalmers. The data consist of processed signal from the detectors in the TIP system which are movable inside tubes located in the 36 channels of the LPRM system. The measurement took place at different core burnups with operating power varies from 108% (nominal level) to 120% (uprated level).

Standalone PARCS calculations were performed with actual core loading and control rods configuration included in the model. The results were compared with the measured TIP data. Also a code-to-code benchmark was performed for comparison with the results from SIMULATE-3. Fig. 4 shows the effective multiplication factors calculated by PARCS and their respective differences (in pcm) against SIMULATE-3. PARCS under-predicts the multiplication factor in all calculation points. The differences, however, are not significant as the largest one is still below 100 pcm.
The axial power profiles at uprated power as calculated by PARCS are shown in Fig. 5. Comparisons were made possible with SIMULATE-3 results and the TIP measurement data at the periphery and center of the core. In the periphery, PARCS tends to over-predict the power at lower heights. On the other hand, at the center of the core, PARCS under-predicts the power at higher elevation. In general, PARCS reproduces the power profile in good agreement with measured data as well with the computational results of SIMULATE-3.

Fig. 4 Multiplication factors calculated by standalone PARCS and their respective differences compared to SIMULATE-3.

Fig. 5 Axial power profile as calculated by PARCS and SIMULATE-3 and compared with the TIP measurement data.
3.2. Validation against Turbine Trip Test

The turbine trip test was performed in May 6, 2013 to show how the plant can handle a turbine trip from 120% power in reactor and turbine. The test should indicate how the systems interact at their new working points as the higher power has achieved and how the reactor power will go down from an extended operating range. The test was performed by tripping one of the two turbines and the turbine stop valve was activated. As a result, a pressure wave was generated in the affected steam lines and propagated toward the reactor core. The trip also triggered the turbine bypass valve to open. This situation actuated a partial scram operation, in which two groups of control rods were inserted into the core and controlled flow reduction occurred in the recirculation pumps. The reactor power decreased and later became steady at around 20%.

To simulate the transient, the control rods movement is modeled as boundary condition in the PARCS model as the available version of PARCS in Chalmers cannot use the signals from RELAP5 to trigger rod movement. Regarding the RELAP5 model, exit pressure boundary is set for turbines and steam bypasses, while the corresponding valves are actuated by control variables. Inlet pressure and temperature boundary conditions are set for feedwater as well as flow condition in our current model. Control system to regulate the feedwater valve will be incorporated in future model.

Comparison of the results from the coupled code simulations are presented in Fig. 6. In general, the nuclear power shows very good agreement up to 7.5 seconds and some discrepancies of maximum 5% occurs between 8 to 10 seconds into the transient (Fig. 6a). The good conformance of the calculated result during the initial part of the transient indicates good modeling of the control rods in PARCS, as control rod reactivity dominates the total reactivity. In later phase of the transient, where the control rod reactivity remains constant, the reactivity feedback from the moderator density is dominant in the temporal behaviour, and hence the accuracy of the power calculation is determined from the thermal-hydraulic side.

The steam dome pressure is shown in Fig. 6b. The coupled codes captured the pressure transient quite well when the turbine stop valves were actuated. The maximum pressure is well represented at the beginning of transient. As the pressure decreases to a minimum at around 3 s, the calculated pressure can not follow to the same value. When the steam dome re-pressurized, the coupled codes also provide a similar trend but not as high as in reality. The most probable cause is that the control parameters for the turbine bypass valves and/or the control valves of undisturbed turbines require fine tuning for uprated power operation. The mismatch in the pressure is also reflected in the total steam flowrate (Fig. 6c). Whereas the general trend of the calculated steam flowrate resembles the measured data, some mismatches are visible and consistent with those of steam dome pressure. The calculated total recirculation flow also resembles the measured data as shown in Fig. 6d, but a consistent lag is quite visible, indicating a need of adjustment to the circulation pump parameters.

The radial power distribution in the core is shown in Fig. 7. At the beginning of transient (Fig. 7a), the power is distributed quite evenly and symmetrically. High power is found around the center, while at the periphery the power is quite low. After 10 seconds (at the end of the simulation) asymmetrical power distribution is visible as shown in Fig. 7b. Some fuel assemblies at the north-west and south-west quadrants have much higher power than the rest. The position of the control rods being inserted to the core for the partial scram operation is not symmetrical; hence the resulting power distribution is quite understandable.
Fig. 6 Comparison of several key parameters.

Fig. 7 Radial assembly power distribution calculated by PARCS.
4. CONCLUSIONS

Development of the coupled PARCS/RELAP5 model for Forsmark BWR has been performed. The model was then validated, first as a standalone PARCS model, against TIP measurement data. The PARCS results show good agreement with the measured data. The coupled model was then validated against a turbine trip transient test. It has been demonstrated that the coupled model was able to capture the main features of the transient with a sufficient level of accuracy. Some discrepancies still persist, indicating the need of fine-tuning of some parameters in the model, especially related to the control systems.

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REFERENCES
