

**Proceedings of the International Conference on Physics of Reactors
(PHYSOR 2014)**

September 28-October 3, 2014, Kyoto, Japan

(Eds.) Kenya SUYAMA, Takanori SUGAWARA, Kenichi TADA, Go CHIBA
and Akio YAMAMOTO

Nuclear Science and Engineering Center
Sector of Nuclear Science Research

JAEA-Conf

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Proceedings of the International Conference on Physics of Reactors (PHYSOR2014)
September 28-October 3, 2014, Kyoto, Japan

(Eds.) Kenya SUYAMA, Takanori SUGAWARA, Kenichi TADA, Go CHIBA^{*1}
and Akio YAMAMOTO^{*2}

Nuclear Science and Engineering Center,
Sector of Nuclear Science Research,
Japan Atomic Energy Agency
Tokai-mura, Naka-gun, Ibaraki-ken

(Received December 15, 2014)

Japan Atomic Energy Agency organized an international conference "PHYSOR2014" on the reactor physics which is one of basic researches in the nuclear engineering, in cooperation with Research Reactor Institute of Kyoto University.

"PHYSOR" is the world's largest scale international conference in the reactor physics field. It originates in the conference held in Marseille, France in 1990, which originally had been organized in the United States as a Physics of Reactors Topical Meeting of the reactor physics division of the American Nuclear Society every two years.

More than 500 papers had been submitted and finally 472 papers were presented in the conference after the paper review process. This report contains the presented papers in CD-ROM, which the PHYSOR organizing committee has decided to publish in an official JAEA report with the permission by authors, except for several selected papers to be published in the Journal of Nuclear Science and Technology of the Atomic Energy Society of Japan.

Keywords: Reactor Physics, International Conference, PHYSOR, Proceedings

*1 Hokkaido University

*2 Nagoya University

炉物理国際会議（PHYSOR2014）論文集

2014年9月28日～10月3日、京都

日本原子力研究開発機構 原子力科学的研究部門 原子力基礎工学研究センター
(編) 須山 賢也、菅原 隆徳、多田 健一、千葉 豪^{*1}、山本 章夫^{*2}

(2014年12月15日 受理)

日本原子力研究開発機構は、京都大学原子炉実験所と共に原子力研究における基礎基盤研究である原子炉物理分野を対象とした炉物理国際会議「PHYSOR2014」を開催した。「PHYSOR」とは、米国原子力学会炉物理部会(ANS/RPD)が2年毎に米国内で開催している炉物理特別会合(Physics of Reactors Topical Meeting)を1990年に「PHYSOR」と命名して仏国(マルセイユ)において開催した会議を起源とする、当該分野における世界最大規模の国際会議である。本会議には総計500件以上の論文が投稿され、査読審査を経て最終的に472件の発表が行われた。本報告書はPHYSOR2014で発表された論文のうち、日本原子力学会欧文誌へ掲載予定のものを除き、組織委員会がJAEAの正式な報告書への掲載を決定して著者の同意が得られたものをCD-ROMに取り纏めたものである。なお、論文は付録のCD-ROMに掲載している。

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*2 名古屋大学

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1. Foreword

The organizing and technical program committees of PHYSOR2014 cordially welcome you. This is the second PHYSOR conference in Japan - the PHYSOR conference returns to Japan after 18 years since it was held in Mito in 1996, hosted by the former Japan Atomic Energy Research Institute (JAERI). For PHYSOR2014, the Japan Atomic Energy Agency (JAEA) and Kyoto University Research Reactor Institute are the official hosts of the conference. A broad selection of people from Japanese universities and industries participate in the organizing committee for the preparation of PHYSOR2014. The technical program committee consists of international experts and has an important role, including high quality reviews for submitted papers.

We would like to host you in the spirit of “Omotenashi,” a spirit of selfless hospitality of Japan, and we would like to provide a well-organized, technically rich, and informative meeting, continuing the traditions of the previous successful PHYSOR meetings, e.g., Mito in 1996, Long Island in 1998, Pittsburgh in 2000, Seoul in 2002, Chicago in 2004, Vancouver in 2006, Interlaken in 2008, Pittsburgh in 2010, and Knoxville in 2012.

The main theme of PHYSOR2014 is the role of reactor physics towards a sustainable future. Japan has the experience of a severe accident, at the Fukushima-Daiichi Nuclear Power Station due to the massive Tsunami triggered by the earthquake, offshore of the north-east part of Japan on March 11, 2011. Many local residents are still evacuated from the region around the Fukushima-Daiichi NPS. It is one of the significant impacts of a nuclear accident. A visit to the Fukushima-Daiichi NPS is planned as a technical tour of the conference. You can directly observe the current state of affairs at the Fukushima-Daiichi NPS through this tour.

We believe that we should reconfirm that safety shall be the first priority in the utilization of nuclear power. Only the reactor physics can explain “why it is a nuclear reactor,” i.e., the fundamental mechanism of the fission chain reaction. In this context, it is clear that nuclear reactor physics is a very important and fundamental area for the safety of nuclear power. During this conference, we would like to reconfirm the importance of reactor physics in the nuclear engineering discipline with you.

We think PHYSOR2014 also provides you a very good opportunity to experience Japan. Kyoto is an impressive, historic city and the conference venue is located near the center of the historic area of Kyoto. Many beautiful shrines, temples, historic architectures, and museums are located within walking distance from the conference venue. You can experience not only the traditional Japanese culture, but also “cool” Japan in Kyoto, including fine dining.

We hope that the PHYSOR2014 conference will be fruitful for you from various aspects.

Shigeaki Okajima and Ken Nakajima
General Chairs

Akio Yamamoto
Technical Program Chair

2. Acknowledgement

The organizers would like to express their heartfelt appreciation to the following organizations and individuals:

- Financial sponsors for their invaluable contributions to support our conference.
- Co-sponsoring technical societies worldwide who publicize our conference.
- International experts who contributed by recruiting technical papers, by reviewing conference papers, by chairing the technical sessions, and/or by being willing to be jury for the best poster and best student paper awards.
- The plenary, banquet, and luncheon speakers, who use their invaluable time to make their address during our conference.
- Organizers of the special sessions and workshops, who provide an invaluable opportunity for technical information exchange on specific specialized topics.
- The hosts of the technical tours: the Kyoto University Research Reactor Institute, the Monju fast reactor of JAEA, the Kumatori Works of Nuclear Fuel Industries, Ltd. and Tokyo Electric Power Company.

We personally would like to offer our special thanks to the members of the organizing and technical program committees who have been doing a lot of work to prepare PHYSOR2014 during the approximately two years since Kyoto was selected as the conference location for PHYSOR2014 in the ANS RPD meeting at the 2012 ANS annual meeting.

Shigeaki Okajima and Ken Nakajima
General Chairs

Akio Yamamoto
Technical Program Chair

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Technical Program Chair

Akio Yamamoto Nagoya University

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Imre Pazsit	Chalmers University of Technology
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J. Zerr

W.S. Yang
I. Zmijarevic

K. Yokoyama
W. Zwermann

K. Yoshioka

R. Yoshioka

6. Track Leaders

Track 1 Reactor Analysis Methods

C. Demazière	H. G. Joo	R. Sanchez	K. Smith	M. Tojo
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Track 2 Deterministic Transport Theory

Y. Azmy	N. Z. Cho	F. Rahnema	K. Yamaji
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Track 3 Monte Carlo Methods

T. Kitada	J. Leppänen	H. J. Shim	K. Wang
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Track 4 Verification, Validation and Uncertainty Analysis

H. Abdel-Khalik	G. Chiba	W. F. G. van Rooijen
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Track 5 Nuclear Criticality Safety

J. Bess	T. Endo	T. Yamamoto
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Track 6 Reactor Physics Experiments

P. Blaise	H. Unesaki
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Track 7 Reactor Concepts and Designs

J. Lee	C. Liangzhi	B. Petrovic	N. Takaki
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Track 8 Reactor Operation and Safety

F. Franceschini	T. Mitsuyasu	G. Sjoden
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Track 9 Transient and Safety Analysis

Y. Ban	S. Dulla	K. Ivanov	P. Ravetto
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Track 10 Nuclear Data

S. Chiba	R. Jacqmin	Y. O. Lee
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Track 11 Research Reactors and Spallation Sources

G. van den Eynde	I. L. Montoya	K. Nishihara
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Track 12 Fuel Cycle and Actinide Management

T. Kim	K. Tsujimoto	A. Worrall
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Track 13 Radiation Applications and Nuclear Safeguards

Y. Kitamura	S. A. Pozzi
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Track 14 Education in Reactor Physics

B. Forget	T. Kameyama	T. Kozlowski
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Track 15 Research Related to Fukushima Accident

A. Haghigiat	S. Kosaka	K. Suyama
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SS1 Molten Salt Reactors

I. Pazsit

SS2 Reactor Physics and Criticality Safety Activities in OECD/NEA Working Party

M. DeHart

SS3 Hybrid Particle Transport Methods for Solving Complex Problems in Real-Time

A. Haghishat F. Rahnema

SS4 Advanced Geometry Processing in Deterministic and Monte Carlo Methods

H. J. Shim

SS5 Multiscale, Multiphysics Approaches in Nuclear Science and Engineering Applications

M. DeHart R. C. Martineau

SS6 Nuclear Criticality Safety of Fuel Debris

N. Takaki

SS7 Control Rod Withdrawal Tests Performed during the PHENIX End-of-Life Experiments

S. Monti

SS8 Reactor Physics of Non-Traditional LWR Fuel Design

B. Petrovic

7. Selection Committee Members

7.1 Selection Committee Members for Best Poster Award

Cao Liangzhi (Xi'an Jiaotong Univ.)	Shinya Kosaka (MHI)
Patrick Blaise (CEA)	Yonghee Kim (KAIST)
Benoit Forget (MIT)	Christophe Demazière (Chalmers Univ. of Tech.)

7.2 Selection Committee Members for Best Student Award

Hyung Jin Shim (Seoul National Univ.)	Andrew Worral (ORNL)
Kan Wang (Tsinghua Univ.)	Wilfred van Rooijen (Univ. of Fukui)
Fausto Franceschini (Westinghouse)	Aldo Dall'Osso (AREVA)

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KYOTO UNIVERSITY
The OECD Nuclear Energy Agency

10. List of Papers

The CD-ROM includes the full papers which were presented in the conference, except for;

1. papers recognized as “canceled” papers by organizing committee,
2. “selected papers” for the publication in Journal of Nuclear Science and Technology of the Atomic Energy Society of Japan, and
3. papers requested by authors to exclude them from JAEA-Conf.

In the conference, there were few presentations by the guest speakers. In this case, paper submission was not required.

The list of papers is shown in following section track by track.

10.1 Track 1 Reactor Analysis Methods

Oral Session

Hitachi's Advanced Technologies

Guest Speaker: H.Soneda

Hitachi-GE Nuclear Energy, Japan

Development of Enhanced SPH Method for Pin-by-Pin Core Calculations ***

S.Takeda, T.Ushio, Y.Ohoka, Y.Kodama

Nuclear Fuel Industries, Ltd., Osaka, Japan

Study on Cross Section Correction Using SPH Method for a Whole Core Heterogeneous MOC Calculation

A.Giho(1), E.Yoshida, K.Miyawaki(2), K.Ohori, Y.Umebara(1), T.Takeda(3)

1)Shikoku Electric Co., Inc., Kagawa, Japan, 2)Yonden Engineering Co., Inc., Kagawa, Japan,

3)Shikoku Instrumentation Co.,Inc., Kagawa, Japan

Asymptotic, Multigroup Flux Reconstruction and Consistent Discontinuity Factors **

T.J.Trahan(1,2), E.W.Larsen(2)

1)Los Alamos National Laboratory, New Mexico, USA, 2)University of Michigan, Michigan, USA

* not included, ** not included for publication in JNST, *** not included by request of authors

A Posteriori Reconstruction of the Flux Profile in the Case of Localized Axial Heterogeneities: An Application to the Modeling of PWR Spacer Grids

E.Girardi, A.Aktogu, H.Leroyer, C.Meriot

Electricité de France-R&D, Clamart, France

Effects of Advanced Radial Submeshing Methods on Pin Power Reconstruction for an EPR Core Design

P.Mala, S.Canepa, H.Ferroukhi, A.Pautz

Paul Scherrer Institute, Switzerland

Study on Robust Energy Group Structure to Spectral Interference for PWR Pin-By-Pin Core Analysis

S.Wada, T.Kitada(1), S.Takeda, T.Ushio(2)

1)Osaka University, Osaka, Japan, 2)Nuclear Fuel Industries, Ltd., Osaka, Japan

Application of the Efficient Consistent Spatial Homogenization Method in Neutron Transport Theory to a Gas Cooled Thermal Reactor Problem

S.Yasseri, F.Rahnema

Georgia Institute of Technology, Atlanta, USA

Application of the Hybrid Diffusion-Transport Spatial Homogenization Method to a High Temperature Test Reactor Benchmark Problem

G.Kooreman, F.Rahnema, S.Yasseri

Georgia Institute of Technology, Atlanta, USA

Normalization Methods for Diffusion Calculations with Various Assembly Homogenizations

C.Brosselard, H.Leroyer, M.Fliscounakis, E.Girardi, D.Couyras

EDF R&D/Sinetics, Clamart, France

On the Practical Feasibility of Continuous-Energy Monte Carlo in Spatial Homogenization^{**}

J.Leppänen(1), R.Mattila(2)

1)VTT Technical Research Centre of Finland,VTT, Finland, 2)Finnish Radiation and Nuclear Safety Authority, Helsinki, Finland

A Dynamic Homogenization Model for Pebble Bed Reactors^{**}

M.Grimod(1), R.Sanchez, F.Damian(2)

1)Fraz. Arpuilles-Aville, Aosta, Italy, 2)CEA de Saclay, Gif-sur-Yvette, France

* not included, ** not included for publication in JNST, *** not included by request of authors

Homogenization of the Step Characteristic Scheme in Phase Space

D.Anistratov, J.Jones

NC State University, North Carolina, USA

Spatial Rehomogenization of Cross Sections and Discontinuity Factors for Nodal Calculations

A.Dall'Osso

AREVA NP, Paris, France

Research Reactor In-Core Fuel Management Optimisation Using the Multiobjective Cross-Entropy Method

E.B.Schlunz, P.M.Bokov(1), J.H.Van Vuuren(2)

1)South African Nuclear Energy Corporation SOC Ltd (Necsa), South Africa, 2)Stellenbosch University, South Africa

Transient Cycle Fuel Management Optimization of a Pressurized Water Reactor

T.K.Park(1), H.G.Joo, H.J.Shim, C.H.Kim(2), J Il Yoon(3)

1)FNC Technology Co. Ltd., Yongin-si, Korea, 2)Seoul National University, Seoul, Korea, 3)KEPCO Nuclear Fuel Co., Daejeon, Korea

A Multi-Level Parallel Computation of Reactor Cores Using GPU for Loading Pattern Optimization

T.Okubo, T.Endo, A.Yamamoto

Nagoya University, Nagoya, Japan

Exact-to-Precision Generalized Perturbation Theory for Reactor Design Calculations

C.Wang, H.S.Abdel-Khalik

North Carolina State University, North Carolina, USA

Depletion GPT-Free Sensitivity Analysis of the TMI Reactor Eigenvalue Model

C.Kennedy, H.Abdel-Khalik

North Carolina State University, North Carolina, USA

The "Virtual Density" Theory of Neutronics: A Generic Method for Geometry Distortion Reactivity Coefficients

M.Reed(1), K.Smith, B.Forget(2)

1)Terra Power, Washington, USA, 2)Massachusetts Institute of Technology, Massachusetts, USA

* not included, ** not included for publication in JNST, *** not included by request of authors

Watts Bar Unit 1 Cycle 1 Zero Power Physics Tests Analysis with VERA-CS

J.C.Gehin, A.T.Godfrey, T.M.Evans, S.P.Hamilton(1), F.Franceschini(2)

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AP1000® PWR Reactor Physics Analysis with VERA-CS and KENO - Part I: Zero Power Physics Tests

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AP1000® PWR Reactor Physics Analysis with VERA-CS and KENO - Part II: Power Distribution

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Solution of the BEAVRS Benchmark Using the nTRACER Direct Whole Core Transport Code^{**}

M.Ryu, Y.S.Jung, H.H.Cho, H.G.Joo

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Calculation of the Pressure Vessel Fluence in the Hungarian VVER-440 Plants for the Lifetime Extension

G.Hegyi, G.Hordósy, A.Keresztúri, C.Maráczy, E.Temesvári

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Automatic Construction of a Simplified Burn-Up Chain Model by the Singular Value Decomposition^{***}

T.Kajihara, M.Tsuji, G.Chiba, Y.Kawamoto, T.Narabayashi(1), Y.Ohoka, T.Ushio(2)

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Generation of Simplified Burnup Chain Using Contribution Matrix of Nuclide Production^{***}

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Important Fission Product Nuclides Identification Method for Simplified Burnup Chain Construction^{**}

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Application of Backtracking Algorithm to Depletion Calculations

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Modeling the Cross Section of Gadolinia Pins in the Depletion for Pin-By-Pin Core Calculations

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The Optimized Algorithm for the Microscopic Depletion Model in the COCAGNE Core Code

A 2-Level Core Partitioning Approach

M.Guillo, D.Couyras, F.Fevotte, F.Hoareau

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Modeling of Shutdown Cooling Reactivity Effects with SIMULATE

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Implementation and Verification of the SDM in the TITAN 3-D Sn Transport Code

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Transport Core Solver Validation for the ASTRID Conceptual Design Study with APOLLO3®

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Methodology Assessment for the Evaluation of the Coolant Void Worth in Sodium Fast Reactors with a Low Void Effect Core Design

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Results of Verification of Computer Codes Used for Analysis of BN-1200 Reactor Core Neutronics*

S.Belov, M.Farakshin, A.Kiselyov, E.Marova(1), P.Alekseev, V.Boyarinov, P.Fomichenko, V.Nevinitsa, A.Timoshinov, M.Zizin(2), I.Malysheva, A.Peregudov, K.Raskach, M.Semenov, V.Stogov, A.Tsibulya(3)

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APOLLO3® Based Method for 3D Warped Cores Calculations – Application to Flowering Tests of PHENIX

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High Order Source Approximation for the EFEN Method

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Extension of Linear Source MOC Methodology to Anisotropic Scattering in CASMO5

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Finite Difference Equations for Neutron Flux and Importance Distribution in a Heterogeneous Reactor without Homogenization and Diffusion Approximation

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Efficient Subspace Construction for Reduced Order Modeling in Reactor Analysis

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Variational Acceleration of Fission Source Iteration for Subcritical Source-Driven Systems

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An Incident Flux Coupling Calculation Study for Nodal Method and Monte Carlo Method

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Modernization Enhancements in SCALE 6.2

B.T.Rearden, R.A.Lefebvre, J.P.Lefebvre, K.T.Clarno, M.A.Williams, L.M.Petrie,

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SALOME-CORE Platform: Uses for EDF R&D Neutronic Studies

H.Leroyer, G.Boulant, A.Calloo, Y.Pora, M.Fliscounakis, T.Clerc, R.Barate

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A Steady-State Core Analysis Code for the Modeling of Accelerator-Driven Subcritical Reactors

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Good Practice in Development of Advanced Assembly/Core Calculation Methods and Implementations of AEGIS/SCOPE2

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Validation of LANCR01/AETNA01 BWR Code Package against FUBILA MOX Experiments and Fukushima Daiichi Nuclear Power Plant Unit 3 MOX Core

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Resonant Upscattering Effects on ^{238}U Absorption Rates

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The Up-Scattering Treatment in the Fine-Structure Self-Shielding Method in APOLLO3®

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Target Motion Sampling Temperature Treatment Technique with Track-Length Estimators in OpenMC - Preliminary Results

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Problem-Dependent Doppler Broadening of Continuous-Energy Cross Sections in the KENO Monte Carlo Computer Code

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Verification of Doubly-Heterogeneous Self-Shielding Method Based on Equivalence Theory

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Quantification of Resonance Interference Effect for Multi-Group Effective Cross-Section in Lattice Physics Calculation

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An Asymptotic Homogenized SP₂ Approximation to the Boltzmann Equation. I. Derivation

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An Asymptotic Homogenized SP₂ Approximation to the Boltzmann Equation. II. Discontinuity Factors and Numerical Testing

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Generalized and Standard Multigroup Neutron Diffusion Equation Eigenvalue Problem with the Finite Volume Method

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A Generalized Multigroup Method Based on Finite Elements

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An Asymptotic, Homogenized, Anisotropic, Multigroup Diffusion Approximation to the Neutron Transport Equation

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Accuracy of the Linear Discontinuous Galerkin Method for Reactor Analysis with Resolved Fuel Pins

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Revisiting the Clio Perturbative Approach for Analyzing Systems in Fundamental Mode Conditions

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A Generalization of λ -Mode Xenon Stability Analysis

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Preliminary Study of the Impact of Xe-135m on the PCR of CANDU

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Verification of the Spectral History Correction Method with Fully Coupled Monte-Carlo Code BGCore

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Core Neutronics Methodologies Applied to the MOX-Loaded KAIST 1A Benchmark: Reference to

Industrial Calculations

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The Role of the Eigenvalue Separation in Reactor Dynamics and Neutron Noise Theory

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Investigation of Conditional Transport Update in Method of Characteristics Based Coarse Mesh Finite Difference Transient Calculation

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Implementation of an a Priori Time Step Estimator for the Multigroup Neutron Diffusion Equation in Asynchronously Coupled RELAP5-3D

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Development of a Three-Dimensional Kinetics Code for Commercial-Scale FBR Full Core Analysis

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Use of Adjoint Functions for Comparing Measured and Calculated Parameters in the Subcritical Systems

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Development of the Neutron Source Evaluation Method and Predictor of SRM/SRNM Count Rate in BWR Simulator^{**}

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Poster Session

A Parametric Study and Comparison of BWR Fuel Depletion Calculations Using CASMO-4, MCNPX, and SCALE/TRITON

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Neutron Noise Induced by Fluctuations of the Boric Acid Content in Pressurized Water Reactors

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Application of Westinghouse NEXUS/ANC9 Cross-Section Model for PWR Accident Analyses

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Numerical Dispersion and Dissipation Analysis of Nodal Expansion Method

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The Integration of Control Rod Calculation and VSOP

J.Guo, F.Li, C.Hao, H.Zhang, X.Zhou, K.Fan, L.Wang, J.Lu

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Reflector Modelling with Multi-Group Nodal Equivalence Theory for the SAFARI-1 Research Reactor

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Relationship Between Computed ANSI/ANS-5.1 and ORIGEN-S Decay Heat Powers for BWR LOCA Safety Analysis

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A New Method to Measuring the α Eigenvalue of a Subcritical Reactor System

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Progress towards an Accurate Lattice-Homogenization Technique for Pressure-Tube Supercritical Water Cooled Reactor Neutronic Calculations

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Proposal of Subcritical PWR Core Benchmark Problems

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A New Monte Carlo-Deterministic Two-Step Method for Fast Reactor Diffusion Analysis ***

W.Heo, Y.Kim

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Explicit Transverse Leakage Treatment Using an Analytic Basis Function Expansion

S.A.Thompson, K.N.Ivanov

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Method for Calculation Capturing Reactions Contribution to Total Energy Release in Nuclear Reactors

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A Preliminary Analysis of the Accuracy of Homogenized 2D Cross Section in 3D Nodal Calculations for BWRs*

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Deterministic Lattice Code Development at UNIST

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Implications of Mesh Refinement in Lattice Physics on BWR Core Analysis and Nuclear Design

P.Forslund Guimarães, D.Bourgin, P.Olivius, J.Aurén

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Generating Multigroup Data Stochastically for a Highly Heterogeneous VHTR Problem

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Reference Solution for Cross Section Parametrization

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Development of a Generalized Cross Section Library Applicable to Various Reactor Types^{*}

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Monte Carlo Analysis of Doppler Reactivity Coefficient for UO₂ Pin Cell Geometry

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Assessment of the Depletion Capability in MPACT

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Monte Carlo Modelling of VR-1 Reactor Core

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Assessment of the 2D/1D Implementation in MPACT

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Coupled Neutronics and Thermal-Hydraulic Solution of a Full-Core PWR Using VERA-CS

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Assessment of the WIMS9A/PARCS/TRACE Code System for Power Density Calculations of the Westinghouse AP1000TM Reactor

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Feasibility of Nodal Equivalence Theory Using Functionalized Discontinuity Factors

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The Multigroup Neutronics Model of NuStar's 3D Core Code EGRET

S.Zhang, D.Lu, T.Wang

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Simulation of Watts Bar Initial Startup Tests with Continuous Energy Monte Carlo Methods

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10.2 Track 2 Deterministic Transport Theory

Oral Session

A Collision Probability Based Method to Compute Cross Sections Sensitivities for the Subgroup Self-Shielding Technique

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Improvement of a Convergence Technique for MOC Calculation with Large Negative Self-Scattering Cross Section

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Boundary Acceleration Techniques for CMFD-Accelerated 2D-MOC

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A Low Order Nonlinear Transport Acceleration Scheme for the Method of Characteristics ^{**}

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p-CMFD Acceleration and Nonoverlapping Local/Global Iterative Transport Methods with 2-D/1-D Fusion Kernel

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Application of the SDD-CMFD Acceleration Method to Parallel 3-D MOC Transport

B.Kochunas, B.W.Kelley, S.G.Stimpson, E.Larsen, T.J.Downar

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Iterative Properties of the Integral Transport Matrix Method for the DD Scheme in 2D Cartesian Geometry

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Neutron Leakage Treatment in Reactor Physics: Consequences for Predicting Core Characteristics

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Revisit Boundary Conditions for the Self-Adjoint Angular Flux Formulation

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Accuracy Preserving Surrogate for Neutron Transport Calculations

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Axial Transport Solvers for the 2D/1D Scheme in MPACT

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Development of Legendre Expansion of Angular Flux Method for 3D MOC Calculation

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Benchmark on Deterministic Time-Dependent Transport Calculations without Spatial Homogenisation

V.F.Boyarinov, A.E.Kondrushin, P.A.Fomichenko

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Coarse-Grained Parallelism for Full-Core Transport Calculations

R.Lenain, E.Masiello, F.Damian, R.Sanchez

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Parallel Performance Results for the OpenMOC Method of Characteristics Code on Multi-Core Platforms

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Making More Precise the Surface Pseudosources Method for RBMK Cluster Cells

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Phase Space Bases for Response Matrix Methods

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The Drift Diffusion Limit of Thermal Neutrons: Theoretical and Numerical Results

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Adequacies of Different Convergence Accuracy Measures in Full-Core Nodal Flux Computations ***

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Flexible Semi-Analytical Calculation Method of Escape Probability

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Corrected Diamond Difference Method for Coupling from the Method of Characteristics to Discrete Ordinates

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Energy Multigroup Spectral Green's Function Constant Nodal Method for Fixed-Source S_N Problems in X,Y-Geometry

W.A.Menezes, H.A.Filho, R.C.Barros

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Poster Session

The Application and Performance of ACMFD Acceleration in 2D/3D Full Core MOC Transport Fuse Method

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A Coupling Method of Subgroup and Wavelet Expansion for the Resonance Parameter Calculation

H.Wu, L.He, Y.Li, T.Zu

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Verification of Ray Effect Elimination Module in the Transport Code ARES

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10.3 Track 3 Monte Carlo Methods

Oral Session

Unstructured Mesh Based Multi-Physics Interface for CFD Code Coupling in the Serpent 2 Monte Carlo Code

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Analysing the Statistics of Group Constants Generated by Serpent 2 Monte Carlo Code

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Theoretical Prediction on Underestimation of Statistical Uncertainty for Fission Rate Tally in Monte Carlo Calculation ***

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Analysis of Tally Correlation in Large Light Water Reactors

B.R.Herman, B.Forget, K.Smith(1), P.K.Romano, T.M.Sutton, D.J.Kelly, III, B.N.Aviles(2)

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Higher-Mode Applications of Fission Matrix Capability for MCNP

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A Symmetric View Hiding the Ugly Truth

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A Monte Carlo Method for Prompt and Delayed Alpha Eigenvalue Calculations

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Geometry Navigation Acceleration Based on Automatic Neighbor Search and Oriented Bounding Box in Monte Carlo Simulation

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Continuous-Energy Monte Carlo Methods for Calculating Generalized Response Sensitivities Using TSUNAMI-3D

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Enhancements in Continuous-Energy Monte Carlo Capabilities for SCALE 6.2

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Leakage-Corrected Fast Reactor Assembly Calculation with Monte-Carlo Code TRIPOLI4® and Its Validation Methodology

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Impact of Nearest Neighbor Distribution of Fuel Particle on Neutronics Characteristics in Statistical Geometry Model***

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Large-Scale Monte Carlo Calculations with Thermal-Hydraulic Feedback

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Sodium Void Reactivity Effect Analysis Using the Newly Developed Exact Perturbation Theory in Monte-Carlo Code TRIPOLI-4®

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Monte Carlo Perturbation Analysis on Isothermal Temperature Reactivity Coefficient of Light-Water Moderated and Reflected Critical Assembly

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Monte Carlo and Thermal-Hydraulic Coupling via PVMEXEC

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Perturbation Based Monte Carlo Criticality Search in Density, Enrichment and Concentration ***

Z.Li, K.Wang, J.Deng

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Monte Carlo Perturbation Method for Geometrical Uncertainty Analysis Using McCARD

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XSBench - The Development and Verification of a Performance Abstraction for Monte Carlo Reactor Analysis

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Development of Neutron Current Connection Method for Whole Core Analysis Based on Monte Carlo Method

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Analysis of Select BEAVRS PWR Benchmark Cycle 1 Results Using MC21 and OpenMC

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Monte Carlo Neutronics Analysis of Sodium-Cooled Fast Reactor Benchmark with OTF Temperature and Burnup Treatment

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Verification of Coupled 3D Fuel Cycle Analysis with Stable Monte Carlo Based Code, BGCore, against the Nodal Diffusion DYN3D Code

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Domain Decomposition and Terabyte Tallies with the OpenMC Monte Carlo Neutron Transport Code

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Effects of Stochastic Noise on a Three-Dimensional Monte Carlo Depletion Analysis of the H.B. Robinson Reactor

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Efficient Estimation of Adjoint-Weighted Kinetics Parameters in the Monte Carlo Wielandt Calculations

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Higher-Order Chebyshev Rational Approximation Method (CRAM)

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Efficiency and Accuracy Evaluation of the Windowed Multipole Direct Doppler Broadening Method^{**}

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Diffusion Monte Carlo Method with Transport Corrections^{*}

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Generation of One-Group Self Shielded Cross Sections with Multi-Group Approach for Monte Carlo Burnup Codes

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Poster Session

3-D Monte Carlo Neutron-Photon Transport Code JMCT and Its Algorithms

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Monte Carlo Calculation of Neutron Generation Time in Critical Reactor and Subcritical Reactor with an External Source

L.Zhang, X.Jiang, X.Zhang, P.Hu, T.Ma, L.Chen, W.Chen

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Neutron Channels Shield Design Analyses of KIPT Neutron Source Facility

Z.Zhong, Y.Gohar

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Development of a New Convergence Criterion for Monte Carlo Simulation with Thermal-Hydraulics Feedback

X.Wu, T.Kozlowski

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JCOGIN: A Parallel Programming Infrastructure for Monte Carlo Particle Transport

B.Zhang, G.Li, L.Deng, Y.Ma, D.Shangguan, A.Zhang, X.Cao, Z.Mo

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Void Transit Time Calculations by Neutron Noise of Propagating Perturbation Using Complex-Valued Weight Monte Carlo

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Simulating Fast Transients with Fuel Behavior Feedback Using the Serpent 2 Monte Carlo Code

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Status of Monte Carlo Code Development at UNIST

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Criticality Benchmarking of ANET Monte Carlo Code

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Effective Diffusion Homogenization of Cross Sections with the Monte Carlo Method*

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Estimating Local In- and Ex-Core Responses within Monte Carlo Source Iteration Eigenvalue Calculations

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10.4 Track 4 Verification, Validation and Uncertainty Analysis

Oral Session

Validation and Benchmarking of Calculation Methods for Photon and Neutron Transport at Cask Configurations

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Recent Advances in the V&V of the New French CEA APOLLO3® Neutron Transport Code : Benchmarks Analysis of the Flux Solvers

J-M.Palau, P.Archier, J.-F.Vidal, G.Rimpault, B.Roque, P.Bourdot, Y.Penelieu, G.Truchet, C.De Saint Jean

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DeCART Code Verifications by Numerical Benchmark Calculations of HTTR

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Development and Verification of Three-Dimensional Hex-Z Burnup Sensitivity Solver Based on Generalized Perturbation Theory

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Validation of HELIOS for ATR Core Follow Analyses

S.E.Bays, E.T.Swain, D.S.Crawford, D.W.Nigg

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PERSENT: Need of a Deterministic Code for Sensitivity Analysis in 3D Geometry and Transport Theory

G.Aliberti, M.A.Smith

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Benchmark Calculation with MOSRA-SRAC for Burnup of a BWR Fuel Assembly

K.Kojima, K.Okumura

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Verification of the COCAGNE Core Code Using Cluster Depletion Calculations

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LWR Fuel Reactivity Depletion Verification Using 2D Full-Core MOC and Flux Map Data

G.A.Gunow, K.S.Smith, B.Forget

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CASMO-4E and CASMO-5 Analysis of the Isotopic Compositions of the LWR-PROTEUS Phase II Burnt PWR UO₂ Fuel Samples

P.Grimm, G.Perret, H.Ferroukhi

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Experimental Validation of Decay Heat Calculations with VESTA 2.1

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Development and Validation of Ad Hoc ORIGEN-ARP Libraries for Very High Burnup UO₂ PWR Fuel with SCALE/TRITON

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Confidence Interval Estimation by Bootstrap Method for Uncertainty Quantification Using Random Sampling Method**

T.Endo, T.Watanabe, A.Yamamoto

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Uncertainty Quantification of Neutronics Characteristics Using Latin Hypercube Sampling Method

K.Kinoshita, T.Endo, A.Yamamoto(1), Y.Kodama, Y.Ohoka, T.Ushio, H.Nagano(2)

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Uncertainty Quantification of BWR Core Characteristics Using Latin Hypercube Sampling Method***

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Applicability of the Cross Section Adjustment Method Based on Random Sampling Technique for Burnup Calculation

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NUSS-RF: Stochastic Sampling-Based Tool for Nuclear Data Sensitivity and Uncertainty Quantification**

T.Zhu, A.Vasiliev, H.Ferroukhi, A.Pautz(1), S.Tarantola(2)

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MOCABA: A General Monte Carlo-Bayes Procedure for Improved Predictions of Integral Functions of Nuclear Data ***

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Pinpower Uncertainty Quantification of LWR-PROTEUS Phase III Experiments

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Quantification of Code, Library and Cross-Section Uncertainty Effects on the Void Reactivity Coefficient of a BWR UO₂ Assembly

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Sensitivity and Uncertainty Analysis of Burnup Reactivity for an Accelerator-Driven System

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Sensitivity/Uncertainty Analysis for BWR Configurations of Exercise I-2 of UAM Benchmark

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Impact of the Fission Yield Nuclear Data Uncertainties in the Pin-Cell Burn-Up OECD/NEA UAM Benchmark

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Deterministic Approach of the Decay Heat Uncertainty Due to JEFF-3.1.1 Nuclear Data Uncertainties with the CYRUS Tool and the DARWIN2.3 Depletion Code

V.Vallet, S.Lahaye, A.Tsilanizara, L.San-Felice, R.Eschbach

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Bias and Uncertainty Assessment of Pressurized Water Reactor Fuel Isotopes Using SCALE

R.N.Bratton, K.N.Ivanov(1), W.A.Wieselquist, M.A.Jessee(2)

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Inventory Uncertainty Quantification and Propagation Using TENDL Covariance Data in FISPACT-II

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Validation of CASMO5 Spent Fuel Isotopes with Decay and Fission Yield Uncertainties

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Uncertainty Propagation and Sensitivity Analysis in the ALEPH Monte Carlo Burnup Code: Applications to Fission Pulse Decay Heat Calculations

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Uncertainty Quantification of Spent Fuel Nuclide Compositions Due to Cross Sections, Decay Constants and Fission Yields

O.Leray, P.Grimm, M.Hursin, H.Ferroukhi, A.Pautz

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Sensitivity and Uncertainty Calculations Methods of Neutronics Parameters in PWR Cores Part I: Theory and Sensitivity Calculations

T.Takeda, B.Foad, H.Katagiri(1), H.Matsumoto, K.Kirimura(2)

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Constrained Quantities in Uncertainty Quantification: Ambiguity and Tips to Follow

Z.Perko, D.Lathouwers, J.L.Kloosterman, T.H.J.J.van der Hagen

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Adjoint-Based Sensitivity and Uncertainty Analysis of Lattice Physics Calculations with CASMO-4

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Nuclear Data Uncertainty Propagation on Power Maps in Large LWR Cores

A.Santamarina

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Are Modeling Uncertainties Properly Considered in Neutronics Data Assimilation Analysis?

P.Athe, H.Abdel-Khalik

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Sensitivity Analysis via Reduced Order Adjoint Method

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Sensitivity and Uncertainty Analysis on Reactor Kinetic Parameters Using Perturbation Theory

C.Bouret, L.Buiron, G.Rimpault

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Selecting Benchmarks for Reactor Calculations

E.Alhassan, H.Sjöstrand.J.Duan, P.Helgesson, S.Pomp, M.Österlund(1), D.Rochman, A.Koning(2)

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OECD NEA Benchmark Database of Spent Nuclear Fuel Isotopic Compositions for World Reactor Designs

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Effective Physics-Based Uncertainty Quantification for ZrH_x Thermal Neutron Scattering in TRIGA

Reactors

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Poster Session

Review of Neutronic Assessments Applied to Small Reactor Core Physics

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A New Neutronics Analysis Code System for Fast Reactors and Validation

T.Takeda, Y.Shimazu

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Uncertainty Analysis of Delayed Neutron Fissile Material Assay Using a Genetic Algorithm

R.P.Kelley, L.M.Rolison, D.Raetz, K.A.Jordan

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IR Approximation for Calculating Sensitivity and Uncertainty of PWR Cells by Taking Account of Self-Shielding Effect

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Uncertainty Quantification of Reactor Kinetics Parameters Using JENDL-4.0 Covariance Data***

G.Chiba, M.Tsuji, T.Narabayashi

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Survey on Effect of Crystal Texture of Beryllium on Total Cross-Section to Improve Neutronic Evaluation in JMTR

N.Takemoto, T.Imaizumi, N.Kimura, K.Tsuchiya(1), J.Hori, T.Sano, K.Nakajima(2)

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Uncertainty and Sensitivity Analysis for an OECD/NEA HTGR Benchmark with XSUSA

A.Aures, K.Velkov, W.Zwermann(1), P.Rouxelin, K.Ivanov(2), J.Lapins, W.Bernnat(3)

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First Verification and Validation Steps of MENDEL Release 1.0 Cycle Code System

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Application of the GRS Method for Estimation of Uncertainties of LMFBR Type Reactor Physics Parameters with Taking into Account Macroscopic Experiments

A.Peregudov, M.Semenov, G.Mantuov, V.Koscheev, K.Raskach, A.Tsibulya

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Benchmarking of Photon and Coupled Neutron and Photon Process of SuperMC 2.0

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In Depth Uncertainty Estimation of the Neutron Computational Tools

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Updated Validation of the PSI Criticality Safety Evaluation Methodology Using MCNPX2.7 and ENDF/B-VII.1

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Validation of Two Monte Carlo Codes for LWR Burnup Calculations

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Benchmarking of DeCART2D against Critical Experiments

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Automated Reactor Records Evaluation Framework

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Monte Carlo Based Equilibrium Cycle Analysis of One-Dimensional Breed and Burn Benchmark Problem

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Verification of the Monte Carlo Code RMC with a Whole PWR MOX/UO₂ Core Benchmark

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An Improved Method for Inverse Uncertainty Quantification for Nuclear Data Assessment

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On How Sensitive the Cross-Section Sensitivity Calculations Are to P_N Order Approximations

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Validation of NuStar's PWR Core Analysis System

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10.5 Track 5 Nuclear Criticality Safety

Oral Session

Nuclear Criticality Safety in the United States: Recent Events, Trends and a Review of the Safety Culture*

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A New OECD/NEA Database of Nuclide Compositions of Spent Nuclear Fuel

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OECD EGBUC Benchmark VIII - Comparison of Calculation Codes and Methods for the Analysis of Small-Sample Reactivity Experiments

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Criticality Calculation of Fuel Debris in Fukushima Daiichi Nuclear Power Station

A.Tsuchiya, T.Kondo, H.Maruyama(1), M.Yamaoka, R.Kimura, Y.Moriki, Y.Hayashi, Y.Takeuchi(2), K.Yamaji, M.Nakano(3), K.Oyama, A.Takagi(4)

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Design of an Efficient Calculation Model of BWR Cold Critical Experiments for Validation

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First Burnup Credit Application for Transport and Storage Cask Using French Experiments^{**}

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Uncertainty Evaluation of Reactivity in Single and Multi-Region TSUNAMI Modeling Analysis for Dry Cask Storage

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Transient Analysis in Super Critical Condition for Several Fuel-Solution Tanks System with Different Layout

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Comparison of Gamma Dose Rate Calculations for PWR Spent Fuel Assemblies

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The KBS-3 Spent Fuel Canister Criticality Calculation during 100,000 Years

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Favorable Features in Kinetics of Fast Reactors with Physically Thick ^{208}Pb -Reflector

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10.6 Track 6 Reactor Physics Experiments

Oral Session

Static Modal Analysis of the Current-to-Flux Subcriticality Monitor for Accelerator-Driven Systems

W.Uyttenhove, P.Baeten, A.Kochetkov, G.Van den Eynde, G.Vittiglio, J.Wagemans(1), D.Lathouwers, J.-L.Kloosterman, T.H.J.J.van der Hagen(2), A.Billebaud, S.Chabod(3), F.Mellier(4), J.-L.Lecouey, F.-R.Lecolley, G.Lehaut, N.Marie(5), X.Doligez(6), M.Carta(7), V.Bécares, D.Villamarin(8)

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12 Years of Franco-Japanese International Programs in EOLE for the Validation of 100%MOX Recycling in LWRs

P.Blaise, P.Alexandre, K.Blandin, JF.Ledoux, JM.Girard, H.Philibert, A.Roche, P.Fougeras, S.Cathala, O.Litaize, A.Santamarina, C.Vaglio, JF.Vidal, N.Thiollay, N.Huot, JP.Chauvin, T.Pont, V.Laval, S.Testanière, B.Vincent

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Application of the Best Representativity Method to a Future PWR Fuel Assembly Calculation Using Four Critical Experiments of Different Facilities

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Characterization of Irradiation Fields for Fuel and Material Irradiation in the Experimental Fast Reactor Joyo

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Reaction Rate, Fission Product Yield, and Rossi-Alpha Measurements Using a HEU Metal, Copper Reflected Critical Assembly^{**}

R.G.Sanchez, J.A.Bounds, T.A.Bredeweg, J.M.Goda, D.K.Hayes, K.R.Jackman, G.E.Mckenzie, W.L.Myers, T.J.Grove

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An Improved Feynman- α Correlation Analysis with a Moving-Bunching Technique

R.Okuda, A.Sakon, S.Hohara, W.Sugiyama, K.Hashimoto

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An Alternative Source Jerk Method Implementation for the Subcriticality Estimation of the VENUS-F Subcritical Core in the FREYA Project

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Pulsed Neutron and Source Jerk Experiments for Reactivity Assessment in Deep Subcritical Configuration: A Case Study within the Framework of the FREYA Project

G.Mila(1), S.Argirò, S.Beolè, M.Masera(2), S.Di Maria, Y.Romanets, P.Teles, P.Vaz(3), M.Osipenko, M.Ripani, P.Saracco(4)

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Reactivity Measurement of the Lead Fast Subcritical VENUS-F Reactor Using Beam Interruption Experiments

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Reactivity Measurements at GUINEVERE Facility Using the Integral k_p Method

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Estimation of the Delayed Neutron Fraction β_{eff} of the MAESTRO Core in MINERVE Zero Power Reactor^{**}

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BNL Metal Fuel Lattice Experiments: Candidates for Reactor Physics Benchmark Evaluation^{***}

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Benchmark Evaluation of the Neutron Radiography (NRAD) Reactor Upgraded LEU-Fuel Core

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CALIBAN and Godiva-IV Measurements Using Helium-3 Detector Systems

J.Hutchinson, A.Sood, M.Smith-Nelson, J.Goda, J.Bounds, W.Myers, T.Cutler, B.Richard, B.Rooney(1),
A.Chapelle, P.Casoli, N.Authier(2)

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Power Spectral Analysis for a Subcritical Reactor System Driven by a Pulsed Spallation Neutron Source

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Equivalency of Open Loop and Closed Loop Reactivity Measurement Techniques

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Overview of the 2014 Edition of the International Handbook of Evaluated Reactor Physics Benchmark Experiments (IRPhEP Handbook)

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Analysis of Tungsten Gray Rods Critical Experiments Using PARAGON with Ultra-Fine Energy Mesh Methodology

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Monte Carlo Assessment of Spatial and Energy Effects in the VENUS-F Subcritical Configurations and Application for Reactivity Determination

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Validation of ORIGEN2 Coupled with JENDL-4.0 Base Libraries for Isotopic Compositions of Irradiated Light Water Reactor Fuels

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The AMMON Experiment in EOLE Facility: A Challenging Program Dedicated to the Experimental Validation of JHR Neutronic and Photonic Calculation Tools^{**}

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Monte Carlo Analysis of Reactivity Effect Measurements in the AMMON Experimental Program Dedicated to JHR Neutron Studies^{**}

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Poster Session

Determination of the ^{58}Ni (n,p) ^{58}Co Reaction Cross Section for both Ground and Isomeric States

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Investigation on Subcriticality Measurement Using Inherent Neutron Source in Nuclear Fuel

T.Shiozawa, T.Endo, A.Yamamoto(1), C.H.Pyeon, T.Yagi(2)

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Benchmark Calculations of Sodium Fast Critical Experiments

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Critical Experiments for BWR Fuel Assemblies with Cluster of Gadolinia Rods

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The Calculation and Measurement of Fast Neutron Reflection in the VVER-1000 Mock-Up Model Placed in the LR-0 Reactor

M.Košťál, B.Jánský, E.Novák, J.Milčák, E.Losa, J.Rejchrt, Z.Lahodová(1), F.Cvachovec(2), M.Veskrna(3)

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Development of Reactivity Meter with Novelty Neutron Source Intensity Evaluation Model for BWR Application

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Boron Carbide Neutron Screen for GRR-1 Neutron Spectrum Tailoring

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Research and Development Activities for Transmutation Physics Experimental Facility in J-PARC

T.Sugawara, H.Iwamoto, K.Nishihara, K.Tsujimoto, T.Sasa, H.Oigawa

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Analysis of TCA Criticality, β_{eff} and β_{eff}/l Using CASMO-4 and CASMO-5

S.Aoki(1), A.Michael, G.Ralph(2)

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Measurement of Subcriticality Using Delayed Neutron Source Combined with Pulsed Neutron Accelerator

T.Misawa, T.Yagi, C.H.Pyeon

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Rossi- α Parameter Measurement of Dalat Nuclear Reactor by Analysis of Cross Power Spectral Density

Obtained from 2 Ion Chambers*

T.M.Nguyen, C.S.Trang

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Analysis of Integral Experiment for Thorium Fuel Cycle at Kyoto University Critical Assembly

Y.Takahashi, T.Sano, J.Hori, H.Unesaki, K.Nakajima

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Studies of Potential for Conversion of the Idaho National Laboratory TREAT Transient Test Reactor to Low-Enrichment Fuel

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10.7 Track 7 Reactor Concepts and Designs

Oral Session

Conceptual Study of a Long-Life Prototype Gen-IV Sodium-Cooled Fast Reactor (PGSFR)***

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Core Design Studies on the Fast Reactor with Flexible Breeding Ratio^{***}

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Enhanced Feedback Effects in Sodium Cooled Fast Reactors Using Moderating Material - The Effect of the Plutonium Content in the Fuel

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Advanced Sodium Cooled Reactor Cores Having Thorium Blankets for Effective Burning of Transuranic Nuclides

W.S.You, S.G.Hong

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PHISICS Improvements and Comparative Study with ERANOS 2.2 on the Gen-IV Lead-Cooled Fast Reactor Concept, ALFRED

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Velocity Characteristic and Stability of Wave Solutions for a CANDLE Reactor with Thermal Feedback^{*}

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SmAHTR-CTC Neutronic Design

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Suppression of Excess Reactivity of Small Long-Life Prismatic HTGR with Passive Decay-Heat Removal

O.Sambuu, T.Obara

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Conceptual Design of a Self-Sustanable Pressurized Water Reactor with Boiling Channels

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Application of the BigT Burnable Absorber to a Soluble Boron-Free PWR Core ***

H.Yu, M.Yahya, Y.Kim

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Axially Homogeneous Thorium Fuel Designs for Transuranic Burning in Reduced- Moderation BWRs

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Neutronic Analysis of a Micro Modular Reactor

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Development of the 900 Second Specific Impulse Carbide Low Enriched Uranium Nuclear Thermal Rocket

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Molybdenum-99 Production in the Oregon State TRIGA Reactor: Analysis of Multiple Smaller Core

Designs Using a New LEU Target as Fuel

A.J.Hummel, T.S.Palmer

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Preliminary Design of the Delft Isotope Production Reactor (DIPR)

J.L.Kloosterman, M.V.Huisman, M.Rohde

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A Study of Safety Core Design on Beam Transient for Accelerator Driven System

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New Inverted Hydride Fuel Design Concept for Pressure Tube Type Super Critical Water Reactors

A.Ahmad, L.Cao, H.Wu

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Preliminary Safety Analysis of a Thorium High-Conversion Pebble Bed Reactor

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Poster Session

Axially Heterogeneous Thorium Fuel Designs for Transuranic Burning in Reduced- Moderation BWRs

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Method Development and Reactor Physics Data Evaluation for Improving Prediction Accuracy of Fast Reactors' Minor Actinides Transmutation Performance

T.Takeda(1), T.Hazama(2), K.Fujimura, S.Sawada(3)

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A Long Life Sodium Cooled Fast Reactor Concept with Radial Shuffling ***

Z.Li, Y.Zheng, L.Cao, H.Wu

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Critical Boron Concentration Reduction Method in a Core Design

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BigT - A New Burnable Absorber Concept for PWR ***

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Preliminary Design of a Spherical Breed/Burn Reactor

E.Y.García-Cervantes, J.L.François, R.C.López-Solís

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Optimization of Ultra-Long Cycle Fast Reactor Core

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The Main Characteristics of the Evolution Project VVER-S with Spectrum Shift Regulation

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Alternative Cores for a Multipurpose Experimental Sodium-Cooled Fast Reactor with U-Zr Fuel

T.Noh, M.H.Kim

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Preliminary Evaluation of Coolant Void Reactivity of a Re-Entrant Channel Pressure-Tube Supercritical Water Cooled Reactor

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10.8 Track 8 Reactor Operation and Safety

Oral Session

Effects of Cross Sections Libraries Parameters on the OECD/NEA Oskarshamn-2 Benchmark Solution

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Evaluation of Operational Experiences and Reactor Physics Tests of MOX Loaded BWR Cores

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Coupling Effects in Large Reactor Cores: The Impact of Heavy and Conventional Reflectors on Power Distribution Perturbations

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Improve the Accuracy of the Power Distribution Reconstruction Using Power Distributions of Different Status as the Fundamental Harmonic

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Advanced Surveillance of Resistance Temperature Detectors in Nuclear Power Plants

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Refined Method for Surveillance and Diagnostics of the Core Barrel Vibrations of the Ringhals PWRs

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Assessment of Flow Induced Vibration Limits in Preliminary I²S-LWR Fuel Designs

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Blockage Index for the Detection of Flow Blockage in a Subassembly of Sodium-Cooled Fast Reactor

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Second Generation Shielding Assemblies - Neutron Flux Impact on Reactor Pressure Vessel and Core Design

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Poster Session

Development and Preliminary Verification of the PWR On-Line Core Monitoring Software System: SOPHORA

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Development of Risk Monitor RiskAngel for Risk-Informed Applications in Nuclear Power Plants

F.Wang, J.Wu, Y.Wu, L.Hu, Y.Li, J.Wang, J.Wang, R.Hu(1), Y.Yin(2), Z.Yang(3)

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Steady-State Subchannel Analysis of Partially Blocked Coolant Channels in a Pool-Type TRIGA Reactor

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10.9 Track 9 Transient and Safety Analysis

Oral Session

Development of the Neutron Kinetics Code for Thermal Molten Salt Reactor

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Study of Neutron Propagation in Multigroup Transport by Space Asymptotic Methods

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Sensitivity Analysis and Performance of the Adiabatic, Theta, and Multigrid Amplitude Function Kinetics Methods in 2D MOC Neutron Transport

S.Shaner, B.Forget, K.Smith

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Computations of Heterogeneous Dilution Transients Using CFX and HEMERA V1

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Prompt Behavior of Generalized-Eigenvalue Point Kinetics Models^{*}

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Recriticality Risk in PWR Spent Fuel Pools

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Demonstration of Fully Coupled Simplified Extended Station Black-Out Accident Simulation with
RELAP-7

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Code Scaling Applicability to a Cold Leg SBLOCA Scenario in a Nuclear Power Plant

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Transient Simulation of Gas Bubble in a Medium Sized Lead Cooled Fast Reactor

C.F.Hellesen, P.Wolniewicz, P.Jansson, A.Håkansson, S.J.Svärd, M.Österlund

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Power Ramp Transient in a Sodium-Cooled Fast Reactor Used for Minor Actinides Transmutation

S.P.Martin, A.Ponomarev, R.Krüßmann, W.Pfrang

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Step towards Integral Validation of Energetic Re-Criticality Prediction for Sodium Cooled Fast Reactor

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SIMMER-III Modeling of Gas Cooled Fast Reactor

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Validation of the Subchannel Code CTF against the Benchmark Data of the OECD/NEA PSBT

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Validation of CASMO5 / SIMULATE-3K Using the Special Power Excursion Test Reactor III E-Core:
Cold Start-Up, Hot Start-Up, Hot Standby and Full Power Conditions

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Validation of the Nodal Kinetics Code System GALAXY/COSMO-K Using the SPERT-III E-Core Experiments

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Propagation of Nuclear Data Uncertainty for a Control Rod Ejection Accident Using the Total Monte-Carlo Method

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Poster Session

Improvement of Space-Time Kinetics Capability in the SNATCH Solver and Comparison to KIN3D/PARTISN Results

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Different Methods to Model the MSLB without Primary Cooling Pumps Using HEMERA V1 System Codes

B.Normand

IRSN, Fontenay-aux-Roses, France

Some Results of Studying of Spatial Kinetics in Fast Reactors

I.S.Panova, E.F.Seleznev, A.A.Belov(1), A.M.Zhukov, I.P.Matveenko(2)

1)Nuclear Safety Institute of Russian Academy of Sciences, Moscow, Russia, 2)Institute for Physics and Power Engineering, Obninsk, Russia

Atucha-2 Obliquely Inserted Control Rods RELAP5-3D/NESTLE Model

R.G.Gonzalez, F.D'Auria(1), C.Parisi(2),M.Pecchia(3), O.Mazzantini(4)

1)GRNSPG, Pisa, Italy, 2)ENEA, Rome, Italy, 3)PSI, Villingen, Switzerland, 4)NA-SA, Buenos Aires, Argentina

The SIMMER/PARTISN Capability for Transient Analysis

M.Marchetti, F.Gabrielli, A.Rineiski, W.Maschek

Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany

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Development of a High-Fidelity Monte Carlo Thermal-Hydraulics Coupled Code System
Serpent/SUBCHANFLOW - First Results

M.Daeubler, J.Jimenez, V.Sanchez

Karlsruhe Institute of Technology, Karlsruhe, Germany

10.10 Track 10 Nuclear Data

Oral Session

COMAC: Nuclear Data Covariance Matrices Library for Reactor Applications

P.Archier, C.De Saint Jean, G.Noguère, O.Litaize, P.Leconte, C.Bouret

CEA Cadarache, Saint Paul-lez-Durance, France

Model-Based Generation of Neutron Induced Fission Yields Up to 20 MeV by the GEF Code

K.Kern, M.Becker, C.Broeders, R.Stieglitz

Karlsruhe Institute of Technology, Leopoldshafen, Germany

Experiments to Improve Uncertainty of the 1st Delayed Neutron Group Abundance in Fast Fissions of ^{238}U
and Sensitivity Studies of the Relative Parameters

H.Chung, K.A.Jordan

University of Florida, Florida, USA

Observation of Neutron Thermalization in Graphite Using the Slowing-Down-Time Technique

A.I.Hawari, B.W.Wehring

North Carolina State University, NC, USA

Measurement of Neutron Capture Cross Section of ^{232}Th in a Low Energy Region

J.Hori, T.Sano, Y.Takahashi, H.Unesaki, K.Nakajima

Research Reactor Institute, Kyoto University, Kumatori, Japan

New Revisions of Reactor Physics Standards

D.Cokinos

Brookhaven National Laboratory, New York, USA

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Impact of the Interpolation Mode on the Secondary Particle Energies for Shielding and Criticality
Benchmarks with TRIPOLI-4® Monte Carlo Code

C.Jouanne, O.Petit

CEA Saclay, Gif-sur-Yvette, France

Impact of the Differences in Nuclear Data on Estimated K-Effective of SFR Cores

N.E.Stauff, G.Aliberti, T.K.Kim, T.A.Taiwo

Argonne National Laboratory, Argonne, USA

Nuclear Data Sensitivity Analysis for Isotopic Generation Using JENDL-4.0, ENDF/B-VII.1 and
JEFF-3.1.1

Y.Kawamoto, G.Chiba, M.Tsuji, T.Narabayashi

Hokkaido University, Sapporo, Japan

Validation of a Pointwise Energy Neutron Cross Section Library Generated by RXSP-BETA2.0 Using
ENDF/B-VII.0

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Criticality Experiments and Analyses of Uranium Zirconium Carbon Nitride LEU Fuel

A.Talamo, Y.Gohar, Z.Zhong(1), S.N.Sikorin, S.G.Mandzik, S.A.Polazau, T.K.Hryharovich,
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Sosny, Minsk, Belarus*

Feedback on ^{239}Pu and ^{240}Pu Nuclear Data and Associated Covariances through the CERES Integral
Experiments**

P.Leconte, G.Truchet, G.Noguere, E.Privas, P.Archier, C.De Saint Jean(1), J.Gulliford(2), D.Hanlon(3)

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3)AMEC, Dorset, UK*

Poster Session

Analysis of Radioactivity Ratios of Fission Product Nuclides Deposited to Soil in Fukushima Dai-Ichi
Nuclear Power Plant Accident***

G.Chiba, Y.Kawamoto, M.Tsuji, T.Narabayashi

Hokkaido University, Sapporo, Japan

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Experimental Uncertainty Estimation on the Effective Capture Cross Sections Measured in the PROFIL Experiments in PHENIX

E.Privas, G.Noguère, C.De Saint Jean, J.Tommasi, P.Archier

CEA Cadarache, Saint-Paul-lez-Durance, France

Updated Multi-Group Cross Sections of Minor Actinides with Improved Resonance Treatment

M.Sohail, O.Shim, M.Kim

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Parameterized Representation of Macroscopic Cross Section for PWR Reactor Considering with 12 Burnable Absorber Fuel Rods in the Fuel Element^{*}

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10.11 Track 11 Research Reactors and Spallation Sources

Oral Session

An Updated Core Design for the Multi-Purpose Irradiation Facility MYRRHA^{**}

G.Van den Eynde, E.Malambu, A.Stankovskiy, R.Fernandez, P.Baeten

SCK-CEN, Belgium

Feasibility Study of Installing a Thermal to 14 MeV Neutron Converter into a Research Nuclear Reactor

L.Snoj, A.Trkov, I.Lengar, A.Kolšek, A.Jazbec, V.Radulović, G.Žerovnik(1), P.Sauvan, F.Ogando, J.Sanz(2)

1)Jozef Stefan Institute, Ljubljana, Slovenia, 2)Universidad Nacional de Educacion a Distancia Ingenieria Energetica, Madrid, Spain

Development and Validation of a New APOLLO2-Based Calculation Scheme Dedicated to Ex-Core Rod Irradiations in the OSIRIS MTR Reactor

F.Chevallier, F.Malouch, S.Santandrea

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Neutronic Designs and Analyses of a New Core-Moderator Assembly and Neutron Beam Ports for the Penn State Breazeale Reactor

D.Ucar, K.Unlu, B.J.Heidrich, K.N.Ivanov, M.N.Avramova

The Pennsylvania State University, Pennsylvania, USA

Design Studies for a Multiple Application Thermal Reactor for Irradiation Experiments (MATRIX)

M.A.Pope, H.D.Gougar(1), J.M.Ryskamp(2)

1)Idaho National Laboratory, Idaho, USA, 2)Retired, USA

Simulated Irradiation of Samples in HFIR for Use as Possible Test Materials in the MPEX (Material Plasma Exposure eXperiment) Facility***

R.Ellis, J.Rapp

Oak Ridge National Laboratory, Oak Ridge, USA

MCNPX Analysis of Delayed Neutron Fraction in Beryllium Reflected Cores

S.Kalcheva

SCK-CEN, Belgium

A Method for Reactivity Monitoring in Subcritical Source-Driven Systems

S.Dulla, M.Nervo, P.Ravetto

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Interpretation of Experimental Measurements on the SC-1 Configuration of the VENUS-F Core

S.Dulla, M.Nervo, P.Ravetto(1), S.Argiro, S.Beole, M.Masera(2), G.Mila(3), G.Bianchini, M.Carta, V.Fabrizio, V.Peluso(4), F.Gabrielli, A.Rineiski(5), A.Kochetkov, P.Baeten, W.Uyttenhove, G.Vittiglio, J.Wagemans(6)

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Casaccia, Galeria (Roma), Italy, 5)Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen,

Germany, 6)Institute of Advanced Nuclear Systems, SCK CEN , Mol, Belgium

Neutronic Characteristics of Solid Targets in Accelerator-Driven System at Kyoto University Critical Assembly***

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Evaluation of Neutron Spectrum at In-Core Irradiation Equipments in KUR with Low Enriched Uranium Fuel

T.Sano, Y.Fujihara, J.Zhang, Y.Takahashi, K.Nakajima

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Accuracy of Thorium-Loaded Accelerator-Driven System Experiments at Kyoto University Critical Assembly^{***}

M.Yamanaka(1), C.Pyeon, T.Yagi, T.Misawa(2)

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Shutdown Transients Analysis for Reflector Devices Power Calculations in Jules Horowitz Material Testing Reactor (JHR)

P.Console Camprini(1), M.Sumini(2), C.Gonnier, B.Pouchin, P.Sireta, S.Bourdon(3)

1)ENEA, Italy, 2)University of Bologna, Italy, 3)CEA, France

Neutronic Analysis of the PULSTAR Reactor Using Monte Carlo Simulations

A.I.Hawari, J.L.Wormald, V.H.Gillette

North Carolina State University, North Carolina, USA

Poster Session

Jules Horowitz Reactor. France Experimental Loop Development According Optimized Irradiation Process.

S.Gaillot, S.Vitry, T.Dousson

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Preliminary Neutronic Design for the Conceptual Fluid Granular Spallation Target

J.Y.Li, L.Gu, D.W.Wang(1), W.Jiang(2)

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Accumulation of Tritium in Beryllium Material under Neutron Irradiation

S.Kunakov, N.Takibayev, N.Kenzhebayev

Physico-Technical faculty, Al-Farabi Kazakh National University, Almaty, Kazakhstan

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A Preliminary Study of an Improved Area Method, Adapted to Short Time Transients in Sub-Critical Systems

P.Saracco(1), R.Marotta, G.Lomonaco, D.Chersola(2), L.Mansani(3)

1)INFN Genova, Genova, Italy, 2)INFN Genova / Universita di Genova, Genova, Italy, 3)Ansaldo Nucleare SpA, Genova, Italy

Preliminary Neutronics Analysis of a Spallation Target for Transmutation

B.Wu, T.Dang, C.Liu, J.Zou, M.Wang, J.Jiang

Institute of Nuclear Energy Safety Technology, Anhui, China

Preliminary Optimization Analysis of the Radiation Shielding of the China Lead-Based Research Reactor

Q.Yang, B.Li, T.Li(1), C.Liu, Y.Bai, J.Song(2)

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Preliminary Analysis of Radioactive Source Term for Normal Operation of China Lead-Based Research Reactor (CLEAR-I)

T.Dang(1), L.Mao(1,2), Q.Zeng(1), Y.Wu(1,2)

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2)University of Science and Technology of China, Hefei, Anhui, China

10.12 Track 12 Fuel Cycle and Actinide Management

Oral Session

Simulation of Fuel Cycles with Minor Actinide Management Using a Fast Burnup Calculation Tool

M.Szieberth, M.Halász, S.Fehér, T.Reiss

Institute of Nuclear Techniques, Budapest, Hungary

Variations in Activity, Toxicity and Decay Heat of Nuclear Waste of Various Fuel Cycles **

N.E.Stauff, T.K.Kim, T.A.Taiwo

Argonne National Laboratory, Argonne, USA

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Effect of Heterogeneity in Plutonium Recycling in Steady State PWR

M.Ernoult, S.David, X.Doligez(1), A.Nuttin, N.Capellan, O.Meplan(2), B.Leniau(3)

1)Institute de Physique Nucléaire d'Orsay, Orsay, France, 2)LPSC, Grenoble, France, 3)SUBATECH, Nantes, France

Evaluation Method of Equivalence Factors for MOX Fuel and Non-Linear "Equivalent Pu-239" Formula

M.Tokashiki, S.Okui

Nuclear Fuel Industries, Ltd., Japan

Development of a Fuel Performance Code for Thorium-Plutonium Fuel

K.L.I.Björk, P.Fredriksson(1,2)

1)Thor Energy, Oslo, Norway, 2)Chalmers University of Technology, Göteborg, Sweden

Two-Stage Fuel Cycles with Accelerator-Driven Systems

F.Heidet, T.K.Kim, T.A.Taiwo

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Thorium-Fueled Breed-and-Burn Fuel Cycle

F.Heidet, T.K.Kim, T.A.Taiwo

Argonne National Laboratory, Argonne, USA

Fuel Cycle Analysis of Self-Sustaining Water Cooled Reactors with $^{232}\text{Th}/^{233}\text{U}$ Fuel and Impact of ^{233}U (n, γ) Cross Section Evaluations ***

N.R.Brown, G.Raitses, M.Todosow

Brookhaven National Laboratory, New York, USA

Production of ^{232}U from Irradiation of Standard and Thorium-Based Fuels in PWR Reactors

B.Leniau, M.Ernoult, X.Doligez, J.N.Wilson

Institut de physique nucléaire d'Orsay, Orsay, France

Fuel Cycle Scheme Design and Evaluation for Thorium-Uranium Breeding Recycle in CANDU Reactors ***

B.Yang, J.Shi, G.Bi, C.Tang

Shanghai Nuclear Engineering Research and Design Institute, Shanghai, China

An Inventory Analysis of Thermal-Spectrum Thorium-Fueled Molten Salt Reactor Concepts

J.J.Powers, J.C.Gehin, A.Worrall, T.J.Harrison, E.E.Sunny

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Agent-Based Dynamic Resource Exchange in CYCLUS

M.J.Gidden, R.W.Carlsen, A.Opotowsky, O.Rakhimov, A.M.Scopatz, P.P.H.Wilson

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Poster Session

Fusion Hybrids for Generation of Advanced ($^{231}\text{Pa}+^{232}\text{U}+^{233}\text{U}+^{234}\text{U}$)-Fuel in Closed (U-Pu-Th)-Fuel Cycle

G.G.Kulikov, A.N.Shmelev, E.G.Kulikov, V.A.Apse, E.F.Kryuchkov

National Research Nuclear University MEPhI (Moscow Engineering Physics Institute), Russia

^{233}U Fuel Production and 30-Year Utilization without Reprocessing and Refuelling Using Heavy Water Coolant

A.Talamo, Y.Gohar

Argonne National Laboratory, Chicago, USA

Uncertainty Analysis for Fuel Flux Calculations of Fast Reactors with External Fuel Cycle

E.F.Seleznev, A.A.Belov(1), G.N.Manturop, A.M.Tsybulya(2)

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Study on Transmutation and Storage of LLFP Using a High-Temperature Gas-Cooled Reactor

K.Kora, H.Nakaya, K.Kubo, H.Matsuura(1), S.Shimakawa, M.Goto, S.Nakagawa(2)

1)Kyushu University, Fukuoka, Japan, 2)Japan Atomic Energy Agency, Ibaraki, Japan

Core Library for Advanced Scenario Simulation, C.L.A.S.S. : Principle & Application.

B.Mouginot, B.Leniau, N.Thiollière(1), M.Ernoult, S.David, X.Doligez(2), A.Bidaud, O.Méplan,

R.Montesanto(3), G.Bellot, J.B.Clavel, I.Duhamel, E.Letang, J.Miss(4)

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Fuel Composition Generation Techniques of Nuclear Fuel Cycle Simulators

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University of Texas at Austin, Texas, USA

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Core Burnup Calculation of Uranium Rock-Like Oxide Fuel PWR for Spent Fuel Composition Estimation

H.Akie, K.Nishihara, Y.Nakano, N.Shirasu(1), T.Iwamura(2)

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10.13 Track 13 Radiation Applications and Nuclear Safeguards

Oral Session

Design of Long Neutron Counter for Intensified D-T Neutron Source

L.Yanan, L.Taosheng, Z.Siwei, Q.Fupeng, S.Gang, S.Jing, FDS Team

Institute of Nuclear Energy Safety Technology, CAS, Anhui, China

Utilizing Simulated Rossi-Alpha Distributions to Develop New Methods of Characterizing Spent Nuclear Fuel^{**}

A.C.Kaplan, V.Henzl, A.P.Belian, M.P.Swinhoe, H.O.Menlove(1), M.Flaska, S.A.Pozzi(2)

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Antineutrino Emission from Fuels with High Proliferation Resistance

T.Shiba, M.Fallot(1), S.Cormon(2)

1)SUBATECH, Nantes, France, 2)Universite de Nantes, Nantes, France

Validation of the Implicit Correlation Method in MCPNX-PoLiMi Using Plutonium Cross-Correlation Measurements

M.J.Marcath, T.H.Shin, S.D.Clarke, J.L.Dolan, M.Flaska, E.Larsen, A.C.Kaplan, S.A.Pozzi(1),
P.Peerani(2), E.Padovani(3)

1)University of Michigan, Michigan, USA, 2)Joint Research Centre, Ispra, Italy, 3)Polytechnic of Milan, Milan, Italy

A Unique Tungsten-Based Tagging Approach for Maintaining of Continuity of Knowledge of Nuclear Waste Copper Canisters

D.Chernikova, K.Axell, A.Nordlund

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10.14 Track 14 Education in Reactor Physics

Oral Session

Education Programs for Students and Graduate Students with Experimental Facilities for Nuclear Energy in Toshiba

Guest Speaker: K.Hiraiwa

TOSHIBA Corporation, Japan

Past, Present and Future of MIT Reactor Physics

B.Forget, K.Smith

Massachusetts Institute of Technology, MA, USA

New Practical Exercises at the JSI TRIGA Mark II Reactor

L.Snoj, S.Rupnik, A.Jazbec

Jožef Stefan Institute, Ljubljana, Slovenia

Developing a Course in Nuclear Reactor Modelling and Going from Campus-Based to Web-Based Teaching

C.Demazière, K.Jareteg

Chalmers University of Technology, Gothenburg, Sweden

Introduction to the Status of Reactor Physics Education in Tsinghua University

K.Wang, G.Yu, Z.Li, G.Shi

Tsinghua University, Beijing, China

Reactor Physics Education at Seoul National University

H.GJoo, H.J.Shim

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Poster Session

Multi-Collision Theory for Educated Pedestrians

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Virtual Labs on Unique Experimental Equipment

I.S.Saldikov, V.V.Afanasyev, E.F.Kryuchkov, V.I.Petrov, M.Y.Ternovskykh, G.V.Tikhomirov

NRNU MEPhI, Moscow, Russian Federation

PINSPEC: A Monte Carlo Code for Pin Cell Spectral Calculations for Educational Applications

W.Boyd, N.Gibson, S.Shaner, L.Li, J.Hunter, K.Smith, B.Forget

MIT, Massachusetts, USA

Unique Approaches in Emphasizing the Role of Reactor Laboratories and Facilities for Training and Education of Future Nuclear Engineers ‘without the Borders’

T.Jevremovic, R.Schow, L.McDonald IV(1), A.Rey(2)

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10.15 Track 15 Research Related to Fukushima Accident

Oral Session

AREVA Dismantling and Decommissioning Experience and Fuel Debris Removal Approach for Fukushima Dai-ichi

Guest Speaker: K.Schauer

AREVA, France

Re-Criticality Potential at Fukushima Dai Ichi Unit 4

A.H.Wells(1), A.J.Machiels, A.G.Sowder(2)

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Why a Criticality Excursion was Possible in the Fukushima Spent Fuel Pool

A.Sargeni, G.Caplin

IRSN, Fontenay-aux-Roses, France

Critical Experiments for Fuel Debris Using Modified STACY

K.Izawa, K.Tonoike, H.Sonov, Y.Miyoshi

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Concept of Capture Credit Based on Neutron Induced Gamma Ray Spectroscopy **

Y.Nauchi, H.Ohta(1), H.Unesaki, T.Sano, T.Yagi(2)

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A Methodology for Fast and Accurate Decay Heat Calculations for In-Pool Used Fuel Assemblies

Developed at AREVA La Hague Reprocessing Facility

A.Launay, G.Grassi, E.Lecampion, J.Lelandais(1), R.Eschbach, L.San Felice(2)

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Poster Session

Spatial Correlation Modeling of Macroscopic Cross Section with Weierstrass Function

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10.16 SS1 Molten Salt Reactors

Oral Session

Experimental Modelling and Numerical Analysis of a Molten Salt Fast Reactor

B.K.Yamaji, A.Aszodi

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Remark on the Propagating Neutron Noise in a MSR

V.Dykin, I.Pazsit(1), R.Sanchez(2)

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The Two-Group Point-Kinetic Component of Neutron Noise in an MSR

V.Dykin, I.Pazsit

Chalmers University of Technology, Gothenburg, Sweden

Neutronics of Fluid Fuel System with Perfect Remmixing

S.Dulla, P.Ravetto(1), A.K.Prinja(2)

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An Innovative Approach to Dynamics Modeling and Simulation of the Molten Salt Reactor Experiment

M.Zanetti, L.Luzzi, A.Cammi(1), C.Fiorina(2)

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Safety Criteria and Guidelines for MSR Accident Analysis

R.Yoshioka, K.Mitachi(1), Y.Shimazu(2), M.Kinoshita(3)

1)International Thorium Molten-Salt Forum, Yokohama, Japan, 2)University of Fukui, Fukui, Japan,

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Reactivity-Insertion-Transient Analysis of a Fluoride Salt Cooled High Temperature Reactor

Y.Yang, F.Yao, Z.Yang, S.Qiang, Z.Jie(1,2)

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Hybrid Spectrum Molten Salt Reactor

J.Krepel, K.Mikityuk(1), B.Hombourger, A.Pautz(2), M.Zanetti, M.Aufiero, L.Luzzi(3)

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3)POLIMI, Milano, Italy

Thorium Conversion Optimization in Two-Fluid Molten-Salt Reactor

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UJV Rez / Czech Technical University in Prague, Husinec-Rez, Czech Republic

Development of Computer Code Packages for Molten Salt Reactor Core Analysis

Y.Jeong, S.Chi, D.Lee

Ulsan National Institute of Science and Technology, Ulsan, Republic of Korea

Use of McDancoff Factor Correction for Multi-Group Fuel Depletion Analyses of Liquid Salt Cooled Reactors

M.Huang, B.Petrovic

Georgia Institute of Technology, Atlanta, USA

Comparative Studies on Plutonium and ^{233}U Utilization in miniFUJI MSR

A.Waris, S.Pramuditya, K.Basar(1), I.K.Aji, R.Wirawan, F.Monado, M.N.Subkhi(2)

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On an Optimized Neutron Shielding for an Advanced Molten Salt Fast Reactor Design

B.Merk, J.Konheiser

HZDR, Dresden, Germany

10.17 SS2 Reactor Physics and Criticality Safety Activities in OECD/NEA Working Party

Oral Session

Activities of OECD/NEA on Scientific Issues of Reactor Systems and Criticality Safety - Current Status and Future Plan

Guest Speakers: J.Gulliford(1), M.C.Brady Raap(2)

1)OECD/NEA, Issy-les-Moulineaux, France, 2)Pacific Northwest National Laboratory, Richland, USA

Uncertainty Analysis of the OECD/NRC Oskarshamn-2 BWR Stability Benchmark

I.Gajev, W.Ma(1), T.Kozlowski(2)

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Analysis of the OECD/NEA Oskarshamn-2 Feedwater Transient and Stability Benchmark with SIMULATE-3K

A.Dokhane, H.Ferroukhi, A.Pautz

Paul Scherrer Institute, Switzerland

Data Assimilation for Kinetic Parameters Uncertainty Analysis

E.Ivanov, T.Ivanova(1), I.Kodeli(2), V.Mastrangelo(3)

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Criticality and Reactor Physics Benchmark Experiments: Influence of Nuclear Data Uncertainties

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The Evaluation of the Subcritical Experiments Performed in the IPEN/MB-01 Research Reactor Facility for the IRPhE Project

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Polaris: A New Two-Dimensional Lattice Physics Analysis Capability for the SCALE Code System

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Evaluation of Large 3600MWth Sodium-Cooled Fast Reactor OECD Neutronic Benchmarks

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Evaluation of Medium 1000 MWth Sodium-Cooled Fast Reactor OECD Neutronic Benchmarks

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SFR Whole Core Burnup Calculations with TRIPOLI-4 Monte Carlo Code

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Summary and Status of OECD/NEA UAM-LWR Benchmark

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Uncertainty and Sensitivity Analysis of OECD/NEA UAM Fuel Thermal Behaviour Benchmark Using a Falcon/URANIE Methodology

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New PSI Methodology for Manufacturing and Technological Uncertainty Quantification^{**}

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Re-Evaluation and Continued Development of Shielding Benchmark Database SINBAD

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Poster Session

Effects of Nuclear Data Library and Ultra-Fine Group Calculation for Large Size Sodium-Cooled Fast Reactor OECD Benchmarks

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Evaluation of OECD/NEA/WPRS Benchmark on Medium Size Metallic Core SFR by Deterministic Code System: MARBLE and Monte Carlo Core: MVP

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Quantifying the Effect of Undersampling in Monte Carlo Simulations Using SCALE

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10.18 SS3 Hybrid Particle Transport Methods for Solving Complex Problems in Real-Time

Oral Session

Discretized Mesh Tools and Related Treatment for Hybrid Transport Application with 3D Discrete Ordinates and Monte Carlo

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A Novel Hybrid Weighting Scheme for Multi-Group Cross Section Collapsing

C.Yi, G.E.Sjoden, C.A.Edgar

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Development of an Iterative Lattice-Core Coupling Method Based on MICROX-2 Cross Section Libraries

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Solution of a Stylized European Pressurized Reactor (EPR) Benchmark Problem Using the Coarse Mesh Radiation Transport Method (COMET)

D.Lago, F.Rahnema, D.Zhang

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Computational Efficiency and Accuracy of the Fission Collision Separation Method in 3D HTTR Benchmark Problems

D.Zhang, F.Rahnema

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Use of the Fission Matrix Method for Solution of the Eigenvalue Problem in a Spent Fuel Pool

W.J.Walters, N.Roskoff, K.K.Royston, A.Haghigat

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10.19 SS4 Advanced Geometry Processing in Deterministic and Monte Carlo Methods

Oral Session

Development of a Multi-Group S_N Transport Calculation Code with Unstructured Tetrahedral Meshes

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Development of an Unstructured Mesh Based Geometry Model in the Serpent 2 Monte Carlo Code

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A CAD Based Automatic Modeling Method for Primitive Solid Based Monte Carlo Calculation Geometry

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Developments and Applications of the Geometry Conversion Tool McCad for Monte Carlo Particle Transport Simulations

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A Memory Efficient Algorithm for Classifying Unique Regions in Constructive Solid Geometries

D.Lax, W.Boyd, N.Horelik, B.Forget, K.Smith

Massachusetts Institute of Technology, Massachusetts, USA

10.20 SS5 Multiscale, Multiphysics Approaches in Nuclear Science and Engineering Applications

Oral Session

Influence of an Sn Solver in a Fine-Mesh Neutronics/Thermal-Hydraulics Framework

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High-Fidelity Multi-Physics Calculations for Light Water Reactors Using Coupled

CTF/TORT-TD/FRAPTRAN

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The Coupling of the Neutronic Transport Application RATTLESNAKE to the Nuclear Fuels Performance Application BISON under the MOOSE Framework

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A Model of Two-Stage Core Calculation Method Coupled with Subchannel Analysis for Boiling Water Reactors

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Subspace Methods for Multi-Physics Reduced Order Modeling in Nuclear Engineering Applications

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Extension of the Entropy Viscosity Method to Flows with Friction Forces and Source Terms

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Efficient Finite Element Field Interpolation for Multiphysics Applications

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Poster Session

Research on SCWR Core Characteristics Utilizing Pin-Wise Neutronics Thermal-Hydraulic Coupling Method

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10.21 SS6 Nuclear Criticality Safety of Fuel Debris

Poster Session

Post-Accident Defueling Procedure and Its Criticality Safety Evaluation of the Fukushima- Daiichi Nuclear Power Plants

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10.22 SS7 Control Rod Withdrawal Tests Performed during the PHENIX End-of-Life Experiments

Oral Session

CEA Contribution to the Analysis of the Control Rod Withdrawal Test Performed During PHENIX End-of-Life Experiments (IAEA Common Research Program)

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IAEA Benchmark Calculations on Control Rod Withdrawal Test Performed During PHENIX End-of-Life Experiments - JAEA's Calculation Results

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Benchmark Analysis of PHENIX Control Rod Withdrawal End-of-Life Experiments

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Calculation of the PHENIX End-of-Life Test "Control Rod Withdrawal" with the ERANOS Code

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Simulation of PHENIX Control Rod Withdrawal Experiments with SIMMER-IV

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Study of the Effect of Heterogeneity of the Control Rods in the PHENIX Reactor

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IAEA Benchmark Calculations on Control Rod Withdrawal Test Performed during PHENIX End-of-Life Experiments - Benchmark Results and Comparisons

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10.23 SS8 Reactor Physics of Non-Traditional LWR Fuel Design

Oral Session

I²S-LWR Equilibrium Cycle Core Analysis

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Uranium Nitride Composite Fuels in a Pressurized Water Reactor: Exploration of Multi-Batch Cycle Length and UB4 Admixture for Reactivity Control***

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Impact of Coating on Nitride Fuel Performance in PWRs

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Optimization of Fully Ceramic Micro-Encapsulated Fuel Assembly for PWR

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Fully Ceramic Microencapsulated Fuels: Characteristics and Potential LWR Applications

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Neutronic Challenges of Advanced Boiling Water Reactor Designs

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国際単位系 (SI)

表1. SI 基本単位

基本量	SI 基本単位	
	名称	記号
長さ	メートル	m
質量	キログラム	kg
時間	秒	s
電流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光度	カンデラ	cd

表2. 基本単位を用いて表されるSI組立単位の例

組立量	SI 基本単位	
	名称	記号
面積	平方メートル	m ²
体積	立方メートル	m ³
速度	メートル毎秒	m/s
加速度	メートル毎秒毎秒	m/s ²
波数	毎メートル	m ⁻¹
密度、質量密度	キログラム毎立方メートル	kg/m ³
面積密度	キログラム毎平方メートル	kg/m ²
比體積	立方メートル毎キログラム	m ³ /kg
電流密度	アンペア毎平方メートル	A/m ²
磁界の強さ	アンペア毎メートル	A/m
量濃度 ^(a) 、濃度	モル毎立方メートル	mol/m ³
質量濃度	キログラム毎立方メートル	kg/m ³
輝度	カンデラ毎平方メートル	cd/m ²
屈折率 ^(b)	(数字の) 1	1
比透磁率 ^(b)	(数字の) 1	1

(a) 量濃度(amount concentration)は臨床化学の分野では物質濃度(substance concentration)ともよばれる。

(b) これらは無次元量あるいは次元をもつ量であるが、そのことを表す単位記号である数字の1は通常は表記しない。

表3. 固有の名称と記号で表されるSI組立単位

組立量	SI 組立単位		
	名称	記号	他のSI単位による表し方
平面角	ラジアン ^(b)	rad	1 ^(b) m/m m ² m ² s ⁻¹
立体角	ステラジアン ^(b)	sr ^(c)	1 ^(b) Hz
周波数	ヘルツ ^(d)		N
力	ニュートン		m kg s ⁻²
圧力、応力	パスカル	Pa	N/m ² m ⁻¹ kg s ⁻²
エネルギー、仕事、熱量	ジュール	J	N m m ² kg s ⁻²
仕事率、工率、放射束	ワット	W	J/s m ² kg s ⁻³
電荷、電気量	クーロン	C	s A
電位差(電圧)、起電力	ボルト	V	W/A m ² kg s ⁻³ A ⁻¹
静電容量	ファラード	F	C/V m ² kg ⁻¹ s ⁴ A ²
電気抵抗	オーム	Ω	V/A m ² kg s ⁻³ A ⁻²
コンダクタンス	ジーメンス	S	A/V m ² kg ⁻¹ s ⁴ A ²
磁束密度	ウェーバ	Wb	Vs m ² kg s ⁻² A ⁻¹
磁束密度	テスラ	T	Wb/m ² kg s ⁻² A ⁻¹
インダクタンス	ヘンリー	H	Wb/A m ² kg s ⁻² A ⁻²
セルシウス温度	セルシウス度 ^(e)	°C	K
光照度	ルーメン	lm	cd sr ^(c) lm/m ² m ² cd s ⁻¹
放射性核種の放射能 ^(f)	ベクレル ^(d)	Bq	lm ⁻¹
吸収線量、比エネルギー分与、カーマ	グレイ	Gy	J/kg m ² s ⁻²
線量当量、周辺線量当量、方向性線量当量、個人線量当量	シーベルト ^(g)	Sv	J/kg m ² s ⁻²
酸素活性	カタール	kat	s ⁻¹ mol

(a) SI接頭語は固有の名称と記号を持つ組立単位と組み合わせても使用できる。しかし接頭語を付した単位はもはやコヒーレントではない。

(b) ラジアンとステラジアンは数字の1に対する単位の特別な名称で、量についての情報をつたえるために使われる。実際には、使用する時には記号rad及びsrが用いられるが、習慣として組立単位としての記号である数字の1は明示されない。

(c) 測光学ではステラジアンという名称と記号srを単位の表し方の中に、そのまま維持している。

(d) ヘルツは周期現象についてのみ、ベクレルは放射性核種の統計的過程についてのみ使用される。

(e) セルシウス度はケルビンの特別な名称で、セルシウス温度を表すために使用される。セルシウス度とケルビンの単位の大きさは同じである。したがって、温度差や温度間隔を表す數値はどちらの単位で表しても同じである。

(f) 放射性核種の放射能(activity referred to a radionuclide)は、しばしば誤った用語で“radioactivity”と記される。

(g) 単位シーベルト(PV,2002,70,205)についてはCIPM勧告2(CI-2002)を参照。

表4. 単位の中に固有の名称と記号を含むSI組立単位の例

組立量	SI 組立単位		
	名称	記号	SI 基本単位による表し方
粘度	パスカル秒	Pa s	m ¹ kg s ⁻¹
力のモーメント	ニュートンメートル	N m	m ² kg s ²
表面張力	ニュートン毎メートル	N/m	kg s ⁻²
角速度	ラジアン毎秒	rad/s	m ⁻¹ s ⁻¹ =s ⁻¹
角加速度	ラジアン毎秒毎秒	rad/s ²	m ⁻¹ s ⁻² =s ⁻²
熱流密度、放射照度	ワット毎平方メートル	W/m ²	kg s ⁻³
熱容量、エンントロピー	ジュール毎ケルビン	J/K	m ² kg s ⁻² K ⁻¹
比熱容量、比エンントロピー	ジュール毎キログラム毎ケルビン	J/(kg K)	m ² s ⁻² K ⁻¹
比エネルギー	ジュール毎キログラム	J/kg	m ² s ⁻²
熱伝導率	ワット毎メートル毎ケルビン	W/(m K)	m kg s ⁻³ K ⁻¹
体積エネルギー	ジュール毎立方メートル	J/m ³	m ¹ kg s ⁻²
電界の強さ	ボルト毎メートル	V/m	m kg s ⁻³ A ⁻¹
電荷密度	クーロン毎立方メートル	C/m ³	m ³ sA
表面電荷密度	クーロン毎平方メートル	C/m ²	m ² sA
電束密度、電気変位	クーロン毎平方メートル	C/m ²	m ² sA
誘電率	ファラード毎メートル	F/m	m ³ kg s ⁻⁴ A ²
透磁率	ヘンリー毎メートル	H/m	m kg s ⁻² A ²
モルエネルギー	ジュール毎モル	J/mol	m ² kg s ⁻² mol ¹
モルエントロピー、モル熱容量	ジュール毎モル毎ケルビン	J/(mol K)	m ² kg s ⁻² K ⁻¹ mol ¹
照射線量(X線及びγ線)	クーロン毎キログラム	C/kg	kg ⁻¹ sA
吸収線量	グレイ毎秒	Gy/s	m ⁻¹ s ⁻³
放射強度	ワット毎メートル	W/sr	m ² kg s ⁻³ =m ² kg s ⁻³
放射輝度	ワット毎平方メートル毎ステラジアン	W/(m ² sr)	m ² m ² kg s ⁻³ =kg s ⁻³
酵素活性濃度	カタール毎立方メートル	kat/m ³	m ⁻³ s ⁻¹ mol

表5. SI接頭語

乗数	接頭語	記号	乗数	接頭語	記号
10 ²⁴	ヨタ	Y	10 ⁻¹	デシ	d
10 ²¹	ゼタ	Z	10 ⁻²	センチ	c
10 ¹⁸	エクサ	E	10 ⁻³	ミリ	m
10 ¹⁵	ペタ	P	10 ⁻⁶	マイクロ	μ
10 ¹²	テラ	T	10 ⁻⁹	ナノ	n
10 ⁹	ギガ	G	10 ⁻¹²	ピコ	p
10 ⁶	メガ	M	10 ⁻¹⁵	フェムト	f
10 ³	キロ	k	10 ⁻¹⁸	アト	a
10 ²	ヘクト	h	10 ⁻²¹	ゼット	z
10 ¹	デカ	da	10 ⁻²⁴	ヨクト	y

表6. SIに属さないが、SIと併用される単位

名称	記号	SI 単位による値
分	min	1 min=60s
時	h	1h=60 min=3600 s
日	d	1 d=24 h=86 400 s
度	°	1°=(π/180) rad
分	'	1'=1(60)'=(π/10800) rad
秒	"	1"=(1/60)"=(π/648000) rad
ヘクタール	ha	1ha=1hm ² =10 ⁴ m ²
リットル	L	1L=1dm ³ =10 ³ cm ³ =10 ⁻³ m ³
トン	t	1t=10 ³ kg

表7. SIに属さないが、SIと併用される単位で、SI単位で表される数値が実験的に得られるもの

名称	記号	SI 単位で表される数値
電子ボルト	eV	1eV=1.602 176 53(14)×10 ⁻¹⁹ J
ダルトン	Da	1Da=1.660 538 86(28)×10 ⁻²⁷ kg
統一原子質量単位	u	1u=1 Da
天文単位	ua	1ua=1.495 978 706 91(6)×10 ¹¹ m

表8. SIに属さないが、SIと併用されるその他の単位

名称	記号	SI 単位で表される数値
バール	bar	1 bar=0.1MPa=100kPa=10 ⁵ Pa
水銀柱ミリメートル	mmHg	1mmHg=133.322Pa
オングストローム	Å	1 Å=0.1nm=100pm=10 ⁻¹⁰ m
海里	M	1 M=1852m
ノット	b	1 b=100fm ² =(10 ⁻¹² cm) ² =10 ⁻²⁸ m ²
ノット	kn	1 kn=(1852/3600)m/s
ネバール	Np	SI単位との数値的な関係は、対数量の定義に依存。
デジベル	dB	

表9. 固有の名称をもつCGS組立単位

名称	記号	SI 単位で表される数値
エルグ	erg	1 erg=10 ⁻⁷ J
ダイーン	dyn	1 dyn=10 ⁻⁵ N
ボアズ	P	1 P=1 dyn s cm ⁻² =0.1Pa s
ストークス	St	1 St=1cm ² s ⁻¹ =10 ⁻⁴ m ² s ⁻¹
スチルブ	sb	1 sb=1cd m ⁻² =10 ⁴ cd m ⁻²
フォート	ph	1 ph=1cd sr cm ⁻² 10 ⁴ lx
ガル	Gal	1 Gal=1cm s ⁻² =10 ⁻² ms ⁻²
マックスウェル	Mx	1 Mx=1G cm ² =10 ⁸ Wb
ガウス	G	1 G=1Mx cm ⁻² =10 ⁻⁴ T
エルステッド(c)	Oe	1 Oe△(10 ³ /4n)A m ⁻¹

(c) 3元系のCGS単位系とSIでは直接比較できないため、等号「△」は対応関係を示すものである。

表10. SIに属さないその他の単位の例

名称	記号	SI 単位で表される数値
キュリ	Ci	1 Ci=3.7×10 ¹⁰ Bq
レントゲン	R	1 R=2.58×10 ⁴ C/kg
ラド	rad	1 rad=1cGy=10 ⁻² Gy
レム	rem	1 rem=1 cSv=10 ⁻² Sv
ガンマ	γ	1 γ=1 nT=10 ⁻⁹ T
フェルミ	fm	1 fm=1 fm=10 ⁻¹⁵ m
メートル系カラット	Torr	1 Torr=(101 325/760) Pa
標準大気圧	atm	1 atm=101 325 Pa
カロリ	cal	1 cal=4.1858J (15°Cカロリー), 4.1868J (ITカロリー)
ミクログラム	μ	1 μ=1μm=10 ⁻⁶ m

