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

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 the 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
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Akio Yamamoto
Technical Program Chair

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J. Zerr
I. Zmijarevic
W. Zwermann

Track Leaders

1	**Reactor Analysis Methods**				
	C. Demazière	H. G. Joo	R. Sanchez	K. Smith	M. Tojo
2	**Deterministic Transport Theory**				
	Y. Azmy	N. Z. Cho	F. Rahnema	K. Yamaji	
3	**Monte Carlo Methods**				
	T. Kitada	J. Leppänen	H. J. Shim	K. Wang	
4	**Verification, Validation and Uncertainty Analysis**				
	H. Abdel-Khalik	G. Chiba	W. F. G. van Rooijen		
5	**Nuclear Criticality Safety**				
	J. Bess	T. Endo	T. Yamamoto		
6	**Reactor Physics Experiments**				
	P. Blaise	H. Unesaki			
7	**Reactor Concepts and Designs**				
	J. Lee	C. Liangzhi	B. Petrovic	N. Takaki	
8	**Reactor Operation and Safety**				
	F. Franceschini	T. Mitsuyasu	G. Sjoden		
9	**Transient and Safety Analysis**				
	Y. Ban	S. Dulla	K. Ivanov	P. Ravetto	
10	**Nuclear Data**				
	S. Chiba	R. Jacqmin	Y. O. Lee		
11	**Research Reactors and Spallation Sources**				
	G. van den Eynde	I. L. Montoya	K. Nishihara		
12	**Fuel Cycle and Actinide Management**				
	T. Kim	K. Tsujimoto	A. Worrall		
13	**Radiation Applications and Nuclear Safeguards**				
	Y. Kitamura	S. A. Pozzi			
14	**Education in Reactor Physics**				
	B. Forget	T. Kameyama	T. Kozlowski		
15	**Research Related to Fukushima Accident**				
	A. Haghighat	S. Kosaka	K. Suyama		

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. Haghighat 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

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.)

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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Nuclear Energy Agency

General Information

Registration

Registration is required for all attendees and presenters. Name plate is required for admission to all events.

Full registration includes: All technical sessions, book of abstracts, CD proceedings, welcome cocktail, hosted lunch, and conference banquet.

Student registration includes: All technical sessions, book of abstracts, CD proceedings, welcome cocktail and hosted lunch (option: conference banquet +JPY10,000).

Conference Proceedings

Conference Proceedings are included with the program book in CD-ROM format.

Guidelines for Speakers

There will be five parallel sessions. Each presentation will last 15 minutes at the maximum, followed by 5 minutes for questions. In order to allow conference participants to attend the presentation of papers in different sessions in a timely manner, we, as organizers, will request the chairpersons to comply with the time schedule rigorously. In view of the given time constraints, please make sure that your presentation fits within the prescribed 15 minutes limit leaving adequate time for questions from the audience.

The laptops going to be used in sessions will have the following programs installed:

- Microsoft Office Word 2013
- Microsoft Office Excel 2013
- Microsoft Office PowerPoint 2013
- Acrobat Reader Version 10

The laptops will be using Microsoft Windows 7 operating system with the Service Pack 1 installed. Default Windows font set will be available on these machines. We highly recommend that you create a PDF version of the presentation so that you can switch to the PDF in case of a problem with the PowerPoint. A microphone will be used for the presentation, please make sure that you keep close to the microphone during your talk. We also request the workshop speakers to adhere to the same rules if they plan to use our computers.

Special Events

General Events

Welcome Cocktail
Place: Mizuho_D
Date: Sunday Sep. 28 16:00-20:00

Conference Banquet
Place: Mizuho_C, D
Date: Tuesday Sep. 30 18:30-21:00
Speakers: Dr. Ronald J. Ellis, Chair, ANS Reactor Physics Division and Prof. Toshikazu Takeda, Honorary Chair, Univ. of Fukui

Hosted Lunches
Place: Mizuho_C, D
Date: Monday Sep. 29 11:30-13:00
Speaker: Etsuro Saji (Mitsubishi Heavy Industries)

Place: Mizuho_C, D
Date: Tuesday Sep. 30 12:00-13:30
Speaker: Sun-Doo Kim (KEPCO Nuclear Fuel)

Technical Tours

Fukushima Dai-ichi Nuclear Power Plant
(35 persons. JPY47,000/person, including train and bus on 10/2 and 10/3; accommodation on 10/2 in Tokyo)
10/2: Conf. Hotel (Afternoon) → Tokyo (Stay)
10/3: Tokyo → Fukushima → Fukushima Dai-ichi NPP → Tokyo (Evening; Dismiss)

Monju in Tsuruga
(20 persons. JPY 6,500/person, including bus transportation and lunch)
10/3: Conf. Hotel (Morning) → Monju → Conf. Hotel (Evening)

Monju is a 280 MWe prototype sodium cooled fast breeder reactor (FBR) using plutonium-uranium mixed oxide fuel. Monju achieved initial criticality in April 1994. In August 1995, Monju became the first Japanese FBR to generate power. It remains the only Japanese facility that can generate electricity using a fast breeder reactor. In December 1995, sodium coolant leaked from a temperature sensor in the secondary system piping. Since then, power generating operation has been suspended. Monju is still in the limelight as an international key facility of the fast breeder reactor R&Ds and is expected to demonstrate its reliability as an operational power plant, and establish sodium handling technology.

Kumatori Area (NFI and KURRI)
(20 persons. JPY 6,500/person, including bus transportation and lunch)
10/3: Conf. Hotel (Morning) → Kumatori → Conf. Hotel (Evening)

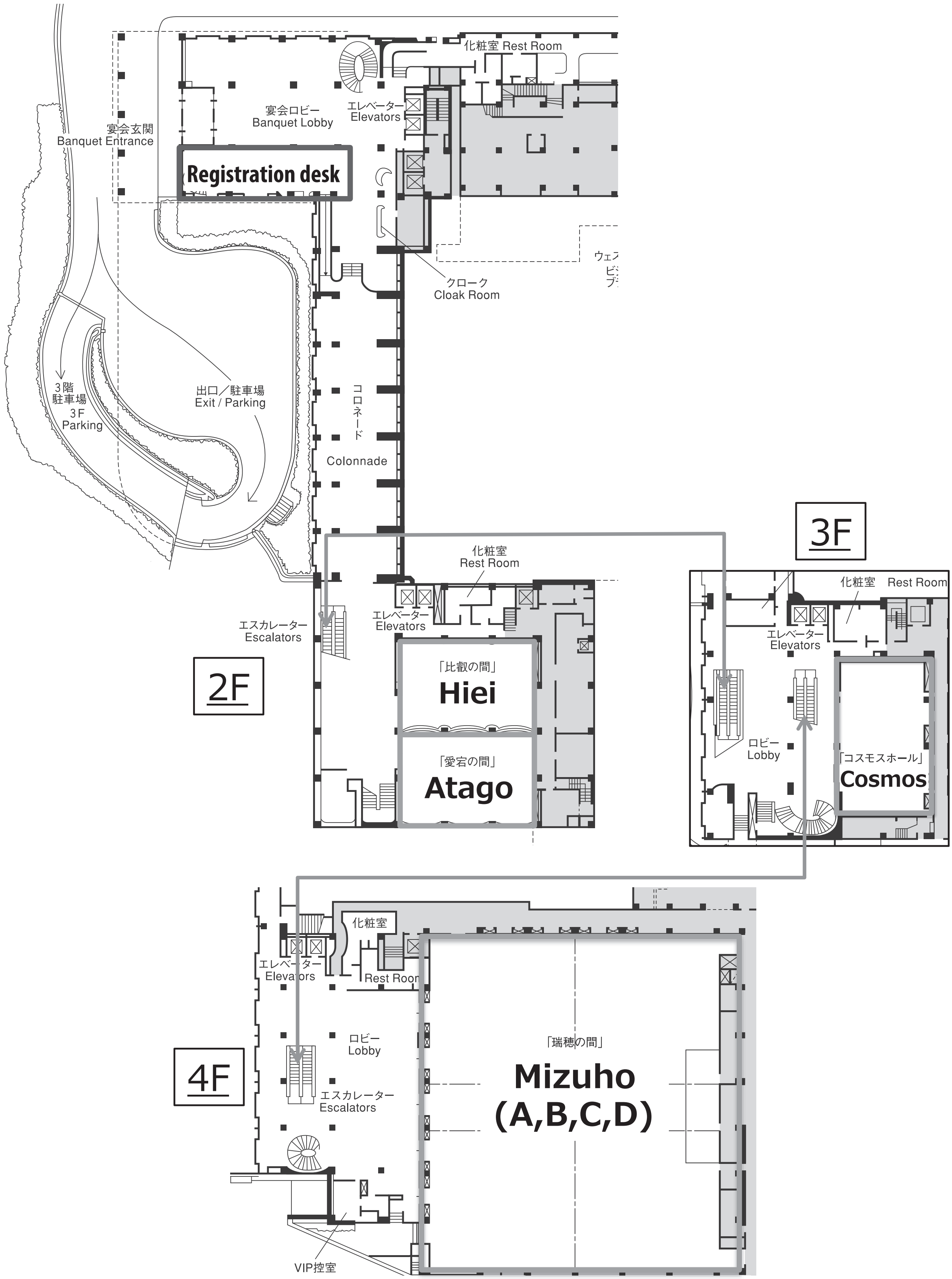
+ Nuclear Fuel Industries, Ltd. (NFI, Kumatori Works)
Nuclear Fuel Industries, Ltd. (NFI) was established in 1972 and provides nuclear fuel and fuel-related services. It operates two fuel processing plants, a PWR fuel plant at Kumatori, and a BWR fuel plant at Tokai. Kumatori works has supplied over 9,000 PWR fuel assemblies for electric power companies. NFI will continue to supply reliable and high-quality products, and work towards a safer and more comfortable society.

+ Kyoto University Research Reactor Institute (KURRI)
The Kyoto University Research Reactor Institute (KURRI) was established in 1963 for the joint use program among Japanese universities to promote the research and education in the fields of nuclear energy and radiation application. Two nuclear reactors, the Kyoto University research Reactor (KUR) and the Kyoto University Critical Assembly (KUCA), and related research facilities have been being used since then, and nowadays greater expectations are being put on the research and education activities at our institute for the issues of energy and environment and for the innovative applications of radiation.

In addition, proton beams from the Fixed Field Alternating Gradient (FFAG) accelerator complex installed in the accelerator room is led to KUCA for the purpose of executing feasibility study on the accelerator-driven system (ADS).

The tour will include the KUCA facilities and the FFAG accelerator.

Meeting Rooms



Technical Workshops

8:00 - 12:00 # *5* **New Features and Capabilities in the Serpent 2 Monte Carlo Code**
Room; Mizuho_A (4F)
Organizer; J. Leppänen (VTT, Finland)

8:00 - 12:00 # *3* **TEPCO's Activity on the Investigation of Fukushima Daiichi Accident / Current Status of Fukushima-Daiichi NPS and Decommissioning Process**
Room; Mizuho_B (4F)
Organizer; S. Mizokami (Tokyo Electric Power Company, Japan),
M. Yamamoto (Tokyo Electric Power Company, Japan)

8:00 - 12:00 # *6* **Attila4MC - CAD Integration, Automated Deterministic Variance Reduction, and GUI Setup for MCNP**
Room; Mizuho_C (4F)
Organizer; G. Failla (Varian Medical Systems, USA)

8:00 - 12:00 # *2* **Workshop on Accelerator-Driven System in PHYSOR 2014**
Room; Hiei (2F)
Organizer; T. Sugawara (JAEA, Japan), C. Pyeon (KURRI, Japan)

13:30 - 17:35 # *1* **Short Course on Uncertainty Characterization in Nuclear Calculations**
Room; Mizuho_A (4F)
Organizer; H. S. Abdel-Khalik (North Carolina State University, USA)

13:30 - 17:35 # *4* **Neutron Noise Techniques for Reactor Diagnostics**
Room; Mizuho_B (4F)
Organizer; I. Pazsit (Chalmers Univ. of Technol., Sweden),
C. Demazière (Chalmers Univ. of Technol., Sweden),
V. Dykin (Chalmers Univ. of Technol., Sweden)

13:30 - 17:35 # *7* **Hybrid Particle Transport Methods for Solving Complex Problems in Real-Time**
Room; Mizuho_C (4F)
Organizer; A. Haghighat (Virginia Tech., USA), F. Rahnema(Georgia Tech., USA),
B. Petrovic (Georgia Tech., USA)

13:30 - 17:35 # *8* **NESTLE 3D Nodal Core Simulator: An Overview of Latest Features and Capabilities**
Room; Hiei (2F)
Organizer; G. I. Maldonado (Univ. of Tennessee, USA),
N. P. Luciano (Univ. of Tennessee, USA), S. D. Hart (Univ. of Tennessee, USA)

Plenary Sessions 1 & 2

8:50 AM

Kord Smith
Professor, Massachusetts Institute of Technology (USA)

Advances in Reactor Physics and Computational Science

9:25 AM

Akira Yamaguchi
Professor, Osaka University (Japan)

Status of Nuclear Power in Japan

10:20 AM

Jun Matsumoto
General Manager, Fukushima Daiichi Decontamination and Decomissioning Engineering Company, Tokyo Electric Power Company (Japan)

Topics on Fukushima Daiichi Decontamination and Decommissioning

10:55 AM

Robin Forrest
Head of Nuclear Data Section, International Atomic Energy Agency (Austria)

Advances in Nuclear Data

Track1-1 Reactor Analysis Methods

Session Chair: Tarmer Bahadir(Studsvik Scandpower, Inc.), Masayuki Tojo(GNF-J)

13:00 PM
Hitachi's Advanced Technologies

Guest Speaker: H.Soneda

Hitachi-GE Nuclear Energy, Japan

13:20 PM
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

Enhanced SPH method is proposed for the accurate calculation in the geometry which has large spectral interference between fuel assemblies. The SPH factor is created in order to preserve the reaction rate of each cell in the pin-by-pin core calculation and reduce homogenization error. Since the discontinuity of the neutron flux between fuel assemblies are not considered in the conventional SPH method, another SPH method which uses discontinuity factor has been developed. Using this SPH method, the discontinuity at the outermost part of fuel assembly can be incorporated to the SPH factor. On the other hand, this SPH method does not consider the spectrum of the outer portion of the fuel assembly. This problem leads the difficulty of accurate evaluation of spectral interference in the system such as UO_2/MOX multi-assembly geometry. In this paper, enhanced SPH method which takes both spectrum and discontinuity at the outermost part of fuel assembly is proposed. And the performance of enhanced SPH method is evaluated by calculation results of 2-dimensional single assembly geometry and multi-assembly geometry. The results show that newly developed method can calculate the multi-assembly geometry accurately even if the spectral interference between the fuel assemblies is large. And it is confirmed that enhanced SPH factor can be created by the proposed method and contribute to accurate prediction of neutronics parameters in pin-by-pin core calculation.

13:40 PM
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

The applicability of the SPH method to a whole core heterogeneous MOC calculation is investigated. The sensitivities to the calculation error of the discretization parameters of MOC regarding the ray trace path width, the number of azimuthal angles and the spatial mesh size are investigated in a two-dimensional whole core configuration which consists of MOX and UO_2 fuels. The cross section sets of MOX and UO_2 fuels are generated in single assembly geometry without SPH method and with SPH method. Although the sensitivities to the calculation error of the ray trace path width and the number of azimuthal angles are quite large without SPH method, these sensitivities are decreased with SPH method. This result reveals that the SPH method is effective to reduce the discretization error in a whole core MOC calculation. On the other hand, the maximum error of power distribution is increased under the coarser condition in spite of SPH correction and this maximum error is obtained near the boundary between MOX and UO_2 fuels. This result means that spectral interference effect among fuel assemblies should be taken into account. The sensitivity to the calculation error of the spatial mesh size is not decreased with SPH method. It is observed that the spatial mesh size in the moderator region has quite large effect on pin power distribution error whether the SPH method is used. These results mean that the spatial distributions of neutron flux and neutron source are quite important in the moderator region near the fuel boundary.

14:00 PM
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

Recent work has led to an asymptotically-derived expression for reconstructing the neutron flux from lattice functions and multigroup diffusion solutions. The leading-order asymptotic term is the standard expression for flux reconstruction, i.e., it is the product of a shape function obtained through a lattice calculation and the multigroup diffusion solution. The first-order asymptotic term is a formally small correction. The correction is negligible when the gradient of the diffusion solution is small, but large when the gradient of the diffusion solution is large. Inclusion of this first-order correction term can significantly improve the accuracy of the reconstructed flux. One may define discontinuity factors (DFs) to make certain angular moments of the reconstructed flux continuous across interfaces between assemblies in 1-D. Indeed, the standard assembly discontinuity factors used often in reactor analysis make the zeroth moment (scalar flux) of the reconstructed flux continuous. The inclusion of the first-order correction term in the flux reconstruction gives us an additional degree of freedom that can be used to make two angular moments of the reconstructed flux continuous across interfaces by using current DFs, in addition to flux DFs, in the multigroup diffusion calculation. Numerical results demonstrate that for certain problems (those not generally requiring colorset calculations to obtain DFs), using flux and current DFs together is more accurate than using only flux DFs. Furthermore, DFs that make the first angular moment (current) and second angular moment of the reconstructed flux continuous across interfaces are more accurate than using DFs that make the zeroth and first angular moments (scalar flux and current) of the reconstructed flux continuous.

14:20 PM
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

In this paper we discuss the possibility to model localized heterogeneities (e.g. fuel assembly spacer grid), within the context of few-group homogeneous diffusion calculation. This work is motivated by the research of a model being able to precisely take into account both global (reactivity effect) and local (axial profile) effect. Based on the assumption that the grid can be seen as a local perturbation, the axial flux profile is then reconstructed a posteriori by incorporating additional local informations to the diffusion core calculation, following the basic principles of the well-known pin-power reconstruction procedure for homogeneous assembly calculations. After describing the main steps allowing to establish the axial reconstruction model, its validity is tested on a simple fuel assembly test-case, and verified with respect to the reference axial profile computed by the Monte Carlo code TRIPOLI4. Then an optimization work, aimed at reducing the axial form factor computing cost, is undertaken. We show that the axial form factor can be computed on a simplified cylindrical fuel cell, and we propose a faster calculation scheme, based on the APOLLO2 SN solver. The final computing time for the form factor is reduced at ~10s, with a very limited precision loss. One-group fission rate discrepancies are within ±0.5% in a wide range around the central part of the assembly, and it increases up to -6% maximum in the outermost part of the assembly, in all likelihood due to the 2-groups homogeneous diffusion hypothesis and not to the factorization approach.

14:40 PM
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

This paper presents an evaluation of the new radial submesh method of the advanced core simulator SIMULATE-5 (S5) and the effects on pin power reconstruction for an EPR core design. The developed S5 core model is first verified both against results from the predecessor SIMULATE-3 (S3) code as well as to a vendor solution. The latter

Track1-1 Reactor Analysis Methods

Session Chair: Tarmer Bahadir(Studsvik Scandpower, Inc.), Masayuki Tojo(GNF-J)

provides confidence in the developed model while the consistency with the former constitutes an intrinsic verification of the new S5 methodology in terms of performing the critical leakage correction of nuclear data that was previously handled by the lattice code. Thereafter, the impact of various submeshes on the S5 assembly and pin power results is studied, starting from a coarse 1x1 assembly-size mesh typical of conventional nodal codes down to a fine 17x17 submesh structure representative of the lattice full pin-by-pin geometry. Since focus is on the radial submeshing, only 2-D axially-integrated pin powers are investigated and it is found that the various submesh subdivisions give comparable results, with differences typically around ±1%. It is however confirmed that submesh effects increase across- and within fuel assemblies as function of stronger flux heterogeneities and gradients (e.g. core/reflector interfaces and/or control rods). An observation is that submesh “mixture” effects can occur, especially if submeshes include gadolinium bearing fuel rods, and compared to a fine mesh solution, this could produce even larger differences than a solution obtained with a simple coarse assembly size mesh.

15:00 PM
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)

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The energy group structure which reduces the error caused by interference effect is studied. The error in pin-power distribution becomes large at interference area of UO2 and MOX fuel assemblies. Although the error could be reduced by increasing the number of collapsed energy groups, the target of this study is to find robust energy group structure against spectral interference without changing the number of collapsed energy groups and without introducing any additional parameters. Considering the characteristics of cross-section curve and the change of neutron flux caused by interference effect, we obtain the robust energy group structure against interference effect and evaluate the robustness of this energy group structure in cases of a multi-assembly and a single-assembly having the different interference effect.

SS2-1 Reactor Physics and Criticality Safety Activities in OECD/NEA Working Party

Session Chair: Mark DeHart(INL), Teruhiko Kugo(JAEA)

13:00 PM
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

13:20 PM
Uncertainty Analysis of the OECD/NRC Oskarshamn-2 BWR Stability Benchmark

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1)Royal Institute of Technology, Stockholm, Sweden, 2)University of Illinois, Urbana-Champaign, USA

On February 25, 1999, the Oskarshamn-2 NPP experienced a stability event which culminated in diverging power oscillations with a decay ratio of about 1.4. The event was successfully modeled by the TRACE/PARCS coupled system code, and further uncertainty analysis of the event is described in this paper. The results show very good agreement with the plant data, capturing the entire behavior of the transient including the onset of instability, growth of the oscillations (decay ratio) and oscillation frequency. This provides confidence in the prediction of other parameters which are not available from the plant records. This paper shows also how an uncertainty method was implemented for the event. Comparing the calculated uncertainty with the measured uncertainty gives confidence in the BWR stability prediction.

13:40 PM
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

The OECD/NEA recently launched an international benchmark on a combined feedwater transient and stability event that occurred at the Swedish Oskarshamn-2 (O2) nuclear power plant (NPP). The primary benchmark objective is to assess advances in coupled neutronic/thermal-hydraulic codes for simulations of challenging transients including the appearance of unstable power oscillations. The Paul Scherrer Institut (PSI) is participating in this benchmark in order to enlarge the validation basis of its advanced stability analysis methodology currently under development for Swiss BWRs and based on the state-of-the-art SIMULATE-3K (S3K) code. This paper presents the development, optimization and validation of a S3K model for the first phase of the O2 benchmark, namely the analysis of the entire event. With the optimized model, the S3K solution is compared to available benchmark data both for at steady-state and transient conditions. For the latter, the qualitative as well as quantitative behavior of S3K results is compared and discussed in relation to the experimental observations. The modeling aspects found in this context to mostly affect the S3K ability to reproduce the event are also presented. Finally, the S3K model indicates that if the reactor scram had not been introduced, the observed diverging oscillations would reach a maximum amplitude before decaying back into a stable state. However, it was also found that the core could instead have evolved into limit-cycle oscillations if a stabilization of the feedwater flow and temperature had occurred just before the scram signal.

14:00 PM
Data Assimilation for Kinetic Parameters Uncertainty Analysis

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

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Several years ago, the OECD/NEA Nuclear Science Committee (NSC) established the Expert Group on Uncertainty Analysis in Modeling (UAM-LWR) after thorough discussions of the demands from nuclear research, industry, safety and regulation to provide the best estimate predictions of nuclear systems parameters with their confidence bounds. UAM objectives include among others, the quantification of uncertainties of neutronic calculations with respect to their value for the multi-physics analysis.
Since the kinetics parameters and their uncertainties are of particular interest for these studies the deterministic approaches for analysis of uncertainties in nuclear reactor kinetic parameters (neutron generation lifetime and delayed neutron effective fraction) have been developed in frame of the UAM-LWR. The approach uses combination of generalization of perturbation theory to reactivity analysis, and the Generalized Perturbation Theory (GPT) for sensitivity computation. It has been applied to the UAM complementary fast neutron SNEAK test case that has unique set of experimental data for βeff. In this example, the covariance matrices of nuclear data have been derived from COMMARA library by Bayesian adjustment upon the set of the fast neutron integral benchmarks with the BERING code package.
Then, the kinetic parameters uncertainties with their correlations have been applied to simplified model of a reactivity insertion transient where relative uncertainty of power peak was taken as figure of merit. The results demonstrate that the uncertainties due to nuclear data impact significantly the energy release in a coupled transient modeling. It was also found that such uncertainties become higher if the correlations between uncertainties of different lumped parameters are taken into account.

14:20 PM
Criticality and Reactor Physics Benchmark Experiments: Influence of Nuclear Data Uncertainties

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GRS, Garching, Germany

A number of LWR-type criticality and reactor physics experiments, mainly from the ICSBEP and IRPhEP Handbooks, many of which were already used in the past for international benchmarks in the framework of OECD/NEA working groups, are being evaluated with respect to uncertainties in the basic nuclear data. For this, the sampling based uncertainty and sensitivity analysis tool XSUSA along with the Monte Carlo code KENO-V.a as transport solver is employed. Particular emphasis is put on experiments where differential quantities, mainly reaction rate distributions, were measured; the uncertainties of such quantities are not directly accessible to tools based on first order perturbation theory. With respect to multiplication factors and reactivity differences, all results are compared with corresponding results obtained with TSUNAMI-3D from the SCALE 6.1 system; the agreement is very good for all assemblies under consideration. With respect to fission rate distributions, the uncertainty analyses yield only moderate uncertainties from nuclear data; therefore, in general the total uncertainty is dominated by measurement uncertainties, which include the uncertainties of technological parameters. The work is continuously being extended; in the future, also non-LWR specific assemblies, mainly relevant for GEN-IV reactors, will be investigated.

14:40 PM
The Evaluation of the Subcritical Experiments Performed in the IPEN/MB-01 Research Reactor Facility for the IRPhE Project

A.D.Santos, S.M.Lee, R.Jerez, R.Diniz

IPEN-CNEN/SP, São Paulo, Brazil

This work presents the evaluation of the subcritical experiments performed at the IPEN/MB-01 research reactor facility. The experimental method developed at the IPEN facility showed itself very valuable in the resolution of the several subcritical parameters employed in the subcritical kinetic model of Gandini and Salvatores. The evaluated data which were acceptable for publication in the IRPhE handbook are of very good quality and suitable for a benchmark problem. The theoretical analyses show that the MCNP5 results, together with the ENDF/B-VII.0 library, for for k_{eff}, ρ_{gen}, $\Sigma\rho_{gen}$, and k_{sub} are in a very good agreement to the benchmark values. In the case of the calculation of ζ MCNP5 does a good job when there is just one additional source, but with the source is distribute like in the case of the intrinsic source, MCNP5 misses part of the physics because the intrinsic source is not weighted by the adjoint flux according to their position. In other words, the intrinsic source that it is in the center of the core is more effective to induce fissions than the one that it is in the periphery. This kind of MCNP5 deficiency imposes restrictions in the determination of ζ and consequently in the determination of φ*. The k_{sub} MCNP5 results were insensitive to the source treatment and only a few cases were slightly outside of the 3σ range of the benchmark values.

SS2-1 Reactor Physics and Criticality Safety Activities in OECD/NEA Working Party

Session Chair: Mark DeHart(INL), Teruhiko Kugo(JAEA)

15:00 PM
Polaris: A New Two-Dimensional Lattice Physics Analysis Capability for the SCALE Code System

M.A.Jessee, W.A.Wieselquist, T.M.Evans, S.P.Hamilton, J.J.Jarrell, K.S.Kim, J.P.Lefebvre, R.A.Lefebvre, U.Mertyurek, A.B.Thompson, M.L.Williams

Oak Ridge National Laboratory, Oak Ridge, USA

Polaris is a new 2-dimensional (2-D) lattice physics capability in the SCALE code system for the analysis of light water reactor fuel designs. In this paper, the Polaris calculational methods are summarized and results are provided for a series of computational benchmarks. The summary includes the implementation of the relatively new resonance self-shielding approach called the embedded self-shielding method, the implementation of a new 2-D method-of-characteristics neutron transport solver, and the integration of the SCALE/ORIGEN depletion and decay solver for depleting material compositions. Polaris calculations are compared with reference continuous energy Monte Carlo solutions for a UO_2 fuel computational benchmark. Because Polaris is integrated into the SCALE code system, Polaris has been utilized as part of the new SCALE/Sampler code sequence, which provides stochastic uncertainty analysis for the impact of cross-section uncertainties on lattice physics calculations. SCALE/Sampler results are summarized for selected benchmark calculations.

SS1-1 Molten Salt Reactors

Session Chair: Imre Pazsit(Chalmers Univ. of Tech.), Ritsuo Yoshioka(International Thorium Molten-Salt Forum)

13:00 PM
Experimental Modelling and Numerical Analysis of a Molten Salt Fast Reactor

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In this paper experimental and numerical investigation of the MSFR (Molten Salt Fast Reactor) concept will be presented. This homogeneous, single region liquid fuelled fast reactor concept uses fluoride-based molten salts with fissile uranium and thorium and other heavy nuclei content with the purpose of applying the thorium cycle and the burn-up of transuranic elements. Molten salt reactors with liquid fuel have a unique safety related property that needs clear understanding. In the core neutron flux and fission distribution is determined by the flow field through distribution and transport of fissile material and delayed neutron precursors. Since the MSFR concept has a single region homogeneous core without internal structures, it is a difficult task to ensure stable flow field, which is also strongly coupled to the volumetric heat generation. These considerations suggest that experimental and numerical modelling (including the option of coupled neutronics-thermal-hydraulics) would be needed to better understand the flow phenomena in such geometry.
A scaled and segmented experimental mock-up of MSFR was designed and built at BME NTI with the purpose of investigating the flow behavior inside the core region using particle image velocimetry. Not only the basic flow behavior inside the core region can be investigated but measurement data can also provide resource for the validation of computational fluid dynamics models, specific problems or phenomena (for example inlet geometry, optional internal structures, mixing) may be studied as well. Measurement results of steady state conditions will be presented with comparison of measurement data and results of numerical analyses.

13:20 PM
Remark on the Propagating Neutron Noise in a MSR

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The neutron noise induced by propagating perturbations in a bare 1-D Molten Salt Reactor (MSR) model is calculated and analyzed using one-group diffusion theory. The neutron noise for different noise sources of which two have not been accounted for, corresponding to the fluctuations of the fission and absorption cross sections as well as to the fuel velocity is calculated and the results are qualitatively compared. Unlike in previous work, the solution is obtained through the matrix Green's function of the flux and precursor equations being kept separate. It is shown that in the case when the noise is represented by the fluctuations of the fission cross-section, the noise source attains a complex structure which is different from that in traditional reactors. On the other hand, in the cases investigated, despite all qualitative differences in the noise calculation procedure as well as in the structure of the noise source, it turns out that the noise induced by the absorption and the fission cross sections follow a similar behaviour. In addition, it is observed that the inclusion of the fluctuations in the fuel velocity examined in this paper slightly suppresses the total neutron noise for low frequency region i.e. below ~ 2 Hz but on the other hand it enhances the latter one by one of order of magnitude for high frequencies i.e. above ~ 2 Hz compared to the effect of other noise sources. The results contribute to the understanding and interpretation of the neutron noise in MSRs.

13:40 PM
The Two-Group Point-Kinetic Component of Neutron Noise in an MSR

V.Dykin, I.Pazsit

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The calculation of the point kinetic component of the neutron noise in two-group diffusion theory in Molten Salt Reactors (MSRs) using different techniques is discussed. First, the point kinetic component of the noise is calculated from the full space-frequency dependent solution analytically by a projection to the static adjoint. Then, the point-kinetic solution is derived by solving the simplified point kinetic equations. Both results are thereafter analyzed and compared quantitatively. This comparison shows that the solution of the simplified point kinetic equations significantly differs from the exact one and cannot reconstruct some important features of the true solution. The similar discrepancies between two methods were also observed and confirmed in earlier one-group MSR calculations.

14:00 PM
Neutronics of Fluid Fuel System with Perfect Remixing

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The neutronics of a homogeneous fluid-fuel nuclear system with perfect remixing implying a uniform distribution of delayed neutron precursors inside the core is studied in this work. A one-group diffusion model is adopted for the neutron balance and an analytical treatment is used throughout thewhole analysis. The critical problemis first solved, allowing the determination of a set of eigenfunctions characteristic of the physico-mathematical problem considered. The time-dependent problem is then solved by expanding the neutron flux in terms of these eigenfunctions and the results are compared with those obtained through a standard Helmholtz eigenfunction expansion. The use of these eigenfunctions show to be advantageous, since the asymptotic state of the system can be represented by the fundamental eigenfunction only.

14:20 PM
An Innovative Approach to Dynamics Modeling and Simulation of the Molten Salt Reactor Experiment

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1)Politecnico di Milano, Milano, Italy, 2)Paul Scherrer Institute, Villigen, Switzerland

The Molten Salt Reactor Experiment (MSRE) was a circulating fuel thermal reactor built and operated in the sixties. As the only Molten Salt Reactor (MSR) testing facility for which extensive experimental data are available, it can be considered as a reference for the development of modeling approaches for the studies related to the Gen-IV MSR. In this work, a geometric multi-scale approach has been adopted for the simulation of the MSRE plant. The data and the experimental results relative to the U-233 fuelled reactor are considered. The neutronic parameters have been determined using the Monte Carlo code Serpent. The reactor core is divided into three radial regions, each one described by a 3D channel in which Navier-Stokes and energy conservation equations plus delayed neutron precursors (DNP) balance equations are solved. Determination of the generated power is obtained employing a point kinetics like equation, fed with importance weighted values of temperatures and DNP concentrations. The remaining part of the plant, that includes the primary and secondary cooling circuits, is modeled by means of zero-dimensional components. The results attained with such modeling approach are compared with experimental data both in time and frequency domain, showing good agreement. The adopted approach, thanks to the punctual, coupled solution of the governing equations in the core, gives better insights into the thermal behavior of the graphite and its effects on MSR dynamics than commonly used correlation-based solvers.

14:40 PM
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, 3)University of Tokyo, Tokyo, Japan

Accident analysis for Molten Salt Reactor (MSR) has been investigated at ORNL for MSRE in 1960s. Since then, safety criteria or guidelines have not been defined for MSR accident analysis. Regarding the safety criteria, the authors showed one proposal in this paper. In order to establish guidelines for MSR accident analysis, we have to investigate all possible accidents. In this paper, the authors describe the philosophy for accident analysis, and show 40 possible accidents. They are at first classified as external cause accidents and internal cause accidents. Since the former ones are generic accidents, we investigate only the latter ones, and categorize them to 4 types, such as power excursion accident, flow decrease accident, fuel-salt leak

SS1-1 Molten Salt Reactors

Session Chair: Imre Pazsit(Chalmers Univ. of Tech.), Ritsuo Yoshioka(International Thorium Molten-Salt Forum)

accident, and other accidents mostly specific to MSR. Each accident is described briefly, with some numerical results by the authors.

15:00 PM
Reactivity-Insertion-Transient Analysis of a Fluoride Salt Cooled High Temperature Reactor

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The Fluoride salt cooled High temperature Reactor (FHR) is an innovative reactor design that uses conventional TRISO high temperature fuel with a low-pressure liquid salt coolant. The design of this reactor is currently in progress both in China and in the United States. An FHR based on ordered pebble bed core design is being planned for construction by the Shanghai Institute of Applied Physics (SINAP). This paper provides a preliminary reactivity insertion transient analysis of an FHR of SINAP's pebble core design, using RELAP5/MOD4.0 code. New models and methodologies are developed for several prototypical facilities that are based on SINAP's pebble bed concept, and different types of reactivity insertion transient are analyzed. SINAP's design is currently in progress; the ultimate goal of the transient analysis is to acquire the capability of RELAP5/MOD4.0 for performing FHR core design.

SS5 Multiscale, Multiphysics Approaches in Nuclear Science and Engineering Applications

Session Chair: Richard Martineau(INL), Takeshi Mitsuyasu(Hitachi)

13:00 PM

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

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In this paper a study on the influence of a neutron discrete ordinates (SN) solver within a fine-mesh neutronic/thermal-hydraulic methodology is presented. The methodology consists of coupling a neutronic solver with a single-phase fluid solver, and it is aimed at computing the two fields on a three-dimensional (3D) sub-pin level. The cross-sections needed for the neutron transport equations are pre-generated using a Monte Carlo approach. The coupling is resolved in an iterative manner with full convergence of both fields. A conservative transfer of the full 3D information is achieved, allowing for a proper coupling between the neutronic and the thermal-hydraulic meshes on the finest calculated scales. The discrete ordinates solver is benchmarked against a Monte Carlo reference solution for a two-dimensional (2D) system. The results confirm the need of a high number of ordinates, giving a satisfactory accuracy in keff and scalar flux profile applying S16 for 16 energy groups. The coupled framework is used to compare the SN implementation and a solver based on the neutron diffusion approximation for a full 3D system of a quarter of a symmetric, 7x7 array in an infinite lattice setup. In this case, the impact of the discrete ordinates solver shows to be significant for the coupled system, as demonstrated in the calculations of the temperature distributions.

13:20 PM

High-Fidelity Multi-Physics Calculations for Light Water Reactors Using Coupled CTF/TORT-TD/ FRAPTRAN

J.W.Magedanz, M.N.Avramova

The Pennsylvania State University, Pennsylvania, USA

This paper describes the development and testing of a high-fidelity multi-physics system consisting of the codes CTF, TORT-TD, and FRAPTRAN. CTF is an improved version of the subchannel code COBRA-TF, while TORT-TD is a multi-group discrete-ordinates neutron kinetics code used for pin-wise analysis of LWR's. FRAPTRAN, the most recent addition to the system, is a fuel performance code, which was incorporated in order to provide more accurate feedback to the thermal-hydraulic and kinetics calculations. The high-fidelity multi-physics system has an increased level of fidelity over current methods by modeling on the pin scale, taking into account the angular dependence of the neutron flux, and taking into account the full range of phenomena that occur inside the fuel rods. It is intended to be used to study the circumstances in which such higher fidelity methods provide a significant benefit. The results obtained here demonstrate the system capability for a set of simple problems based on fuel pin properties obtained from the OECD/NRC PWR MOX/UO2 Core Transient Benchmark.

13:40 PM

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

F.N.Gleicher, J.Ortensi, B.W.Spencer, D.Gaston, Y.Wang, S.R.Novascone, J.D.Hales, R.L.Williamson, R.C.Martineau

Idaho National Laboratory, Idaho, USA

The MOOSE based reactor physics tool MAMMOTH provides the capability to seamlessly couple the neutron transport application RATTLESNAKE to the fuels performance application BISON to produce a higher fidelity tool for fuel performance simulations. The ultimate purpose of this coupling is to provide a tool with the predictive capabilities to gain new knowledge and help resolve fundamental questions in the fuel performance arena, i.e. high-burnup structures, pellet-cladding interaction, missing pellet surface, etc. RATTLESNAKE solves the self-adjoint angular flux transport equation, derived from the linearized Boltzmann transport equation, and provides a sub-pin level resolution of the multigroup neutron flux. BISON solves the coupled thermomechanical equations for the fuel on a sub-millimeter scale. The coupling within the MOOSE framework allows both applications to solve their respective systems on aligned and unaligned unstructured finite element meshes. MAMMOTH uses the power density calculated by RATTLESNAKE to compute the local burnup evolution. Subsequently, MAMMOTH transfers the power density and burnup distribution to BISON with the MOOSE Multiapp transfer system. BISON in turn is able to provide sub-pin level temperature for cross section feed back effects. Multiple depletion cases were run with one-way and two-way data transfer in MAMMOTH for RATTLESNAKE-BISON. The one-way eigenvalues obtained show good agreement with the reference values obtained from the lattice physics code DRAGON4 while the two-way eigenvalue show expected differences. The power distributions obtained are consistent with both DRAGON4 and the SERPENT Monte Carlo code. The one-way and two-way calculations produce power density results that are comparable with those of the internal, static, Lassmann style model in BISON. Differences in the power densities arise from the use of better neutron energy deposition parameters obtained from the DRAGON4 tabulations, and differences in the fuel temperature arise from a different thermal expansion models in the fuel.

14:00 PM

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

T.Mitsuyasu, K.Ishii, N.Nakadozono, M.Aoyama

Hitachi, Ltd., Japan

The two-stage core analysis method is widely used for BWR core analysis. The purpose of this study is to develop the core analysis model coupled with subchannel analysis within the two-stage calculation scheme using an assembly-based thermal-hydraulics calculation in the core analysis. This model does not change the thermal-hydraulics scheme of the core analysis. Rather, it appends the subchannel void distribution to the previous uniform analysis in lattice physics, and couples that with the subchannel analysis which axially calculates full assembly and uses the flow condition that produces the maximum void fraction in the operation core. The subchannel void distribution of one node from the subchannel analysis is only normalized and used for the lattice physics. The developed model was evaluated for the heterogeneous problem with multiple enrichments. The developed model could decrease the eigenvalue differences by more than half of that of the uniform case and made the differences of assembly power the same as the uniform case. Furthermore, it could reduce the root mean square differences to more than half of those of the uniform case in the low and high enrichment fuels. The computation times of the lattice physics become 2.3 times longer. The extended computing time does not prevent core analysis because the nuclear data are prepared in advance of the core analysis. As the result of the evaluation, the model can incorporate the subchannnel effect to the core analysis.

14:20 PM

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

B.A.Khuwaileh, J.M.Hite, H.S.Abdel-Khalik

North Carolina State University, Raleigh, USA

This manuscript proposes a new extension to a reduced order modeling algorithm, previously introduced for single-physics models, to multi-physics models. This manuscript focuses on loosely-coupled physics models wherein the output of one physics model is fed as input to the next physics model, and each physics model is solved separately while freezing all other physics models. The idea is to perform three reductions at each physics-to-physics interface, one based on the upstream physics, another for the downstream physics, and a third for the interaction thereof. Accurately capturing the interaction between the reduced physics models is an essential feature of the proposed algorithm, and will be the key measure for its success. For standard model execution, this interaction is often captured using an iterative technique that loops over the different physics until convergence or meeting some stopping criterion. We adopt a similar approach in which the effective dimensionality of each physics-to-physics interface is updated iteratively until a user-defined error tolerance is met. A quarter PWR fuel assembly depleted to 32 GWD/MTU by iteratively solving the quasi-static transport-depletion approximation is used to exemplify the application of the proposed algorithm. Active subspaces for the nuclear cross-sections and neutron flux are determined, and compared to the active subspaces obtained without the physics coupling.

SS5 Multiscale, Multiphysics Approaches in Nuclear Science and Engineering Applications

Session Chair: Richard Martineau(INL), Takeshi Mitsuyasu(Hitachi)

14:40 PM
Extension of the Entropy Viscosity Method to Flows with Friction Forces and Source Terms

M.Delchini, J.C.Ragusa(1), R.A.Berry(2)

1)Texas A&M University, College Station, USA, 2)INL, Idaho Falls, USA

In this paper, we extend the entropy viscosity method to 1-D Euler equations with source terms present. The entropy viscosity method has been successfully applied to hyperbolic equations such as Burgers equation and the Euler equation system. This method consists in adding dissipative terms to the governing equations so as to ensure the entropy minimum principle. The dissipative terms contain a viscosity coefficient (function) that locally modulates the amount of dissipation. This viscosity coefficient is based on the entropy production that occurs in the wiggles, discontinuities, and shocks of hyperbolic equation systems. By adding source terms to the Euler equations (friction and gravity forces to the momentum equation and heat sources/sinks in the energy equation), the entropy viscosity method must be modified to account for the entropy production due to these additional terms. Tests are run for a 1D channel, using pressurized water reactor (PWR) conditions, with the RELAP-7 code based on the MOOSE framework. The equations are discretized with a continuous Galerkin finite element method (FEM) using linear polynomials along with a second-order, implicit temporal scheme (BDF2).

15:00 PM
Efficient Finite Element Field Interpolation for Multiphysics Applications

D.Lebrun-Grandie(1), J.C.Ragusa(1), R.Sampath(2)

1)Texas A&M University, College Station, USA, 2)Houston, TX, USA

We have proposed a two-stage algorithm for data field interpolation for parallel multiphysics simulations. In such simulations, source and target meshes between physics components are not necessarily coincident (for example, neutronic/thermal-hydraulic coupled computations) or non-penetration conditions between multiple bodies must be enforced (for instance, pelletclad interaction in nuclear fuel performance). Our approach employs a coarse parallel search to determine the subset of elements where data points of interest are located. This stage uses an octree representation and space-filling curve. In the second stage, a local fine search is performed by inverse the coordinate mapping. This nonlinear system is solved by means of a Newton technique and several initial guess options are proposed to improve the robustness of the nonlinear solve. Weak and strong scaling studies were performed.

Track11-1 Research Reactors and Spallation Sources

Session Chair: Ronald J.Ellis(ORNL), Gaillot Stephan(CEA)

13:00 PM
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

MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) is a multipurpose research facility currently being developed at SCK-CEN. It will be able to work in both critical and subcritical modes and, cooled by lead-bismuth eutectic. It will play a key role in the development of the Pb-alloy technology needed for the LFR (Lead Fast Reactor) GEN IV concept. MYRRHA will demonstrate the ADS (Accelerator Driven System) full concept by coupling a proton accelerator, a spallation target and a subcritical reactor at a reasonable power level to allow operation feedback. MYRRHA will also contribute to the study of partitioning and transmutation of high-level waste. Recently, a new core design with a longer active core has been proposed. This paper presents the first neutronic analyses for this design improvement.

13:20 PM
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

A deuterium-tritium (DT) based thermal-to-fast neutron converter will be installed into the TRIGA reactor at Jožef Stefan Institute. This paper presents preliminary results of a feasibility study where different aspects have been addressed. The thermal column is the most appropriate irradiation position for the DT converter in the reactor. From potential active converter materials, LiD yields the most 14 MeV neutrons. Lithium enrichment affects the required thickness of the active converter material, but not significantly the 14 MeV neutron yield. For coupled neutron-tritium transport calculations, the modified MCUNED code was used. Relevant DT neutron activation monitor materials for spectrum unfolding have been identified and neutron flux criteria established.

13:40 PM
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

Alternative Energies and Atomic Energy Commission (CEA), Gif sur Yvette Cedex, France

This paper describes a new neutron calculation scheme dedicated to long-duration fuel-rod irradiations carried out with the GRIFFONOS device in the periphery of the core of the OSIRIS material testing reactor (CEA/Saclay Center). With this calculation scheme, based on the APOLLO2 neutron transport code, the neutron flux and fission power can now be computed taking into account most irradiation parameters such as the core-loading map, the control rod effects or the fuel depletion in the core assemblies. Validation has been performed by comparisons to reference calculations using TRIPOLI-4 Monte-Carlo code and measured quantities. Obtained discrepancies on neutron flux and fission power in the fuel rod are less than 5%.

14:00 PM
Neutronic Designs and Analyses of a New Core-Moderator Assembly and Neutron Beam Ports for the Penn State Breazeale Reactor

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A new core-moderator assembly and five new neutron beam ports are modeled and designed for the Penn State Breazeale Reactor (PSBR). The PSBR is an open pool, light water cooled, and moderated 1-MW research reactor with seven neutron beam ports. The existing core-moderator assembly design does not allow simultaneous utilization of all the available beam ports; only two beam ports, namely #4 and #7, are currently in use for research and education in the facility. Moreover, the prompt gamma-rays produced at the back side of the heavy water moderator tank shine into neutron beam tube #4. Subsequently that is hampering the quality of the experimental data at the existing beam port facilities. The proposed design eliminates all the limitations of the existing design and provides multiple high-intensity and clean neutron beams to a new and expanded beam hall utilizing various instruments and techniques. The new design features a crescent-shaped moderator tank, which couples the reactor core to four thermal ports and one cold neutron beam port with three curved guide tubes for various cold neutron beam techniques. The modeling of the new PSBR design was achieved with highly detailed neutronics simulations using several stochastic simulation tools developed for the PSBR. The simulation results revealed the optimal design parameters and neutronics performance of the new beam ports, such that the thermal neutron beam intensity was significantly increased and the total prompt gamma dose was drastically decreased in the new beam port facilities.

14:20 PM
Design Studies for a Multiple Application Thermal Reactor for Irradiation Experiments (MATRIX)

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

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The Advanced Test Reactor (ATR) is a high power density test reactor specializing in fuel and materials irradiation. For more than 45 years, the ATR has provided irradiations of materials and fuels plus secondary missions such as the production of radioisotopes. Should unforeseen circumstances lead to the decommissioning of ATR, the U.S. Government would be left without a large-scale materials irradiation capability to meet the needs of its nuclear energy and naval reactor missions. In anticipation of this possibility, work was performed under the Laboratory Directed Research and Development (LDRD) program to investigate test reactor concepts that could serve the current missions of the ATR along with an expanded set of secondary missions. A survey was conducted in order to catalogue the anticipated needs of current and potential customers. Concepts were then evaluated to fill the role for this reactor, dubbed the Multi-Application Thermal Reactor for Irradiation eXperiments (MATRIX). This evaluation indicates that the baseline MATRIX design described herein can achieve longer cycle lengths than ATR given a particular batch scheme and using uranium enriched to less than 20%. The volume of test space in In-Pile-Tubes (IPTs) in MATRIX is larger than those in ATR with comparable magnitude of neutron flux. Furthermore, MATRIX has a greater number and volume of irradiation spaces having high fast neutron flux than ATR. From the analyses performed in this work, it appears that a new materials test reactor, based upon the MATRIX concept, can meet the anticipated needs of current and new high flux irradiation programs for the foreseeable future.

14:40 PM
Simulated Irradiation of Samples in HFIR for Use as Possible Test Materials in the MPEX (Material Plasma Exposure eXperiment) Facility

R.Ellis, J.Rapp

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The importance of Plasma Material Interaction (PMI) is a major concern in fusion reactor design and analysis. The Material-Plasma Exposure eXperiment (MPEX) facility will explore PMI under fusion reactor plasma conditions. Samples with accumulated displacements per atom (DPA) damage produced by irradiations in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) will be studied in the MPEX facility. The project presented in this paper involved performing assessments of the induced radioactivity and resulting radiation fields of a variety of potential fusion reactor materials. The scientific code packages MCNP and SCALE were used to simulate irradiation of the samples in HFIR; generation and depletion of nuclides in the material and the subsequent composition, activity levels, gamma radiation fields, and resultant dose rates as a function of cooling time. These state-of-the-art simulation methods were used in addressing the challenge of the MPEX project to minimize the radioactive inventory in the preparation of the samples for inclusion in the MPEX facility.

Track11-1 Research Reactors and Spallation Sources

Session Chair: Ronald J.Ellis(ORNL), Gaillot Stephan(CEA)

15:00 PM

MCNPX Analysis of Delayed Neutron Fraction in Beryllium Reflected Cores

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The behavior of the effective delayed neutron and effective delayed photo neutron fractions in beryllium reflected cores is investigated. An MCNPX analysis of the dependence of the effective beta fractions on the fuel depletion is performed for different fuel types.
An alternative method for estimation of the delayed neutron fraction during the reactor operation, avoiding double criticality simulation on prompt neutrons only (k_p), is compared to the standard MCNP approach ($1 - k_p/k_{p+d}$). The methodology is based on MCNP tally calculations of fission integrals and actual delayed neutron fraction, B^k_m , for each fissile isotope m (^{235}U, ^{239}Pu, ^{241}Pu) in a given reactor configuration and fuel type *k*. The methodology is verified on a generic infinite lattice and after that applied for the MCNP whole core model of the BR2 reactor at different depletion steps during the reactor operation.
The influence of the poisoning of the beryllium reflector on the effective beta fraction is also studied. Criticality and spectra calculations taking into account delayed neutrons and delayed photo neutrons are performed using MCNPX 2.7.0.

Track1-2 Reactor Analysis Methods

Session Chair: Richard Sanchez(CEA), Hideki Matsumoto(MHI)

15:45 PM
Application of the Efficient Consistent Spatial Homogenization Method in Neutron Transport Theory to a Gas Cooled Thermal Reactor Problem

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Georgia Institute of Technology, Atlanta, USA

In this paper, the accuracy and computational efficiency of the efficient consistent spatial homogenization method (ECSH) in neutron transport theory is assessed in a 1D benchmark problem characteristic of gas cooled thermal systems that are extremely challenging for conventional homogenization methods because of their longer neutron mean free path than water-based thermal reactors. The ECSH method is an extension of the consistent spatial homogenization method by using: (1) B-spline instead of Fourier series for the expansion of the spatial domain in the auxiliary cross section term and (2) a source iteration scheme instead of local fixed-source calculations in the re-homogenization procedure. Furthermore, the effect of the angular expansion order in the definition of the auxiliary cross section is studied. This method can be viewed as a significant improvement in accuracy of standard homogenization methods used for VHTR whole core analysis in which core environment effects are pronounced. It is shown that the ECSH method can reproduce the heterogeneous transport solution with up to 4 times faster computational speed, depending on the configuration of the control rods while maintaining reasonable accuracy and robust re-homogenization procedure.

16:05 PM
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

The recently developed Hybrid Diffusion-Transport Spatial Homogenization (DTH) Method was previously tested on a benchmark problem typical of a boiling water reactor. In this paper, the DTH method is tested in a 1-D benchmark problem based on the Japanese High Temperature Test Reactor (HTTR). This acts as a verification of the method for a reactor that is optically thinner than the original BWR test benchmark.

16:25 PM
Normalization Methods for Diffusion Calculations with Various Assembly Homogenizations

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When considering diffusion core calculations, a standard transport-diffusion SPH equivalence procedure is performed to preserve infinite lattice reaction rates. Since the corresponding equivalence coefficients are defined up to a multiplicative constant, a normalization is applied aiming at preserving global leakage at assemblies boundary. Supposing a linearly anisotropic flux in the transport reference calculation, preserving partial currents is equivalent to preserving boundary fluxes in a periodic lattice. This hypothesis may be questionned when dealing with rodded or MOX assemblies that present strong anisotropic fluxes. In this prospect, this paper compares a new normalization method based on boundary partial currents conservation to more familiar normalization techniques such as flux-volume conservation and Selengut normalization. The comparison has been carried out on various 2D 3x3 assembly clusters involving several assembly types (UOX, MOX, UOX-Gadolinium and UOX-Pyrex), control rods and burn-up gradients, for 2-group diffusion calculations performed with the new EDF calculation scheme (Gabv2-Cocagne) for different homogenization models (homogeneous, heterogeneous and pin-by-pin) with respect to Apollo-2 transport reference calculations. The new current normalization has proven to be accurate over the whole range of configurations, all the more as the flux is anisotropic at boundary assemblies, when associated with any homogenization model.

16:45 PM
On the Practical Feasibility of Continuous-Energy Monte Carlo in Spatial Homogenization

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This paper aims to evaluate the practical feasibility of using the continuous-energy Monte Carlo method for producing homogenized group constants for deterministic core simulators. The calculations are carried out using the Serpent 2 Monte Carlo code and ARES nodal diffusion fuel cycle simulator. A test case from a previous validation study is repeated with varying number of neutron histories in group constant generation. The impact of statistical variation in the results of ARES simulations is evaluated, and the corresponding calculation times used to provide an order-of-magnitude estimate for the overall computational cost for generating the full set of group constants covering all state points. It is concluded that, while computationally expensive, Monte Carlo based spatial homogenization involving burnup and thousands of state points per assembly type is within the range of feasibility using modern computer clusters.

17:05 PM
A Dynamic Homogenization Model for Pebble Bed Reactors

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We have used a macro stochastic model to provide a new homogenization paradigm for pebble bed core calculations. Our method results in an iterative core homogenization which for each homogenized region accounts for interactions between different pebble types and the neighboring regions. Hence, at convergence pebble depletion and power computation are done using the fine-group transport fluxes. We discuss the validation of the method and give a new and accurate estimation of the critical height for the HTR-10 international benchmark. Finally, we introduce a general formulation for the search for the equilibrium core with a multipass fuel management where the pebble recycling strategy is defined by a limit of the actual pebble burnup in the core, as opposed to a fixed number of passes through the core. We also extend this formulation to account for a radial loading distribution.

17:25 PM
Homogenization of the Step Characteristic Scheme in Phase Space

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We apply homogenization methodology to formulate a transport discretization scheme for solving k-eigenvalue problems on coarse spatial and angular grids in 1D slab geometry. The proposed homogenized scheme is based on the step characteristic (SC) method. It is algebraically consistent with fine-mesh SC equations. We analyze sensitivity of this scheme to perturbations in homogenized cross sections and other coecients of the scheme. The presented results show numerical stability of the developed method to small perturbations in its parameters.

17:45 PM
Spatial Rehomogenization of Cross Sections and Discontinuity Factors for Nodal Calculations

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Cross sections used in nodal calculations (interpolated from multi parameterized tables) come from infinite medium flux homogenization. For highly heterogeneous compositions this average value does not permit to represent correctly the reaction rate when the flux shape is different with respect to the infinite medium shape. The error induced is not negligible, for example, for assemblies with control rod under flux gradient. Several approaches have been proposed to reduce this kind of error. A class of them, called rehomogenization, aims to determine a homogenization correction to the cross sections corresponding to the difference between flux shapes in actual

Track1-2 Reactor Analysis Methods

Session Chair: Richard Sanchez(CEA), Hideki Matsumoto(MHI)

environment and in infinite medium. This kind of approach neglects the discontinuity factors, which remain at the value computed in infinite medium condition. The purpose of the method developed here is to extend the rehomogenization method to the discontinuity factors. The accuracy of the method and its limitations are shown in several significant configurations. Its implementation in the AREVA NP reactor core simulation code ARTEMIS™ has shown better evaluation of control rod worth, which is often overestimated if no correction is applied.

SS2-2 Reactor Physics and Criticality Safety Activities in OECD/NEA Working Party

Session Chair: Mark DeHart(INL), Teruhiko Kugo(JAEA)

15:45 PM
Evaluation of Large 3600MWth Sodium-Cooled Fast Reactor OECD Neutronic Benchmarks

L.Buiron, G.Rimpault, B.Fontaine(1), T.K.Kim, N.E.Stauff, T.A.Taiwo(2), A.Yamaji, J.Gulliford(3), E.Fridmann(4), A.Keresztur i, I.Pataki, A.Tota(5), K.Kugo, K.Sugino, M.M.Uematsu(6), R.Lin Tan, T.Kolowski(7), C.Parisi(8), A.Ponomarev(9)

1)CEA, Saint Paul-lez-Durance, France, 2)ANL, Argonne, USA, 3) OECD/NEA, Issy-les-Moulineaux, France, 4)HZDR, Dresden, Germany, 5)CER, Budapest, Hungary, 6)JAEA, Ibaraki, Japan, 7)UIUC, Urbana, USA, 8)ENEA, Rome, Italy, 9)KIT, Eggenstein-Leopoldshafen, Germany

Within the activities of the Working Party on Scientific Issues of Reactor Systems (WPRS) of the OECD, an international collaboration is ongoing on the neutronic analyses of several Generation-IV Sodium-cooled Fast Reactor (SFR) concepts. This paper summarizes the results obtained by participants from institutions of different countries (ANL, CEA, ENEA, HZDR, JAEA, CER, KIT, UIUC) for the large core numerical benchmarks. These results have been obtained using different calculation methods and analysis tools to estimate the core reactivity and isotopic composition evolution, neutronic feedbacks and power distribution. For the different core concepts analyzed, a satisfactory agreement was obtained between participants despite the different calculation schemes used. A good agreement was generally obtained when comparing compositions after burnup, the delayed neutron fraction, the Doppler coefficient, and the sodium void worth. However, some noticeable discrepancies between the k-effective values were observed and are explained in this paper. These are mostly due to the different neutronic libraries employed (JEFF3.1, ENDFB7.0 or JENDL-4.0) and to a lesser extent the calculations methods.

16:05 PM
Evaluation of Medium 1000 MWth Sodium-Cooled Fast Reactor OECD Neutronic Benchmarks

N.E.Stauff, T.K.Kim, T.A.Taiwo(1), L.Buiron, G.Rimpault, Y.K.Lee, E.Brun(2), A.Yamaji, J.Gulliford(3), N.Guilliard, W.Bernnat(4), M.Uematsu, K.Sugino, T.Kugo(5), I.Pataki, A.Keresztur i, A.Tota(6), A.Ponomarev(7), N.Messaoudi(8), R.Tran, T.Kozlowski(9)

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Within the activities of the Working Party on Reactor and System (WPRS), an international collaboration is ongoing on the neutronic analysis of several Generation-IV Sodium-cooled Fast Reactor (SFR) concepts. Eleven institutions are participating in the analysis of a set of four SFR cores. Two “large” SFR core designs that were proposed by CEA are included in the set. These large SFR cores generate 3,600 MW(th) and employ oxide and carbide fuel technologies. Two “medium” SFR core designs proposed by ANL complete the set. These medium SFR cores generate 1,000 MW(th) and employ oxide and metallic fuel technologies. This paper summarizes the results obtained by the benchmark participants for the medium cores. Nine participating institutions (ANL, CEA/Cadarache, CEA/Saclay, CER, JAEA, KIT, SCK-CEN, UIUC, and IKE) modeled these two SFR cores while using different calculation methods and systems to estimate the k-effective, reactivity feedbacks, isotopic composition evolution, and power distribution. For the different core concepts analyzed, a satisfactory agreement between participants was obtained despite the different calculation schemes used. A good agreement is generally obtained when comparing the burnup composition evolution, the delayed neutron fraction, the Doppler coefficient, and the sodium void worth. However, some noticeable discrepancies between the k-effective values were observed and are explained in this paper. These are mostly due to the different neutronic libraries employed (JEFF3.1, ENDF/B-VII.0 or JENDL-4.0) and in a less extend calculations methods.

16:25 PM
SFR Whole Core Burnup Calculations with TRIPOLI-4 Monte Carlo Code

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Under the Working Party on Scientific Issues of Reactor Systems (WPRS) of the OECD/NEA, an international collaboration benchmark was recently established on the neutronic analysis of four Sodium-cooled Fast Reactor (SFR) concepts of the Generation- IV nuclear energy systems. As the whole core Monte Carlo depletion calculation is one of the essential challenges of current reactor physics studies, the continuous-energy TRIPOLI-4 Monte Carlo transport code was firstly used in this study to perform whole core 3D neutronic calculations for these four SFR cores. Two medium size (1000 MWt) and two large size (3600 MWt) SFR of GEN-IV systems were analyzed. The medium size SFR concepts are from the Advanced Burner Reactors (ABR). The large size SFR concepts are from the self-breeding reactors. The TRIPOLI-4 depletion calculations were made with MOX and metallic U-Pu-Zr fuels for the ABR cores and with MOX and Carbide (U,Pu)C fuels for the self-breeding cores. The whole core reactor physics parameters calculations were performed for the BOEC and EOEC (Beginning and End of Equilibrium Cycle) conditions. This paper summarizes the TRIPOLI-4 calculation results of Keff, βeff, sodium void worth, Doppler constant, control rod worth, and core power distributions for the BOEC and EOEC conditions. The one-cycle depletion calculation results of the core inventory of U and TRU (Pu, Am, Cm, and Np) are also analyzed, after 328.5 days depletion irradiation for the ABR cores, 410 days for the large MOX core, and 500 days for the large carbide core.

16:45 PM
Summary and Status of OECD/NEA UAM-LWR Benchmark

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In recent years there has been an increasing demand from nuclear research, industry, safety and regulation for best estimate predictions to be provided with their confidence bounds. Consequently OECD/NEA has initiated an international Uncertainty Analysis in Modeling (UAM) benchmark focused on uncertainty analysis in best-estimate coupled code calculations for design, operation, and safety analysis of LWRs. The title of this benchmark is: “Benchmarks for uncertainty analysis in modelling (UAM) for the design, operation and safety analysis of LWRs”, or often referred to as “OECD/NEA UAM-LWR Benchmark”. Reference systems and scenarios for coupled code analysis are defined to study the uncertainty effects for all stages of the system calculations. Measured data from plant operation are available for the chosen scenarios. The general frame of the OECD/NEA UAM LWR benchmark consists of three phases with different exercises for each phase. This paper summarizes the benchmark activity and discusses the status of each phase and exercise. Selected results are shown to illustrate the technical approach in defining different exercises and findings from benchmark analysis. For the first time this benchmarks aims to determine the uncertainty in LWR system calculations at all stages of a coupled reactor physics/thermal-hydraulics/fuel performance calculation. The full chain of uncertainty propagation from basic data, engineering uncertainties, across different scales (multi-scale), and physics phenomena (multi-physics) are tested on a number of benchmark exercises for which experimental data are available and for which the power plant details have been released. As part of this effort, the development and assessment of different methods or techniques to account for the uncertainties in the calculations are investigated and reported, which contributes to the state-of-the-art in this area.

17:05 PM
Uncertainty and Sensitivity Analysis of OECD/NEA UAM Fuel Thermal Behaviour Benchmark Using a Falcon/URANIE Methodology

C.Cozzo, Y.Yun, O.Leray, H.Ferroukhi

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Within the framework of the OECD UAM Phase 2 benchmark, an uncertainty and sensitivity analysis of the fuel thermal behavior is performed using the Falcon code along with the URANIE platform. In a

SS2-2 Reactor Physics and Criticality Safety Activities in OECD/NEA Working Party

Session Chair: Mark DeHart(INL), Teruhiko Kugo(JAEA)

first part, focus is on an uncertainty quantification of the predicted fuel temperature pertaining to input parameters specified as manufacturing tolerances of the fuel rod design. It is observed that the mean value of the fuel maximum temperature slightly increases with burnup and that the standard deviation and its variation remain relatively small. In a second part, a global sensitivity analysis method based on Sobol indices is applied in an attempt to estimate the relative contributions of the various input parameters to the fuel temperature uncertainty. It is found that the uncertainty is essentially dependent on the pellet-gap behavior. At lower burnups when the gap is open, the pellet OD and cladding ID are found as the key parameters while after gap closure, all parameters contribute in a collaborative manner to the fuel temperature.

17:25 PM
New PSI Methodology for Manufacturing and Technological Uncertainty Quantification

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A new methodology, referred to as Manufacturing and Technological parameters Uncertainty Quantification (MTUQ), is under development at PSI. Based on global uncertainty and sensitivity analysis methods, MTUQ aims at advancing state-of-the-art for the treatment of geometrical/material uncertainties in LWR computations in general and for the MCNPX Monte-Carlo neutron transport code in particular. For the latter, the development is currently focused primarily on Criticality Safety Evaluations (CSE). In that context, the key components are a dedicated modular interface with the MCNPX code and a userfriendly interface to model functional relationship between system variables. A unique feature is an automatic capability to parameterize variables belonging to so-called ‘repeated structures’ such as to allow for perturbations of each individual element of a given system modelled with MCNPX. Concerning the statistical analysis capabilities, these are currently implemented through an interface with the ROOT platform to handle the random sampling design. This paper presents the current status of the MTUQ methodology development as well as a first assessment on the basis of an on-going OECD/NEA benchmark dedicated to uncertainty analyses for CSE. The presented results illustrate the overall capabilities of MTUQ and underline its relevance in predicting more realistic results compared to a methodology previously applied at PSI for this particular benchmark.

17:45 PM
Re-Evaluation and Continued Development of Shielding Benchmark Database SINBAD

I.A.Kodeli(1), P.Ortego(2), A.Milocco(3), G.Žerovnik(1), R.E.Grove(4), A.Yamaji(5), E.Sartori(6)

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The Expert Group on Radiation Transport and Shielding was started at the OECD/NEA in June 2011 with the mandate to, among others, monitor, steer and support the continued development of the Shielding Integral Benchmark Archive and Database (SINBAD), in cooperation with RSICC. The present status of this database is presented, which grew since its beginnings in the 1990’s to an international reference database containing at present 100 benchmarks relevant for fission, fusion and accelerator shielding applications. As part of its activities a thorough revision of 45 benchmark experiments was completed recently in order to verify in details the completeness and consistency of the benchmark information, in particular concerning the evaluation of the experimental sources of uncertainty. This review process is expected to provide users with a proper choice and help them make better use of the experimental information and is planned to be extended to other available benchmarks.

SS1-2 Molten Salt Reactors

Session Chair: Imre Pazsit(Chalmers Univ. of Tech.), Ritsuo Yoshioka(International Thorium Molten-Salt Forum)

15:45 PM
Hybrid Spectrum Molten Salt Reactor

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The Molten Salt Reactor (MSR) concept has a unique feature, if compared to the majority of other reactor designs, that its fuel is liquid. This property creates, on one hand, several technical challenges; on the other hand, it offers flexibility in shaping and designing of the active core. Accordingly, single fuel fluid can, for instance, circulate through several core zones with different moderation ratios. This possibility was already considered in the past; however, in relation to the thermal MSR. In the presented study, an extreme case of hybrid spectrum MSR is proposed and preliminarily analyzed. It is concluded that hybrid spectrum may provide several advantages and could be applicable especially during the initial phase of the fuel cycle or by the transition to equilibrium cycle.

16:05 PM
Thorium Conversion Optimization in Two-Fluid Molten-Salt Reactor

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Molten-Salt Reactors (MSR) are an attractive reactor system for various purposes. They can be designed to be operated in a fast neutron spectrum for spent fuel transmutation or in a thermal spectrum. Thermal MSRs provide an ideal platformfor conversion of thorium to ^{233}U. Flowing salt can be continuously reprocessed to minimize neutron losses due to neutron absorption in fission products. This study deals with a static neutronic optimization of a Two-Fluid MSR concept. Such a reactor features two separated molten-salt streams in the reactor core. One salt contains fissile material ^{233}U, the other thorium. Separation of these streams improves the conversion capabilities of MSRs. Such a design was analysed for Molten-Salt Breeder Reactor (MSBR) development. This reactor was not realized, but it is used as a reference for this study.
Monte-Carlo code MCNP5 was used to model a simplified MSBR core and for calculation of the breeding capabilities of this design. Several basic geometric parameters were selected for evaluation of their effect on characteristics of the reactor. Based on this analysis, an improved designed was prepared with shorter fissile material doubling time. The whole analysis was carried out for fresh fuel composition. It is possible to expect that on-line fuel reprocessing will limit fuel composition changes during reactor operation. Only effect of ^{233}Pa accumulation on the thorium conversion was studied for several fuel reprocessing rates.

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Development of Computer Code Packages for Molten Salt Reactor Core Analysis

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This paper presents the implementations of the Oak Ridge National Laboratory (ORNL) approach for Molten Salt Reactor (MSR) core analysis with two nuclear reactor core analysis computer code systems. The first code system has been set up with the MCNP6 Monte Carlo code, its depletion module CINDER90 and the PYTHON script language. The second code system has been set up with the NEWT transport calculation module and ORIGEN depletion module connected by TRITON sequence in SCALE code, and the PYTHON script language. The PYTHON script language is used for implementing the online reprocessing of molten-salt fuel, and feeding new fertile material in the computer code simulations. In this paper, simplified nuclear reactor core models of a Molten Salt Breeder Reactor (MSBR), designed by ORNL in the 1960's, and FUJI-U3 designed by Toyohashi University of Technology (TUT) in the 2000's, were analyzed by the two code systems. Using these, various reactor design parameters of the MSRs were compared, such as the multiplication factor, breeding ratio, amount of material, total feeding, neutron flux distribution, and temperature coefficient.

16:45 PM
Use of MCDancoff Factor Correction for Multi-Group Fuel Depletion Analyses of Liquid Salt Cooled Reacotors

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Liquid Salt Cooled Reactors (LSCRs) are high temperature reactors, cooled by liquid salt, with a TRISO-particle based fuel in a solid form (stationary fuel elements or circulating fuel pebbles); this paper is focusing on the former. In either case, due to the double heterogeneity, core physics analysis require different considerations with more complex approaches than LWRs core physics calculations. Additional challenges appear when using the multi-group approach. In this paper we examine the use of SCALE6.1.1. Double heterogeneity may be accounted for through the Dancoff factor, however, SCALE6.1.1 does not provide an automated method to calculate Dancoff Factors for fuel planks with TRISO fuel particles. Therefore, depletion with continuous energy Monte Carlo Transport (CE depletion) in SCALE6.2 beta was used to generate MC Dancoff factors for multi-group calculations. MCDancoff corrected multi-group depletion agrees with the results for CE depletion within ±100 pcm, and within ±2σ. Producing MCDancoff factors for multi-group (MG) depletion calculations is necessary to LSCR analysis because CE depletion runtime and memory requirements are prohibitive for routine use. MG depletion with MCDancoff provides significantly shorter runtime and lower memory requirements while providing results of acceptable accuracy.

17:05 PM
Comparative Studies on Plutonium and ^{233}U Utilization in miniFUJI MSR

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Molten salt reactor (MSR) has many merits such as safety enhancement and capability to be used for hydrogen production. A comparative evaluation of plutonium and ^{233}U utilization in miniFUJI MSR has been performed. Reactor grade plutonium (RGPu), weapon grade plutonium (WGPu), and super grade plutonium (SGPu) have been utilized in the present study. The reactors can obtain their criticality condition with the ^{233}U concentration in the Th-^{233}U fuel, RGPu concentration in Th-RGPu fuel, WGPu concentration in Th-WGPu fuel, and SGPu concentration in Th-SGPu fuel of 0.52%, 5.76%, 2.16%, and 1.96%, respectively. The Th-^{233}U fuel results in the soft neutron spectra of miniFUJI reactor. The neutron spectra turn into harder with the enlarging of plutonium concentration in loaded fuel where Th-RGPu fuel gives the hardest neutron spectra.

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

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The molten salt reactor technology has gained renewed interest. In contrast to the historic molten salt reactors, the current projects are based on designing a molten salt fast reactor. Thus the shielding becomes significantly more challenging than in historic concepts. One very interesting and innovative result of the most recent EURATOM project on molten salt reactors – EVOL – is the fluid flow optimized design of the inner core vessel using curved blanket walls. The developed structure leads to a very uniform flow distribution. The design avoids all core internal structures. On the basis of this new geometry a model for neutron physics calculation is presented and applied for a shielding optimization. Based on these results an optimized shielding strategy is developed for the molten salt fast reactor to keep the fluence in the safety related outer vessel below expected limit values. A lifetime of 80 years can be assured, but the size of the core/blanket system has to be significantly increased and will finally be comparable to a sodium cooled fast reactor. The HELIOS results are verified against Monte-Carlo calculations with very satisfactory agreement for a deep penetration problem.

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

Session Chair: Stefano Monti(IAEA), Shigeo Ohki(JAEA)

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

V.Pascal, G.Prulhiere, M.Vanier, B.Fontaine(1), F.Varaine(2)

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In 2007 the IAEA, within the framework of the Technical Working Group on Fast Reactors (TWG-FR) activities, decided to launch a Coordinated Research Project (CRP), devoted to benchmarking analyses on “Control Rod Withdrawal Test” performed during the “PHENIX End-of-Life Experiments”. This test program was conducted by the CEA, EDF and AREVA before the final shutdown of the prototype power fast reactor PHÉNIX in order to gather important data and knowledge about several aspects of the operation and safety of pool-type sodium-cooled fast reactors. The overall CRP objective was to improve the participants’ analytical capabilities in various fields of research and design of sodium-cooled fast reactors.
Among the accident sequences that are to be taken into account, inadvertent withdrawal of a control rod is considered. During operation at nominal power, such a sequence induces a general power increase and local deformations of the power shape. Afterwards, the local fuel temperature increases can thereby lead to fuel melting and clad failure. The quasi-static control rod withdrawal test was especially designed to gather power local data on fissile sub-assemblies and to complete validation databases of neutronic codes. The maximal deformation of the power shape reached ±12% and was obtained when two control rods were shifted in opposite directions.
After a description of the test and the measurement methods, this paper presents some results obtained in the course of the test with special emphasis on control rod efficiencies and power deformation by subassemblies. This paper also discusses CEA results obtained in the course of the benchmark with the European neutronic code used for fast reactors design, ERANOS-2.2.

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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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This paper describes details of the IAEA/CRP benchmark calculation by JAEA on the control rod withdrawal test in the Phenix End-of-Life Experiments.
The power distribution deviation by the control rod insertion/withdrawal, which is the major target of the benchmark, is well simulated by calculation. In addition to the CRP activities, neutron and photon heat transport effect is evaluated in the nuclear heating calculation of the benchmark analysis. It is confirmed that the neutron and photon heat transport effect contributes to the improvement of the absolute power calculation results in the breeder blanket region.

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

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The post-analysis of control rod withdrawal end-of-life experiments, performed in the PHENIX reactor during June 2009, were carried out at IGCAR as a part of IAEA Coordinated Research Project (CRP) on “Control rod withdrawal and sodium natural circulation tests performed during the PHENIX end-of-life experiments”. The main objective of this CRP was to assess the prediction capability of distorted radial power distribution due to absorber rod movement (insertion and/or withdrawal) at nominal power. In addition, core reactivity, absorber rod worth, S-curve, maximum neutron flux, SA-wise power and sodium heating deviation with respect to the reference core for various critical core configurations were analyzed. 3-D diffusion theory calculations using FARCOB (IGCAR) and ERANOS-2.1 (European) code systems were made for this analysis. IGCAR prediction of radial power distribution due to absorber rod movement is very close to the measured values. No significant deviation is observed between the results of FARCOB and ERANOS-2.1 and also between the results of other CRP participants. This benchmark exercise has provided a wide international forum to verify the method, computer codes and cross section data employed at IGCAR for the physics design calculations of sodium cooled fast reactors, by keeping the experimental data as the reference.

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

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The Institute of Radioprotection and Nuclear Safety (IRSN) being established as technical support organization for French public authorities is in charge of safety assessment of both operating and under construction reactors and nuclear facilities. It provides safety studies of advanced and innovative projects like fast sodium cooled reactors as well. In this context, one of the IRSN objectives is to evaluate comprehensively the accuracy of numerical tools and their performance on studies of safety relay items. Reactor physics studies step in the safety assessment support from different points of view, among which the design of core and its protection system. They are essential in the cores behavior analysis in normal, perturbed and accidental conditions in order to assess the integrity of the first barrier and the exclusion of prompt criticality and re-criticality risks. The codes capability to compute in an accurate manner the fission power distribution in the core during the whole reactor lifetime could indicate the codes’ accuracy for many so-called spatial dependent values calculations. The IAEA Coordinated Research Project on the Phénix end-of-life test “Control Rod Withdrawal” has been a good opportunity to check the capability of calculation tools by comparison with the measured radial power distributions on fast reactor. IRSN participated to this benchmark with the ERANOS code package developed by CEA for fast reactors studies. The challenge for this code package was that in the considered core configurations the neutron fields were notably deformed. This paper presents the results obtained in the framework of the benchmark activity. A relatively good agreement has been found with available measures considering the approximations done in the modeling. The work underlines the importance of precise knowledge of the details of burn-up distribution as it could impact the calculations of the power distribution.

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Simulation of PHENIX Control Rod Withdrawal Experiments with SIMMER-IV

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The “end-of-life” tests performed in the PHÉNIX reactor in 2009, in particular the Control Rod (CR) withdrawal experiments provide an excellent opportunity for validation and verification of the reactor physics computer codes and modelling approaches. SIMMER-IV, a modern three-dimensional reactor safety code, has been recently employed at KIT for simulating these experiments in the framework of a benchmark exercise organized under the IAEA project. In this paper, we report and discuss main results obtained with SIMMER at KIT. The reactor reactivity, power and neutron flux distributions calculated with SIMMER-IV are in good agreement with advanced neutronics codes, such as ERANOS, while the CR reactivity worth is overestimated due to neglecting heterogeneity effects. We show that SIMMER neutronics model can be improved by employing a correction that is based on the results of cell calculations performed with ERANOS. The study confirms that the 3D SIMMER-IV code can accurately predict major fast reactor neutronics parameters, provided that a special treatment is employed for CR modelling.

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

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Neutron cross-section processing for fast reactor sub-regions containing control rods has to take into account heterogeneity effects

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in order to get a reliable assessment of the control rod reactivity worth. Several numerical methods have been developed in the past with a support of experimental campaigns and are employed nowadays to reasonably treat such effects by using sophisticated neutronics codes. The SIMMER code is employed at KIT for severe accident analyses of metal-cooled fast reactors and other reactor systems. Neutron cross-sections are processed in the original SIMMER version by approximating each reactor region as a homogeneous medium. This simplified treatment results in the overestimation of the control rod worth. Efforts are therefore going on to extend the code in order to take into account heterogeneity effects in the control rod subassemblies. In this paper a technique proposed in the past was applied to take into account these effects as basis for further SIMMER extensions. With this aim a 3D (HEX-Z) ERANOS model of the PHÉNIX reactor has been assessed. Further, the corresponding 3D (XYZ) PARTISN model has been employed, the latter code being now introduced in SIMMER as a new neutronics solver. Results show that the employed technique improves the capability of SIMMER and ERANOS codes to predict the control rod reactivity worths. Results also show that the HEX-Z ERANOS and XYZ PARTISN models for PHÉNIX reasonably agree.

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

V.Pascal, G.Prulhiere, M.Vanier, B.Fontaine(1), K.Devan, P.Chellapandi(2), V.Kriventsev(3), S.Monti(4), K.Mikityuk(5), A. Chenu(5), M.Semenov(6), T.Taiwo(7), K.Takano(8), V.Tiberi(9)

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In 2007, the IAEA, within the framework of the Technical Working Group on Fast Reactors (TWG-FR) activities, decided to launch a Coordinated Research Project (CRP) devoted to benchmark analyses of the “Control Rod Withdrawal Test performed during the PHÉNIX End-of-Life Experiments”. This test program was conducted by the CEA, EDF and AREVA before the final shutdown of the French prototype power fast reactor PHÉNIX in order to gather important data and knowledge about several aspects of the operation and safety of pool-type sodium-cooled fast reactors. The overall CRP objective was to improve the participants’ analytical capabilities in various fields of research and design of sodium-cooled fast reactors.
Among the accident sequences that are to be taken into account, the inadvertent withdrawal of a control rod is considered. During operation at nominal power, such a sequence induces a general power increase and local deformations of the power shape. Afterwards, the local fuel temperature increases can thereby lead to fuel melting and clad failure. The quasi-static control rod withdrawal test was specifically designed to gather power dataof fissile sub-assemblies and to complete validation databases of neutronic codes. The description of the test results are presented in reference. The maximal deformation of the power shape reached ±12% and was obtained when two control rods were shifted in opposite directions.
After a description of the neutronic codes used in the benchmark and their respective assumptions, this paper presents the results obtained in the course of the benchmark as well as the comparisons with experimental parameters with special emphasis on reactivity estimations, control rod efficiencies, power estimations and power deformation by sub-assemblies.

Track11-2 Research Reactors and Spallation Sources

Session Chair: Gert Van den Eynde(SCK/CEN), Cheolho Pyeon(KURRI)

15:45 PM
A Method for Reactivity Monitoring in Subcritical Source-Driven Systems

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For the operation of subcritical source-driven systems it is very important to develop reliable techniques to monitor the subcriticality level, in order to be able to establish the effective distance from criticality and promptly detect changes in the reactivity that may be relevant for the safety of the plant. In this paper, a new method to interpret flux measurements in a subcritical system is presented, generalizing a technique that has been recently proposed for source-free nuclear systems. The method is based on the mathematical relationship between the power, its derivative, the convolution integral appearing in the delayed neutron precursor balance equations and the stable period of the multiplying system within the point kinetic model. The assessment of the method is carried out interpreting flux evolutions obtained from numerical transient simulations. The results presented prove that the method can yield accurate reactivity predictions for various physical situations and can be of interest for accelerator-driven system technology.

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Intepretation 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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Within the FREYA Project, methods to interpret flux measurements in the VENUS-F core are being studied in order to reconstruct the subcriticality level of the facility. In this work, after the presentation of results obtained with standard techniques such as the Area Method, we introduce an alternative approach to the experimental determination of the reactivity. This method, whose validity has been tested by computational exercises, makes use of general mathematical properties of the point kinetics system of equations and has been recently extended for subcritical system analysis. The evaluation of spatial correction factors is also carried out using deterministic transport evaluations (ERANOS code). The statistical and systematic uncertainty of the results in terms of reactivity is discussed and numerical results are presented.

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

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1)Kyoto University, Research Reactor Institute, Osaka, Japan, 2)Kyoto University, Graduate School of Energy Science, Kyoto, Japan

At the Kyoto University Critical Assembly, a series of the accelerator-driven system (ADS) experiments are being carried out by coupling with the fixed-field alternating gradient (FFAG) accelerator, and the spallation neutrons generated by 100 MeV protons from the FFAG accelerator are successfully injected into the cores. In the ADS experiments, the neutron characteristics are investigated through the static and kinetic analyses, when the source neutron spectrum is varied by the kind of solid target (W, W+Be and Pb-Bi). From the results of the experiments, the neutron yield is found to be large in the W target, whereas the discrepancy is observed in the comparison between the experiments and the calculations. The static parameters are estimated to be fairly well in the analyses of neutron multiplication and subcritical multiplications factor, with the use of ^{115}In(n, γ)^{116m}In reaction rate distribution in the core region. From the kinetic experiments, the dependence on source neutron spectrum is significantly observed in the prompt neutron decay constant and subcriticality, when the kind of solid target is varied.

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

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Kyoto University Research Reactor Institute, Osaka, Japan

The neutron spectrum in the in-core irradiation equipments between LEU and HEU core are compared and irradiation characteristics of the equipments in LEU core are evaluated by using the MVP2.0 with the JENDL-4.0. Compared with the HEU core, the thermal neutron flux in the LEU fuel is decreased by about 14% because of the thermal neutron capture by U-238. The thermal neutron flux in the HYD and the SSS is decreasing about 4 to 8% compared with the HEU core because those irradiation equipments are located at the center of core and near the fuel region. On the other hand, since the pneumatic tubes and the SLANT are located at the graphite reflector or the outside of the reflector, the differences in the thermal neutron fluxes at the irradiation equipments between the LEU core and the HEU core are about 1 to 5 %. The difference in the thermal flux between LEU core and HEU core becomes smaller, if an irradiation field is positioned at a reflector region. In addition, the experimental validation via irradiation of MoO3 pellet is carried out. As the result, the quantity of Mo-99 produced by the (n, γ) method is depend on the neutron spectrum in the irradiation field.

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Accuracy of Thorium-Loaded Accelerator-Driven System Experiments at Kyoto University Critical Assembly

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1)Graduate school of Energy Science, Kyoto University, Kyoto, Japan, 2) Nuclear Science Engineering Division, Research Reactor Institute, Kyoto University, Osaka, Japan

Thorium plate irradiation and thorium-loaded accelerator-driven system (ADS) experiments are carried out at the Kyoto University Critical Assembly (KUCA) to examine the accuracy of reaction rates, subcriticality and neutron profiles on thorium capture reactions in the critical and subcritical states for the ADS experiments with 14 MeV neutrons and 100 MeV protons. In the previous study at KUCA, thorium capture and fission reaction rates were measured, and showed a large discrepancy between the calculations and experiments. From the lessons learned from previous study, the thorium foil is employed in this study because the thorium fuel plate could not be suitable for the improvement of the experimental errors caused by the effects of self-shielding, volume source, and perturbation of neutron flux from its large dimension. The difference between the reaction rate C/E (calculation / experiment) values, with the use of thorium foil, is within the relative difference of 10%. Here, the numerical analysis is conducted by MCNPX-2.5.0 with ENDF/B-VII.0 and JENDL/D-99. Subcritical measurements with 14 MeV neutrons and 100 MeV protons indicate different prompt neutron decay constants at the same core, when the neutron source is varied. The neutron profiles on thorium capture reactions reveal good agreement between the calculation and the experiments.

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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

Jules Horowitz Material Testing Reactor (JHR) is planned to be the first European nuclear experimental facility of next decades thanks to its testing capacity. High flux level according to 100 MW power is exploited through many test slots. Fast core spectrum allows high dose rates for material testing and thermal neutron flux is achieved inside a large reflector. Here fuel samples are irradiated inside experimental devices – namely MADISON, ADELINE and MOLFI – and each specific power is then worth to be evaluated for safety reasons. Moreover, devices transients require particular analyses for reactor shutdown conditions, in order to evaluate power behavior. All nuclear heating effects are concerned and related time-dependent description is carried out in this work. First, thermal hydraulic and

Track11-2 Research Reactors and Spallation Sources

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neutronic core model is implemented through DULCINEE code to obtain core transients. Then, detailed power calculations for reflector devices are obtained through an enhanced multi-point kinetics model accounting for every device which is now thought of as a single lumped system - coupled with reactor core as external source. Core-device coupling coefficients to define this model are finally obtained by means of Monte Carlo simulations with TRIPOLI 4.8 code, about different core fuel compositions – namely Beginning of Cycle (BOC), Xenon Saturation Point (XSP), Middle of Cycle (MOC) and End of Cycle (EOC). Complete power deposition in devices is obtained through TRIPOLI simulations considering promp

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Neutronic Analysis of the PULSTAR Reactor Using Monte Carlo Simulations

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Neutronic analysis of the PULSTAR nuclear reactor was performed in support of its utilization and power upgrade from 1-MWth to 2-MWth. The PULSTAR is an open pool research reactor that is currently fueled with UO2 enriched to 4% in U-235. Detailed models were constructed of its core using the MCNP6 Monte Carlo code and its standard nuclear data libraries. The models covered all eight variations of the core starting with the first critical core in 1972 to the current core that was configured in 2011. Three dimensional heterogeneous models were constructed that faithfully reflected the geometry of the core and its surroundings using the original as-built engineering drawings. The Monte Carlo simulations benefited extensively from measurements that were performed upon the loading of each core and its subsequent operation. This includes power distribution and peaking measurements, depletion measurements (reflecting a core's excess reactivity), and measurements of reactivity feedback coefficients. Furthermore, to support the PULSTAR's fuel needs, the simulations explored the utilization of locally existing inventory of fresh UO_2 fuel that is enriched to 6% in U-235. The analysis shows reasonable agreement between the results of the MCNP6 simulations and the available measured data. In general, most discrepancies between simulations and measurements may be attributed to the limited knowledge of the exact conditions of the historical measurements and the procedures used to analyze the measured data. Nonetheless, the results indicate the ability of the constructed models to support safety analysis and licensing action in relation to the on-going upgrades of the PULSTAR reactor.

Track1-3 Reactor Analysis Methods

Session Chair: Han Gyu Joo(SNU), Wei Shen(CCSN)

8:00 AM

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

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

The in-core fuel management optimisation (ICFMO) problem has been studied for several decades. Very little research has, however, been aimed at multiobjective optimisation involving the fundamental notion of Pareto optimality. In this paper, the recently developed multiobjective optimisation using the cross-entropy method (MOO CEM) algorithm is applied to a multiobjective ICFMO problem for the first time. A derivation of the MOO CEM algorithm is presented for ICFMO, along with a constraint handling technique. The algorithm is applied to a biobjective test problem for the SAFARI-1 nuclear research reactor. The Pareto set approximated by the algorithm is compared to solutions obtained by typical operational reload strategies. The results indicate that the MOO CEM algorithm for multiobjective ICFMO is a robust and efficient method which is able to obtain a good spread of trade-off solutions. The method may therefore greatly aid in the decision making of a reactor operator tasked with designing reload configurations.

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Transient Cycle Fuel Management Optimization of a Pressurized Water Reactor

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This paper concerns with how to optimally determine enrichments of fuel assembly (FA) batches of beginning-of-life (BOL) and reload cycle cores of a pressurized water reactor (PWR) plant which runs on a multi-batch, multi-cycle fuel management scheme. As a way to determine the optimum FA enrichments, a multi-cycle, multi-objective FA loading pattern (LP) optimization problem for the transient cycle cores involving the BOL and the reload cycle cores of the PWR plant is solved by the adaptively constrained discontinuous penalty function-based (ACDPF-based) multi-objective simulated annealing (MOSA) algorithm in combination with the commercial core neutronics design code ASTRA (Advanced Static and Transient Reactor Analyzer). The applicability and the effectiveness of the ACDPF-based MOSA algorithm is examined in terms of its solution to the first three transient cycle FA LP optimization problem of Yonggwang Nuclear Unit 4 (YGN4) a PWR plant in Korea. The practicality and usefulness of the ACDPF-based MOSA algorithm as an optimizer to determine optimum enrichments of BOL and reload cycle cores are discussed.

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

T.Okubo, T.Endo, A.Yamamoto

Nagoya University, Nagoya, Japan

Efficient and rapid computation of multiple loading patterns using GPU is studied aiming application to loading pattern optimization of LWR. The loading pattern has significant impacts on safety and economy of a reactor. However, design of loading pattern is a combinatorial optimization problem, thus it is computationally intensive task. In order to address this issue, efficient and rapid computation method of loading patterns using massively parallel computing capability of GPU is studied in the present paper. Though GPU has higher computational performance than CPU, but different computational algorithm and coding approach are necessary to maximize the performance of GPU, due to different architecture of GPU. In the present study, a multi-level parallel computing approach is examined considering hardware architecture of GPU, i.e., parallel computing is carried out not only in spatial mesh-wise, but also in loading pattern-wise. In other words, multiple loading patterns are simultaneously computed and domain (mesh-wise) decomposition is applied to each loading pattern. With the present approach, computational efficiency using GPU is approximately four times higher than that of CPU. The present core analysis algorithm can be used for screening of poor loading patterns in optimization process.

9:00 AM

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

C.Wang, H.S.Abdel-Khalik

North Carolina State University, North Carolina, USA

In this manuscript we have presented the application of exact-to-precision generalized perturbation theory (E_PGPT) we recently developed to the reactor design calculations. E_PGPT approach can be used to reconstruct the neutron flux and the responses of interest by means of a Neumann series expansion. Their application to reactor design calculations can be performed by pre-computing a set of forward fluxes and GPT fluxes, in this work the main details about how this is done are presented. The E_PGPT approach makes use of both GPT approach and state reduction methods, i.e. proper orthogonal decomposition or POD of snapshots, to reduce the number of model executions and to allow one to calculate the higher-order variations in the responses of interest due to general parameters perturbations without having to re-execute the forward model. The computational cost is also inexpensive as it only involves simple inner products operations. This provides an enabling tool to analyze the impact of parameters variations on the responses variations as many times as needed for reactor design calculations.

9:20 AM

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

C.Kennedy, H.Abdel-Khalik

North Carolina State University, North Carolina, USA

For reactor design calculations, generalized perturbation theory (GPT) has been recognized as the most computationally efficient approach to performing sensitivity analyses of models with many input parameters, i.e. the majority of reactor design problems during at least one component of the design phase. Most of these design problems are multi-component models where one or more of the codes lack GPT functionality. Additionally, stochastic Monte Carlo codes may not have a well-understood GPT formulation that can be implemented into the software. Without GPT support, the computationally efficient approach to performing a sensitivity analyses is unavailable or computationally infeasible. The sensitivity analysis is a critical step in determining a preferred solution or propagating uncertainties through a series of design codes.

This paper uses a fundamental adjoint algorithm, denoted GPT-Free, that emulates depletion perturbation theory (DPT) without the need to set up or solve the GPT equations. Specifically, this algorithm constructs a reduced order model (ROM) by developing a subspace from the set of response sensitivity profiles formed using perturbation theory. This algorithm determines a ROM that is independent of the number of responses and is instead dependent on the rank or complexity of the underlying physical model.

This paper demonstrates the GPT-Free algorithm for depletion reactor calculations performed in SCALE6 using a TMI 9 assembly mini-core model. The reactor, which is depleted to 8.1 GWd, is used to evaluate the algorithm's ROM reduction error with the κ-metric. The algorithms results are then benchmarked against forward sensitivities.

9:40 AM

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

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Geometry distortions (bowing, flowering, and axial swelling of fuel assemblies) provide crucial negative reactivity feedback in fast reactors. Currently, no generic, accurate, and efficient method exists for computing distortion reactivities. Our solution is the "virtual density" theory of neutronics, which alters material density (isotropically or anisotropically) instead of explicitly changing geometry. While geometry is discretized, material densities occupy a continuous domain; this allows density changes to obviate any re-meshing required for geometry changes. Although rudimentary forms of this theory exist in Russian literature, they are only applicable to cases in which entire cores swell uniformly. Thus, we conceive a generic and pragmatic form of "virtual density" theory to model non-uniform

Track1-3 Reactor Analysis Methods

Session Chair: Han Gyu Joo(SNU), Wei Shen(CCSN)

and localized geometry distortions. We implement this theory in finite difference diffusion for 3-D coarse mesh models of the Fast Flux Test Facility (FFTF) and Jōyō benchmarks. We model a panoply of non-uniform anisotropic swelling scenarios, including axial swelling of individual assemblies, axial swelling of each mesh cell in proportion to its fission power, and radial core flowering with arbitrary axial dependence. In 3-D coarse mesh Cartesian cores with explicit coolant gaps, we model assembly axial swelling and assembly row motion with arbitrary axial dependence. We validate “virtual density” theory in two ways: (1) by building distorted geometries in continuous energy Monte Carlo or (2) by slightly perturbing dimensions of mesh cells in diffusion. In all examined cases, distortion reactivity coefficients agree with Monte Carlo well within 2σ uncertainty and with diffusion references precisely. We culminate with the VirDenT code, which computes reactivity insertions for arbitrary 3-D assembly distortions in realistic fast reactor designs. VirDenT and Monte Carlo typically agree within a few pcm.

Track5-1 Nuclear Criticality Safety

Session Chair: John Bess(INL), Toshihiro Yamamoto(KURRI)

8:00 AM
Nuclear Criticality Safety in the United States: Recent Events, Trends and a Review of the Safety Culture

M.S.Hodges, C.E.Sanders

University of Nevada, Las Vegas, USA

If nuclear energy is to play an important role in our economies in the future, fissile materials must be handled safely over the whole fuel cycle. Although a wealth of information is available from more than 50 years of cumulative knowledge acquired, case-specific analyses will be needed and will dominate criticality safety. There have been no nuclear criticality accidents in the United States since 1978, although several significant near misses have occurred. Previously, information on these near misses has not been compiled into a single source, which has left a missing piece of the puzzle in criticality safety. To alleviate this, these events are reviewed and a new system of criticality accident near miss categorization has been developed. Analysis of the information shows the nuclear criticality accident safety culture practiced in the United States has been successful in eliminating serious events. In the coming years there will likely be further clarification of potential nuclear fuel cycle strategies, each one with its specific needs in criticality safety. Criticality safety calls for constant support and attention. A sound understanding and correct application of the principles of nuclear criticality safety are vital to the nuclear industry. The objective is to pursue an accidentfree goal, while keeping in mind the repercussions that an avoidable criticality excursion could have. Diversity in national policies has existed and is likely to exist into the near future. Therefore, it is worth sharing criticality information beyond national policies.

8:20 AM
A New OECD/NEA Database of Nuclide Compositions of Spent Nuclear Fuel

F.Michel-Sendis, M.Bossant, N.Soppera(1), I.Gauld(2)

1)OECD Nuclear Energy Agency, Issy-Les-Moulineaux, France, 2)Oak Ridge National Laboratory, Tennessee, USA

The SFCOMPO database of nuclide compositions of spent nuclear fuel is hosted by the OECD Nuclear Energy Agency since 2001. Since 2011, a collaborative effort led by the OECD/NEA Data Bank and Oak Ridge National Laboratory, under the guidance of the NEA Expert Group on Assay Data of Spent Nuclear Fuel, has resulted in the creation of a new enhanced relational database structure and a significant expansion of SFCOMPO, now containing experimental assay data for a wide selection of international reactor designs. This paper aims at describing the new SFCOMPO Database developed at NEA in terms of functionalities, contents and foreseen developments. This new database is expected for public release in 2014.

8:40 AM
OECD EGBUC Benchmark VIII - Comparison of Calculation Codes and Methods for the Analysis of Small-Sample Reactivity Experiments

P.Leconte, A.Santamarina

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Small-sample reactivity experiments are relevant to provide accurate information on the integral cross sections of materials. One of the specificities of these experiments is that the measured reactivity worth generally ranges between 1 and 10 pcm, which precludes the use of Monte Carlo for the analysis. As a consequence, several papers have been devoted to deterministic calculation routes, implying spatial and/or energetic discretization which could involve calculation bias. Within the Expert Group on Burn-Up Credit of the OECD/NEA, a benchmark was proposed to compare different calculation codes and methods for the analysis of these experiments. In four Sub-Phases with geometries ranging from a single cell to a full 3D core model, participants were asked to evaluate the reactivity worth due to the addition of small quantities of separated fission products and actinides into a UO_2 fuel. Fourteen institutes using six different codes have participated in the Benchmark. For reactivity worth of more than a few tens of pcm, the Monte-Carlo approach based on the eigen-value difference method appears clearly as the reference method. However, in the case of reactivity worth as low as 1 pcm, it is concluded that the deterministic approach based on the exact perturbation formalism is more accurate and should be preferred. Promising results have also been reported using the newly available exact perturbation capability, developed in the Monte Carlo code TRIPOLI4, based on the calculation of a continuous energy adjoint flux in the reference situation, convoluted to the forward flux of the perturbed situation.

9:00 AM
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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In the Fukushima-Daiichi Nuclear Power Station units 1, 2 and 3, we have estimated that the nuclear fuel bundles have become non-uniform fuel debris, which could not remain at the normal core location in the reactor pressure vessel (RPV), but could have fallen to the RPV bottom head and accumulated in the pedestal floor. At the present, the fuel debris is estimated to be sub-critical by measurements of short life time fission product (FP) gas and water temperature. In the phase of introducing small circulation loop or fuel debris removal, the part of fuel debris may flow into the liquid waste treatment and cooling facilities. Thus, we have studied the criticality scenarios on each location where fuel debris is estimated to exist. The criticality evaluation based on the criticality scenarios has been performed with conservative conditions, because the debris properties are unidentified. The criticality evaluation shows the range of condition for criticality.

9:20 AM
Design of an Efficient Calculation Model of BWR Cold Critical Experiments for Validation

A.Ranta-aho

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The term burnup credit is used when the calculated spent fuel composition is credited in the criticality safety analysis as opposed to the fresh fuel assumption. Applicable standards place requirements for the validation of the burnup codes that are used in the analysis. Unfortunately, there is a lack of high quality BWR radiochemical assay data suitable for validation. In order to circumvent this difficulty, BWR cold critical experiments could be used for the validation.
A disadvantage in the use of reactor measurements is the number of detail that needs to be fed into the calculation model. An accurate modelling would require thousands of assembly burnup calculations and setting up a core model with hundreds of thousands of fuel material compositions and different control rod designs present in the core. Clearly, a simplified approach would be very valuable for the modelling of cold critical experiments with Monte Carlo codes.
A simplified way of modelling BWR cold critical experiment has been considered in this work. In this approach, only the most relevant part of the core is described in a detailed manner and suitable boundary conditions are applied in other parts of the core by replacing the assembly and control rod data with representative designs.
In this work BWR cold critical measurements of Olkiluoto 1 and Olkiluoto 2 units were used to demonstrate the quality of the approach. SIMULATE-3 calculations were made in order to compare different calculation models for 58 cold critical experiments. The results show that the simplified core model with suitable boundary conditions is robust, accurate and neutronically equivalent with the detailed model.
The results suggest that instead of modelling all 500 assemblies in the core including nodal burnup calculations with a depletion code, only 48 assemblies need to be considered. Futhermore, instead of modelling all control rod types in the core, considering one or two rod designs is sufficient for validation purposes. This means that the computational effort can be reduced by more than an order of magnitude without significant effect on the final results. For the same reason the work needed to prepare benchmarks from the experiments is reduced significantly. The procedure can be looked upon as being a conservative calculation model for criticality code validation or a way to create simplified benchmarks from cold critical experiments.

Track5-1 Nuclear Criticality Safety

Session Chair: John Bess(INL), Toshihiro Yamamoto(KURRI)

9:40 AM

First Burnup Credit Application for Transport and Storage Cask Using French Experiments

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The burnup credit (BUC) methodology for a transport and storage cask application including actinides and fission products is implemented at AREVA TN using the French BUC calculation route for PWR UO_2 used fuel. The methodology is based on the connection of the French depletion code DARWIN2 and the French Criticality Safety Package CRISTAL V1. The BUC methodology includes the experimental validation of the computation codes dedicated to the calculation of the used fuel inventory calculations. Indeed, the results of the comparison Calculation-Experiment (C-E)/E allow to determine either a set of isotopic correction factors (ICF) for the BUC nuclides considered in the criticality calculation or keff-penalty terms directly used for the definition of the keff-acceptance criterion for the criticality assessment of the transport and storage cask. These ICFs or keff-penalty terms are one of the key of the BUC method to guarantee the conservativeness of the fuel reactivity in safety-criticality calculations using BUC approach.
A French BUC program has been developed at CEA/Cadarache in the framework of the CEA-AREVA collaboration in order to validate fuel inventory calculations. This program involves two kinds of experiments: chemical analyses and microprobe measurements of PWR irradiated fuel pins (French P.I.E. program) on one hand; and reactivity worth measurements of the BUC nuclides in the MINERVE reactor on the second hand.
This paper highlights, through a first industrial AREVA TN's application of the BUC method including fission products, that the French P.I.E. program and reactivity worth measurements in MINERVE reactor are suitable for the implementation of burnup credit in transport and storage cask applications loaded with PWR-UOX used fuels assemblies.

Track2-1 Deterministic Transport Theory

Session Chair: Nam Zin Cho(KAIST), Masato Tabuchi(NEL)

8:00 AM
A Collision Probability Based Method to Compute Cross Sections Sensitivities for the Subgroup Self-Shielding Technique

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École Polytechnique de Montréal, Montréal, Canada

We present a method to evaluate self-shielded cross sections sensitivity to light isotopes densities. Our method is based on a subgroup self-shielding model, using physical probability tables. We propose a simplified way to evaluate the derivatives of the collision probabilities, and show these can be used to compute the derivatives of the weight function required to average resonant cross sections in a multigroup scheme. We present self-shielded cross sections sensitivity coefficients based on a variation of the hydrogen density for a relatively simple lattice, and we compare with reference values using complete calculations.

8:20 AM
Improvement of a Convergence Technique for MOC Calculation with Large Negative Self-Scattering Cross Section

M.Tabuchi, M.Tatsumi(1), A.Yamamoto, T.Endo(2)

1)Nuclear Engineering, Ltd., Osaka, Japan, 2)Nagoya University, Nagoya, Japan

In the method of characteristics, convergence behavior of inner iteration would be degraded by the transport correction. A previous study revealed that large negative self-scattering cross section due to the transport correction worsens convergence of inner iteration. In order to resolve this issue, a technique to improve the convergence behavior based on the successive over relaxation (SOR) method was proposed in the previous study. However, considerable calculation time is necessary to estimate the optimum acceleration factor for the SOR method especially in large scale geometries. In order to address this drawback, a fast estimation scheme of the acceleration factors is proposed in this study. The present approach can provide acceleration factors which ensure convergence of iterative calculation. In the present approach, the equation to obtain the appropriate acceleration factor is derived by using Gerschgorin's theorem. Based on the equation, the acceleration factors can be easily obtained in each energy group with negligible calculation time. Verification calculations are performed in various critical experiments, in which degradation of convergence by the transport correction is significant. When the transport correction is used without the SOR method, all iterative calculations diverged. On the other hand, when the SOR method with the acceleration factors based on the present approach is applied, convergent solutions are successfully obtained in all cases. From these results, validity of the present approach is confirmed.

8:40 AM
Boundary Acceleration Techniques for CMFD-Accelerated 2D-MOC

S.G.Stimpson, B.S.Collins, B.M.Kochunas, T.J.Downar

University of Michigan, Michigan, USA

Coarse mesh finite difference (CMFD) has been used successfully for accelerating the eigenvalue and scalar fluxes for transport problem solutions. In the conventional use of CMFD to accelerate transport solution methods, it provides homogenized scalar fluxes on the pin- or assembly-basis and drives the eigenvalue calculation. The transport sweeper provides currents at coarse mesh boundaries which are used to inform the CMFD current coupling coefficients. In this paper we present three methods that use the CMFD solutions to modify the incoming flux boundary condition of the 2D-MOC sweeps that are the primary driver of the transport solution. The first technique (P0) uses a ratio of the scalar surface fluxes to scale the incoming angular flux. This is the most basic technique, since all incoming angular fluxes are scaled isotropically. The second technique (DP0) uses a ratio of the partial currents to scale the incoming angular flux, so the update is angularly dependent in half-space. The third technique (P1) uses the P1 approximation of the angular flux to compute the scaling factor for the angular flux. This requires both the scalar flux and net current on each coarse mesh surface. These techniques can be applied to both system and parallel boundaries, which will reduce the additional outer iterations incurred in certain spatially decomposed parallel transport sweep algorithms. Results for various parallel configurations are also presented for the C5G7 application and 2D quarter core slice AMA Problem 5.

9:00 AM
A Low Order Nonlinear Transport Acceleration Scheme for the Method of Characteristics

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This study presents a new physics-based multi-grid nonlinear acceleration method, henceforth referred to as the Low-Order Operator (LOO) method, which uses a coarse space-angle multi-group Method of Characteristics (MOC) neutron transport calculation to accelerate the fine space-angle MOC calculation. LOO is designed to capture more angular effects than diffusion-based acceleration methods through a transport-based low-order solver. LOO differs from the two existing transport-based acceleration schemes in that it emphasizes simplified coarse space-angle characteristics and preserves physics in quadrant phase-space. The details of the method, including the restriction step, the low-order iterative solver, and the prolongation step are discussed in this work. The performance of LOO is compared to the Coarse-Mesh Finite Difference method (CMFD) in benchmark problems. LOO shows comparable convergence compared to CMFD, and does not require under-relaxation to converge, making it a robust acceleration scheme.

9:20 AM
p-CMFD Acceleration and Nonoverlapping Local/Global Iterative Transport Methods with 2-D/1-D Fusion Kernel

S.Yuk, N.Z.Cho

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Conventional reactor core analysis, based on isolated single-assembly lattice calculation and diffusion nodal core calculation, has the limitation due to the absence of treatment of accurate inter-assembly transport effect. We consider two approaches in this paper: 1) whole-core fine-group deterministic transport calculations are accelerated by the partial-current based coarse-mesh (in space, angle, and energy) finite-difference (p-CMFD) method, and 2) whole-core domain is decomposed into nonoverlapping local problems and the local problem transport solutions are wrapped around by the p-CMFD methodology in a two-level iterative scheme to provide whole-core transport solution. In both approaches, the 2-D/1-D fusion method is used as solution kernel, which employs two-dimensional method of characteristics (MOC) in radial direction and one-dimensional S_N-like method in axial direction. We report results of the two approaches applied to the three whole-core configurations of the C5G7 OECD/NEA three-dimensional benchmark problem.

9:40 AM
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

University of Michigan, Michigan, USA

In this paper the spatial domain decomposed coarse mesh finite difference (SDD-CMFD) method is applied as an acceleration technique to a parallel implementation of the 3-D method of characteristics (MOC) for a series of problems to assess the effectiveness of the method for practical applications. The SDD-CMFD method assumes the problem domain is divided into independent parallelizable sweep regions globally linked within the framework of a CMFD-like system. Results obtained with the MPACT code are examined for three problems. The first analysis is of multi-dimensional, 1-group, infinite homogeneous media problems that compare the numerically-measured rate of convergence to that predicted by the 1-D Fourier analysis performed in previous work. It is observed that the rate of convergence of the numerical experiments has similar behavior to that predicted by the Fourier analysis for variations of optical thickness in the coarse cell and spatial sub-domain. However, the rate of convergence is measured to be slightly less than that predicted by Fourier analysis. The algorithm is applied to the Takeda 3-D neutron transport benchmark, and compared to a standard source

Track2-1 Deterministic Transport Theory

Session Chair: Nam Zin Cho(KAIST), Masato Tabuchi(NEL)

iteration. In the analysis of this problem, the method is observed to speed up convergence, significantly reducing the number of outer iterations by a factor of nearly 20x and reducing the overall run time by a factor of about 10x. Finally, the method is applied to a realistic PWR assembly, which is observed to converge in 7 outer iterations, a factor of 150x less than source iteration, using the SDD-CMFD acceleration method, and have an estimated speedup of ~34x over conventional source iteration.

Track3-1 Monte Carlo Methods

Session Chair: Jaakko Leppänen(VTT), Takanori Kitada(Osaka Univ.)

8:00 AM
Unstructured Mesh Based Multi-Physics Interface for CFD Code Coupling in the Serpent 2 Monte Carlo Code

J.Leppänen, V.Valtavirta, T.Viitanen(1), M.Aufiero(2)

1)VTT Technical Research Centre of Finland, Espoo, Finland, 2) Politecnico di Milano, Milano, Italy

This paper presents an unstructured mesh based multi-physics interface implemented in the Serpent 2 Monte Carlo code, for the purpose of coupling the neutronics solution to componentscale thermal hydraulics calculations, such as computational fluid dynamics (CFD). The work continues the development of a multi-physics coupling scheme, which relies on the separation of state-point information from the geometry input, and the capability to handle temperature and density distributions by a rejection sampling algorithm. The new interface type is demonstrated by a simplified molten-salt reactor test case, using a thermal hydraulics solution provided by the CFD solver in OpenFOAM.

8:20 AM
Analysing the Statistics of Group Constants Generated by Serpent 2 Monte Carlo Code

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An important topic in Monte Carlo neutron transport calculations is to verify that the statistics of the calculated estimates are correct. Undersampling, non-converged fission source distribution and inter-cycle correlations may result in inaccurate results. In this paper, we study the effect of the number of neutron histories on the distributions of homogenized group constants and assembly discontinuity factors generated using Serpent 2 Monte Carlo code. We apply two normality tests and a so-called “drift-in-mean” test to the batch-wise distributions of selected parameters generated for two assembly types taken from the MIT BEAVRS benchmark. The results imply that in the tested cases the batch-wise estimates of the studied group constants can be regarded as normally distributed. We also show that undersampling is an issue with the calculated assembly discontinuity factors when the number of neutron histories is small.

8:40 PM
Theoretical Prediction on Underestimation of Statistical Uncertainty for Fission Rate Tally in Monte Carlo Calculation

T.Endo, A.Yamamoto, K.Sakata

Nagoya University, Nagoya, Japan

A theoretical model to predict underestimation of standard deviation for the mean of fission rate in each tally region is proposed on the basis of the behavior of higher order modes in fission source distribution and the autoregressive (AR) model. The predicted underestimation of standard deviation for the mean of fission rate is compared with that obtained by actual Monte Carlo calculations and they agree well each other. Dependency of underestimation of standard deviation on number of tally regions and spatial position is clarified by the proposed theoretical model. The present theoretical model can be used to quantitatively predict underestimation of standard deviation of local fission rate tally.

9:00 AM
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)

1)Massachusetts Institute of Technology, Massachusetts, USA, 2)Knolls Atomic Power Laboratory - Bechtel Marine Propulsion Corporation, New York, USA

A study of tally correlation was performed for a 2-D radial slice of the BEAVRS light water reactor. This reactor has a dominance ratio of approximately 0.995. Convergence rates of fission source tallies of an assembly and pin mesh were analyzed. Results of this study indicate that these tallies do not converge at the ideal rate of the inverse of the square root of number of tally realizations. Ideal convergence rates are achieved only when tally realizations are uncorrelated. Correlation exists because source sites of one generation of neutrons are from fission sites produced from a previous generation. The degree of correlation was investigated by computing autocorrelation coefficients. High autocorrelation coefficients of about 0.7 for lag one were observed for assembly mesh tallies. Pin mesh tallies were much less correlated. In addition, autocorrelation coefficients were not affected by increasing number of neutrons in a tally batch. Three methods are discussed to reduce correlation: increasing number of initial discard batches, multiple generations per tally batch and CMFD feedback. The only significant decrease in correlation was observed when applying CMFD feedback. Lag one autocorrelation coefficients decreased from about 0.7 to 0.4. No significant difference in tally convergence rates were observed when applying CMFD feedback.

9:20 AM
Higher-Mode Applications of Fission Matrix Capability for MCNP

S.E.Carney(1), F.B.Brown, B.C.Kiedrowski(2), W.R.Martin(1)

1)University of Michigan, Ann Arbor, USA, 2)Los Alamos National Laboratory, NM, USA

The fission matrix method, implemented into the MCNP6 Monte Carlo code, can be used to provide estimates of the fundamental mode fission distribution, the dominance ratio, the eigenvalue spectrum, and higher mode forward and adjoint eigenfunctions of the fission neutron source distribution. It can also be used to accelerate the convergence of the power method iterations. The higher-mode fission sources can be used in MCNP6 to determine higher mode fluxes and tallies, both forward and adjoint. These higher mode fluxes are necessary for important applications such as second-order perturbation theory and quasi-static calculations. The flux calculations are theoretically justified here, along with a justification of eigenmode expansions during source convergence. Forward fluxes and the relative uncertainties for a 2D PWR are shown, both of which qualitatively agree with expectations. Adjoint-weighted flux for a 3-group slab problem is calculated and found to agree with discrete ordinates results. Lastly, eigenmode expansions are performed during source convergence from two intial distributions for the 2D PWR problem; observed decay rates of coefficients agree closely with expectation.

9:40 AM
A Symmetric View Hiding the Ugly Truth

D.Mennerdahl

E Mennerdahl Systems, Täby, Sweden

The general recommendation to use symmetry in Monte Carlo code keff calculations is questioned. A solution using a symmetric model provides less information and may hide serious errors that would be apparent when using a full model calculation. Convergence indicators such as dominance ratio and Shannon entropy have appropriate applications but they do not always give correct and reliable information about symmetric model calculations. Further, the methods do not see the symmetry in a full model, apparent to the user of the code. The user often accounts for symmetry in a full model by having the same material or geometry unit in the symmetric regions. If average tallies are requested, they can be obtained from the full model results. The integral values are the same but the differences between local reactions in symmetric regions are informative.
The symmetry recommendation, together with false convergence indicators, encourages the user to believe that the symmetric model calculations are more reliable than full model calculations, when the same Monte Carlo statistics are applied. Examples are given in the paper, demonstrating the points made. The most basic example of symmetry is a calculation model for an infinite material. Another example is a cylinder with azimuthal symmetry. In both cases it is clear that more symmetry is not better. This is demonstrated for cubes with different dimensions and mirror reflection all around. Application of symmetry is necessary for a simulation of such geometry due to computer memory storage limitations (less than an infinite value). Better convergence is not a relevant point.

SS8 Reactor Physics of Non-Traditional LWR Fuel Design

Session Chair: Bojan Petrovic(Georgia Tech.), Yoichiro Shimazu(Univ. of Fukui)

8:00 AM
I^{2}S-LWR Equilibrium Cycle Core Analysis

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1)Westinghouse, Pennsylvania, USA, 2)Georgia Tech., Atlanta, USA

This paper presents the preliminary core design developed for the Integral Inherently Safe LWR (I^{2}S-LWR). The I^{2}S-LWR is a conceptual design of a ~1,000 MWe (2,850 MWt) integral PWR with inherent safety features. The baseline core configuration contains 121 fuel assemblies with a 19×19 square lattice with 144-in active fuel height. The reference fuel option is U_3Si_2 in advanced stainless steel cladding, which is envisioned to enhance accident tolerance compared to the current UO_2/Zr system. The viability of UO_2 fuel is also under investigation as an alternative option for accelerated deployment. A 12-month cycle 3-batch fuel management scheme has been found to be the economic optimum for the I^{2}S-LWR with the current ^{235}U licensed enrichment. The U_3Si_2 and UO_2 core designs implementing this option are presented in this paper together with a comparison of their reactor physics performance at the equilibrium cycle.

8:20 AM
Uranium Nitride Composite Fuels in a Pressurized Water Reactor: Exploration of Multi-Batch Cycle Length and UB4 Admixture for Reactivity Control

N.R.Brown, M.Todosow, K.J.Mcclellan

Brookhaven National Laboratory, New York, USA

In the aftermath of the Fukushima accident, uranium nitride (UN)-based fuel composites have been presented as potential light water reactor fuels with enhanced accident tolerance. Enhanced accident tolerance implies improved accident performance versus the present UO_2-Zr fuel system as well as similar or improved nominal performance under operational conditions. UN-based composite fuels consist of a primary UN phase that is “shielded” from water via a secondary phase. This allows the potential benefits of the UN fuel, including enhanced thermal conductivity, to be realized while significantly reducing hydrolysis of the nitride phase. We compared performance of several nitride composites in a PWR. Performance parameters included quantities like cycle length for three-, four-, and five-batch fuel management schemes as well as reactivity coefficients. Although a nitride composite fuel would most likely be paired with an advanced cladding, our analyses assume Zr-based cladding. However, many of the qualitative comparisons from our analyses have relevance for proposed advanced cladding materials, especially those that are relatively neutron transparent (e.g. silicon carbide). We studied UN/U_3Si_5, UN/U_3Si_2, UN/UB_4, UN/ZrO_2, and UN/U_3Si_2/UB4. In all cases UN is the primary phase, with small fractions of U3Si5, U_3Si_2, UB4, or ZrO_2 as a secondary phase. For the UN/UB_4 cases, the boron content is 100% enriched in ^{11}B, but for UN/U_3Si_2/UB_4 cases natural boron is used for reactivity control. This paper presents a summary of some of the key results and an overview of the research philosophy. In general, the full-core equilibrium cycle reactivity coefficients for the nitride and nitride composite fuels were within the design limits for the reference PWR. Increasing the number of batches appears to be a viable alternative to increasing the cycle length relative to the nominal UO_2-Zr fuel system. We found that the reactivity suppression of integral fuel burnable absorber rods could be effectively reproduced using small fractions of UB_4 admixture homogeneously distributed in the fuel.

8:40 AM
Impact of Coating on Nitride Fuel Performance in PWRs

F.Heidet, A.M.Yacout

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In order for uranium nitride fuel to be used in light water reactors and benefit from its high thermal conductivity and high density, it is necessary to develop a mitigation strategy to prevent uranium nitride oxidation with water/steam at high temperatures. One possible strategy is the use of a protective layer between the fuel and cladding (coating fuel pellet), preventing the water/steam to contact with the fuel. Another strategy consists in using thin-layer coating applied to uranium nitride fuel particles before the fuel sintering step. It is important for the materials used not to have a detrimental effect on the neutronics performance of the fuel. The impact on the neutronics performance of using Al_2O_3 coatings has been assessed in this work. The main effect is the reduction of the heavy metal mass, as compared to pure uranium nitride fuel, which results in slightly softer spectrum. However, when compared with uranium oxide fuel which has a similar heavy metal mass, the achievable cycle length and discharge burnup are found to be nearly identical since para- sitic absorption in Al_2O_3 is relatively small. Further development and application of Al_2O_3 coatings (and possibly other materials) can ultimately lead to the deployment of uranium nitride fuel in light water reactors.

9:00 AM
Optimization of Fully Ceramic Micro-Encapsulated Fuel Assembly for PWR

R.A.Shapiro, M.J.Vincenzi(1), M.Fratoni(2)

1)The Pennsylvania State University, Pennsylvania, USA, 2)University of California, Berkeley, USA

Fully ceramic micro-encapsulated (FCM) fuels consist of TRISO particles embedded in a ceramic (SiC) matrix to form fuel pellets and rods, and are expected to largely improve fuel performance in light water reactors during beyond design basis accident conditions. The viability of FCM fuel is hindered by a considerable lower heavy metal load compared to conventional fuel. This deficiency is compensated using high-density fuel (UN), large kernels, large fuel rods, and higher enrichment. This study evaluated the assembly design requirements for FCM fuel to match the neutronics performance of conventional fuel, in particular to reach similar cycle length while major safety criteria—reactivity coefficients, and control system worth, are met. It was found that with a traditional 1717 assembly FCM fuel must be enriched close to 20% in order to achieve similar cycle length as conventional fuel, but an optimization of the assembly design (99) could reduce such enrichment to about 16%.

9:20 AM
Fully Ceramic Microencapsulated Fuels: Characteristics and Potential LWR Applications

J.J.Powers, A.Worrall, K.A.Terrani, J.C.Gehin, L.L.Snead

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This paper summarizes the characteristics of the fully ceramic microencapsulated (FCM) fuel concept and two potential light water reactor (LWR) applications of FCM fuels: for actinide management and as an accident tolerant fuel (ATF). Recent progress in FCM fuel development includes production of uranium mononitride kernels, fabrication of FCM pellets and pins, and irradiation testing of matrix samples and FCM pellets. Potential applications of FCM fuel in LWRs appear promising based upon studies performed by several organizations; however, further efforts are needed to investigate various design aspects in further detail and explore promising new areas of research such as new fuel pin and assembly designs or alternate materials of interest. Current challenges in FCM fuel development and LWR applications for FCM fuels include low heavy metal fuel loading densities and increased uncertainties in analysis due to several different factors. Overall, LWR FCM concepts appear feasible for both actinide management and as an ATF.

9:40 AM
Neutronic Challenges of Advanced Boiling Water Reactor Designs

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The advancement of Boiling Water Reactor technology has been under investigation at the Center for Advance Nuclear Energy Systems at MIT. The advanced concepts under study provide economic incentives through enabling further power uprates (i.e. increasing vessel power density) or better fuel cycle uranium utilization. The challenges in modeling of three advanced concepts with focus on neutronics are presented. First, the Helical Cruciform Fuel rod has been used in some Russian reactors, and studied at MIT for uprating the power in LWRs through increased heat transfer area per unit core volume. The HCF design requires high fidelity 3D tools to assess its reactor physics behavior as well as thermal and fuel performance. Second, an advanced core design, the BWR-HD, was found to promise 65% higher power density over existing BWRs, while using current licensing tools and existing technology. Its larger assembly size requires stronger coupling between neutronics and thermal hydraulics compared to the current practice. Third is the reduced moderation BWRs, which

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had been proposed in Japan to enable breeding and burning of fuel as an alternative to sodium fast reactors. Such technology suffers from stronger sensitivity of its neutronics to the void fraction than the traditional BWRs, thus requiring exact modeling of the core conditions such as bypass voiding, to correctly characterize its performance.

Track1-4 Reactor Analysis Methods

Session Chair: Jess Gehin(ORNL), Scott Palmtag(Core Physics Inc.)

10:20 AM
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.Francheschini(2)

1)Oak Ridge National Laboratory, Oak Ridge, USA, 2)Westinghouse Electric Corporation, Pennsylvania, USA

The Consortium for Advanced Simulation of Light Water Reactors (CASL) is developing a collection of methods and software products known as VERA, the Virtual Environment for Reactor Applications, including a core simulation capability called VERA-CS. A key milestone for this endeavor is to validate VERA against measurements from operating nuclear power reactors. The first step in validation against plant data is to determine the ability of VERA to accurately simulate the initial startup physics tests for Watts Bar Nuclear Power Station, Unit 1 (WBN1) cycle 1. VERA-CS calculations were performed with the Insilico code developed at Oak Ridge National Laboratory using cross section processing from the SCALE system and the transport capabilities within the Denovo transport code using the SPN method. The calculations were performed with ENDF/B-VII.0 cross sections in 252 groups (collapsed to 23 groups for the 3D transport solution). The key results of the comparison of calculations with measurements include initial criticality, control rod worth critical configurations, control rod worth, differential boron worth, and isothermal temperature reactivity coefficient (ITC). The VERA results for these parameters show good agreement with measurements, with the exception of the ITC, which requires additional investigation. Results are also compared to those obtained with Monte Carlo methods and a current industry core simulator.

10:40 AM
AP1000® PWR Reactor Physics Analysis with VERA-CS and KENO - Part I: Zero Power Physics Tests

F.Franceschini(1), A.Godfrey, J.C.Gehin(2)

1)Westinghouse, Pennsylvania, USA, 2)ORNL, Tennessee, USA

Westinghouse has applied the Core Simulator of the Virtual Environment for Reactor Applications, VERA-CS, under development by the Consortium for Advanced Simulation of LWRs (CASL) to the core physics analysis of the AP1000® PWR. The AP1000 PWR features an advanced first core with radial and axial heterogeneities, including enrichment zoning, multiple burnable absorbers, and a combination of light and heavy control banks to enable the MSHIM™ advanced operational strategy. These advanced features make application of VERA-CS to the AP1000 PWR first core especially relevant to qualify VERA performance. A companion paper at this conference describes the power distribution analysis of the AP1000 PWR with VERA-CS and the KENO Monte-Carlo code. This paper describes the results obtained for the startup physics tests simulations of the AP1000 PWR first core (critical boron, rod worth and reactivity coefficients), supporting the excellent numerical agreement reported in the companion paper for the power distribution.

11:00 AM
AP1000® PWR Reactor Physics Analysis with VERA-CS and KENO - Part II: Power Distribution

F.Franceschini(1), J.C.Gehin, A.T.Godfrey(2)

1)Westinghouse, Pennsylvania, USA, 2)ORNL, Tennessee, USA

Westinghouse has applied the Core Simulator of the Virtual Environment for Reactor Applications, VERA-CS, under development by the Consortium for Advanced Simulation of LWRs (CASL) to the core physics analysis of the AP1000® PWR. The AP1000 PWR features an advanced first core with radial and axial heterogeneities, including enrichment zoning, multiple burnable absorbers, and a combination of light and heavy control banks to enable the MSHIM™ advanced operational strategy. These advanced features make application of VERA-CS to the AP1000 PWR first core especially relevant to qualify VERA performance. A companion paper at this conference describes the results obtained with VERA-CS and the KENO Monte-Carlo code for startup physics tests simulations of the AP1000 PWR first core (critical boron, rod worth and reactivity coefficients). This paper describes the results of detailed power distribution comparisons between VERA-CS and KENO, and confirms the excellent numerical agreement reported in the companion paper for the startup physics tests simulations.

11:20 AM
Solution of the BEAVRS Benchmark Using the nTRACER Direct Whole Core Transport Code

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

Seoul National University, Seoul, Korea

The BEAVRS benchmark is solved by the nTRACER direct whole core transport code to assess its accuracy and to examine the solution dependence on modeling parameters. A sophisticated nTRACER core model representing the BEAVRS core is prepared after a series of study to ensure solution accuracy. The resulting solutions for several hot-zero-power (HZP) states are compared first with the corresponding Monte Carlo solutions and then with the measured data which include the control rod worths as well as the critical boron concentrations (CBC). The core depletion calculation is performed for the initial and second cycles with a set of approximated power histories and the calculated CBCs are compared with the measured data. The comparison results show that the criticality, control rod bank worths at HZP and the boron let-down curves of two cycles agree well with the measurements within ~200 pcm and ~20 ppm, respectively.

11:40 AM
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

Magyar Tudományos Akadémia Energiatudományi Kutatóközpont, Budapest, Hungary

Originally the design operational lifetime of the VVER-440 plants was 30 years, which was based on a rather conservative estimation. To extend its service life is an economically desirable goal, keeping in mind safety precautions. One of the limiting factors for the plant lifetime is the embrittlement of the reactor pressure vessel (RPV) due to the interaction of the solutes in the vessel with the vacancies and other products created by the incident neutrons. In order to qualify this effect the neutron fluence at the RPV must be well known. For the proper analysis of this quantity the detailed operational history (from initial irradiation to near future) needs to be known and a large number of shielding calculations have to be performed. In Hungary a coupled calculation tool was employed. The outgoing current and the scalar flux at the periphery of the core are calculated by the KARATE code. The neutron source for calculation of RPV fluence by the MCNP code is based on these quantities. Considerable research has been performed on validating this method. At Paks NPP, Hungary, a major project was launched to investigate the possibility of lifetime extension up to 60 years while new fuel containing gadolinium pins and corresponding reloading schemes are being introduced. The paper details the evaluation of the fluence using the above outlined procedure, the data used, the fluence obtained, and the uncertainty of the result.

Track5-2 Nuclear Criticality Safety

Session Chair: Bo Feng(ANL), Tomohiro Endo(Nagoya Univ.)

10:20 AM
Uncertainty Evaluation of Reactivity in Single and Multi-Region TSUNAMI Modeling Analysis for Dry Cask Storage

Q.T.Newell, C.E.Sanders

University of Nevada, Las Vegas, USA

Uncertainties in the reaction probabilities have the potential to affect every major facet of uranium use in operational lifecycles. One of these operations includes disposal of used nuclear fuel. Evaluation of the uncertainties in a given data library can provide assurances in the area of dry cask storage. This study investigates single-region versus multi-region modeling of a large cask system with the TSUNAMI-3D (Tools for Sensitivity and Uncertainty Analysis Methodology Implemented in Three Dimensions) control module (for sensitivity and uncertainty analysis) in the SCALE6 computer code. The difference in the sensitivity coefficients for three nuclides (U-235, Pu-239, and Eu-153) are considered by modeling single and multi-region fuel assemblies within a single region storage cask. Overall, the study shows that there are no distinct advantages of using a multi-region assembly compared to a single region assembly to calculate the sensitivity coefficients of a storage cask. However, developing multiple regions in the problem provides users a different way of modeling, than simply running more particles.

10:40 AM
Transient Analysis in Super Critical Condition for Several Fuel-Solution Tanks System with Different Layout

H.Kikuchi(1), T.Obara(2)

1)Department of Nuclear Engineering, Tokyo Institute of Technology, Tokyo, Japan, 2)Research Laboratory for Nuclear Reactors, Tokyo Institute of Technology, Tokyo, Japan

To confirm that changes of power and the total integral power in critical accidents are affected by different layouts of fuel-solution tanks, transient analyses were performed for a single-tank system and different layouts of a three-tank system using an integral kinetic analysis method for a weakly coupled system. The changes of power and the integral power were analyzed with different layouts of fuel-solution tanks. The code used for the analysis took into account the neutronic coupling between the tanks. The results show that different layouts of fuel-solution tanks cause different transients of power in the supercritical condition by different neutronic coupling and different increases of integral power compared to the integral power in a single-tank system. The integral power can be underestimated if neutronic coupling between the tanks is ignored.

11:00 AM
Comparison of Gamma Dose Rate Calculations for PWR Spent Fuel Assemblies

B.Feng, R.N.Hill(1), R.Girieud, R.Eschbach(2)

1)Argonne National Laboratory, Argonne, USA, 2)CEA Cadarache, Saint-Paul-lez-Durance, France

Gamma dose rate calculations from the U.S. Department of Energy and the French Commissariat à l'Énergie Atomique et aux Énergies Alternatives were compared for PWR spent fuel assembly models with UO_2 fuel at 33 MWd/kg burnup and MOX fuel at 60 MWd/kg. Both sides used their reference physics codes in a three-step calculation procedure involving depleting fresh assemblies to obtain the discharge compositions, obtaining the isotopic gamma source after 30 years of simulated radioactive decay, and applying the source to heterogeneous 3-D transport models to tally the flux at one meter away from the axial midpoint through air. The fluxes were ultimately converted to dose rates with different energy-dependent conversion factors. This study was able to pinpoint the intermediate calculations that exhibited the largest sources of discrepancy between the two approaches. Most notably, the U.S. gamma source calculation accounts for Bremsstrahlung and adjusts its multi-group photon release rates to conserve energy. Despite these differences, very similar dose rates were calculated by both approaches. For the UO_2 case, which was intended to benchmark a frequently-cited reference study, both the DOE and CEA calculated 30-year dose rates between 4.6 and 5.8 Sv/h with various conversion factors, which are roughly three times lower than the reference study's results. For the MOX case, the calculated dose rates ranged from 8.5 to 11 Sv/h. Given the same conversion factor, the largest difference between DOE and CEA values was about 1.5 Sv/h.

11:20 AM
The KBS-3 Spent Fuel Canister Criticality Calculation during 100,000 Years

S.Wu, B.Yang(1), H.Wu, Q.Wei, Y.Liu(2)

1)East China Institute of Technology, Ministry of Education, Nanchang, China, 2)East China Institute of Technology, Nanchang, China

KBS-3 system is a mature disposal system, the spent fuel is directly stored in canister without reprocessing. This paper studies the KBS-3 spent fuel canister criticality security issue under groundwater penetration accident from 100 to 100,000 years. The composition of the material PWR spent fuel assemblies are calculated by using MCBurn software, which is a coupling of MCNP and ORIGEN2.1. The calculation results are used in MCNP5 to calculate the effective multiplication factor k_{eff} of the assemblies in different volumes of groundwater penetration. The k_{eff} maximum value in initial period is 1.0968, the error is about 0.735%. And then, four PWR spent fuel assemblies are set in the canister, The paper study finds that the maximum k_{eff} of canister is 0.79609 in groundwater penetration accident. The calculation results show that the KBS-3 spent fuel canister is safe in criticality security issue.

11:40 AM
Favorable Features in Kinetics of Fast Reactors with Physically Thick ^{208}Pb-Reflector

G.G.Kulikov, A.N, Shmelev, V.A.Apse, M.Y.Ternovykh

National Research Nuclear University MEPhI, Moscow, Russia

Light materials with small atomic mass (light or heavy water, graphite and so on) are usually used as a neutron reflector and moderator. The present paper proposes to use a new, very heavy neutron moderator and reflector, namely "radiogenic lead" with dominant content of isotope ^{208}Pb. Radiogenic lead is a stable natural lead contained in the ore deposits with thorium component, usually, they are poly-metallic ores of rare earth elements. The lead is actually a final product of radioactive decay chains starting from uranium and thorium (see appendix bellow). This isotope is characterized by extremely low micro cross-section of radiative neutron capture (~0.23 mb for thermal neutrons, smaller than graphite and deuterium cross-sections) and large albedo for thermal neutrons.

The following mathematical models of neutron kinetics are applied to analyze the quick-proceeding processes initiated by reactivity jump in the reactor cores surrounded by physically thick, weakly absorbing neutron reflector: one-point, two-point, multi-point and continuous models. It was revealed that the results of one-point and two-point models differed substantially from those obtained by multi-point and continuous kinetic models which were developed by authors and described in the paper. Also, the paper demonstrates that only multi-point and continuous models are able to follow up adequately neutron kinetics in the fast reactor core surrounded by a physically thick and weakly absorbing neutron reflector. The summand of the "inverse-hour" equation defining kinetic effects of the neutron reflector was analyzed in detail.

It is evaluated that the use of radiogenic lead makes it possible to slow down the chain fission reaction on prompt neutrons in the fast reactor. This can improve the fast reactor safety and reduce some requirements to the technologies used to fabricate fuel for the fast reactor.

Track2-2 Deterministic Transport Theory

Session Chair: Farzad Rahnema(Georgia Tech. Univ.), Wu Hongchun(Xi'an Jiaotong Univ.)

10:20 AM

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

D.Anistratov, Y.Azmy

NC State University, Raleigh, USA

We study convergence of the integral transport matrix method (ITMM) based on block-Jacobi strategy for solving diamond-differenced S_N equations for two-dimensional transport problems. This is a spatial domain decomposition method applied in massively parallel computations. We consider the case of one cell per subdomain. A Fourier analysis of the equations for S_2 is performed. The analysis shows that the iteration method in this particular case loses its effectiveness. Numerical results of finite-medium problems are presented to demonstrate the behavior of the ITMM for the DD scheme that was theoretically predicted.

10:40 AM

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

G.Rimpault, J.-F.Vidal(1), W.F.G.van Rooijen(2)

1)CEA, Saint Paul-lez-Durance, France, 2)University of Fukui, Fukui, Japan

New generations of simulation tools responding to the challenges brought by the advanced features of both 3rd+ generation Pressurized Water Reactor (PWR) cores and 4th generation sodium fast neutron reactor (SFR) cores are taking shape. The developments of new simulations tools are also motivated by strict requirements of nuclear safety authorities. The new tools have the objective of setting new reference standards for neutronic prediction and will take advantage of innovative algorithms which have been implemented in existing CEA codes, such as ERANOS (fast reactors) and APOLLO2 (PWR); the new codes should at the same time remove remaining calculation errors. Although innovative algorithms have been filling the gaps which did exist 40 years ago between tools specifically dedicated to either thermal neutron cores or fast neutron ones, there remains a series of algorithms which deserve particular attention: the treatment of leakage in cell calculations.
This paper describes methods for treating neutron leakage in self-shielding calculations with the sub-group method, and in the cell balance calculation. Applications of the MOC method of solution to treat neutron leakage are described. The application of the MOC can eliminate approximations at the cell interfaces while maintaining precise neutron leakage treatment. The new APOLLO3® code, presently under development at CEA, is candidate for hosting such algorithms.

11:00 AM

Revisit Boundary Conditions for the Self-Adjoint Angular Flux Formulation

Y.Wang, F.N.Gleicher

Idaho National Laboratory, Idaho, USA

This paper revisits the boundary conditions applied to the self-adjoint angular flux formulation in a more rigorous variation derivation. It includes a new way of weakly imposing the reflecting boundary condition for preserving the symmetry of the final system of equations. Completely equivalent parity variational forms are given. Numerical results with the combination of these boundary conditions are presented.

11:20 AM

Accuracy Preserving Surrogate for Neutron Transport Calculations

C.Wang, H.S.Abdel-Khalik

North Carolina State University, Raleigh, USA

Recent advances in reduced order modeling and exact-to-precision generalized perturbation theory are combined in a novel algorithm that constructs a surrogate model for the Boltzmann equation, commonly used in assembly calculations to functionalize the few-group cross-sections in terms of the various assembly types, depletion characteristics, and thermal-hydraulics conditions. First, the algorithm employs reduced order modeling to determine the dominant input parameters, aggregated in the so-called active subspace, using a random sample of first-order derivatives calculated using an adjoint model. Next, exact-to-precision generalized perturbation theory identifies an active subspace for the state solution (i.e., angular flux) and constructs a surrogate model that is parameterized over the active subspace of the input parameters. This approach is shown to significantly reduce computational time needed for the analysis of a large number of model variations, while meeting the user-defined accuracy requirements. Numerical experiments are employed to demonstrate the mechanics and application of the proposed approach to assembly calculations commonly used in reactor physics analysis.

Track13 Radiation Applications and Nuclear Safeguards

Session Chair: Alexis C. Kaplan(Michigan Univ.), Yasunori Kitamura(JAEA)

10:20 AM
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

A low sensitivity long neutron counter was designed as a standard directional flow detector to monitor neutron fluence reference values of an accelerator-based 14 MeV D-T neutron source with yield about 10^{13}n/s. The energy response over 6 MeV was improved using a tungsten radiator, which acts as an energy converter via the (n,xn) reaction. Different parameters were optimized to flatten the neutron energy response over a wide energy range. A simulation of the designed long neutron counter using the Monte Carlo codes MCNP was undergone. The response function is relatively flat in the energy range of 1 keV-20 MeV. The results show the maximal relative variation is about 7.8%.

10:40 AM
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)
1)Los Alamos National Laboratory, New Mexico, USA, 2)University of Michigan, Michigan, USA

List-mode data collection for high-activity sources such as spent nuclear fuel has been made possible recently by advances in data collection and data-storage capabilities. While the shift-register technique has been utilized primarily in the past, list-mode data collection and subsequent Rossi-alpha distribution (RAD) production offers the same analysis approach, as well as increased flexibility post-measurement. Simulating the RADs allows us to examine different time domains and the effect of fissile and fertile materials on the shape of the measured RADs. New analysis methods possible only with RADs allow us to determine assembly multiplication with a mean difference of 0.7% when tested against a benchmark set of 44 different simulated spent fuel assemblies. This method also enables us to determine total Pu mass with fewer calibration constants and a-priori knowledge than previous methods. Here we consider different techniques for simulating and analyzing Rossi-alpha distributions for applications in spent nuclear fuel characterization.

11:00 AM
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

A study on the derivation of antineutrino spectra from a normal mixed oxide fuel and fuels with high proliferation resistance has been conducted for the future study of the feasibility of antineutrino detector to be used for nuclear non-proliferation purpose. Due to the large isotopic proportion of ^{238}Pu in fuels with high proliferation resistance, the antineutrino detection method could be used as a verification tool of the use such fuels in fast reactors. Therefore, we have performed simulations of the time evolution of a fast breeder reactor loading a few types of nuclear fuels and obtained the inventory of fission products. A subsequent summation calculation based on nuclear data derives the antineutrino energy spectra from a normal mixed oxide fuel and a fuel with high proliferation resistance. The results show a 3% difference of the number of emitted antineutrinos between the normal mixed oxide fuel and the fuels with high proliferation resistance.

11:20 AM
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

Monte Carlo particle transport codes used to accurately model organic liquid scintillator detector response are traditionally run in fully analog mode. Analog simulations of cross-correlation measurements with these codes are extremely time-consuming because the probability of correlated detection is extremely small, approximately equal to the product of the probabilities of a single detection in each detector. The new “implicit correlation” method described here increases the number of correlation event tallies thereby, reducing variance and required computation times. The cost of the implicit correlation method is comparable to the cost of simulating single event detection in the lowest efficiency detector. This method is especially useful in the nuclear nonproliferation and safeguards fields for simulating correlation measurements of shielded special nuclear material.
A plutonium metal sample was measured by a fast-neutron multiplicity counter for verification of MCNPX-PoliMi calculations and the new method. The fast-neutron multiplicity counter was used with pulse shape discrimination techniques to produce a neutron-neutron cross-correlation distribution. The new method was implemented in MCNPX-PoliMi with a Pu-240 spontaneous fission source in the measurement geometry. The method demonstrated good agreement with analog simulation and measurement results and a speed-up of a factor of 500 over analog calculations.

11:40 AM
A Unique Tungsten-Based Tagging Approach for Maintaining of Continuity of Knowledge of Nuclear Waste Copper Canisters

D.Chernikova, K.Axell, A.Nordlund
Chalmers University of Technology, Göteborg, Sweden

A new approach to the unique tagging of nuclear waste copper canisters is suggested. In this new method a combination of two different techniques, radiation and ultrasonic measurements, is used in order to get the same unique identifier of the cask. The necessary component of the method is a tungsten/lead insert marked with a binary or bar code and placed inside the container. The paper discusses results of the radiation measurements performed in lab as the first proof of the concept, as well as results of Monte-Carlo evaluation of the feasibility of proposed approach. The method makes it possible to maintain continuity of knowledge of nuclear waste for a time scale up to a few hundred years without comprising the environmental safety of casks.

Track9-1 Transient and Safety Analysis

Session Chair: Kostadin Ivanov(Pennsylvania Univ.),Yuichiro Ban(Toshiba)

10:20 AM
Development of the Neutron Kinetics Code for Thermal Molten Salt Reactor

Y.Zheng, K.Zhuang, L.Cao, H.Wu

Xi'an Jiaotong University, ShannXi, China

In the molten salt reactor, the delayed neutron precursors continuously change their position both in the core and in the external loop due to the flow of fuel, which lead to the loss of reactivity and decrease of effective delayed neutron fraction. Therefore, the neutron kinetics of molten salt reactor is significantly different from that of conventional reactors using solid fuels. In this study, a 3D neutron kinetics model of molten salt reactor considering the flow effects of delayed neutron precursors was established. The analytic basis functions expansion nodal method for arbitrary triangular-z node was performed to solve the three dimensional neutron diffusion equations and the method of characteristics was used to find the solution of delayed neutron precursor equations within the whole primary circuit. To verify the code, calculations were preformed on a homogeneous reactor model. The results were in good agreement with the reference results. Besides, the influence of fuel circulation on the kinetic characteristic of reactor was investigated. The results showed some special phenomenon in the molten salt reactor.

10:40 AM
Study of Neutron Propagation in Multigroup Transport by Space Asymptotic Methods

J.C.L.Fernandes(1), S.Dulla, P.Ravetto(2), M.T.Vilhena(3)

1)Politecnico di Torino / Universidade Federal do Rio Grande do Sul, Torino, Italy, 2)Politecnico di Torino, Torino, Italy, 3)Universidade Federal do Rio Grande do Sul, Porto Alegre, Brasil

In this work the space asymptotic theory is used for the study of the propagation of neutron pulses in multigroup transport. Pulsed experiments are of interest in the study and characterization of source-driven subcritical systems. The model allows for an exact analytic treatment, although it can be applied only to highly idealized configurations. Therefore, the objective of the investigation is to obtain a good physical insight into the phenomena and to derive highly accurate results to be used as benchmarks. The solution is obtained by a combined application of the Fourier and Laplace transforms. The inversion of the Laplace transform is attained by the use of the residue theorem, which, through the study of the singularities of the transformed solution, allows interesting physical considerations. The solution is exact until the particles, moving with a finite velocity, reach the external boundary of the system. Numerical results for some typical propagation problems are presented for one-dimensional plane systems.

11:00 AM
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

Massachusetts Institute of Technology, MA, USA

A sensitivity study of the solution accuracy for the transport version of the 2D LRA benchmark was conducted for the Adiabatic, Theta, and Multigrid Amplitude Function (MAF) methods. Transient analysis has been extensively studied with neutron diffusion theory, but only recently has the focus changed to investigating reactor transients using fine grained neutron transport theory calculations due to the computational cost of transport methods. The Method of Characteristics (MOC) was chosen as the transport method to investigate due to the computational efficiency and broad use of the method in academia and industry. The Adiabatic, Theta, and MAF methods have been used for diffusion theory transient calculations and appear well suited for implementation in transport theory. Understanding how these methods scale in transport theory will be essential in developing codes to model 3D problems of industrial interest on current computer architectures. This study seeks to validate the transient analysis approach employed in the our implementation of the Adiabatic, Theta, and MAF methods as being computationally efficient and scalable and thus deserving further investigation in transport theory transient analysis.

11:20 AM
Computations of Heterogeneous Dilution Transients Using CFX and HEMERA V1

L.Maas, B.Normand

Institut de Radioprotection et de Sûreté Nucléaire (IRSN), Fontenay-aux-Roses, France

Heterogeneous boron dilution can occur in a PWR in case of a leakage of a primary pump exchanger. Unborated cold water can accumulate in the U-leg during shutdown states and form a plug. When the primary pump restarts, this plug is transported toward the core and may lead to the core re-criticality. To evaluate the consequences of this transient, a sequential calculation involving the CFD code CFX (for the vessel) and the fully coupled 3D computational chain HEMERA V1 (for the core) is used.
This paper is divided into three parts. The first one describes the codes used for the transport of the unborated water (CFX) and for the core response (HEMERA). The second part focuses on the modeling of the heterogeneous boron dilution transient and the third one presents the results. In order to demonstrate the capability of this calculating scheme, two responses of the core have been calculated. The first one keeps the core sub-critical during whole dilution transient and the second one assumes a prompt critical jump in order to demonstrate the capability of the calculation scheme.

11:40 AM
Prompt Behavior of Generalized-Eigenvalue Point Kinetics Models

B.C.Kiedrowski

Los Alamos National Laboratory, NM, USA

The point kinetics model is used as a first approximation for modeling transients in nuclear systems. Point kinetics is accurate enough in many situations, but its performance can degrade far from criticality. The classic approach uses a k or multiplication eigenvalue as the basis for developing the underlying model. This work generalizes the point kinetics equations for any multiplicative eigenvalue. The collision and leakage eigenvalues are studied, and preliminary results show that in some cases the collision eigenvalue provides a more accurate representation of the prompt period.

Track1-5 Reactor Analysis Methods

Session Chair: Myung Hyun Kim(Kyung Hee University), Masahiro Tatsumi(NEL)

13:30 PM

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)

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

Nuclear reactor design analysis often requires a simplified burn-up chain model to reduce computation time. It is difficult to construct the simplified burn-up chain because it requires engineers to have highly skilled techniques and in-depth knowledge of burn-up processes. This paper develops an algorithm for automatically constructing a reduced-order burn-up chain model from the detailed model using the singular value decomposition (SVD). In our approach, we prepare a detailed burn-up chain matrix A, and an extraction matrix C, which extracts important nuclides for specific purposes such as the evaluation of neutron multiplication factor. By applying SVD to C, we can obtain the first candidate nuclides of the reduced-order burn-up chain model. Then, by applying SVD to 1st information transfer matrix $F_{12}^{(1)}$, which defines the relationship between the first candidate nuclides and remaining nuclides, we can obtain additional candidate nuclides for the reduced-order burn-up chain model from the remaining nuclides. We repeat this process until the norm of the information transfer matrix is sufficiently close to zero. Finally, all candidate nuclides chosen through these simplification processes are adopted as a reduced-order burn-up chain model. As the test case, we reduce a detailed burn-up chain model consisting of 1421 nuclides to a model of 219 nuclides. We can use the resulting reduced-order model to calculate the burn-up with a high degree of accuracy.

13:50 PM

Generation of Simplified Burnup Chain Using Contribution Matrix of Nuclide Production

R.Katano, T.Endo, A.Yamamoto(1), Y.Kamiyama, K.Kirimura, S.Kosaka(2)

1)Nagoya University, Nagoya, Japan, 2)Mitsubishi Heavy Industries, Ltd., Kobe, Japan

A procedure to automatically generate a simplified burnup chain is proposed, which preserves prediction accuracy of target nuclides number density using calculation results of a detail burnup chain. In the present procedure, the contributions to production of the target nuclides are quantitatively evaluated at first using the off-diagonal elements of a burnup matrix and nuclides number density. Then, by considering the contributions to nuclide production, which is represented as a “contribution matrix”, the nuclides in detail burnup chain are selected according to the importance represented as the contribution matrix. Finally, a simplified burnup chain is generated with the target nuclides and the selected nuclides from the contribution matrix. Since the present method only utilizes the burnup matrix, which is usually used in the common lattice physics computation, its implementation to production codes is easy. In order to examine the effectiveness of the present method, burnup calculation in a PWR fuel pin-cell problem with some simplification on calculation conditions is carried out. The calculation results suggest that the present method will be a good candidate for an automated generation method for a simplified burnup chain.

14:10 PM

Important Fission Product Nuclides Identification Method for Simplified Burnup Chain Construction

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

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

A method of identifying important fission product nuclides which should be included in a simplified burnup chain is proposed. This method utilizes adjoint nuclide number densities and contribution functions which quantify the importance of nuclide number densities to the target nuclear characteristics: number densities of specific nuclides after burnup. This method is tested against light water reactor fuel pin-cell problems, and it is shown that this method successfully identifies important fission product nuclides which should be included in a simplified burnup chain with which nuclide number densities of target nuclides after burnup are well reproduced.

14:30 PM

Application of Backtracking Algorithm to Depletion Calculations

M.Wu

China Institute of Atomic Energy, Beijing, China

Based on the theory of linear chain method for analytical depletion calculations, the burn-up matrix is decoupled by the divide and conquer strategy and the linear chain with Markov characteristic is formed. The density, activity and decay heat of every nuclide in the chain can be calculated by analytical solutions. Every possible reaction path of the nuclide must be considered during the linear chain establishment process. To confirm the calculation precision and efficiency, the algorithm which can cover all the reaction paths of the nuclide and search the paths automatically according to the problem description and precision restrictions should be sought. Through analysis and comparison of several kinds of searching algorithms, the backtracking algorithm was selected to search and calculate the linear chains using Depth First Search (DFS) method. The depletion program can solve the depletion problem adaptively and with high fidelity. The solution space and time complexity of the program were analyzed. The new developed depletion program was coupled with Monte Carlo program MCMG-II to calculate the benchmark burn-up problem of the first core of China Experimental Fast Reactor (CEFR). The initial verification and validation of the program was performed by the calculation.

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

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SCOPE2 is a next generation core calculation code based on the semi-analytic simplified P3 nodal method in 3-dimensional pin-by-pin geometry with microscopic-correction depletion capability. SCOPE2 uses pin-wise homogenized multi-group cross-section library generated by AEGIS. SCOPE2 can track number densities of major nuclides explicitly in each pin and capture effects under off-nominal condition directly with microscopic-correction model. In addition, the special treatment at Gd-bearing UO2 pins is implemented due to its large spatial self-shielding. It is the method that reconstructs cross-section trends in depletion under arbitrary neutron spectrum using number density of gadolinium in spite of burnup. To prove the effectiveness of this method, depletion calculations in single-assembly geometry are performed. The results show that this model can reconstruct cross-sections accurately in depletion under different spectrum from base condition such as RCC-in depletion. Thus, the high effectiveness of this model is proved.

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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

EDF R&D, Clamart, France

EDF/R&D is developing a new calculation scheme based on the transport-Simplified Pn (SPn) approach. The lattice code used is the deterministic code APOLLO2, developed at CEA with the support of EDF and AREVA-NP. The core code is the code COCAGNE, developed at EDF R&D. The latter can take advantage of a microscopic depletion solver which improves the treatment of spectral history effects. The accuracy of this model has been proven to be very useful for industrial uses but until now, being very time consuming as well as memory greedy, this model could not be used in production and needed to be re-written from the ground with speed and memory constraints in mind. This paper introduces the new methodology, a 2-levels core partitioning paradigm. Theses partitions, based on physics considerations, are also well suited for modern multi-core CPU architecture, and parallelizing in shared memory shows good speedup. Compared to the previous version, a total speedup of around 100 has been obtained: roughly 20 due to the algorithm and 5 due to the parallelization on a 8 cores chip. These results have been obtained on a machine similar to those that will be used in production.

Track4-1 Verification, Validation and Uncertainty Analysis

Session Chair: Hany Abdel-Khalik(NSCU), Kenji Yokoyama(JAEA)

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

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The reliability of calculation tools to evaluate and calculate dose rates appearing behind multi-layered shields is important with regard to the certification of transport and storage casks. Actual benchmark databases like SINBAD do not offer such configurations because they were developed for reactor and accelerator purposes. Due to this, a benchmark-suite based on own experiments that contain dose rates measured in different distances and levels from a transport and storage cask and on a public benchmark to validate Monte-Carlo-transport-codes has been developed. The analysed and summarised experiments include a ^{60}Co point-source located in a cylindrical cask, a ^{252}Cf line-source shielded by iron and polyethylene (PE) and a bare ^{252}Cf source moderated by PE in a concrete-labyrinth with different inserted shielding materials to quantify neutron streaming effects on measured dose rates. In detail not only MCNP™ (version 5.1.6) but also MAVRIC, included in the SCALE 6.1 package, have been compared for photon and neutron transport. Aiming at low deviations between calculation and measurement requires precise source term specification and exact measurements of the dose rates which have been evaluated carefully including known uncertainties. In MAVRIC different source-descriptions with respect to the group-structure of the nuclear data library are analysed for the calculation of gamma dose rates because the energy lines of ^{60}Co can only be modelled in groups. In total the comparison shows that MCNP™ fits very well to the measurements within up to two standard deviations and that MAVRIC behaves similarly under the prerequisite that the source-model can be optimized.

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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

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This paper presents a synthesis of the latest advances in the Verification and Validation (V&V) process of the new French (CEA) deterministic neutron transport code APOLLO3® developed within the framework of a common CEA, AREVA and EDF project. It focuses more precisely on the generic V&V of the main transport flux solvers of the code (namely IDT, Minaret, Pastis, TDT and Minos,) through 1D to 3D international benchmarks (ZPR-1D, Stepanek, C5G7, Takeda). Precise criteria have been defined to assess the quality of each solver by comparison with TRIPOLI4® multigroup Monte-Carlo calculations that have been performed for each configuration. We show that pure transport flux solvers (IDT, Minaret, Pastis and TDT-MOC) based on S_n , P_n and characteristics methods meet the k_{eff} target precision criteria (100 pcm) whereas SP_n solver (Minos) give satisfactory results within reasonable computation time. The complementary of the APOLLO3® flux solvers set is globally highlighted.

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DeCART Code Verifications by Numerical Benchmark Calculations of HTTR

C.J.Jeong, H.C.Lee, J.M.Noh

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DeCART code verifications have been performed through the numerical benchmark calculations of HTTR. The reference calculations have been carried out using the Monte Carlo McCARD code in which a double heterogeneity model was used. Verification results show that the DeCART code gives less negative MTC and RTC than the McCARD code does and thus the DeCART code underestimates the multiplication factors at states with high moderator and reflector temperatures. However, the DeCART code predicts more negative FTC than McCARD code does. In the depletion calculation for the HTTR single cell and single block, the error of the DeCART code increases with burnup. While the DeCART code error in a 2-dimensional core depletion calculation decreases with burnup up to around 500 FPD.

14:30 PM

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

K.Yokoyama

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A burnup sensitivity analysis solver based on generalized perturbation theory has been developed in a multi-purpose analysis framework MARBLE. The new solver has capability to calculate sensitivity coefficients of transmuted nuclear fuel composition after burnup, i.e. atomic number density, with respect to nuclear data. Based on the diffusion theory, the new solver calculates the burnup sensitivity coefficients not only in 2-dimensional R-Z model but also in 3-dimensional Hexagonal-Z model, that is not supported in a existing code system in JAEA. The new solver has been verified by comparing with results of so-called direct calculation. According to the generalized perturbation theory, it is possible to calculate a breakdown of the burnup sensitivity coefficient by using five terms which appear in the formulation. In the present paper, a new numerical verification method named component-wise direct calculation is proposed to check the results of each term. A numerical experiment of the new verification method was performed, and it shows that the new verification method is valid. In addition, a sample application result is shown in order to evaluate calculation model effects due to difference between 2- and 3-dimensional models. The sample application result suggests that 3-dimensional calculation model is necessary for some heterogeneous core configuration.

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Validation of HELIOS for ATR Core Follow Analyses

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This work summarizes the validation analyses for the HELIOS code to support core design and safety assurance calculations of the Advanced Test Reactor (ATR). Past and current core safety assurance is performed by the PDQ-7 diffusion code; a state of the art reactor physics simulation tool from the nuclear industry's earlier days. Over the past twenty years, improvements in computational speed have enabled the use of modern neutron transport methodologies to replace the role of diffusion theory for simulation of complex systems, such as the ATR. More exact methodologies have enabled a paradigm-shift away from highly tuned codes that force compliance with a bounding safety envelope, and towards codes regularly validated against routine measurements. To validate HELIOS, the 16 ATR operational cycles from late-2009 to present were modeled. The computed power distribution was compared against data collected by the ATR's on-line power surveillance system. It was found that the ATR's lobe-powers could be determined with ±10% accuracy. Also, the ATR's cold startup shim configuration for each of these 16 cycles was estimated and compared against the reported critical position from the reactor log-book. HELIOS successfully predicted criticality within the tolerance set by the ATR startup procedure for 13 out of the 16 cycles. This is compared to 12 times for PDQ (without empirical adjustment). These findings, as well as other insights discussed in this report, suggest that HELIOS is highly suited for replacing PDQ for core safety assurance of the ATR. Furthermore, a modern verification and validation framework has been established that allows reactor and fuel performance data to be computed with a known degree of accuracy and stated uncertainty.

15:10 PM

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

G.Aliberti, M.A.Smith

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A new code PERSENT was recently developed at ANL for sensitivity analysis of relevant integral parameters in 3D geometry and transport theory. The goal of this paper is to demonstrate the computational capabilities of PERSENT with respect to the deterministic codes commonly used for sensitivity analysis and particularly to address the impact on sensitivity coefficients when a simplified RZ model was generally adopted in past studies for the analysis of complicated reactor designs.

Track2-3 Deterministic Transport Theory

Session Chair: Yousry Azmy(North Carolina State Univ.), Kazuya Yamaji(MHI)

13:30 PM
Axial Transport Solvers for the 2D/1D Scheme in MPACT

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The MPACT code being developed collaboratively at the University of Michigan (UM) and Oak Ridge National Laboratory (ORNL) provides users with a variety of deterministic methods for solving the 2D and 3D Boltzmann transport equation. One of these methods, the 2D/1D technique, decomposes 3D problems into a 1D axial stack of 2D radial planes. In this scheme, the 2D planes are typically solved using a method such as the Method of Characteristics (MOC) to preserve the geometric heterogeneity in the radial direction. These planes are incorporated into a 1D axial solver, which can use a variety of methods. This work demonstrates the use of the traditional nodal methods for solving the 1D axial problem (finite difference, NEM, SANM, SP3), but also introduces a discrete ordinates (Sn) solver which uses up to cubic Legendre expansion spatially and can also incorporate higher order angular distributions of the radial transverse leakage.
Several test cases are presented to demonstrate the accuracy of the solvers for various axial sizes. The first three are the 3D-C5G7 extension benchmark cases. The fourth case is a single quarter assembly benchmark problem with explicit nozzle, plenum, and core plate modelling known as AMA Problem 3. The final case is a quarter core benchmark problem that is an extension of the quarter assembly problem known as AMA Problem 5. In general, the diffusion-based axial solvers perform very well, though higher-order solvers provide some benefit in more difficult problems, particularly rodded cases.

13:50 PM
Development of Legendre Expansion of Angular Flux Method for 3D MOC Calculation

Y.Kato, T.Endo, A.Yamamoto

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A three-dimensional transport calculation method, Legendre Expansion of Angular Flux Method (LEAF method) has been proposed for practical MOC calculations in three-dimensional geometry. In the LEAF method, transport calculation is carried out on “characteristics planes(CP)” rather than conventional ray traces, whose concept is similar to that of the ASMOC3D method. By utilizing the characteristics plane, required memory and computational load are significantly reduced. Each characteristics plane forms rectangular regions, thus efficient transport calculation scheme based on transmission probability can be adopted. Consequently, the LEAF method can practically perform MOC calculation in three-dimensional geometry. In the LEAF method, spatial distribution of incoming, outgoing and average angular fluxes in a CP are expressed as a series of the Legendre polynomials in order to accurately capture their spatial dependence. Calculation results in a test problem indicate the accuracy and effectiveness of the LEAF method.

14:10 PM
Benchmark on Deterministic Time-Dependent Transport Calculations without Spatial Homogenisation

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

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The space-time neutron kinetics benchmark on deterministic transport calculations without spatial homogenization C5G7-TD has been developed and proposed for verification of codes solving the time-dependent neutron transport equation. The well-known C5G7 benchmark has been chosen as the base for new benchmark. The proposed benchmark has been calculated by SUHAM-TD code, which realizes the surface harmonic method (SHM). Authors hope to attract the attention of other researchers in order to involve them to participate in calculations of the proposed benchmark.

14:30 PM
Coarse-Grained Parallelism for Full-Core Transport Calculations

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

CEA, Saclay, France

In this paper we analyze the synergy between the Domain Decomposition Method and the Coarse-Mesh Finite Difference technique. In contrast to massively parallel computations, we construct a coarse-grained parallelism for daily run calculations on standard SIMD workstations based on shared memory architecture. We evaluate the effectiveness of the algorithm for several high-fidelity calculations spanning different types of color-sets up to the full-core. We show that CPU times for a best-estimate 2D calculation of the EPR can be reduced from several days to few hours using a standard workstation.

14:50 PM
Parallel Performance Results for the OpenMOC Method of Characteristics Code on Multi-Core Platforms

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Over the past decade, multi-core processors have become a fixture of modern computing systems. The shift towards multi-core architectures has ushered in a new era of shared memory parallelism for scientific applications. This paper describes and analyzes two parallel transport sweep implementations in the OpenMOC method of characteristics neutron transport code for multi-core platforms using OpenMP. Strong and weak scaling studies are performed for Intel Xeon multi-core processors using the GNU and Intel C++ compilers. In addition, variations of both implementations using exponential intrinsics and linear interpolation techniques to evaluate the exponential in the MOC equations are compared and contrasted. The results for Open-MOC demonstrate 100% parallel efficiency for 12 threads on 12 cores on Intel Xeon platforms. These results illustrate the potential for hardware acceleration for MOC neutron transport on modern multi-core architectures. In addition, they provide insights as to the benefits and shortcomings of the OpenMP parallel programming model with respect to MOC for both multi-core processors as well as future many-core platforms.

15:10 PM
Making More Precise the Surface Pseudosources Method for RBMK Cluster Cells

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At present the KLARA code is being used for the multigroup calculations of the VVER and RBMK cluster cells and for preparation of a few group cell characters. Some calculation instability has appeared in the 69-group calculations of the RBMK cluster cells. Detailed analysis of these calculations showed that this instability has been connected with divergence of a type pole in a few integrals in time of a calculation of angular moments of the sine component of the Green’s function. We removed this peculiarity by means of addition of the two lines of angular moments of the sine component of the neutron distribution function, when these peculiarities compensated each other and a calculation became stable. Difference of the Kinf. values between the options KLARA in G_3^1- approximation and PIJ became less or equal to 0.2% for the RBMK cluster cells both with a fuel and without.

Track3-2 Monte Carlo Methods

Session Chair: Kan Wang(Tsinghua Univ.), Yasushi Nauchi(CRIEPI)

13:30 PM
A Monte Carlo Method for Prompt and Delayed Alpha Eigenvalue Calculations

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Several technological issues, such as for instance kinetics studies of accelerator-driven systems, reactor start-up analysis, or reactivity measurements, demand the asymptotic time behaviour of neutron transport to be assessed. Typically, this amounts to solving an eigenvalue equation associated to the Boltzmann operator, whose precise nature depends on whether delayed neutrons are taken into account. In this work, we propose a Monte Carlo method for determining the dominant eigenvalue of the Boltzmann operator and the associated fundamental mode for arbitrary geometries, materials, and boundary conditions. This approach can be applied to configurations involving both prompt and delayed neutrons. Extensive verification tests of the algorithm are performed.

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

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Geometry navigation plays the most fundamental role in Monte Carlo particle transport simulation. It's mainly responsible for locating a particle inside which geometry volume it is and computing the distance to the volume boundary along the certain particle trajectory during each particle history. Geometry navigation directly affects the run-time performance of the Monte Carlo particle transport simulation, especially for large scale complicated systems. Two geometry acceleration algorithms, the automatic neighbor search algorithm and the oriented bounding box algorithm, are presented for improving geometry navigation performance. The algorithms have been implemented in the Super Monte Carlo Calculation Program for Nuclear and Radiation Process (SuperMC) version 2.0. The FDS-II and ITER benchmark models have been tested to highlight the efficiency gains that can be achieved by using the acceleration algorithms. The exact gains may be problem dependent, but testing results showed that runtime of Monte Carlo simulation can be considerably reduced 50%~60% with the proposed acceleration algorithms.

14:10 PM
Continuous-Energy Monte Carlo Methods for Calculating Generalized Response Sensitivities Using TSUNAMI-3D

C.M.Perfetti, B.T.Rearden

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This work introduces a new approach for calculating the sensitivity of generalized neutronic responses to nuclear data uncertainties using continuous-energy Monte Carlo methods. The GEneralized Adjoint Responses in Monte Carlo (GEAR-MC) method has enabled the calculation of high resolution sensitivity coefficients for multiple, generalized neutronic responses in a single Monte Carlo calculation with no nuclear data perturbations or knowledge of nuclear covariance data. The theory behind the GEAR-MC method is presented here and proof of principle is demonstrated by calculating sensitivity coefficients for responses in several 3D, continuous-energy Monte Carlo applications.

14:30 PM
Enhancements in Continuous-Energy Monte Carlo Capabilities for SCALE 6.2

B.T.Rearden, L.M.Petrie, D.E.Peplow, K.B.Bekar, D.Wiarda, C.Celik, C.M.Perfetti, M.E.Dunn(1), S.W.D.Hart(2)

1)Oak Ridge National Laboratory, Oak Ridge, USA, 2)University of Tennessee, Knoxville, USA

SCALE is a widely used suite of tools for nuclear systems modeling and simulation that provides comprehensive, verified and validated, user-friendly capabilities for criticality safety, reactor physics, radiation shielding, and sensitivity and uncertainty analysis. For more than 30 years, regulators, industry, and research institutions around the world have used SCALE for nuclear safety analysis and design. SCALE provides a "plug-and-play" framework that includes three deterministic and three Monte Carlo radiation transport solvers that are select- ed based on the desired solution. SCALE includes the latest nuclear data libraries for continuous-energy and multigroup radiation transport as well as activation, depletion, and decay calculations. SCALE's graphical user interfaces assist with accurate system modeling, visualization, and convenient access to desired results. SCALE 6.2 provides several new capabilities and significant improvements in many existing features, especially with expanded continuous-energy Monte Carlo capabilities for criticality safety, shielding, depletion, and sensitivity/uncertainty analysis, as well as improved fidelity in nuclear data libraries. A brief overview of SCALE capabilities is provided with emphasis on new features for SCALE 6.2.

14:50 PM
Leakage-Corrected Fast Reactor Assembly Calculation with Monte-Carlo Code TRIPOLI4® and Its Validation Methodology

L.Cai, Y.Pénéliau, J.Tommasi, J-F Vidal(1), C.M.Diop(2)

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A leakage model based on B1 Homogeneous Equations has been recently implemented in continuous-energy Monte Carlo code TRIPOLI4®. This leakage model algorithm iterates between the point-wise Monte Carlo simulation and a B1 Homogeneous Equations solver till reaching a final critical state in Monte Carlo simulation. The two advantages of our leakage model compared with the others are: we use critical flux spectrum to generate the multi-group constants for solving the B1 Homogeneous Equations; the leakage coefficients calculated are considered in point-wise Monte Carlo simulation. This leakage model is validated by a predesigned numerical experiment simulated with continuous-energy TRIPOLI4® and the obtained results are also compared with those from SERPENT leakage model and deterministic leakage model in ECCO code. Finally, our leakage-corrected multi-group constants are used in transport theory based core calculation and they give out consistent multiplicative factor and neutronic balance.

15:10 PM
Impact of Nearest Neighbor Distribution of Fuel Particle on Neutronics Characteristics in Statistical Geometry Model

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Effect of nearest neighbor distribution of fuel particle used in the statistical geometry model (STGM) for the neutronics characteristics calculation is investigated. The statistical uniform distribution is sometimes assumed as a nearest neighbor distribution (NND) in STGM. This assumption is considered to be valid for the high temperature gas reactor (HTGR) fuel compact, where small fine fuel particles are distributed in the graphite matrix. However, the validity of this assumption is not confirmed for relatively large fuel debris distributed in light water, which is one of the plausible situations in a light water reactor core with severe accident. In order to verify the validity of NND of statistical uniform distribution, two different types of NNDs are numerically calculated: One is determined by the simple cubic lattice where a single fuel sphere is located in the center of unit lattice; another is obtained by random arrangement of fuel spheres in three dimensional space. Based on these NNDs, mean neutron transmission probability values in moderator are evaluated. As a result, we find that the mean transmission probability depends on the NNDs even if the packing fraction of fuel particle and fuel particle radius are identical, and the difference of mean transmission probability among NNDs becomes larger as the optical path length in moderator increases. Therefore, when the optical path length in moderator is large, the simple cubic lattice geometry and the statistical uniform distribution give significantly different neutron multiplication factors, even if the packing fractions in both NNDs are identical. Consequently, in order to properly estimate the neutronics characteristics such as the neutron multiplication factor, the appropriate NND should be provided in the STGM calculation for the light water moderated system.

Track12-1 Fuel Cycle and Actinide Management

Session Chair: Nicholas Brown(BNL), Naoyuki Takaki(Tokyo City Univ.)

13:30 PM

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

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The paper presents a fast and flexible burnup model for fuel cycle simulations which is based on the description of the one-group cross-sections as analytic functions of the isotopic composition. This was accomplished by multi-dimensional regression based on the results of numerous core calculations. The developed model is able to determine the spent fuel composition in reasonable CPU time, and was integrated into a simplified fuel cycle model containing Gas Cooled Fast Reactors (GFR) and conventional light water reactors (LWRs). The fuel cycle simulations revealed an advantageous effect of increased minor actinide content in the GFR core on the fuel utilization parameters. In order to explore the processes that lay behind this effect the neutronics balance of the GFR was investigated in equlibrium cycle conditions.

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

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

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The variations in the values of activity, toxicity and decay heat of nuclear waste, particularly spent nuclear fuel (SNF) and high-level waste (HLW), have been assessed for 40 fuel-cycle examples at 10, 100, and 100,000 years after reactor discharge. A correlation analysis was performed to identify the parameters that have common trends for the purpose of reducing the number of evaluation metrics required for comparing the fuel cycles. Such an analysis in this paper allowed focusing on the activity and inhalation toxicity at 10 years and 100,000 years after discharge. The variation in the 10 years activity is primarily due to the variation in the quantity and specific activity of the fission products. The specific activity of the fission products is consistently higher for thorium fuel cycles and smaller when the fuel residence time is long. The variation in the activity 100,000 years after discharge is primarily explained by the quantity of U-233 and Pu-239 sent to nuclear waste, which is directly linked to the type of fuel used and to the reprocessing scheme employed for these elements. The difference between the inhalation toxicity and the activity is directly explained by the variations in the effective dose coefficients. The heavy actinides such as Pu, Am and Cm have a predominant effect, by 3 to 4 orders of magnitude, on the inhalation toxicity. As a consequence, thorium fuel-cycles always display low values of inhalation toxicity 10 years after discharge. Uranium-plutonium fuel cycles display low 10 years inhalation toxicity values only when their TRU content is being reprocessed. Thorium fuel cycles that send U-233 into nuclear waste are responsible for some of the largest values in the inhalation toxicity at 100,000 years after discharge. However, the largest values in the inhalation toxicity 100,000 years after discharge are dominated by uranium fuel-cycles that send large amount of Pu-239 into the waste.

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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

The possible delay of decades for the deployment of fourth generation reactors brings up new issues. Countries like France which are storing Plutonium for years in prediction of starting FBR could have to adapt their plutonium management strategy. We have compared different strategies for plutonium recycling in PWR reactors. We have chosen to limit the scope of this study to PWRs considering the standard uranium cycle, and we investigated the influence of the heterogeneity in the assembly that would contain the recycled plutonium. The comparison of different strategies is made at steady-state. This paper present a specific method developed to allow complete and detailed studies of equilibrium scenarios and how it has been implemented and integrated inside the opensource MURE package. We will then discuss of the results obtained through this method apply to PWRs loaded with homogeneous and heterogeneous assemblies using the following criteria: resource consumption, Pu inventories in the cycle and waste production.

14:30 PM

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

M.Tokashiki, S.Okui

Nuclear Fuel Industries, Ltd., Japan

The systematic method with least squares method to evaluate high accurate equivalence factors, and non-linear "Equivalent Pu-239" formula which is applicable to broad range of Pu isotopic composition, have been developed. And the method to evaluate each MOX fuel rod type's equivalence factors with high accurate equivalence for local power peaking has been developed. The applicability of the methods was confirmed for BWR MOX fuel.

14:50 PM

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

Thorium-plutonium Mixed OXide fuel (Th-MOX) is considered for use as light water reactor fuel. Both neutronic and material properties show some clear benefits over those of uranium-oxide and uranium-plutonium mixed oxide fuel, but for a new fuel type to be licensed for use in commercial reactors, its behaviour must be possible to predict. For the thermomechanical behaviour, this is normally done using a well validated fuel performance code, but given the scarce operation experience with Th-MOX fuel, no such code is available today.

In this paper we present the ongoing work with developing a fuel performance code for prediction of the thermomechanical behaviour of Th-MOX for light water reactors. The wellestablished fuel performance code FRAPCON is modified by incorporation of new correlations for the material properties of the thorium-plutonium mixed oxide, and by develoment of a new subroutine for prediction of the radial power profiles within the fuel pellets. This paper lists the correlations chosen for the fuel material properties, describes the methodology for modifying the power profile calculations and shows the results of fuel temperature calculations with the code in its current state of development. The code will ultimately be validated using data from a Th-MOX test irradiation campaign which is currently ongoing in the Halden research reactor.

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Two-Stage Fuel Cycles with Accelerator-Driven Systems

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

Argonne National Laboratory, Argonne, USA

As part of ongoing efforts to assess nuclear fuel cycle options, four fuel cycle options based on the same two reactor technologies have been studied. All four options are composed of two stages, one which contains pressurized-water reactors (PWRs), and the other, fast spectrum accelerator-driven systems (ADS). The performance characteristics and material mass flows have been determined for the fuel cycle options considered, and compared. The three major difficulties encountered when modeling and analyzing these fuel cycle options have been to maintain the PWR fuel temperature reactivity coefficient negative when multi-recycling MOX fuel, to design the ADS core to be a breeder, and to achieve a high enough k_{eff} in the ADS to avoid the accelerator power consumption to be larger than the power generated by the ADS core. The differences observed in the performance characteristics and mass flows between the four fuel cycle options analyzed are discussed in this paper. Overall it is found that despite the four fuel cycle options being based on the same reactor technologies and seemingly similar at first sight, they perform differently and offer different features: resource utilization, need for uranium enrichment, required reprocessing capacity, and material type to be stored.

Track1-6 Reactor Analysis Methods

Session Chair: Mohamed Elsawi(Khalifa University), Yunzhao Li(Xi'an Jiaotong University)

15:55 PM
Modeling of Shutdown Cooling Reactivity Effects with SIMULATE

T.Bahadir

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During plant outages and subsequent return to power operations, the fuel isotopic inventory, relative to that of steady-state operation, changes as nuclides go through radioactive decay. Accurate predictions of essential core neutronic parameters require that the reactivity impact of these isotopic changes be taken into account. Extended outages, as experienced in Japan following the Fukushima accident, as well as consecutive outages, which are close in time such that the residual effects from the first one influence the reactivity of the subsequent one, may challenge the reactivity models implemented in three-dimensional (3D) core simulators. This paper studies the modeling issues with the core simulators for analyzing nominal and extended refueling shutdown outages as well as combination of such with mid-cycle outages. The shutdown cooling (SDC) reactivity models that are implemented based on macroscopic (lumped) depletion and explicit isotope tracking are described for the Studsvik 3D core simulators, SIMULATE-3 and SIMULATE5. Using relevant numerical test cases, it is demonstrated that the explicit isotope-tracking model offers robust and accurate prediction capabilities for computing shutdown-cooling reactivity, regardless of the outage time.

16:15 PM
Implementation and Verification of the SDM in the TITAN 3-D Sn Transport Code

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The subgroup decomposition method (SDM) has recently been developed as an improvement over the consistent generalized energy condensation theory for treatment of the energy variable in deterministic particle transport problems. By explicitly preserving reaction rates of the fine-group energy structure, the SDM directly couples a consistent coarse-group transport calculation with a set of fixed-source “decomposition sweeps” to provide a fine-group flux spectrum. This paper will outline the implementation of the SDM into a three-dimensional, S_N deterministic transport code such as TITAN. The new version of TITAN, TITAN-SDM, is tested using a 1-D benchmark problem based on the High Temperature Engineering Test Reactor (HTTR) in Japan. In addition to accuracy, this study examines the efficiency of the SDM algorithm in a 3-D S_N transport code.

16:35 PM
Transport Core Solver Validation for the ASTRID Conceptual Design Study with APOLLO3®

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In this study, we are questioning the capability of 3D deterministic transport solvers available at CEA and EPM to accurately predict the void effect of the sodium plenum of the CFV core (GEN IV Sodium Fast Reactor concept selected for the ASTRID project at CEA). Discrete Ordinates, Spherical Harmonics and Simplified Spherical Harmonics transport solvers have been tested against Monte Carlo method on Takeda Benchmark Model 4 and on a new variant with a sodium plenum. Sn solvers offer the best accuracy on multiplication factor, void effect and control rod worth, with values very close to reference Monte Carlo ones. However, even using acceleration techniques, computing time are quite consequent. This difficulty sould be solved by the parallelization of the MINARET Sn solver in the APOLLO3® platform. SPn solvers are very fast, but also quite distant from the reference, especially for the calculation of the void effect where they prove largely inadequate. A good compromise between accuracy and computational time is obtained with the Pn solver VARIANT. The corresponding nodal method is being implemented in APOLLO3®.

16:55 PM
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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PSI, Villigen, Switzerland

Due to the recent renewed interest in Sodium Fast Reactor cores featuring a low coolant void worth, an attempt was made to perform a methodology assessment for the evaluation of such non-conventional designs' performance characteristics, which might not be accurately predicted by traditional reactor physics deterministic methods. The Monte Carlo code Serpent was employed to provide the reference solutions, thanks to both its capability to model complex 3D geometries and its continuous energy representation of cross-sections. Serpent results were then compared with ERANOS multi-group, 2D and 3D, transport and diffusion theory predictions. A reasonably good agreement was found among the calculated nominal reactor parameters (e.g., reactivity, effective delayed neutron fraction, prompt neutron generation time) and coolant void worth involving perturbations of the active core regions. Major discrepancies occur when voiding both fuel zones and the large sodium plenum above the core, though. In such cases, further investigations were carried out concerning the impact of computational options at a cross-section generation level, for instance relative to spectrum and leakage approximations for sub-critical regions. It may be concluded that the former (i.e., 33 vs. 175 energy groups condensation) have an impact of the order of 200 pcm on absolute reactivity values, but does not affect reactivity variations, whereas the latter (i.e., imposed buckling values) play a substantial role, inducing differences between Serpent and ERANOS of up to 35 %. Finally, the limits of variational nodal methods in treating very low density zones were confirmed, being particularly critical when employing fine energy group structures.

17:15 PM
Results of Verification of Computer Codes Used for Analysis of BN-1200 Reactor Core Neutronics

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The paper presents the results of neutronic calculations of a test model of the BN-1200 reactor core with the use of engineering codes and codes based on the Monte Carlo method.

17:35 PM
APOLLO3® Based Method for 3D Warped Cores Calculations – Application to Flowering Tests of Phenix

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In the frame of multiphysics coupling tools development, a methodology for deformed cores calculations with APOLLO3®, a neutronic code in development at CEA Saclay, was developed. This methodology, based on three dimensional pixels, was applied to Phénix flowering tests performed in 2009 and 2012, during which reactivity and displacement were measured.
This paper first presents these flowering tests and the results of previous study, using RZ symmetry and ERANOS, which only reproduces roughly the measurements. Then, our methodology is explained, assessed with different pixel sizes and validated by a comparison with TRIPOLI-4® results. Some academic deformations are also calculated in order to analyze the impacts of heterogeneities, and we confirm, in particular, that the bending of assemblies can be neglected. Afterwards, mechanical calculations of flowering tests, done with CAST3M, are very briefly described. Finally our results are given, for central and peripheral flowerings, and with different hypotheses on gaps between assemblies.
The paper concludes that real core compaction is probably driven by irradiation deformations of assemblies, and can be approximated by gaps between assemblies.

Track4-2 Verification, Validation and Uncertainty Analysis

Session Chair: Bassam Khuwaileh(NCSU), Kensuke Kojima(JAEA)

15:55 PM
Benchmark Calculation with MOSRA-SRAC for Burnup of a BWR Fuel Assembly

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The Japan Atomic Energy Agency has developed the Modular Reactor Analysis Code System MOSRA to improve the applicability of neutronic characteristics modeling. The cell calculation module MOSRA-SRAC is based on the collision probability method and is one of the core modules of the MOSRA system. To test the module on a real-world problem, it was combined with the benchmark program “Burnup Credit Criticality Benchmark Phase IIIC.” In this program participants are requested to submit the neutronic characteristics of burnup calculations for a BWR fuel assembly containing fuel rods poisoned with gadolinium (Gd_2O_3), which is similar to the fuel assembly at TEPCO’s Fukushima Daiichi Nuclear Power Station. Because of certain restrictions of the MOSRA-SRAC burnup calculations part of the geometry model was homogenized. In order to verify the validity of MOSRA-SRAC, including the effects of the homogenization, the calculated burnup dependent infinite multiplication factor and the nuclide compositions were compared with those obtained with the burnup calculation code MVP-BURN which had already been validated for many benchmark problems. As a result of the comparisons, the applicability of MOSRA-SRAC module for the BWR assembly has been verified. Furthermore, it can be shown that the effects of the homogenization are smaller than the effects due to the calculation method for both multiplication factor and compositions.

16:15 PM
Verification of the COCAGNE Core Code Using Cluster Depletion Calculations

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EDF/R&D is developing a new calculation scheme based on the transport-Simplified Pn (SPn) approach. The lattice code used is the deterministic code APOLLO2, developed at CEA with the support of EDF and AREVA-NP. The core code is the code COCAGNE, developed at EDF R&D. The latter can take advantage of a microscopic depletion solver which improves the treatment of spectral history effects. This solver can resort to a specific correction based on the use of the Pu239 concentration as a spectral indicator. In order to evaluate the improvements brought by this Pu239 correction model, one uses (3x3 assemblies) cluster depletion calculations as test-cases. UOX and UOX/MOX clusters are both considered. As a reference, APOLLO2 depletion calculations of these clusters, using a critical boron (CB) search scheme at each calculation step, are performed. This choice of methodology (using CB search instead of a fixed average CB) enables to highlight historical spectral effects related to the boron concentration. This methodology is also more consistent with the depletion calculation of real cores. Pin by pin COCAGNE calculations are performed and compared with the APOLLO2 results. The analysis of the results obtained shows that the boron concentration computed by COCAGNE gets more consistent with APOLLO2 when the Pu239 corrector is used, especially for UOX/MOX clusters. As for pin power distribution, the use of the Pu239 model also enables to reduce slightly the gap between APOLLO2 and COCAGNE. This work will be extended to clusters with gadolinium-poisoned fuel assemblies and reflector regions.

16:35 PM
LWR Fuel Reactivity Depletion Verification Using 2D Full-Core MOC and Flux Map Data

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Experimental quantification of PWR fuel reactivity (k-infinity) burnup decrement biases and uncertainties using in-core flux map data from operating power reactors has previously been conducted employing analytical methods to systematically determine experimental fuel reactivities that best match measured fission rate distributions. This optimal core reactivity distribution that best matches the measured fission rate distribution is assumed to be associated with the true fuel reactivity distribution. Some parties have questioned whether fortuitous cancellation of errors between various approximations inherent in the 3D nodal diffusion core analysis models might have caused reactivity decrement biases and uncertainties to be unrealistically small. In this study, the BEAVRS benchmark is modeled with both 2D, full-core, multi-group transport calculations and 2-group nodal diffusion calculations. The calculated in-core detector U-235 fission rates are compared with measured fission rates supplied in the benchmark. The same analytical methods previously mentioned are again used to obtain fuel reactivity biases and uncertainties. Results demonstrate that fuel batch reactivities inferred from flux map data using full-core transport calculations are nearly identical to those inferred using nodal diffusion calculations. Consequently, nodal methods do not contribute significantly to reactivity decrement biases in the BEAVRS cores. It is recommended that fuel reactivity biases and uncertainties inferred from 3D nodal diffusion calculations remain valid.

16:55 PM
CASMO-4E and CASMO-5 Analysis of the Isotopic Compositions of the LWR-PROTEUS Phase II Burnt PWR UO_2 Fuel Samples

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The isotopic compositions of 7 UO_2 samples irradiated in a Swiss PWR power plant, which were investigated in the LWR-PROTEUS Phase II programme, were calculated using the CASMO-4E and CASMO-5 assembly codes and cross-section libraries based on ENDF/B-VI and ENDF/B-VII release 1, respectively. The burnups of the samples span a very wide range from ~40 to ~120 MWd/kg. The results for a large number of actinide and fission product nuclides were compared to those of chemical analyses performed using a combination of chromatographic separation and mass spectrometry. A satisfactory overall agreement of calculated and measured nuclide concentrations has been found for both codes. Significant improvements over CASMO-4E can be seen in the CASMO-5 results for the main Pu isotopes and for the lanthanides Sm, Eu and Gd. Considering the very high burnups of some of the samples, it is important to note that the differences between calculated and measured concentrations do generally not show a trend to increase for higher burnups.

17:15 PM
Experimental Validation of Decay Heat Calculations with VESTA 2.1

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The removal of decay heat is a significant safety concern in nuclear engineering for the operation of a nuclear reactor both in normal and accidental conditions and for intermediate and long term waste storage facilities. The correct evaluation of the decay heat produced by an irradiated material requires first of all the calculation of the composition of the irradiated material by depletion codes such as VESTA 2.1, currently under development at IRSN in France. The use of this software for decay heat applications requires a consistent experimental validation effort. For this purpose, PWR assembly decay heat measurements have been selected from the CLAB, GE and HEDL data as well as fission experiments. Using the JEFF 3.1 and ENDF/BVII. 0 nuclear data libraries, VESTA 2.1 calculates the assembly decay heat for almost all cases within 4 % of the measured decay heat. On average, the JEFF 3.1 calculated decay heat value appear to give a systematic underestimation of about 2 %. When using ENDF/B-VII.0, this is limited to an average underestimation of 0.5 %.

17:35 PM
Development and Validation of Ad Hoc ORIGEN-ARP Libraries for Very High Burnup UO_2 PWR Fuel with SCALE/TRITON

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The estimation of the nuclide inventory of the spent fuel to be disposed of in the high-level waste geological repository in Switzerland is fundamental for the realization of the repository itself. The fuel characterization is relevant not only for safety-relevant issues concerning the repository, but also for all the issues related to the

Track4-2 Verification, Validation and Uncertainty Analysis

Session Chair: Bassam Khuwaileh(NCSU), Kensuke Kojima(JAEA)

previous phase, consisting of fuel handling and encapsulation. In order to keep the uncertainties sufficiently low, a fuel characterization methodology needs to be defined and validated. The large heterogeneity of the fuel used in the five Swiss reactors makes this task highly complex.
UO_2 PWR fuels have been simulated using SCALE/TRITON employing the latest ENDF/B-VII cross-section data, aiming to produce ORIGEN-ARP libraries suitable for very high burnup fuels. The results obtained from TRITON and ORIGEN-ARP calculations are compared to the results of previous experimental studies. The experimental results considered here were selected from chemical isotopic analysis and passive neutron emission performed earlier at the Paul Scherer Institute in Switzerland. A set of fuel rod samples with different burnups was used for the analysis. Very high burnup samples have been investigated, particularly for the neutron output. The results show that the developed ORIGEN-ARP libraries provide a valid alternative to very long and detailed fuel simulations without loss of accuracy in the depletion capabilities.

Track2-4 Deterministic Transport Theory

Session Chair: Ricardo Barros(Univ. do Estado do Rio de Janeiro), Tetsuo Matsumura(CRIEPI)

15:55 PM
Phase Space Bases for Response Matrix Methods

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New basis functions in space, angle, and energy are developed for use in the eigenvalue response matrix method (ERMM) . ERMM solves reactor eigenvalue problems by spatially decomposing a model and linking the resulting independent nodes through approximate boundary conditions. The conditions are defined via a set of nodal transport calculations for which the forcing function is a tensor product of basis functions in all phase space variables on one nodal surface (with vacuum elsewhere). Traditional implementations of ERMM have used the standard bases of mathematical physics, including Legendre polynomials and their discrete analogs. Alternatively, new bases that incorporate some physics a priori are shown to yield satisfactory results with far lower expansion orders than is possible with more traditional bases. In particular, a very simple spatial basis incorporating pin-dependent spatial variation reduces partial current errors by nearly an order of magnitude over a standard basis. In angle, conservative bases with appropriate angular quadratures outperform simpler bases. In energy, incorporating representative spectral information from infinite medium calculations results in a basis that can achieve sub-1% pin fission rate errors with as low as a 5th order expansion in energy for a 44- group problem. The results indicate that highly-accurate, low-order response matrix solutions should be feasible for reactor physics analysis.

16:15 PM
The Drift Diffusion Limit of Thermal Neutrons: Theoretical and Numerical Results

P.A.Vaquer, R.G.McClarren, M.L.Adams

Texas A&M University, TX, USA

This study presents a new approach to modeling neutron thermalization which can greatly simplify nuclear reactor simulations. A drift-diffusion equation is derived through an asymptotic analysis of the 1-D energy-dependent transport equation. Two test problems were used to provide numerical comparisons of this drift-diffusion approximation to MCNP6 calculations. Both of these test problems include a large 1-D graphite moderator with a steady-state temperature gradient. The results demonstrated that the drift-diffusion is able to capture energydependent effects and can successfully be used for modeling neutron thermalization in a large moderating material.

16:35 PM
Adequacies of Different Convergence Accuracy Measures in Full-Core Nodal Flux Computations

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This article presents a computational analysis that emphasizes the importance of achieving convergence rate invariance in deriving truly dependable error estimates for use in convergence monitoring in iterative nodal flux solution processes. It is argued and shown that this property is not fulfilled by every known formula in common use for estimation of the convergence accuracy. Based on a rigorous theoretical convergence analysis context, a generalized convergence error measure concept is derived that avoids remaining computational uncertainties that may otherwise arise due to convergence rate influences combined with the use of a deficient error measure. A pursued in-depth convergence analysis has confirmed that the use of this derived convergence error measure concept enables an optimum convergence monitoring process. With this generalized convergence control concept in place, the delivered nodal flux solution can be expected to truly and dependably adhere to any imposed convergence accuracy requirement. Simultaneously, the computational effort (i.e. numbers of iterations) required for establishing that accuracy is automatically avoided to evolve beyond minimal necessity. This concept is in standard use in AREVA's core simulation code ARTEMIS™ in which it enforces a significantly enhanced convergence accuracy.

16:55 PM
Flexible Semi-Analytical Calculation Method of Escape Probability

T.Matsumura

CRIEPI, Tokyo, Japan

A new semi-analytical calculation method of escape probability is developed. Application for an infinite cylinder is introduced in present paper. Present calculation method has advantage in flexibility of application for various shapes of calculation cells. By approximating the integration result of the direction of z by simple quadratic formula divided into ranges, it became possible to perform integration to y direction analytically. Numerical result of escape probability shows high precision (less than 0.1%) by present semi-analytical calculation method. Present calculation method has flexible applicability for various two-dimensional shapes. Also it is thought that possibility of an analytical solution for the rector physics calculation is still large.

17:15 PM
Corrected Diamond Difference Method for Coupling from the Method of Characteristics to Discrete Ordinates

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A new S_N formulation has been developed which is effective at preserving the behavior of a coupled fine-mesh transport solver on a coarse, pin homogenized Cartesian mesh. The method uses angleand energy-dependent correction factors and a modified formulation for the diamond difference equations, called corrected diamond difference (CDD). An extension of the CDD method from previous work includes a new cross section correction step. It is shown that the new CDD approach is equivalent in two-dimensional problems in theory and practice. The CDD method is then applied to several three-dimensional benchmark cases using one-way 2-D/3-D coupling, and is shown to provide more accurate results as compared to the previous CDD formulation and diamond difference alone.

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

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Presented here is a spectral nodal method for the numerical solution of energy multigroup, fixed-source, discrete ordinates (S_N) problems in two-dimensional rectangular geometry. This analytical coarse-mesh method is referred to as the multigroup spectral Green´s function-constant nodal (SGF-CN) method as it uses the multigroup SGF method for numerically solving the one-dimensional transverse-integrated S_N nodal equations with constant approximations for the transverse leakage terms. The only approximations in the present X,Y-geometry nodal method occur in these transverse leakage terms, as the energy-group transfer scattering source terms are treated analytically within the offered method. Numerical results to a typical model problem are given to illustrate the method´s accuracy.

Track3-3 Monte Carlo Methods

Session Chair: Daniel J. Kelly(Knolls Atomic Power Laboratory), Zeguang Li(Tsinghua University)

15:55 PM

Large-Scale Monte Carlo Calculations with Thermal-Hydraulic Feedback

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Monte Carlo based codes provide the most accurate solution of the particle transport problem. Individual particle trajectories are followed, and the interaction physics is simulated using detailed modeling of the physical reaction. The calculations are usually done using uniform temperature and density distributions. This is a significant approximation and leads to significantly distorted solution when applied to hot full power conditions. In this paper a method for introducing the thermal-hydraulic feedback by dynamic material distributions is introduced. The global variance reduction technique has been used to optimize the power tallying. The fission source convergence was accelerated by applying the Wielandt's acceleration method. Since the aim of this work is to solve coupled neutronic/thermal-hydraulic problems a convergence acceleration strategy based on stochastic approximation was proposed. The coupled system was applied to a quarter PWR core at pin and sub-channel level resolution.

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

G.Truchet, P.Leconte, J.M.Palau, P.Archier, J.Tommasi, A.Santamarina, Y.Peneliau(1), A.Zoia, E.Brun(2)

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The analysis of void reactivity effect is prominent interest for Sodium-cooled Fast Reactor (SFR) safety. Indeed, in case of sodium leakage of the primary circuit, void reactivity represents the main passive negative feedback to ensure reactivity control. The core can be designed to maximize neutron leakage and lower the average neutron multiplication factor in the event of sodium disappearing from within assemblies. Thus, the nuclear chain reaction is stopped. The most promising solution is to place a sodium region above the fuel in order for neutrons to be reflected when the region is filled and escape when the region is empty. In terms of simulation, this configuration is a challenge for usual calculation schemes:
1. Deterministic codes are typically limited in their ability to homogenize a sub-critical medium as the sodium plenum.
2. Monte Carlo codes are typically not able to split the total reactivity effect on different components, which prevents to achieve straightforward uncertainty analysis. Furthermore, since experimental values can sometimes be small, Monte Carlo codes may not converge within a reasonable computation time.
A new feature recently available in the Monte Carlo TRIPOLI-4® based on the Exact Perturbation Theory allows very small reactivity perturbations to be computed accurately as well as reactivity effect to be estimated on distinct isotopes cross-sections. In the first part of this paper, this new feature of the code is described and then applied in the second part to a core configuration composed of several layers of fuel and fertile zones below a sodium plenum. Reactivity and its contributions from specific reactions and energy groups are calculated and compared with the results of the deterministic code ERANOS. The aim of this work is twofold: (1) Achieve a numerical validation of the new TRIPOLI-4® features and (2) Identify where deterministic codes might be less accurate and why – even when using them at full capacity (S_{16} transport, 1968 energy groups or more).

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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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1)Seoul National University, Seoul, Korea, 2)Kyoto University, Kumatori, Japan

Experiments have been carried out on the isothermal temperature reactivity coefficient (ITRC) for the light-water moderated core at the Kyoto University Critical Assembly. The temperature effect on reactivity is analyzed by the Seoul National University Monte Carlo (MC) code, McCARD, which well reproduce experimental data. The contributions of the each isotope by the density changes of the core and reflector regions and the microscopic cross section changes to the ITRCs are quantified by sensitivity analyses based on the MC adjoint-weighted perturbation methods.

16:55 PM

Monte Carlo and Thermal-Hydraulic Coupling via PVMEXEC

D.F.Gill, D.L.Aumiller, D.P.Griesheimer

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Successful high-fidelity coupling between a Monte Carlo neutron transport solver and a subchannel thermal-hydraulics solver has been achieved using PVMEXEC, a coupling framework developed for analysis of transient phenomenon in nuclear reactors. The PVMEXEC framework provides a generic program interface for exchanging data between solver kernels for different physical processes, such as radiation transport, heat conduction, and fluid flow. In this study, PVMEXEC was used to couple the in-house Monte Carlo radiation transport code, MC21, with a locally modified version of COBRA-TF. In this coupling scheme, MC21 is responsible for calculating three-dimensional power distributions and COBRA-TF for calculating local fluid temperatures and densities, as well as fuel temperatures. The coupled system was used to analyze 3D single-pin and assembly models based on the Calvert Cliffs commercial PWR. Convergence properties of the coupled simulations are examined and results are compared to simulations conducted using the existing integrated thermal feedback kernel in MC21.

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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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Criticality search is a very important aspect in reactor physics analysis. Due to the advantages of Monte Carlo method and the development of computer technologies, Monte Carlo criticality search is becoming more and more necessary and feasible. Existing Monte Carlo criticality search methods need large amount of individual criticality runs and may have unstable results because of the uncertainties of criticality results. In this paper, a new perturbation based Monte Carlo criticality search method is proposed and discussed. This method only needs one individual criticality calculation with perturbation tallies to estimate keff changing function using initial keff and differential coefficients results, and solves polynomial equations to get the criticality search results. The new perturbation based Monte Carlo criticality search method is implemented in Monte Carlo code RMC, and criticality search problems in density, enrichment and concentration are taken out. Results show that this method is quite inspiring in accuracy and efficiency, and has advantages compared with other criticality search methods.

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Monte Carlo Perturbation Method for Geometrical Uncertainty Analysis Using McCARD

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Monte Carlo geometrical uncertainty analysis method based on the Monte Carlo number density perturbation method is introduced. In the method, all the regions of a given system are divided into the perturbed region and non-perturbed regions. In the perturbed region, the MC number density perturbation calculations are performed. In order to examine the performance of the MC geometrical uncertainty analysis, the accuracy of σ(keff) estimated by the MC number density perturbation techniques are validated in Godiva, PWR fuel pin, and HMF-030 ICSBEP benchmark problem.

Track12-2 Fuel Cycle and Actinide Management

Session Chair: Alberto Talamo(ANL), Kazufumi Tsujimoto(JAEA)

15:55 PM
Thorium-Fueled Breed-and-Burn Fuel Cycle

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The feasibility to sustain a breed-and-burn mode of operation with thorium fuel in a sodium-cooled fast reactor is assessed. It is necessary to use fissile support to sustain the breed-and-burn mode with thorium based fuel due to the poor neutron economy of thorium. A thorium-fueled B&B core concept with low-enriched uranium (LEU) support is proposed and the performance characteristics of this core and the associated fuel cycle are assessed. The core is fueled with natural thorium and LEU. The thorium fuel is irradiated for seven cycles and is discharged with a burnup of ~38%. The LEU fuel is irradiated for only one cycle and discharged with a burnup of ~13%. About 72% of the energy is produced in the thorium fuel and the remaining 28% in the LEU fuel. The resource utilization is improved compared to light-water reactors but is not as good as that of conventional once-through sodium-cooled fast reactors. Other benefits typically observed in thorium-based fuel cycles, such as more favorable reactivity coefficients, have not been assessed as part of this study.

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

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This paper presents the analysis of a nuclear fuel cycle based on an intermediate spectrum water-cooled reactor with $^{232}Th/^{233}U$ oxide fuel. The objective of the analyzed fuel cycle is to attain equilibrium operation with only fertile ^{232}Th resource feed while using water reactor technology. The analysis is performed in the context of a large multi-laboratory effort to develop a compendium of fuel cycle performance data. Several potential reactor systems are identified in the literature and evaluated for deployment in the proposed fuel cycle. The investigation considers concepts where more than 50% of fissions occur in the intermediate energy regime, defined for the purposes of the study as 1 eV to 0.1 MeV. One configuration, a reduced moderation BWR proposed by UC Berkeley, meets the minimum criteria for discharge fissile inventory ratio when accounting for losses (>1.012). The natural resource required is 165.5 t of fertile feed per 100 GWe-yr, more than two orders of magnitude lower than the natural resource requirement for a typical light water reactor in a once through cycle. The Berkeley RBWR-Th concept has been identified as having potential issues due to lack of acceptable shutdown margin, so we also investigated an alternative D_2O-cooled PWR-like concept and showed that it has very similar fuel cycle performance. We also found that the neutronic performance of intermediate spectrum $^{232}Th/^{233}U$ fuelled reactors are highly sensitive to the ^{233}U radiative capture cross section in the intermediate energy range. This behavior had been previously observed in the literature for molten salt fast reactors. The discrepancy between different cross sections evaluations coupled with the lack of available measured data in this energy regime indicates that measurements of the ^{233}U radiative capture cross section would be desirable. Integral criticality experiments are also warranted for $^{232}Th/^{233}U$ systems with intermediate spectra.

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

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The production of small quantities of ^{232}U can induce radiation protection issues in the back end of the fuel cycle, particularly for thorium-based fuels. This is due to its relatively short half life (69 years) and the emission of a high energy gamma ray of 2.6 MeV at the end of its decay chain. With the depletion code MURE, we determine the different reactions pathways, and their proportions, leading to the synthesis of ^{232}U in UO_2 and $(Th,Pu)O_2$ fuels irradiated in a PWR. Moreover, the impact, on the ^{232}U production, of cycle times such as time separating the fabrication of the fuel and its irradiation as well as influence of the fissile content has been investigated for UO_2 fuel. The impact of the thorium ore provenance and of the plutonium quality has been studied for the $(Th,Pu)O_2$ case.

16:55 PM
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

This work is aimed to develop a fuel cycle scheme for thorium-uranium breeding recycle in CANDU reactors. According to the neutronics theory on the breeding of fissile nuclides in nuclear fuel system, the prerequisite of ^{232}Th-^{233}U breeding is revealed in CANDU fuel system and a roadmap of thorium-uranium breeding recycle in CANDU reactors is promoted. Following the roadmap, the thorium-based fuel cores were designed and the neutronics performance was evaluated. A CANDU-6 core with rated thermal power of 2061 MW was taken as the reference core. Based on the fuel management strategy, the spent-fuel characteristics for employed fuel types were assessed, including radiotoxicity, decay heat and gamma rays. Finally, the thorium-uranium breeding recycle schemes were established and evaluated in contrast with natural uranium once-through cycle, widely applied in operational CANDU reactors. The design and evaluation results indicated that the Th-U breeding recycles is technically feasible in current CANDU reactors.

17:15 PM
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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Inventory analyses of thermal-spectrum, thorium-fueled molten salt reactors (MSRs) have been performed to support US Department of Energy fuel cycle screening and evaluation activities within the. A single-fluid, single-zone 2250 MW_{th} (1000 MW_e) MSR concept with a fuel-bearing molten fluoride salt moderated by graphite was used as the basis for this work. Depletion calculations were performed using SCALE 6.1.1 with ENDF/B-VII.0 nuclear data. Equilibrium conditions were evaluated for several design parameter sets using a methodology developed at Oak Ridge National Laboratory (ORNL) that enables MSR analysis by performing multiple SCALE/TRITON depletion calculations with material flow modeling calculations between time steps. Adequate modeling approximations were identified by comparing results obtained from calculations that used different modeling choices and levels of fidelity. Parametric analyses examined the performance sensitivity of a thorium MSR to different separations approaches and elemental removal efficiencies. Finally, an inventory analysis for a thorium-fueled MSR with full recycling demonstrated how these insights can be applied and showed that such a system appears feasible from a mass flow and reactivity basis.

17:35 PM
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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A novel methodology for modeling the dynamic exchange of resources among actors in the CYCLUS fuel cycle simulator is described. The methodology is comprised of an information gathering step, a bipartite-graph-based supply-demand matching step, and a trade execution step. Its implementation in CYCLUS allows the simulator to model generic fuel cycles, i.e., those in which the types of facilities and possible resource flows is not known *a prior.* The flow of resources at each time step is algorithmically determined based on facilities' resource flow preferences. The dynamicsim of both the preference-based flows and quantity and quality constraints on those flows are demonstrated via two simple scenarios. The first scenario exemplifies situations in which actors' preferences change over time and the second typifies situations in which quality-based constraints limit resource flow. This generic capability provides a sufficient framework and basis for the CYCLUS simulation engine to model complex, advanced fuel cycles.

Track1-7 Reactor Analysis Methods

Session Chair: Eleodor Nichita(University of Ontario Institute of Technology), Chang Joon Jeong(KAERI)

8:00 AM
High Order Source Approximaion for the EFEN Method

Y.Li, L.Cao, X.Yuan
Xi'an Jiaotong University, Shaanxi, China

The flat source approximation in one dimensional Exponential Function Expansion Nodal (EFEN) method is extended to a high order polynomial approximation while maintaining the simplicity of the nodal response matrix. By applying the new method to a one dimensional PWR pin-by-pin problem, it has been observed that quadratic source approximation is good enough for PWR pin-by-pin calculation, while the flat source approximation causes about 5% of relative error to the thermal flux. By applying the new method to a one dimensional assembly homogenized problem, it has been found that the EFEN method with cubic source approximation can be employed to handle PWR core diffusion problems. Numerical results suggest the optimization of source approximation order for different energy groups and different spacial locations to achieve more accurate results with less computing effort.

8:20 AM
Extension of Linear Source MOC Methodology to Anisotropic Scattering in CASMO5

R.M.Ferrer, J.D.Rhodes
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The Linear Source (LS) spatial approximation, previously implemented in the Method of Characteristic (MOC) transport solver under the assumption of isotropic scattering, is extended to account for anisotropic scattering. The generalized formulation results in the representation of the angular moments of the flux as a linearly varying function within each source region and is fully compatible with Coarse-Mesh Finite Difference (CMFD) acceleration. The main set of equations for this anisotropic-source LS MOC method are derived in this work and numerical results are presented for critical assembly test cases. The results show improvement in the accuracy of the transport solution, but also an increase in run time and storage relative to the LS isotropic scattering implementation.

8:40 AM
Finite Difference Equations for Neutron Flux and Importance Distribution in a Heterogeneous Reactor without Homogenization and Diffusion Approximation

A.V.Elshin
Alexandrov Research Institute of Technology, Russia

This paper describes an application of the surface harmonics method to derivation of fewgroup finite difference equations for neutron flux distribution in a 3D square-lattice reactor model. The Boltzmann neutron transport equation is used as the original equation. No homogenization is used because the method first solves a system of finite difference equations (with a spatial step corresponding to a lattice pitch) and then constructs a detailed neutron flux distribution in the reactor cells with assumed approximations. Non-diffusion approximations apply to calculation of a whole reactor core if we increase the number of trial functions for describing the neutron flux distribution in each cell and the size of the matrices of the few-group coefficients for finite difference equations. Few-group finite difference equations are derived, which describe the neutron importance distribution (the multiplication factor in the homogeneous eigenvalue problem) in the reactor core. The derived finite difference equations remain adjoint to each other like the original equation of neutron transport and its adjoint equation. Difficulties and possibility of calculating the neutron flux distribution (trial functions) in cells of the reactor core model are discussed.

9:00 AM
Efficient Subspace Construction for Reduced Order Modeling in Reactor Analysis

B.A.Khuwaileh, C.Wang, Y.Bang, H.S.Abdel-Khalik
North Carolina State University, Raleigh, USA

Subspace-based reduced order modeling (ROM) is a powerful technique for reducing the computational burden required for the analysis of complex engineering models, such as nuclear reactor core calculations. It identifies a relatively small subspace that represents the variations for the quantities of interest with quantifiable user-defined accuracy. Focusing on neutron transport calculations employed in reactor analysis, the subspace is defined by a set of basis vectors that are extracted from the converged flux solution associated with a set of randomized forward model executions. In each execution, the input cross-sections are randomly perturbed, and the corresponding converged flux solution, referred to as snapshot, is recorded. This work mathematically proves and numerically demonstrates that the non-converged flux iterates can be used to approximate the reduction subspace without compromising the user-defined accuracy. It is shown that although this is expected to be bigger than (and inclusive of) the one based on converged flux snapshots, the computational savings resulting from the early termination of the iterative solution are significant even for situations where the convergence is slow, such as for systems with high dominance ratio (e.g., BWR). A quarter BWR fuel assembly was modeled and the variations of multiplication factor and neutron multi-group flux distribution were used to assess the adequacy of the proposed approach.

9:20 AM
Variational Acceleration of Fission Source Iteration for Subcritical Source-Driven Systems

B.Ozgener, H.A.Ozgener
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Numerical evaluation of the neutron flux distribution in slightly subcritical source-driven systems by means of multigroup diffusion theory is considered. It has been shown that the classical fission source iteration results in inordinately large number of iterations in such cases. To overcome the severe convergence problems met, a variational method, coarse mesh rebalance technique, is formulated, implemented and tested in a number of problems characterizing slightly subcritical systems. The numerical studies of such problems show that it is possible to accomplish superior convergence rates by utilizing the proposed variational technique. The method also has the potential for being applicable in space-time kinetics problems.

9:40 AM
An Incident Flux Coupling Calculation Study for Nodal Method and Monte Carlo Method

X.Wang, H.Yu, L.Xu, Y.Hu, X.Yang
China Institute of Atomic Energy, China

Conventional nodal method can not calculate fine distribution of intro-assembly neutron flux , especially for the assemblies with complex geometry and materials. Fine distribution of neutron flux and the other related parameters, however, are very useful for reactor core design and neutronic simulation in some special conditions, for example, simulation of neutron irradiation in test assemblies. A method named Incident Flux Nodal and Monte-carlo Coupling method (IFNMC), is studied and proposed to couple nodal transportation calculation with Monte-carlo calculation by using incident flux on the coupling interface. The IFNMC method, first use nodal transport method to do whole core neutronic calculation and get neutron flux distribution in full-core sacle and the incident interface flux for the specially identified assembly, and then use Monte-carlo method to simulate neutron behavior and get fine neutron flux distribuion within the special assembly with fine geometry description. By this way, fine distribution of neutron flux and other related information can be calculated quickly. According to the basic algorithm and the calculation flow of IFNMC, a computer code has been developed to achieve the coupling calculation. Two cases have been calculated and used to verify the method, one is fuel pin power distribution of China Experimental Fast Reactor (CEFR) and the other is radial distribution of ^{197}Au capture reaction rate in CEFR which is actually measured in CEFR physical startup test. The calculate results are consistent quit well with reference data, which preliminarily proves the prosed IFNMC method is effective and accurate, and is particularly suitable for fine distribution of neutron flux calculation in specially identified assembly.

Track4-3 Verification, Validation and Uncertainty Analysis

Session Chair: Nuria Garcia Herranz(Univ. Politecnica de Madrid), Go Chiba(Hokkaido Univ.)

8:00 AM
Confidence Interval Estimation by Bootstrap Method for Uncertainty Quantification Using Random Sampling Method

T.Endo, T.Watanabe, A.Yamamoto

Nagoya University, Nagoya, Japan

The random sampling method is simple and practical technique to quantify uncertainties of target core parameters calculated by a core analysis code. It is noted that statistical errors are inevitably involved in the estimated uncertainties, since the random sampling method is a probabilistic method using pseudorandom numbers. In the present study, the applicability of bootstrap method is investigated in order to estimate the confidence interval of the estimated uncertainty using the random sampling method. The bootstrap method enables to evaluate the histogram, variance, and confidence interval of sample estimates (e.g., mean, variance, median) by the resampling technique, i.e., a series of resampling are carried out using random sampling with replacement from an original data. In other words, limited number of samples can be used to estimate the confidence interval of uncertainty. Through a lattice burnup calculation for a simplified BWR fuel assembly, it is verified that the bootstrap confidence interval is a reasonable estimator to evaluate appropriate number of samples from the statistical point of view. If the distribution of target parameter is well approximated by a normal distribution, the confidence interval of uncertainty can be estimated by the conventional method based on the chi-squared distribution. The merit using the bootstrap method is to simply estimate the confidence interval without the assumption of normal distribution.

8:20 AM
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)

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

The Latin Hypercube Sampling (LHS) method is applied to the uncertainty quantification of neutronics characteristics based on the Monte-Carlo based sampling method. The Monte-Carlo based sampling method is one of the uncertainty quantification methods of neutronics characteristics (e.g. neutron multiplication factor) due to uncertainty cross section. The sampling method has the advantage that burnup and thermal-hydraulics feedback effects are easily considered in complicated light water reactor core analysis. Contrary, the sampling method has the disadvantage that statistical errors are involved in estimated uncertainties of neutronics characteristics since the sampling method is a probabilistic approach. Although the statistical errors can be reduced by increasing the number of samples (e.g. number of perturbed cross section libraries), computational cost also increases. Therefore development of an efficient uncertainty quantification method, which provides smaller statistical error of estimated uncertainty while reducing number of samples, is highly desirable. In the present study, adequacy and performance of the LHS method is investigated as an efficient Monte-Carlo based sampling method. Uncertainty quantification of multiplication factor for a BWR fuel assembly is carried out with the LHS method. The results indicate that uncertainty quantification using the LHS method is more efficient than that using the conventional sampling method, which utilizes simple random sampling of cross sections.

8:40 PM
Uncertainty Quantification of BWR Core Characteristics Using Latin Hypercube Sampling Method

A.Yamamoto

Nagoya University, Nagoya, Japan

Uncertainties of neutronics characteristics in a commercial BWR core due to cross section covariance are evaluated by the Latin Hypercube Sampling (LHS) method, which is a Monte-Carlo based efficient sampling algorithms. The Peach Bottom Unit 2, which is a BWR core, is a target for the uncertainty quantification in the present study. Thermal hydraulics feedback and burnup effects are fully and explicitly taken into account using a licensing-grade core simulator. Uncertainties for various core characteristics and correlations among them are evaluated by the statistical processing of core calculation results based on the LHS method. The calculation results indicate that uncertainty of critical eigenvalue (i.e., core reactivity) in the BWR core is comparable to that of typical PWR. On the other hand, uncertainties of assembly relative power distribution and maximum assembly burnup in the present BWR core are much smaller than those of typical PWR. The strong thermal hydraulics feedback effect of BWR would significantly contribute the difference of uncertainties in BWR and PWR.

9:00 AM
Applicability of the Cross Section Adjustment Method Based on Random Sampling Technique for Burnup Calculation

T.Watanabe, T.Endo, A.Yamamoto(1), Y.Kodama, Y.Ohoka, T. Ushio(2)

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

Applicability of the cross section adjustment method based on random sampling (RS) technique to burnup calculations is investigated. The cross section adjustment method is a technique for reduction of prediction uncertainties in reactor core analysis and has been widely applied to fast reactors. As a practical method, the cross section adjustment method based on RS technique is newly developed for application to light water reactors (LWRs). In this method, covariance among cross sections and neutronics parameters are statistically estimated by the RS technique and cross sections are adjusted without calculation of sensitivity coefficients of neutronics parameters, which are necessary in the conventional cross section adjustment method. Since sensitivity coefficients are not used, the RS-based method is expected to be practically applied to LWR core analysis, in which considerable computational costs are required for estimation of sensitivity coefficients. Through a simple pin-cell burnup calculation, applicability of the present method to burnup calculations is investigated. The calculation results indicate that the present method can adequately adjust cross sections including burnup characteristics.

9:20 AM
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)

1)Paul Scherrer Institut, Villigen Switzerland, 2)European Commission Joint Research Council, Ispra (VA), Italy

The “blackbox” approach of stochastic sampling (SS) methods for simultaneous nuclear data uncertainty quantification is powerful except it reveals little of the individual uncertainty contributions. In this work, the SS-based tool “NUSS” developed at PSI is updated to “NUSS-RF” which can estimate how much individual nuclear data uncertainty contributes to the total output uncertainty. The new capability is based on the Random Balance Design and FAST methods, both belonging to the so-called Global Sensitivity Analysis. First, the implementation of NUSS-RF is tested using a mathematical function, followed by the sensitivity and uncertainty analysis for ^{239}Pu(n,f), ^{235}U(n,f) and ^{238}U(n,γ) cross sections in Jezebel, Godiva and BWR pincell benchmarks respectively. The results are compared to the deterministic Sensitivity/ Uncertainty “Sandwich Rule” approach (which is local). For uncorrelated inputs, both methods have the equivalent interpretation of the input uncertainty contribution (in terms of variance fraction and Sensitivity Index), hence producing good agreement in the results. For correlated inputs, the discrepancy between the two methods broadens with the extent of the correlations which currently limits the validation of NUSS-RF to inputs with weak correlations.

Track4-3 Verification, Validation and Uncertainty Analysis

Session Chair: Nuria Garcia Herranz(Univ. Politecnica de Madrid), Go Chiba(Hokkaido Univ.)

9:40 AM

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

A.Hoefer, O.Buss, M.Hennebach, M.Schmid(1), D.Porsch(2)

1)AREVA GmbH Offenbach, Offenbach, Germany, 2)AREVA GmbH Erlangen, Erlangen, Germany

MOCABA is a combination of Monte Carlo sampling and Bayesian updating algorithms for the prediction of integral functions of nuclear data, such as reactor power distributions or neutron multiplication factors. Similarly to the established Generalized Linear Least Squares (GLLS) methodology, MOCABA offers the capability to utilize integral experimental data to reduce the prior uncertainty of integral observables. The MOCABA approach, however, does not involve any series expansions and, therefore, does not suffer from the breakdown of first-order perturbation theory for large nuclear data uncertainties. This is related to the fact that, in contrast to the GLLS method, the updating mechanism within MOCABA is applied directly to the integral observables without having to “adjust” any nuclear data. A central part of MOCABA is the nuclear data Monte Carlo program NUDUNA, which performs random sampling of nuclear data evaluations according to their covariance information and converts them into libraries for transport code systems like MCNP or SCALE. What is special about MOCABA is that it can be applied to any integral function of nuclear data, and any integral measurement can be taken into account to improve the prediction of an integral observable of interest. In this paper we present two example applications of the MOCABA framework: the prediction of the neutron multiplication factor of a water-moderated PWR fuel assembly based on 21 criticality safety benchmark experiments and the prediction of the power distribution within a toy model reactor containing 100 fuel assemblies.

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

Session Chair: Alireza Haghighat(Virginia Tech Univ.), Kazuya Yamaji(MHI)

8:00 AM
Discretized Mesh Tools and Related Treatment for Hybrid Transport Application with 3D Discrete Ordinates and Monte Carlo

K.L.Manalo, G.E.Sjoden, C.A.Edgar
Georgia Institute of Technology, Atlanta, USA

Hybrid methods of neutron transport have increased greatly in use, for example, in applications of using both Monte Carlo and deterministic transport methods to calculate quantities of interest, such as the flux and eigenvalue in a nuclear reactor. Many 3d parallel Sn codes apply a Cartesian mesh, and thus for nuclear reactors the representation of curved fuels (cylinder, sphere, etc.) are impacted in the representation of proper fuel inventory, resulting in both a deviation of mass and exact geometry in the computer model representation. In addition, we discuss auto-conversion techniques with our 3d Cartesian mesh generation tools to allowfor full generation of MCNP5 inputs (Cartesian mesh and Multigroup XS) from a basis PENTRAN Sn model. For a PWR assembly eigenvalue problem, we explore the errors associated with this Cartesian discrete mesh representation, and perform an analysis to calculate a slope parameter that relates the pcm to the percent areal/volumetric deviation (areal → 2d problems, volumetric → 3d problems). This paper analysis demonstrates a linear relationship between pcm change and areal/volumetric deviation using Multigroup MCNP on a PWR assembly compared to a reference exact combinatorial MCNP geometry calculation. For the same MCNP multigroup problems, we also characterize this linear relationship in discrete ordinates (3d PENTRAN). Finally, for 3D Sn models, we show an application of corner fractioning, a volume-weighted recovery of underrepresented target fuel mass that reduced pcm error to < 100, compared to reference Monte Carlo, in the application to a PWR assembly.

8:20 AM
A Novel Hybrid Weighting Scheme for Multi-Group Cross Section Collapsing

C.Yi, G.E.Sjoden, C.A.Edgar
Georgia Institute of Technology, Atlanta, USA

Multi-group cross section library generation plays an important role in deterministic transport simulations. In this paper, a new fine-group to broad-group cross section collapsing method is introduced. Rather than a traditional flux weighting, the new method uses a hybrid weighing scheme to collapse the scattering cross section matrix. Based upon a matrix analysis approach, we generalize different weighting schemes and derive the new hybrid weighting scheme, which mathematically shows that it is rational for the scattering cross section to be weighted by the (i) forward fluxes of the incoming/in-bound neutron groups and (ii) the adjoint functions of the outgoing/out-bound neutron energy groups. This approach also makes physical sense, since it conserves the “importance flow” of particles through scattering while collapsing cross sections. To conserve the reaction rates at the same time, we re-normalize the hybrid weighted scattering cross section to the original library total scattering reaction rate. We demonstrate that the hybrid weighting scheme is more accurate, especially for the detector response simulation problem in a Dual-Range Coincidence Counter (DRCC) 3-D S_N transport model.

8:40 PM
Development of an Iterative Lattice-Core Coupling Method Based on MICROX-2 Cross Section Libraries

J.Hou(1), H.Choi(2), K.Ivanov(1)
1)The Pennsylvania State University, PA, USA, 2)General Atomics, CA, USA

This paper describes an innovative online cross section generation method, developed based on Iterative Diffusion-Transport (IDT) calculation to minimize the inconsistency and inaccuracy in determining physics parameters by feeding actual reactor core conditions into the cross section generation process. A 2-dimensional (2-D) pin-by-pin lattice program, NEMA, was also developed to generate assembly lattice parameters using the embedded MICROX-2 cross section libraries and Nodal Expansion Method (NEM). The proposed methods were validated against a 2-D miniature core benchmark problem for both NEMA itself and its coupling to a reactor code by the IDT method (NEMA- DIF3D). The computational benchmark calculations have shown that the IDT improves the eigenvalue and power distribution predictions when compared with the conventional offline method.

9:00 AM
Solution of a Stylized European Pressurized Reactor (EPR) Benchmark Problem Using the Coarse Mesh Radiation Transport Method (COMET)

D.Lago, F.Rahnema, D.Zhang
Georgia Institute of Technology, Atlanta, USA

In this paper, as additional verification of its accuracy and efficiency,the coarse mesh radiation transport code COMET is compared to Monte Carlo solutions in a stylized benchmark problembased on the European Pressurized Reactor (EPR). The core specifications were taken directly from the Final Safety Analysis Report (FSAR) submitted to the Nuclear Regulatory Commission (NRC) and the reactor was modeled in a stylized manner while maintaining full heterogeneity at the pin and assembly level. Detailed results including assembly eigenvalues, core eigenvalues, and pin fission densities using a 2-group cross section library are presented. COMET results are in excellent agreement with MCNP with eigenvalue relative differences on the order of 10 pcm and average pin fission density relative differences on the order of 1-5%. Some maximum errors were on the order of 10% due to poor statistics on the periphery of the core in the reference results.

9:20 AM
Computational Efficiency and Accuracy of the Fission Collision Separation Method in 3D HTTR Benchmark Problems

D.Zhang, F.Rahnema
Georgia Institute of Technology, Georgia, USA

A fission collision separation method has been recently developed to significantly improve the computational efficiency of the COMET response coefficient generator. In this work, the accuracy and efficiency of the new response coefficient generation method is tested in 3D HTTR benchmark problems at both lattice and core levels. In lattice calculations, the surface-to-surface and fission density response coefficients computed by the new method are compared with those directly calculated by the Monte Carlo method. In whole core calculations, the eigenvalues and bundle/pin fission densities predicated by COMET based on the response coefficient libraries generated by the fission collision separation method are compared with those based on the interpolation method as well as the Monte Carlo reference solutions. These comparisons have shown that the new response coefficient generation method is significantly (about 3 times) faster than the interpolations method while its accuracy is close to that of the interpolation method.

9:40 AM
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.Haghighat
Virginia Tech, Arlington, USA

In this paper, we examine the use of the fission-matrix method to calculate the eigenvalue for a spent fuel pool. The fission matrix coefficients are calculated using MCNP5. In order to make the method as efficient as possible for real-time calculations, these coefficients are pre-calculated. Certain simplifying assumptions are made based on geometric considerations, greatly reducing the amount of pre-computation, and allowing the coefficients to be used on a variety of problems beyond that for which the coefficients were calculated. This methodology is applied to a reference pool that is being designed for the I2S-LWR project. Typically, the eigenvalue calculation in Monte Carlo is very difficult for loosely coupled problems such as a spent fuel pool due to source convergence issues, which are not present using the fission matrix. The fission matrix method has shown accuracy very close to that of MCNP5, while giving results in several orders of magnitude less time. Total pre-calculation time was less than a single eigenvalue calculation and can be used across many different pool configurations.

Track3-4 Monte Carlo Methods

Session Chair: Ho Jin Park(KAERI), Toshihiro Yamamoto(KURRI)

8:00 AM
XSBench - The Development and Verification of a Performance Abstraction for Monte Carlo Reactor Analysis

J.R.Tramm, A.R.Siegel(1), T.Islam, M.Schulz(2)

1)Argonne National Laboratory, Argonne, USA, 2)Lawrence Livermore National Laboratory, Livermore, USA

We isolate the most computationally expensive steps of a robust nuclear reactor core Monte Carlo particle transport simulation. The hot kernel is then abstracted into a simplified proxy application, designed to mimic the key performance characteristics of the full application. A series of performance verification tests and analyses are carried out to investigate the low-level performance parameters of both the simplified kernel and the full application. The kernel's performance profile is found to closely match that of the application, making it a convenient test bed for performance analyses on cutting edge platforms and experimental next-generation high performance computing architectures.

8:20 AM
Development of Neutron Current Connection Method for Whole Core Analysis Based on Monte Carlo Method

N.Nakadozono, K.Ishii, M.Aoyama, T.Mitsuyasu, T.Hino

Hitachi Research Laboratory, Hitachi, Ltd., Ibaraki, Japan

The whole core analysis with a Monte Carlo neutron transport method gives more accurate neutronics analysis of a heterogeneous core than the conventional two-stage core analysis method with lattice physics and the core analysis. The whole core Monte Carlo analysis, however, needs vast amounts of computational time and memory, so generally parallelization is necessary. In this study, the neutron current connection method is developed for the whole core Monte Carlo analysis to improve its parallelization efficiency. When the whole core analysis is spatially decomposed into many calculation nodes, this method connects nodes by the neutron current instead of using the information of each neutron. While this can reduce the amount of communication data between nodes, there is a discretization error when compressing the data into macro neutron currents and iteration of the calculation between the node analysis and update of boundary conditions is required. The validation of the method feasibility in view of this error is evaluated. This method is installed into the multi-group Monte Carlo code. Heterogeneous problems are evaluated and their solutions are compared with reference case solutions which calculates the whole problem by direct Monte Carlo calculation. By setting adequate division, after a sufficient number of current iterations, the calculated neutron infinite multiplication factors are agreed with reference cases within 0.1%dk and the root-mean-square differences of pin power distribution of each fuel assembly with reference cases are found to be within 1%.

8:40 PM
Analysis of Select BEAVRS PWR Benchmark Cycle 1 Results Using MC21 and OpenMC

D.J.Kelly, B.N.Aviles, P.K.Romano(1), B.R.Herman, N.E.Horelik, B. Forget(2)

1)Bechtel Marine Propulsion Corporation, New York, USA, 2) Massachusetts Institute of Technology, Massachusetts, USA

MC21 and OpenMC Monte Carlo results have been compared with hot zero power measurements from an operating pressurized water reactor (PWR), as specified in the recently released BEAVRS (Benchmark for Evaluation And Validation of Reactor Simulations) flux map data. Included in this comparison are axially-integrated full core detector measurements and axial detector profiles.
In addition, MC21 was used in the 3-D analysis of cycle 1 using integrated reactor feedback capabilities. A quarter core model was chosen for this study because of difficulties converging a full core model. This depletion encompassed quarter assembly temperature feedback with more than 200 fission products in more than 379,000 regions. An approximate power history was utilized to model the actual ragged power history present in cycle 1. Detailed analysis of in-core detector data was performed with MC21 at three different exposure levels. These also included comparisons with SIMULATE-3 in order to further validate the MC21 code. The MC21 results agreed well with experiment and were comparable to SIMULATE-3 indicating that these initial analysis methods in MC21 provide a highly accurate solution to the analysis of a 3-D PWR.

9:00 AM
Monte Carlo Neutronics Analysis of Sodium-Cooled Fast Reactor Benchmark with OTF Temperature and Burnup Treatment

N.S.Guilliard, W.Bernnat, J.Starflinger(1), W.Zwermann, I. Pasichnyk(2)

1)Universität Stuttgart - IKE, Stuttgart, Germany, 2)GRS - Garching, Germany

The sodium-cooled fast reactor design is one of six reactor technologies proposed by the Generation-IV International Forum for deployment during the next years. The concept combines advances in sustainability, safety and proliferation-resistance. Therefore four advanced fast reactor cores had been proposed by the Working Party on Reactor and System (WPRS). These cores are the base for neutronic analysis in a sodium-cooled fast reactor OECD Benchmark. Eleven international organizations investigate two large cores with 3600 MWth (oxide or carbide fuel) as well as two small cores generating 1000 MWth with either metallic or oxide fuel. Besides the k-effective value, reactivity feedbacks, isotopic composition at the end of equilibrium cycle (EOC) and power distributions were calculated. The benchmark partners used different code systems and different data libraries. The latter had been identified as main source for the discrepancies in the results. This paper shows selected parts of IKE results for the large oxide fueled core performed with MCNP6. The obtained IKE MCNP6 results are compared with the benchmark results. The large oxide core was analyzed with both the JEFF-3.1.2 and the ENDF/B-7.1 data set. Additionally, the k-effective was calculated with varying treatment of the Doppler broadening. The methods used were interpolation between two temperature dependent data sets, a polynomial representation (Fit-OTF) and the direct generation of data sets for the problem temperatures by the MAKXSF code. Detailed burnup calculations were done with pin-wise assignment of burnup dependent nuclide compositions in the hexagonal core lattice. The benchmark calculations were performed for constant coolant and fuel temperatures. Additionally, by external coupling of MCNP6 with the thermal-hydraulic code ATHLET realistic coolant and fuel temperatures were calculated to show the influence of thermal hydraulic coupling.

9:20 AM
Verification of Coupled 3D Fuel Cycle Analysis with Stable Monte Carlo Based Code, BGCore, against the Nodal Diffusion DYN3D Code

D.Kotlyar, E.Shwageraus(1), M.Margulis(2), Y.Bilodid, E.Fridman(3)

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Previous studies suggested that different schemes for coupling Monte Carlo (MC) neutron transport with burnup and thermal hydraulic feedbacks may potentially be numerically unstable. This issue can be resolved by application of new Stochastic Implicit Mid-Point (SIMP) methods. In order to assure numerical stability, the new methods do require additional computational effort. The instability issue however, is problem-dependent and does not necessarily occur in all cases. Therefore, blind application of the unconditionally stable coupling schemes, and thus incurring extra computational costs, may not always be necessary. In this paper, we attempt to develop an intelligent diagnostic mechanism, which will monitor numerical stability of the calculations and, if necessary, switch from simple and fast explicit coupling scheme to more computationally expensive but unconditionally stable one. To illustrate this diagnostic mechanism, we performed a coupled burnup and TH analysis of a single BWR fuel assembly. The reference solution was obtained by the state of the art nodal diffusion code – DYN3D. Very good agreement was observed in all neutronic and TH parameters. The results indicate that the developed algorithm can be easily implemented in any MC based code for monitoring of numerical instabilities. The proposed monitoring method has negligible impact on the calculation time even for realistic 3D multi-region full core calculations.

Track3-4 Monte Carlo Methods

Session Chair: Ho Jin Park(KAERI), Toshihiro Yamamoto(KURRI)

9:40 AM
Domain Decomposition and Terabyte Tallies with the OpenMC Monte Carlo Neutron Transport Code

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Memory limitations are a key obstacle to applying Monte Carlo neutron transport methods to high-fidelity full-core reactor analysis. Billions of unique regions are needed to carry out full-core depletion and fuel performance analyses, equating to terabytes of memory for isotopic abundances and tally scores - far more than can fit on a single computational node in modern architectures. This work introduces an implementation of domain decomposition that addresses this problem, demonstrating excellent scaling up to a 2.39TB mesh-tally distributed across 512 compute nodes running a full-core reactor benchmark on the Mira Blue Gene/Q supercomputer at Argonne National Laboratory.

Track9-2 Transient and Safety Analysis

Session Chair: Miriam Daeubler(KIT), Shigeaki Aoki(MNF)

8:00 AM
Recriticality Risk in PWR Spent Fuel Pools

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In this paper we investigated the situation in a PWR Spent Fuel Pool (SFP) following a long-term loss of power / loss of cooling accident. In the SFP there is a large amount of water with soluble boron between the fuel assemblies. There may be a problem from the point of view of criticality safety if the water of the SFP starts to boil and evaporate. A thermal-hydraulic analysis was performed using a simplified model of the SFP. The thermal-hydraulic analysis shows that in all cases a chaotic boiling phenomenon develops. This indicates that even if there is an issue of (near-)criticality, it will have a very intermittent nature. The multiplication factor of the SFP was evaluated with a Monte Carlo calculation. The neutronic analysis was performed for several representative cooling situations. In all cases, the system remains (deeply) subcritical.

8:20 AM
Demonstration of Fully Coupled Simplified Extended Station Black-Out Accident Simulation with RELAP-7

H.Zhao, H.Zhang, L.Zou, D.Andrs, R.Martineau

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The RELAP-7 code is the next generation nuclear reactor system safety analysis code being developed at the Idaho National Laboratory (INL). A number of physical components with simplified two phase flow capability have been developed to support the simplified boiling water reactor (BWR) extended station blackout (SBO) analyses. The demonstration case includes the major components for the primary system of a BWR, as well as the safety system components for the safety relief valve (SRV), the reactor core isolation cooling (RCIC) system, and the wet well. Three scenarios for the SBO simulations have been considered. Since RELAP-7 is not a severe accident analysis code, the simulation stops when fuel clad temperature reaches damage point. Scenario I represents an extreme station blackout accident without any external cooling and cooling water injection. The system pressure is controlled by automatically releasing steam through SRVs. Scenario II includes the RCIC system but without SRV. The RCIC system is fully coupled with the reactor primary system and all the major components are dynamically simulated. The third scenario includes both the RCIC system and the SRV to provide a more realistic simulation. This paper will describe the major models and discuss the results for the three scenarios. The RELAP-7 simulations for the three simplified SBO scenarios show the importance of dynamically simulating the SRVs, the RCIC system, and the wet well system to the reactor safety during extended SBO accidents.

8:40 AM
Code Scaling Applicability to a Cold Leg SBLOCA Scenario in a Nuclear Power Plant

A.Querol, S.Gallardo, G.Verdú

Universitat Politècnica de València, Valencia, Spain

The knowledge of thermalhydraulic phenomena occurring in Nuclear Power Plants (NPPs) during an accident is crucial in the assessment of nuclear safety. Several experimental facilities have been built to reproduce some accidental scenarios obtaining measured data to be compared with simulation results and to test the capability of the thermalhydraulic codes. This paper presents the scaling method used to obtain a NPP scaled-up TRACE5 input from a Large Scale Test Facility (LSTF) model to reproduce a cold leg Small Break Loss-of-Coolant Accident (SBLOCA) transient. The scaled-up TRACE5 model has been developed conserving the power-to-volume scaling ratios of LSTF components, initial and boundary conditions. A comparison of the simulation results between LSTF and NPP scaled-up TRACE5 models is provided throughout different graphs, which represent the main thermalhydraulic variables. Results show that the main physical phenomena produced during a cold leg SBLOCA are qualitatively reproduced with the NPP scaled TRACE5 model.

9:00 AM
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

Uppsala University, Uppsala, Sweden

A common problem for many liquid metal cooled fast reactor designs is the positive void worth of the coolant. In this context, an advantage of lead cooled fast reactors is the high temperature of coolant boiling. In contrast to sodium cooled fast reactors this, in practice, precludes coolant boiling. However, partial voiding of the core could result from e.g. gas bubbles entering the core from below. This would introduce a positive reactivity, and if the bubble is large enough. In this paper we model this type of event using a point kinetics code coupled to a heat transport code. The reactivity parameters are obtained from a Monte Carlo code. The 300 MWth reactor design Alfred is used as a test case. We show that in general the reactor design studied is robust in such events, and we conclude that small bubbles a measureable Power oscillation would occur. For very large bubbles there exist a possibility of core damage. The cladding is the most sensitive part.

9:20 AM
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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In this work the relevance of MOX fuel bearing minor actinides (MA) on the behavior of a SFR reactor under an unprotected transient overpower (UTOP) event has been studied. An updated version of the SAS-SFR code able to analyze the initial phase of accident scenarios for sodium-cooled fast reactors fuelled with innovative oxide mixtures $(U,Pu,Np,Am,Cm)O_2$ has been used. We have considered the effect of MA on reactor kinetics (i.e. reactivity coefficients, power, and kinetic parameters), as well as on fuel thermal and mechanical properties (i.e. thermal conductivity, heat capacity, thermal expansion and melting temperature). The core design used in this work is the so-called Reference Oxide core used in the CP-ESFR project at End Of Equilibrium Cycle.

9:40 AM
Step towards Integral Validation of Energetic Re-Criticality Prediction for Sodium Cooled Fast Reactor

T.Ivanova, E.Ivanov

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A specific feature of sodium cooled fast reactors is the energetic re-criticality risk resulting from relocation of core materials during a hypothetical core disruptive accident. It may lead to a severe nuclear power excursion. The risk of mechanical energy release during the accident is determined by whether the prompt criticality is reached or not, or in other words, whether the re-criticality is predicted with accuracy under delayed neutron fraction ($ß_{eff}$) value or not. The released energy depends on reactivity excess above $ß_{eff}$ value. Accordingly, for safety studies, the k_{eff} uncertainty should be well established through accident evolution. A large portion of this uncertainty originates from nuclear data. This paper presents work on evaluation of nuclear data uncertainties for the selected snapshots of unprotected loss of flow accident modeled for a large sodium fast reactor and propagation of this uncertainty through the transient process.

Track1-8 Reactor Analysis Methods

Session Chair: Youqi Zheng(Xi'an Jiaotong University), Tatsuya Iwamoto(GNF-J)

10:20 AM
Modernization Enhancements in SCALE 6.2

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

Oak Ridge National Laboratory, Oak Ridge, USA

SCALE is a widely used suite of tools for nuclear systems modeling and simulation that provides comprehensive, verified and validated, user-friendly capabilities for criticality safety, reactor physics, radiation shielding, and sensitivity and uncertainty analysis. For more than 30 years, regulators, industry, and research institutions around the world have used SCALE for nuclear safety analysis and design. However, the underlying architecture of SCALE is based on a 40-year-old design with dozens of independent functional modules and control programs, primarily implemented in the Fortran programming language, with extensive use of customized intermediate files to control the logical flow of the analysis. Data are passed between individual computational codes using custom binary files that are read from and written to the hard disk. The SCALE modernization plan provides a progression towards SCALE 7, which will provide an object-oriented, parallel-enabled software infrastructure with state-of-the-art methods implemented as reusable components. This paper provides a brief overview of the goals of SCALE modernization and details some modernized features available with SCALE 6.2.

10:40 AM
SALOME-CORE Platform: Uses for EDF R&D Neutronic Studies

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

EDF R&D, Clamart, France

EDF R&D is currently developing a new, state of the art, calculation chain, ANDROMEDE, which relies on APOLLO-2 for assembly calculation and on the in-house core code COCAGNE. Engineers using these new neutronic tools have generic needs in terms of numerical simulation: pre-processing, post-processing, physics coupling, uncertainty propagation, data assimilation, calculation supervision and distribution,... These needs, shared with other general fields (solid mechanics, fluid dynamics, thermalhydraulics) are not covered by the perimeter of APOLLO-2 nor COCAGNE. In order to mutualize the development effort in those generic modules, and to provide a common framework for all their numerical simulation engineers, EDF and the CEA co-developed the open-source SALOME platform, a toolbox with generic modules for numerical simulation. This paper presents the SALOME-CORE platform, which packages the core code COCAGNE and the SALOME platform, and also uses of this platform for reactor physics studies: meshing of assemblies for APOLLO-2 calculation, data assimilation for the computation of an appropriate reflector model, calculation supervision for the optimization of energy meshes for SPn or Sn multigroup calculations, calculation distribution on supercomputers, efficient postprocessing and visualization of the results.

11:00 AM
A Steady-State Core Analysis Code for the Modeling of Accelerator-Driven Subcritical Reactors

S.Zhou, H.Wu, L.Cao, Y.Zheng

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In order to analyze and evaluate Accelerator Driven Subcritical Reactors (ADSR) efficiently, a neutronics/thermal-hydraulics coupling analysis code named LAVENDER has been developed. In the neutronics calculation, the three dimensional deterministic neutron transport method is adopted. The nuclides transmutation analysis is implemented by the micro-depletion method with the Transmutation Trajectory Analysis algorithm (TTA). In the thermal-hydraulics calculation, a heat transfer model is established to consider thermal feedback and examine thermal-hydraulics design. The validations are performed based on several benchmarks. Numerical results indicate that LAVENDER is reliable and efficient to be applied for the design and steady-state analysis of ADSR.

11:20 AM
Good Practice in Development of Advanced Assembly/Core Calculation Methods and Implementations of AEGIS/SCOPE2

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This paper reviews the history of development of AEGIS/SCOPE2, an advanced in-core fuel management code for PWRs. The initial project, development of a proto-type code, was started in 1996 as a feasibility study of the advanced calculation method/algorithm for advanced computation environments such as distributed parallel computers like PC-clusters which are commonly used nowadays. With success of development of the prototype code, a production-level advanced core calculation code, SCOPE2, was developed followed by AEGIS, an advanced assembly calculation code. These codes have been developed on the basis of the object-oriented programming approach and the agile software development. The authors extracted the key factors for success of the project as good practices from the viewpoint of code design, implementation, project management and verification and validation. Those practices are universal and may be applicable to any projects in the future.

11:40 AM
Validation of LANCR01/AETNA01 BWR Code Package against FUBILA MOX Experiments and Fukushima Daiichi Nuclear Power Plant Unit 3 MOX Core

T.Iwamoto, T.Ikehara, M.Ono, T.Yamana, H.Suzuki

Global Nuclear Fuel-Japan, Yokosuka, Japan

LANCR01 assembly code and AETNA01 core simulator are the advanced BWR package developed by Global Nuclear Fuel (GNF). In order to establish the applicability of the package to the plutonium containing mixed oxide (MOX) fuel, validation tests have been conducted against BASALA, FUBILA MOX critical experiments and the operational data from Fukushima Daiichi Nuclear Power Plant Unit 3 (1F3) with MOX assemblies loaded at cycle 25. For BASALA and FUBILA, critical eigenvalue and the pin-by-pin fission rate distribution by LANCR01 were compared with the experimental data. For 1F3, AETNA01 predictions with LANCR01 assembly cross sections were compared with the measured control blade worth and the moderator temperature coefficient in the reactor physics tests, as well as the cold/hot eigenvalues and the in-core instrument readings during the operation. It is concluded that LANCR01/AETNA01 system has a comparable accuracy for the MOX cores with that for the uranium cores.

Track4-4 Verification, Validation and Uncertainty Analysis

Session Chair: Oscar Cabellos(Univ. Politecnica de Madrid), Hiroki Iwamoto(JAEA)

10:20 AM

Pinpower Uncertainty Quantification of LWR-PROTEUS Phase III Experiments

M.Hursin, M.Scriven, G.Perret, P.Grimm, A.Pautz

Paul Scherrer Institut, Villigen, Switzerland

CASMO-5MX is a set of Perl-based tools built around the lattice code CASMO-5 and developed at the Paul Scherrer Institut, used to perform sensitivity analysis (SA) and uncertainty quantification (UQ). The paper describes the recent developments to the CASMO-5MX methodology to perform pinpower UQ with the Stochastic Sampling methodology while perturbing nuclear data and technological parameters.
The CASMO-5MX SS pinpower UQ is applied to the re-analysis of the LWR-PROTEUS Phase III campaign which featured SVEA-96 Optima2 fresh BWR fuel assemblies in an attempt to explain some discrepancy between the fission rate measurements (E) and their associated predictions with CASMO-5M (C).
The pinpower UQ results have shown that the pinpower uncertainty due to nuclear data and technological parameters is in the range of 0.4% to 0.8%. This additional source of uncertainty, added to the existing experimental uncertainty, do not fully explain the remaining few C/E discrepancies in the pinpower map. The magnitude of nuclear data and technological parameters uncertainty is either underestimated or an additional source of uncertainty has not been taken into account.

10:40 AM

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

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

Paul Scherrer Institut, Villigen, Switzerland

A study of BWR Void Reactivity Coefficient (VRC) calculations with state-of-the-art lattice physics codes is presented in this paper. Although the study is performed with the CASMO code, since used at PSI for analyses of the Swiss reactors, a first objective is to quantify the VRC range of variation that can be expected from code/library modifications. A second objective is to start assessing the VRC uncertainty related to nuclear data in modern lattice physics codes. To these aims, a benchmark case is presented based on a representative BWR lattice design and described in details in order to be readily available for further assessments and comparisons with other lattice codes and/or libraries. The VRC behavior in the 0%-100% void range and at low/high burnups is then studied as function of CASMO code/library updates in general and of a transition from CASMO-4 to it advanced successor CASMO-5 in particular. Through this, a quantitative assessment of the VRC range of variation resulting among other things from a much finer group structure and from updates to the most recent neutron data evaluations is made. Qualitatively, distinct trends are observed but mainly in the high-to-very-high void range and especially with CASMO-4 combined with the oldest libraries. To understand this, a reactivity breakdown of the VRC is performed, indicating that these trends are driven primarily by a non-linear ^{238}U capture reaction rate behavior in the epithermal to fast range. At higher burnups, compensating effects from other nuclide/reactions are pointed out. Finally, the methodology under development at PSI for nuclear data uncertainty propagation with CASMO-5 is applied. Although only cross-section uncertainties based on the SCALE-6 44-group variance-covariance matrices are at this stage considered, the VRC uncertainty is in this manner estimated and compared to the range of variation due to code/library updates.

11:00 AM

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

H.Iwamoto, M.Maier, K.Nishihara

Japan Atomic Energy Agency, Ibaraki, Japan

A burnup calculation is carried out for an accelerator-driven system (ADS) proposed by the Japan Atomic Energy Agency (JAEA) with the fourth version of JENDL, JENDL-4.0 and the previous one, JENDL-3.3. Considerable differences are seen in burnup reactivity between the nuclear data libraries for an initial phase (first burnup cycle) and an equilibrium phase (tenth burnup cycle). The differences in these values are investigated using two methods: a method by replacing a nuclear data library by nuclide and a sensitivity analysis technique. Among many contributors to them for the both phases, we identify major ones; (i) the initial phase: fission cross section and fission neutron multiplicity of ^{238}Pu, capture cross section of ^{241}Am, and (ii) the equilibrium phase: capture cross section of ^{244}Cm and ^{241}Am, and inelastic scattering cross section of $^{206;207}$Pb. The uncertainty analysis shows that uncertainties in the bunrup reactivity deduced from the JENDL-4.0 covariance data are comparable in magnitude to the differences between the nuclear data libraries, and are dominated by nuclear data parameters of ^{238}Pu. Finally, we show the necessity of uncertainty evaluation of the branching ratio of ^{241}Am capture reaction.

11:20 AM

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

N.Garcia-Herranz, O.Cabellos

Universidad Politècnica de Madrid, Madrid, Spain

In order to evaluate the uncertainties in prediction of lattice-averaged parameters, input data of core neutronics codes, Exercise I-2 of the OECD benchmark for uncertainty analysis in modeling (UAM) was proposed. This work aims to perform a sensitivity/uncertainty analysis of the BWR configurations defined in the benchmark for the purpose of Exercise I-2. Criticality calculations are done for a 7x7 BWR fresh fuel assembly at HFP in four configurations: single unrodded fuel assembly, rodded fuel assembly, assembly/reflector and assembly in a color-set. The SCALE6.1 code package is used to propagate cross section covariance data through lattice physics calculations to both k-effective and two-group assembly-homogenized cross sections uncertainties. Computed sensitivities and uncertainties for all configurations are analyzed and compared. It was found that uncertainties are very similar for the four test-problems, showing that the influence of the assembly environment on uncertainty prediction is very small.

11:40 AM

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

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The prediction of fission products and the impact of their uncertainties to different safety-related spent fuel applications (burn-up credit, decay heat generation, radiological safety, waste management, burn-up prediction) are required for the evaluation of spent fuel system designs and safety analysis options. One of the nuclear data needs to this prediction is the independent fission yields. The mostly used general-purpose evaluated nuclear data libraries provide these data including their uncertainties as standard deviation, with no-correlation between fission yields. However, new developments in the theory and measurements of fission product yields are expected to result in new evaluated files in the next coming years. These files will include considerably more accurate yields including neutron energy dependence combined with new covariance information allowing realistic uncertainty estimates. In this paper, we focused on the effect of fission yield covariance information on criticality and depletion calculations. A LWR pin-cell burn-up benchmark, proposed in the general framework of the OECD/UAM Benchmark is analyzed to address the impact of independent fission yield uncertainties. Calculations were performed with the SCALE6 system and the ENDF/B-VII.1 fission yield data library, adding covariance data obtained from including covariance info of mass yields. Results are compared with those obtained with the uncertainty data currently provided by ENDF/B-VII.1. The uncertainty quantification is performed with a Monte Carlo sampling and then compared with linear perturbation.

Track8-1 Reactor Operation and Safety

Session Chair: Gerald Rimpault(CEA), Hiroshi Akie(JAEA)

10:20 AM

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

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1)ENEA, "Casaccia" Research Center, Rome, Italy, 2)"Sapienza" University, Rome, Italy

The OECD/NEA proposes a new international benchmark based on the data collected from an instability transient occurred at the Oskarshamn-2 NPP with the aim to test the coupled 3D Neutron Kinetic/Thermal Hydraulic codes on challenging situations. The ENEA "Casaccia" Research Center is participating to this benchmark, developing a computational model using RELAP5-3D© code. The 3DNK model was developed starting from the cross sections datasets calculated by OKG, the Oskarshamn-2 licensee, using the CASMO lattice code. Integration of neutron cross sections database in RELAP5-3D© required data fitting by a n-dimensional polynomials, calculations of the various polynomial coefficients and of the base cross sections values. An ad-hoc tool named PROMETHEUS has been developed for automatically generate the RELAP5-3D©-compatible cross sections libraries. Thanks to this software it has been easily possible to visualize the complex structure of the neutronic data sets and to derive different cross sections libraries for evaluating the effects of some neutronic parameters on the prediction of the reactor instability. Thus, the effects of the fuel temperature and control rod history, of the discontinuity factors (averaged/not averaged), and of the neutron poisons has been assessed. A ranking table has been produced, demonstrating the relevance of the not-averaged discontinuity factors and of the on-transient neutron poisons calculations for the correct prediction of the Oskarshamn-2 event.

10:40 AM

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

Y.Kuroda, Y.Miki, K.Nozaki, A.Fukao(1), N.Yanagisawa(2), T.Makigami(3)

1)TEPCO SYSTEMS CORPORATION, Tokyo, Japan, 2)Electric Power Development Co.,Ltd., Tokyo, Japan, 3)Tokyo Electric Power Company, Tokyo, Japan

Boiling water reactor (BWR) cores with various MOX fuel loading ratios (i.e., ratios of the number of MOX fuel assemblies to the total number of fuel assemblies in the core) were operated for many cycles at two commercial BWRs in Germany. The evaluation of operational experiences and reactor physics tests of those two plants is useful for enhancing the reliability of in-core fuel management for MOX loaded cores. The evaluation of those experiences of MOX loaded cores has been performed using CASMO-4 and SIMULATE-3 in order to confirm that the evaluation results with various MOX fuel loading ratios are equivalent to those of UO_2 loaded cores from the point of view of in-core fuel management. This paper introduces the evaluation results of the critical eigenvalue in hot condition (i.e., rated and partial power condition), the traversing in-core probe (TIP) response, the cold critical eigenvalue, the differential control rod worth, and the moderator temperature coefficient. The MOX fuel loading ratio has no significant effect to those parameters in the evaluation results. Hence, it has been confirmed that the in-core fuel management for MOX loaded cores can be performed equally with that for UO_2 loaded cores.

11:00 AM

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

A.Sargeni(1), K.W.Burn(2), G.B.Bruna(1)

1)IRSN, Fontenay-aux-Roses, France, 2)ENEA, Bologna, Italy

The paper is intended to provide a simple but general overview of the problem of large-size nuclear reactor coupling, through an application that demonstrates the sensitivity of the system perturbations to the eigenvalue dominance ratio. As an example, we have investigated the sensitivity of the core radial power distribution to the nature of the reflector (conventional vs. a "heavy" stainless-steel one), by applying a perturbation to some peripheral assemblies of an EPR-like reactor system (as defined in the UAM GEN-III benchmark, a fresh fuel core). The perturbation was selected to simulate possible assembly bowing effects, engendering a variation of the inter-assembly water gap, which can generate an azimuthal asymmetry in the power distribution. This is conventionally referred to as a 'power tilt'. The amplification effect of the reflector on the power tilt has been addressed in two steps. Firstly, we have compared the power tilt, computed with the deterministic code CRONOS2 (the neutronics diffusion module of the HEMERA chain) and with the probabilistic transport code MCNP, to validate the CRONOS2 response against a Monte-Carlo reference and obtain a quantification of the amplification effect at zero power. Then the feedbacks have been taken into account in the diffusion calculations to evaluate the sensitivity of the tilt to the power level. The explanation of the difference in the core behavior in the presence of either a conventional or a heavy reflector has been investigated through the examination of the power distributions and the eigenvalue splitting of the two systems. The results explain the higher sensitivity to a peripheral perturbation of a heavy-reflector-core compared to a conventionally- reflected one.

11:20 AM

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

F.Kai, L.Fu, Z.Xuhua, G.Jiong

Tsinghua University, Beijing, China

Modular high temperature gas cooled reactor HTR-PM demonstration plant, designed by INET, Tsinghua University, is being built in Shidaowan, Shandong province, China. HTR-PM adopts pebble bed concept. Like other HTR, to measure or monitor the core power distribution is very important but challenging for HTR-PM, as there is no in-core neutron detectors. There are some proposals to reconstruct the power distribution from the readings of ex-core neutron detectors. One method is harmonic synthesis method. In this paper, improved harmonic synthesis method is proposed and tested to increase the accuracy of the power distribution reconstruction, by using better fundamental harmonic according to different reactor operation conditions. The numerical result shows the improvement in the reconstruction accuracy, contrasting to harmonics' coefficients polynomial expansion method.

Track6-1 Reactor Physics Experiments

Session Chair: Patrick Blaise(CEA),Takuya Umano(Toshiba)

10:20 AM
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)

1)SCK-CEN, Belgian Nuclear Research Centre, Boeretang, Belgium, 2) Delft University of Technology, The Netherlands, 3)CNRS-IN2P3/UJF/INPG, France, 4)CEA/DEN/DER/SPEX/LPE Cadarache, France, 5) ENSICAEN/Université de Caen, CNRS-IN2P3, France, 6)CNRS-IN2P3/Université Paris Sud, Orsay, France, 7)ENEA, Rome, Italy, 8)CIEMAT, Madrid, Spain

With Accelerator-Driven Systems (ADS), the incineration of large quantities of minor actinides is envisaged by coupling a particle accelerator to a subcritical core. Despite the subcritical operation of the core, a monitoring tool is indispensable in an ADS for both safety and operational purposes. Several experimental techniques are under investigation today within the European FP7 project FREYA (Fast Reactor Experiments for hYbrid Applications) at the VENUS facility at SCK-CEN, where the lead fast zero-power VENUS-F core is coupled to the GENEPI-3C deuteron accelerator. The proposed monitoring concept combines an initial reactivity determination, e.g. by the pulsed neutron source technique, a long-term relative reactivity monitor, by the current-to-flux (CTF) technique, and an interim cross-checking, by the source jerk technique. In this paper the CTF technique is presented as a candidate for continuous reactivity monitoring. Static modal analysis is applied to evaluate the point kinetics relationship between accelerator source neutrons and the neutron flux. As a first test case, the zero-power VENUS-F SC1 core is considered. Spatial correction on the CTF monitor is evaluated by measurement results for the control rods worth at different detector positions. Good agreement between the modal and experimental spatial correction factor (SCF) is found with 9 modes. Limited SCF are found in the VENUS-F reflector, far away from the perturbation. Modal analysis is envisaged for the monitoring of burn-up in demonstrator ADS like MYRRHA.

10:40 AM
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.Cathalau, O.Litaize, A.Santamarina, C.Vaglio, JF.Vidal, N.Thiollay, N.Huot, JP.Chauvin, T.Pont, V.Laval, S.Testanière, B.Vincent

CEA Cadarache, Saint Paul-lez-Durance, France

In France, recycling was decided in 1985 and the first mixed oxide (MOX) reload was introduced in the Saint Laurent B1 PWR French EdF Nuclear Power Plant in 1986. In order to validate the current calculation schemes and to reduce the uncertainties, exhaustive experimental programs were launched in the French EOLE zero power critical facility. The first programs were focused on physical properties of 30%MOX recycling in French PWRs. Later, specific studies progressed in Japan and in France in the mid 90's to verify the possibility of introducing high moderation 100%MOX reloads in Light Water Reactors. These studies were performed in the framework of development of Advanced LWR concepts (ALWRs) which include BWRs and PWRs. They relied on current calculation schemes that require additional validation in order to achieve similar accuracies for high moderation 100%MOX cores as it had already been obtained for UO_2 and 30%MOX recycling in P- and BWRs. The paper details the 3 experimental programs that have been conducted in the French EOLE critical facility of the Cadarache Research Centre between 1995 and 2007, and the associated experimental techniques applied to reach target accuracies. These programs are :

MISTRAL: *(**M**OX **I**nvestigation of **S**ystems which are **T**echnically **R**elevant of **A**dvanced **L**ight water reactors) for 100%MOX investigation in overmoderated homogeneous and 17x17 PWR mock-up lattices*

BASALA: *(**B**oiling water reactor **A**dvanced core physics **S**tudy **A**imed at mox fuel **LA**ttice) for 100%MOX investigation in 9x9 Advanced BWR mock-up lattices of various moderation ratios*

FUBILA: *(**FU**ll mox core physics experiments of **B**wr **I**nitiated for **L**attice **A**nalysis method verification and improvement) for 100%MOX investigation in High Burn 9x9 and 10x10 BWR mock-up lattices with void increase.*

After a description of those main experimental programs performed in EOLE, an overview of their feedback on experimental uncertainty improvement and code V&V is given.

11:00 AM
Application of the Best Representativity Method to a Future PWR Fuel Assembly Calculation Using Four Critical Experiments of Different Facilities

T.Umano, K.Yoshioka, M.Yamaoka(1), T.Obara(2)

1)Toshiba Corporation Power Systems Company, Kawasaki, Japan, 2) Tokyo Institute of Technology, Tokyo, Japan

Since the inception of the nuclear industry, for the qualification and the validation of the nuclear calculation codes, precise measurement data from the critical experiments are both necessary and profitable. To judge the experiment similarities to the objective of the actual reactor conditions or actual reactor equipment, the concept of a "representativity factor" has recently been adopted in the critical experiment field, particularly for fast breeder reactors (FBRs) and future reactor studies. Based on this concept, a new numerical evaluation method has been developed to improve the representativity of critical experiments for the target reactor conditions. With this method, it is also possible to evaluate more reliable value of a physical property of the target system under combining two or more critical experiments. As an industrial application, this new method was utilized to correct the infinite neutron multiplication factor (k-infinity) of a future pressurized water reactor (PWR) fuel assembly. This PWR 17×17 type fuel assembly has 6.0 wt% of ^{235}U enrichment and 24 UO_2-Gd_2O_3 rods. Based on the method, four kinds of critical experiments were combined. Two of them were the critical experiments for PWR simulations at the Toshiba NCA facility. Other two were due to ICSBEP 2010 data disk. The SCALE 6.1.2 system was utilized for performing calculations. After combining four critical experiments, the representativity factor became larger and the similarity was improved. Also k-infinity of the PWR fuel assembly was corrected. In this study, the calculation procedures are explained. After that, calculation results are shown with physical explanations and discussions.

11:20 AM
Characterization of Irradiation Fields for Fuel and Material Irradiation in the Experimental Fast Reactor Joyo

S.Maeda, H.Saito, H.Naito, C.Ito, T.Sekine

Japan Atomic Energy Agency, Ibaraki, Japan

The Joyo MK-III core is a worldwide fast neutron irradiation field not only for FBR development but also for use in other fields such as light water reactor (LWR) and fusion reactor studies, and in the non-nuclear industry. The characterization of these neutron and gamma ray fields is most important to utilize for irradiation tests. This paper describes the details of distributions of neutron flux, reaction rate and gamma heating in the MK-III core. The calculation accuracy of the core management codes HESTIA, TORT and MCNP, was also evaluated by the measured data. The calculated results in neutron calculation agreed well with the measured one. The calculation method was validated and correction factors were identified. In case of gamma heating evaluation, the calculated result is underestimated with respect to the experimental value especially in the upper and lower SS reflector region. Further investigations in gamma heating evaluation are needed.

11:40 AM
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

Los Alamos National Laboratory, Los Alamos, USA

A critical experiment was performed on the Comet assembly to provide nuclear data in a non-thermal neutron spectrum and to re-establish experimental measurement capabilities relevant to the

Track6-1 Reactor Physics Experiments

Session Chair: Patrick Blaise(CEA),Takuya Umano(Toshiba)

United States Department of Energy's general purpose nuclear criticality experiments capability and to the Technical Nuclear Forensic program. Activation and fission foils were placed at specific locations in the Zeus all-oralloy core, copper reflected critical experiment to infer spectral indices data and obtain reaction rate data. After the irradiation, passive gamma ray measurements were performed on all the foils and several of them were packaged and shipped to Los Alamos National Laboratory for further radiochemical analysis. The results from the non-destructive and radiochemical analyses are presented. Finally, Rossi-α measurements were performed on a slightly modified configuration from the configuration used for the activation measurements. The Rossi-α results are presented and compared to past measurements performed using other critical assemblies.

Track9-3 Transient and Safety Analysis

Session Chair: Silva Kalcheva(SCK/CEN), Mikio Tokashiki(NFI)

10:20 AM
SIMMER-III Modeling of Gas Cooled Fast Reactor

X.N.Chen, L.Andriolo, A.Rineiski, W.Maschek

Karlsruhe Institute of Technology (KIT), Karlsruhe, Germany

This paper deals with extension and application of the SIMMER-III code for safety studies of a gas cooled fast reactor. The equation of state of the helium gas and its thermal physical properties have been prepared and implemented in the code. The geometric, thermal hydraulic and neutronic models have been set up for the ALLEGRO reactor. The code and the associated model are verified by comparing steady state and unprotected loss of flow with 20% remained flow rate (ULOF-20%) results with those done by other project partners. Reasonable or good agreements have been achieved for major physical variables.
The unprotected loss of coolant accident (ULOCA) case is a severe transient case with core melting and degradation that was simulated only by SIMMER in the project. In the initiating phase the clad becomes molten; this triggers the first power excursion. Then the fuel becomes more mobile and further power excursions take place, which lead to core melting and degradation. The fuel is ejected by power excursion and then moves relatively slowly to the lower part of vessel. Finally there are only a few kilograms of fuel escaping to the vessel outside (into reactor container) and the released thermal energy is about 6 GJ within a period of one minute. The final power stays below one MW and the reactor is in a deep sub-criticality state, since 1/2 fuel becomes noneffective.

10:40 AM
Validation of the Subchannel Code CTF against the Benchmark Data of the OECD/NEA PSBT

A.Abarca, R.Miró, P.Hidalga, T.Barrachina, G.Verdú

Universitat Politècnica de València (UPV), Valencia, Spain

Nuclear reactor safety analysis has been increased rapidly in the last decades. Due to this, advanced methods for core phenomena predictions are developed, specially related to thermal-hydraulics using subchannel codes. CTF is a version of the subchannel code COBRA-TF improved by the ISIRYM. This code must be validated against full-scale high-quality experimental data, which allow comparing results for several case types. The benchmark of the OECD/NEA PWR Subchannel and Bundle Test (PSBT) has been carried out in order to validate the features of CTF during steady state and transient cases in Light Water Reactors (LWR). The objective of the benchmark is to analyze the void fraction predicted by CTF in 4 different cases for steady state and 4 transient cases. Benchmark cases are performed with 3 different bundle types.

11:00 AM
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

G.Grandi

Studsvik Scandpower, Inc., ID, USA

The Special Power Excursion Reactor Test (SPERT) III E-core was a pressurized-water, nuclear research facility constructed to analyze the reactor's kinetic behavior under initial conditions similar to those of commercial LWRs. The SPERT III E-core, except for its size, resembles a PWR. The initial test conditions were representative of a wide range of initial and transient conditions: cold start-up, hot start-up, hot standby, and full power. This paper is a sequel to a paper presented in PHYSOR 2012. In the previous work, only the cold start-up tests were analyzed. The aim of the present work is to extend the previous work and validate the code system CASMO5 / SIMULATE-3K for the full range of initial and transient conditions expected in reactivity initiated accidents.
The CASMO5 / SIMULATE-3K model was developed using data from the original SPERT experiments and the feedback models recommended by Studsvik for LWR applications. The CASMO5 / SIMULATE-3K results are summarized in terms of: peak power, reactivity compensation at the time of peak power and the energy release to time of peak power. The CASMO5 / SIMULATE-3K transient simulations are in good agreement with the experimental data. The main uncertainty in the transient calculations is the initial position of the control rods. Differences in the control rod positions, and hence in the initial reactivity insertion, can explain the differences between the calculated and experimental results.

11:20 AM
Validation of the Nodal Kinetics Code System GALAXY/COSMO-K Using the SPERT-III E-Core Experiments

K.Yamaji, Y.Takemoto, K.Kirimura, S.Kosaka, H.Matsumoto

Mitsubishi Heavy Industries, Ltd., Kobe, Japan

This paper describes validation of the GALAXY/COSMO-K system to simulate reactivity insertion accidents (RIA) in the Special Power Excursion Reactor Test III (SPERT-III) E-core tests. Mitsubishi Heavy Industries, Ltd. (MHI) has developed the new neutronics and thermal-hydraulic (T/H) coupling code system SPARKLE-2. SPARKLE-2 consists of the core neutronic, the T/H and the plant system calculation codes. Under the present circumstances, the SPARKLE-2 code system has been applied to the Anticipated Transient without Scram (ATWS) in Design Extension Condition (DEC) analyses, which are required by new Japanese regulation. GALAXY/COSMO-K is used as the neutronic code in SPARKLE-2. In the previous study, GALAXY/COSMO-K has been verified and validated using several well-known benchmark problems. In this study, GALAXY/COSMO-K simulates the RIA analyses in cold zero power, hot startup, hot stand-by and hot-operation conditions of the SPERT-III. It is found that the GALAXY/COSMO-K system represents the experiment results well against any initial conditions and insertion reactivity cases in the SPERT-III experiments. In addition, by using the SPERT-III experiment analyses, we performed sensitivity analyses about Doppler reactivity.

11:40 AM
Propagation of Nuclear Data Uncertainty for a Control Rod Ejection Accident Using the Total Monte-Carlo Method

D.F.D.Cruz, D.Rochman, A.J.Koning

Nuclear Research and Consultancy Group NRG, Petten, The Netherlands

The Total Monte-Carlo method was applied to a full 3-D core of a typical pressurized water reactor (PWR) for the uncertainty analysis of key reactor parameters as result of uncertainties in nuclear data of the isotopes $^{235,238}U$, ^{239}Pu, and thermal scattering data for H in H_2O. Both steady-state and transient reactor states were considered. Uncertainty evaluation of key physical quantities is of importance in the safety assessment during licensing of new nuclear power plants (NPP), and upgrading of operating conditions of NPP already in operation. This work focus on the uncertainties in important reactor parameters followed during the evolution of a control rod ejection accident, where the control rod with the highest worth is supposed to be ejected from the core within 0.1 seconds. Among other parameters studied, the clad-to-coolant heat flux (and the associated uncertainty) was considered, which is an important parameter in relation with the departure of nucleate boiling (DNBR) safety criteria used for this type of reactivity insertion accident scenarios.

Track1-9 Reactor Analysis Methods

Session Chair: Richard Sanchez(CEA), Baocheng Zhang(WH)

13:30 PM
Resonant Upscattering Effects on ^{238}U Absorption Rates

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Commissariat a l'energie atomique et aux energies alternatives, Gif-sur-Yvette Cedex, France

The new requirements of accuracy in reactor physics calculations justify a review of the current models, like that one adopted in the cross-section processing, for a better evaluation of the keff and the resonant absorption rates. In this context, the aim of this paper is to investigate the effects of the free gas kernel developed by Sanchez on the ^{238}U absorption rates. Homogeneous medium tests point out the increase of the absorption rates in the left wing of the resonances due to the upscattering produced by the new kernel. Heterogeneous tests show that the absorption in the left wing of the resonances is mostly affected by the scattering anisotropy in the laboratory system.

13:50 PM
The Up-Scattering Treatment in the Fine-Structure Self-Shielding Method in APOLLO3®

L.Lei-Mao, I.Zmijarevic

CEA, Gif-sur-Yvette Cedex, France

The use of the exact elastic scattering in resonance domain introduces the neutron upscattering which must be taken into account in the deterministic transport code. We present the newly implemented up-scattering treatment in the fine-structure self-shielding method of APOLLO3® . Two pin cell calculations have been carried out in order to evaluate the impact of the up-scattering treatment. The results are compared to those obtained by the Monte Carlo code TRIPOLI-4® with its newly implemented DBRC model. The comparison of k-eff values on the examples of single cell calculations shows a very good agreement between the APOLLO3® up-scattering treatment and the TRIPOLI-4® DBRC model, which is less than 30 pcm for UOX fuel and less than 110 pcm for MOX. Also, the differential effects of asymptotic versus exact kernel produced by APOLLO3® compared to TRIPOLI-4®, do not exceed 20 pcm for the UOX cell and 40 pcm for the MOX cell. A detailed comparison of the U238 absorption rates shows clearly the influences of the first four big resonances of U238 to the calculation results.

14:10 PM
Target Motion Sampling Temperature Treatment Technique with Track-Length Estimators in OpenMC - Preliminary Results

T.Viitanen, J.Leppanen(1), B.Forget(2)

1)VTT Technical Research Centre of Finland, VTT, Finland, 2) Massachusetts Institute of Technology, MA, USA

This paper examines the applicability of the Target Motion Sampling (TMS) temperature treatment method together with track-length estimators. Several track-length estimator based quantities are calculated in four test cases using a preliminary implementation of the method in OpenMC. The results are compared to an NJOY-based reference. The study reveals a statistically significant bias in the estimator results, but the errors were found to only affect limited energy regions, and their magnitude is relatively small as long as energy-integrated estimators are considered.

14:30 PM
Problem-Dependent Doppler Broadening of Continuous-Energy Cross Sections in the KENO Monte Carlo Computer Code

S.W.D.Hart, G.I.Maldonado(1), C.Celik, L.C.Leal(2)

1)The University of Tennessee, Knoxville, USA, 2)Oak Ridge National Laboratory, Oak Ridge, USA

This paper discusses recent progress in the SCALE KENO Monte Carlo code to generate problem-dependent, Doppler-broadened cross sections. The approach investigated in this study uses a finite-difference method to calculate the temperature-dependent cross sections for the 1D data, and a simple linear-logarithmic interpolation in the square root of temperature for the probability tables. With the current approach, the temperature-dependent cross sections are Doppler broadened before transport starts, and the impact on cross section loading is negligible for all but a few isotopes. Results are compared with those obtained by using multigroup libraries. Current results compare favorably with these expected results.

14:50 PM
Verification of Doubly-Heterogeneous Self-Shielding Method Based on Equivalance Theory

S.Choi, D.Lee(1), M.L.Williams(2)

1)UNIST, Ulsan, Korea, 2)ORNL, Oak Ridge, USA

A new methodology has been developed to treat resonance self-shielding in double heterogeneity systems such as VHTR. The new method adopts equivalence theory in both micro- and macro- level heterogeneities and uses the resonance integral table with a modified interpolation parameter, i.e., background cross section. This paper presents the implementation of the new method and verification results for various pin cell problems. The verification results show acceptable accuracy compared to the results of Monte Carlo code. A formulation of correction factor for Dancoff factor has also been introduced to improve the accuracy.

15:10 PM
Quantification of Resonance Interference Effect for Multi-Group Effective Cross-Section in Lattice Physics Calculation

H.Koike, K.Yamaji, K.Kirimura, S.Kosaka, H.Matsumoto

Mitsubishi Heavy Industries, Ltd., Kobe, Japan

This paper presents the quantification of resonance interference effect for multi-group effective cross-section in lattice physics calculation. In the resonance self-shielding method based on the equivalence theory, the resonance interference effect among multiple nuclides cannot be treated directly to the multi-group effective cross-section. The continuous energy or the ultra-fine-group treatment can directly consider the effect, but the application to the fuel assembly geometry is not realistic with practical computation time. In the present study, the resonance interference effect to the multi-group effective cross-section is simply quantified by the resonance interference factor (RIF) in order to confirm the benefit for considering the effect. The RIF is generated for the typical pin-cell geometry of water moderated system. The multi-group effective cross-sections with and without RIFs are compared with the continuous energy Monte-Carlo result. As a result, the significant impact for considering the resonance interference effect is confirmed to the limited nuclide, reaction type and energy group. Fortunately, these have small effect on k-infinity because the resonance interference effect is mainly induced by the wide resonances of ^{238}U to the other minor nuclides (e.g., ^{235}U, ^{239}Pu) in the limited resonance energy ranges. The results also show that the effect is small to the absorption cross-section of ^{238}U, which is the dominant resonance nuclide in the fuel. The quantification results in the present study indicate a useful material to investigate the more advanced resonance treatment for the next generation lattice physics code.

Track4-5 Verification, Validation and Uncertainty Analysis

Session Chair: Maria Avramova(Penn State University), Willem F.G.van Rooijen(Univ. of Fukui)

13:30 PM

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

CEA/DEN, Gif sur Yvette Cedex, France

The knowledge of the decay heat value and its associated uncertainty is of paramount importance for safety concerns in nuclear power plants, storage and transport facilities. The CYRUS tool has been developed in order to perform sensitivity analysis and uncertainty propagation on fuel inventories and decay heat due to the propagation of the variance and covariance data of all the nuclear data involved in the decay heat calculation (cross-sections, branching ratios, radioactive decay constants, fission yields, mean decay energies), for all kind of fuels and reactors. This tool works together with the reference depletion code DARWIN2.3 and is based on a deterministic method. In the meantime, the CEA/DEN also developed a probabilistic approach based on the URANIE uncertainty platform and the MENDEL depletion code.
The purpose of this paper is to set up the bases of the deterministic method in CYRUS, to discuss the choice of the different physical correlations between the parameters, and to validate CYRUS by a numerical comparison with the probabilistic approach implemented in URANIE/MENDEL. Two preliminary applications of elementary fission bursts have been investigated: the thermal fission of 235U and the fast fission of 239Pu. All the nuclear data come from the JEFF-3.1.1 evaluation. Eventually, the comparison of the deterministic approach of CYRUS and the probabilistic approach reveals good agreement between the two methods.

13:50 PM

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

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

1)The Pennsylvania State University, Pennsylvania, USA, 2)Oak Ridge National Laboratory, Tennessee, USA

The purpose of this study is to investigate bias and uncertainty in fuel isotopic calculations for a well-defined radiochemical assay benchmark with Sampler, the new sampling-based uncertainty quantification tool in the SCALE code system. Isotopic predictions are compared to measurements of fuel rod MKP109 of assembly D047 from the Calvert Cliffs Unit 1 core at three axial locations, representing a range of discharged fuel burnups. A methodology is developed which quantifies the significance of input parameter uncertainties and modeling decisions on isotopic prediction by comparing to isotopic measurement uncertainties. The SCALE Sampler model of the D047 assembly incorporates input parameter uncertainties for key input data such as multigroup cross sections, decay constants, fission product yields, the cladding thickness, and the power history for fuel rod MKP109. The effects of each set of input parameter uncertainty on the uncertainty of isotopic predictions have been quantified. In this work, isotopic prediction biases are identified and an investigation into their sources is proposed; namely, biases have been identified for certain plutonium, europium, and gadolinium isotopes for all three axial locations. Moreover, isotopic prediction uncertainty resulting from only nuclear data is found to be greatest for Eu-154, Gd-154, and Gd-160.

14:10 PM

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

J.C.Sublet(1), J.W.Eastwood, J.G.Morgan(2)

1)United Kingdom Atomic Energy Authority, Abingdon, UK, 2)Culham Electromagnetics Ltd., Abingdon, UK

The new inventory code FISPACT-II provides predictions of inventory, radiological quantities and their uncertainties using nuclear data covariance information. Central to the method is a novel fast pathways search algorithm using directed graphs. The pathways output provides (1) an aid to identifying important reactions, (2) fast estimates of uncertainties, (3) reduced models that retain important nuclides and reactions for use in the code's Monte-Carlo sensitivity analysis module. Described are the methods that are being implemented for improving uncertainty predictions, quantification and propagation using the covariance data that the recent nuclear data libraries contain. In the TENDL library, above the upper energy of the resolved resonance range, a Monte Carlo method in which the covariance data come from uncertainties of the nuclear model calculations is used. The nuclear data files are read directly by FISPACT-II without any further intermediate processing. Variance and covariance data are processed and used by FISPACT-II to compute uncertainties in collapsed cross-sections, and these are in turn used to predict uncertainties in inventories and all derived radiological data.

14:30 PM

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

J.Hykes, J.Rhodes

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We examine the effects of the uncertainty in decay and fission yield data on CASMO5- computed nuclear fuel isotopics, comparing the results to measurements. The uncertainties of radioactive decay constants, decay and (n,γ) capture branching fractions, and fission yields are extracted from the ENDF/B-VII.1 data files or estimated where they are lacking. These nuclear data uncertainties are propagated through the CASMO5 lattice depletions via Monte Carlo sampling. The distribution of computed isotopic compositions are compared to the Fukushima Daini-2 measurements of Nakahara. The uncertainties in the calculated values are able to explain some of the calculation-to-measurement discrepancies, particularly for Am-242m and the samarium isotopes.

14:50 PM

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

L.Fiorito(1), G.van den Eynde, A.Stankovskiy(2), O.Cabellos, C. Diez(3)

1)SCKCEN, Mol, Belgium / ULB, Brussels, Belgium, 2)SCKCEN, Mol, Belgium, 3)UPM, Madrid, Spain

Uncertainty propagation and sensitivity analysis for burnup codes are key topics in the nuclear field because of their huge involvement in safety-related problems. Consequently, a correct evaluation of a given response variation after propagating the system parameter uncertainties is a primary feature for any burnup code.
In this paper we described the options of linear sensitivity analysis, either forward or adjoint, recently implemented in the ALEPH Monte Carlo burnup code. The testing phase was carried out against the fission pulse decay heat problem for 239Pu thermal fission and compared to total Monte Carlo results, also produced by the code through its dedicated routines. The different techniques give comparable results thus supporting the consistency of the methods. To conclude, we also calculated and listed the parameters that contribute the most to decay heat perturbations.

15:10 PM

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

Paul Scherrer Institut, Villigen, Switzerland

Uncertainty Quantification (UQ) helps to understand discrepancies between predictions and experiments by providing confidence intervals in the results. A tool, referred to as SharkX, is under development at PSI for UQ studies in relation to the CASMO-5 lattice physics code. In this paper, calculation-to-experimental ratios (C/Es) for the nuclide compositions of a UO_2 spent fuel sample from the LWR-PROTEUS Phase II program are presented with both experimental and nuclear data uncertainties. The uncertainty propagation of the nuclear data is performed using a stochastic sampling method for cross sections and decay constants but also for fission yields. For the latter, a new perturbation methodology taking into account physical constraints is developed and detailed in this paper. This methodology helps to quantify the uncertainty on kinf and the nuclides compositions with respect to the fission yields uncertainties. Its results are compared with that of straightforward perturbations and basic normalization methodologies without consideration of correlations. These results also highlight the very strong impact of the uncertainty of the I-133 yield due to U-235 fissions (σ = 64% in ENDF/B-VII.1) on both kinf

Track4-5 Verification, Validation and Uncertainty Analysis

Session Chair: Maria Avramova(Penn State University), Willem F.G.van Rooijen(Univ. of Fukui)

and caesium composition uncertainties. An alternative source of uncertainty on this fission yield has also been considered and shows more reasonable results.

Track7-1 Reactor Concepts and Designs

Session Chair: Bojan Petrovic(Georgia Tech. Univ.), Akiyuki Tsuchiya(Hitachi GE)

13:30 PM

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

D.Hartanto, Y.Kim

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A long-life Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR) has been investigated from the neutronic perspective. This reactor will serve as an alternative plan while waiting for the availability of reprocessing technology in Korea. The objective of this study is to extend the lifetime of the PGSFR to more than 20 EFPYs and have an average core discharge burnup that is greater than 100 GWd/MTHM. The objective has been achieved by significantly increasing the fuel volume fraction, the use of a better reflector, and the utilization of an axial blanket while minimizing modifications from the current PGSFR design. After fulfilling the design target, the long-life PGSFR has been characterized. Several parameters such as the conversion ratio, coolant void reactivity, power distribution, and fast neutron fluence have been calculated. The safety concern due to the positive reactivity insertion due to fuel compaction of the long-life PGSFR core has also been investigated. All neutronics calculations have been performed using the Monte Carlo code McCARD.

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Core Design Studies on the Fast Reactor with Flexible Breeding Ratio

Y.Xiao, H.Wu, Y.Zheng

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This paper presents the conceptual design of a sodium cooled fast reactor with flexible breeding ratio. It can achieve variable breeding ratios without significant changes in core design. This concept enables fast reactor effectively responds to the industrial needs. The U-Pu-Zr alloy with better breeding and thermal performance is chosen as fuel. Blankets contain PWR spent fuels are axially and radially arranged. The number and layout of assemblies are adjusted to achieve the target breeding ratios. The achieved breeding ratios varied from 1.0 to 1.4. All the reactor cores use the same fuel design. Three enrichment are appropriate selected to ensure all reactors maintain critical during the operation. The assembly number and their layout are adjusted to reduce the maximum linear power. Two control systems are designed and evaluated to guarantee that the capability of either system fulfilled its control function. Concerned reactivity coefficients are calculated at the equilibrium cycle. The temperature distribution is obtained by the single channel heat transfer calculation and the preliminary safety analysis is evaluated by the quasi-static methodology to indicate the inherent safety features.

14:10 PM

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

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Helmholtz-Zentrum Dresden-Rossendorf, Dresden, Germany

The use of fine distributed moderating material to enhance the negative feedback effects and to reduce the sodium void affecting sodium cooled fast reactor cores is described. The influence of the moderating material on the neutron spectrum is given and evaluated through impact on the capture cross sections of major materials (U-238, Pu-239, and Pu- 240). The influence of the variation of the Pu content on the efficiency of the enhancement of the Doppler effect and on the reduction of the positive coolant and sodium void effect in a representative SFR fuel assembly configuration is analyzed. Additionally the influence of the moderating material combined with the variation of the Pu content on the infinite multiplication factor is studied.

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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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In this paper, a design concept of 400 MWe sodium cooled fast reactor (SFR) cores having thorium blankets for effective burning of TRU (Transuranics) from LWR spent fuel is described. Specifically, we considered two recycling options of thorium blankets : 1) no recycling and 2) fully recycling. The thorium blankets are loaded in the axially central regions of the core regions and their axial heights are adjusted so as to increase TRU burning rate and to reduce burnup reactivity swing. Also, we analyzed the performances of the cores having different fuel management batch sizes and different recycling options for the searched core configuration. The results show that the axial thorium blankets with no recycling option can be effectively used to increase TRU burning rate with a significant reduction of burnup reactivity swing in comparison with typical SFR burner cores having no blankets while the recycling of thorium blanket degrades TRU burning rate and burnup reactivity swing but it leads to a reduction of sodium void worth and more negative Doppler coefficient.

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

F.Lodi, M.Sumini(1), G.Grasso(2), C.Rabiti(3)

1)University of Bologna, Bologna, Italy, 2)ENEA, Bologna, Italy, 3)Idaho National Laboratory, Idaho Falls, USA

Recent improvements in the PHISICS code are listed and discussed. Also, the assessment of the PHISICS code for fast reactor design applications is considered. For this purpose a comparison with the deterministic code ERANOS2.2 and the stochastic MCNPX is performed on the lead-cooled concept ALFRED. The results indicate good general agreement between the codes on the integral quantities usually of interest in reactor design. Also the analysis highlighted PHISICS's possibility of keeping low the computational time even when using higher order approximations.

15:10 PM

Velocity Characteristic and Stability of Wave Solutions for a CANDLE Reactor with Thermal Feedback

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We study nuclear burning wave (NBW) of a steady shape in the U-Pu CANDLE reactor analytically and numerically taking into account nuclear density mechanisms which are responsible for the wave velocity formation and thermal feedback to power. One dimensional model of the infinite cylindrical reactor with the absorber which does not burn out includes equations for nuclear number densities and one group diffusion equation for neutron flux which includes quadratic in flux term responsible for instant thermal feedback in the simplest form. We show that thermal feedback and mechanism related to finite life-time of the short-living ^{239}Np additively contribute to velocity formation and compete together with the ^{241}Pu mechanism which dominates at lower velocities and leads to instability of the corresponding stationary wave solutions. Negative feedback decreases the wave velocity and lowers possible absorber concentrations. Under realistic conditions both neptunium and plutonium nuclear density mechanisms and thermal feedback may be important for the wave velocity formation in CANDLE reactors.

Track3-5 Monte Carlo Methods

Session Chair: Bradley Rearden(ORNL), Marco Pecchia(PSI)

13:30 PM

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

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Monte Carlo depletion calculations for nuclear reactors are affected by the presence of stochastic noise in the local flux estimates produced during the calculation. The effects of this random noise and its propagation between timesteps during long depletion simulations are not well understood. To improve this understanding, a series of Monte Carlo depletion simulations have been conducted for a 3-D, eighth-core model of the H.B. Robinson PWR. The studies were performed by using the in-line depletion capability of the MC21 Monte Carlo code to produce multiple independent depletion simulations. Global and local results from each simulation are compared in order to determine the variance among the different depletion realizations. These comparisons indicate that global quantities, such as eigenvalue (k_{eff}), do not tend to diverge among the independent depletion calculations. However, local quantities, such as fuel concentration, can deviate wildly between independent depletion realizations, especially at high burnup levels. Analysis and discussion of the results from the study are provided, along with several new observations regarding the propagation of random noise during Monte Carlo depletion calculations.

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

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The effective delayed neutron fraction, β_{eff}, and the prompt neutron generation time, Λ, in the point kinetics equation are weighted by the adjoint flux to improve the accuracy of the reactivity estimate. Recently the Monte Carlo (MC) kinetics parameter estimation methods by using the adjoint flux calculated in the MC forward simulations have been developed and successfully applied for reactor analyses. However these adjoint estimation methods based on the cycle-by-cycle genealogical table require a huge memory size to store the pedigree hierarchy. In this paper, we present a new adjoint estimation method in which the pedigree of a single history is utilized by applying the MC Wielandt method. The algorithm of the new method is derived and its effectiveness is demonstrated in the kinetics parameter estimations for infinite homogeneous two-group problems and critical facilities.

14:10 PM

Higher-Order Chebyshev Rational Approximation Method (CRAM)

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The burnup equations can in principle be solved by computing the exponential of the burnup matrix. However, due to the difficult numerical characteristics of burnup matrices, the problem is extremely stiff, and the matrix exponential solution was long considered infeasible for an entire burnup system containing over a thousand nuclides. After discovering that the eigenvalues of burnup matrices are generally confined to a region near the negative real axis, the Chebyshev rational approximation method (CRAM) was introduced as a novel method to solve the burnup equations. It can be characterized as the best rational function on the negative real axis and it has been shown to be capable of simultaneously solving an entire burnup system both accurately and efficiently. The main difficulty in using CRAM for computing the matrix exponential is determining the coefficients of the rational function for a given approximation order. Some polynomial CRAM coefficients have been published in 1984, and based on these literature values, CRAM approximations up to the order 16 have been thus far applied in burnup calculations. The topic of this paper is the computation of CRAM approximations and their application to burnup equations. A Remez-type method utilizing the equioscillation property of best approximations is used to construct the CRAM approximants for approximation orders 1, ..., 50. Numerical results are presented for a large burnup system and for a decay system. It is demonstrated that higher-order CRAM can be used to accurately solve the burnup equations even with time steps of the order of millions of years.

14:30 PM

Efficiency and Accuracy Evaulation of the Windowed Multipole Direct Doppler Broadening Method

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It was recently proposed to use the multipole formalism for direct Doppler broadening by reducing the number of poles that need broadening and approximating the contributions of distant poles. The multipole representation takes the Reich-Moore formalism and restructures it into a pole and residue form that can be analytically Doppler broadened using the Solbrig kernel. The windowed multipole method reduces the number of calculations required by approximating the contribution of far away poles with a fitting function. Additionally, approximate Faddeeva function evaluations can be used to reduce computational costs. A detailed study was performed on ^{238}U to determine the tradeoff between efficiency, memory requirements and accuracy. It was determined that accuracy could be gained at little to no additional cost by fitting the nonfluctuating poles in windows that were equally spaced in momentum instead of energy as had been done prior. It was also determined that outer window sizes can be tailored to a specific accuracy level. Two sets of promising parameters for accuracy and memory requirements were selected and tested at two different temperatures on a slowing down problem in ^{238}U and hydrogen over the resolved resonance range. Various approximations of the Faddeeva function evaluations were also compared to determine the tradeoff between accuracy and efficiency. The results indicated that a performance comparable to table lookup is achievable. Table lookup with temperature interpolation required 0.68 μs per ^{238}U cross-section evaluation, while windowed multipole ranged from 0.35 to 2.55 μs depending mostly on the Faddeeva approximation used. The windowed multipole required roughly 300 kB of storage for the resolved resonance range covering all temperatures of interest and three cross-sections of interest, while a typical single temperature piece-wise linear library in the resonance range required roughly 5 MB for the same data.

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Diffusion Monte Carlo Method with Transport Corrections

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Analytical Kernel Based Diffusion Monte Carlo method has been developed for simulating proposed reactor noise experiments for measuring the degree of sub-criticality in accelerator driven systems. Although this approach has several advantages such as speed, elegance and exactitude, but it was applicable to a rather restricted class of problems, such as the bare homogeneous reactor. This approach has been further developed to demonstrate its utility in a wider class of problems involving heterogeneous media and forms the subject of this paper. We also investigate whether and to what extent the diffusion based Monte Carlo can be improved to give results closer to transport theory, particularly in situations wherein diffusion theory methods are otherwise inapplicable.

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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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Allowing Monte Carlo (MC) codes to perform fuel cycle calculations requires coupling to a point depletion solver. In order to perform depletion calculations, one-group (1-g) cross sections must be provided in advance. This paper focuses on generating accurate 1-g cross section values that are necessary for evaluation of nuclide densities as a function of burnup. The proposed method is an alternative to the conventional direct reaction rate tally approach, which requires extensive computational efforts. The method presented here is based on the multi-group (MG) approach, in which pre-

Track3-5 Monte Carlo Methods

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generated MG sets are collapsed with MC calculated flux. In our previous studies, we showed that generating accurate 1-g cross sections requires their tabulation against the background cross-section (σ0) to account for the self-shielding effect. However, in previous studies, the model that was used to calculate σ0 was simplified by fixing Bell and Dancoff factors. This work demonstrates that 1-g values calculated under the previous simplified model may not agree with the tallied values. Therefore, the original background cross section model was extended by implicitly accounting for the Dancoff and Bell factors. The method developed here reconstructs the correct value of σ0 by utilizing statistical data generated within the MC transport calculation by default. The proposed method was implemented into BGCore code system. The 1-g cross section values generated by BGCore were compared with those tallied directly from the MCNP code. Very good agreement (<0.05%) in the 1-g cross values was observed. The method does not carry any additional computational burden and it is universally applicable to the analysis of thermal as well as fast reactor systems.

Track15 Research Related to Fukushima Accident

Session Chair: Alireza Haghighat(Virginia Tech Univ.), Akio Yamamoto(Nagoya Univ.)

13:30 PM
AREVA Dismantling and Decommisioning Experience and Fuel Debris Removal Approach for Fukushima Dai-ichi

Guest Speaker: K.Schauer
AREVA, France

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Re-Criticality Potential at Fukushimi Dai Ichi Unit 4

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The damage to Fukushima Daiichi Unit 4 reactor buildings discovered on March 15, 2011, led to concerns that the Unit 4 spent fuel pool had leaked and might have gone dry. The possibility of a re-criticality event occurring following reflooding of a fully drained Unit 4 spent fuel pool was a concern. Uncertainties regarding the status of the fuel and the pool made it difficult to apply conventional criticality safety methods. However, EPRI was able to apply an alternative criticality safety methodology, developed for the analysis of spent fuel criticality during transportation accidents, to show that the most likely configuration of damaged fuel and pool would be a decrease in reactivity. Thus it was safe to reflood the pool.

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

A.Sargeni, G.Caplin
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During the Fukushima event, IRSN performed some calculations to assess whether a criticality excursion was likely to occur in case of a loss of coolant of the spent fuel pools (dry-out). The results of these calculations show that the k-eff of the spent fuel pool can increase when a water-air mixture is modeled within the storage (with subsequent water density decrease), compared to the k-eff value with a full water density within the pool. This k-eff increase reaches more than 5% for an optimal range of water-air mixture, which is more than usual safety margins kept to demonstrate the safety of the spent fuel pools. This initial study showed that the pitch between the fuel assemblies, the materials (steel with more or less boron) composing the storage rack and the water densities are the main parameters driving the magnitude of the k-eff increase.
All the above calculations were performed with the standard route of the CRISTAL package and, in order to interpret and understand the here-above results, investigations were performed in parallel using pure Monte-Carlo pointwise calculations and pure deterministic calculations so as to first confirm, then to explain the observed results using a reactivity splitting via the four factors formula. This paper presents these complementary studies showing how, when water density decreases and inter-assembly gap increases, the balance between increasing fast fissions and water absorption may justify a reactivity increase.

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Critical Experiments for Fuel Debris Using Modified STACY

K.Izawa, K.Tonoike, H.Sonov, Y.Miyoshi
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Critical assemblies of the thermal neutron system are decreasing in number in spite of their important roles in the reactor physics research, the design of nuclear facilities and the human resource training. On the other hand, the utilization term extension of the light water reactors brings new research themes requiring critical experiments of the thermal neutron system; e.g., the new fuel design with higher burn up, the introduction of burn up credit into the criticality safety control of spent fuel, the criticality safety control of fuel debris generated in a severe accident of a reactor, etc. Japan Atomic Energy Agency is modifying the Static Experiment Critical Facility (STACY) to revive the critical experiments. The modified STACY will be an infrastructure, a multipurpose critical assembly, for the experimental research of the reactor physics on thermal neutron system. The primary mission of the modified STACY at present is the critical experiments for fuel debris to contribute to the criticality safety control of the fuel debris generated by the severe accident of the Fukushima Daiichi Nuclear Power Station. This report introduces the plan of criticality safety research in Japan Atomic Energy Agency following the accident, and describes the role of the modified STACY in the retrieval work of fuel debris from the damaged reactor.

14:50 PM
Concept of Capture Credit Based on Neutron Induced Gamma Ray Spectroscopy

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Capture credit (CapC) based on neutron induced gamma ray spectroscopy (NIGS) is proposed to estimate the sub-criticality margin of fuel debris in which nuclear fuel and structural materials are co-melted or mixed. By NIGS, rates of some capture reactions can be measured in relation to fission reactions. By the ratio, we can credit the sub-criticality margin of the fuel debris caused by the capture reactions. The theory of CapC is described. In order to demonstrate how CapC is beneficial, numerical simulations are performed for a hypothetical array of canisters in which the fuel debris is stored. An application procedure of the CapC is also proposed, which consists of several technologies: 1) NIGS, 2) simulations of a response and an efficiency of the γ ray detection, 3) unfolding of the γ ray spectrum to obtain reaction rates. Experimental studies of NIGS have been launched in Kyoto university critical assembly facility. NIGS is firstly studied for simulated fuel debris of a few kinds of mixture of stainless-steel and uranium. The measured γ ray spectra and preliminary analysis give promising results for CapC based on NIGS.

15:10 PM
A Methodology for Fast and Accurate Decay Heat Calculations for In-Pool Used Fuel Assemblies Developed at AREVA La Hague Reprocessing Facility

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The evaluation of the decay heat released from used fuels stored in an interim storage pool is a crucial issue for safety and operation. This paper presents a methodology developed at the AREVA NC La Hague reprocessing facility in order to perform a fast and accurate assessment of the decay heat for a pool containing a large variety of fuel assemblies with different features (kind of reactor, initial composition, number of irradiation cycles, operating power, outage length) and cooling times ranging from 6 months to 60 years. The key aspect of the methodology relies on the fact that, for cooling times higher than 6 months, only a very short number of isotopes significantly contribute to the overall decay heat. The methodology uses results from calculations performed by the industrial depletion code CESAR and takes benefit from the experimental validation of CEA reference depletion package DARWIN. A full CESAR calculation at a given cooling time for each assembly allows building up a reference database: for any cooling time higher than the reference one, a simplified on-line calculation involving less than 30 isotopes is performed, and all approximations made are covered by the mean of biases relying on pre-evaluated coefficients. Results for an application to a 10,000-used-fuel-assembly interim storage pool are given.

Poster Session

Track1: Reactor Analysis Methods

PAPER 1089476

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

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CASMO-4 is a multigroup two-dimensional transport code for LWR lattice physics calculations. MCNPX and TRITON/T6-Depl are two general-purpose transport codes with depletion capability for various fuel designs. MCNPX can use continuous-energy cross sections while TRITON currently only supports multigroup depletion calculations. This study presented a systematic comparison of these three codes for depletion calculations of a typical BWR fuel assembly. Key parameters for sensitivity studies were neutron cross-section libraries, burnup steps, modeling of poison rods, inclusion of additional nuclides for depletion, thermal expansion, pin-by-pin depletion, and Dancoff factors. The CASMO-4 results were arbitrarily taken as a reference base on which the differences of MCNPX or TRITON calculations were evaluated. Useful observations from the comparisons were as follows: The ENDF/B-VII cross-section library gave the most consistent result with CASMO-4. At least five radially subdivided zoning of a Gd-bearing rod was necessary for depletion calculations. MCNPX calculations were more sensitive to choices of burnup steps and numbers of nuclides being traced in fuel inventory than TRITON did. Applying the same thermal expansion corrections in TRITON reduced its differences with CASMO-4 in the middle of cycle. Pin-by-pin depletion is necessary but only slightly changed k_∞ profiles in this case compared with average depletion. Using more accurate Dancoff factors in TRITON resulted in an excellent agreement of k_∞ values with CASMO-4 at the early stage of burnup, but they still gradually deviated at later burnups. Overall, both MCNPX and TRITON predicted k_∞ profiles in this problem were within 500 pcm agreement with CASMO-4 in the entire burnup period.

PAPER 1092007

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

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Propagation noise induced by fluctuations of the boron content of the coolant, in pressurized water reactors (PWRs) is addressed in this paper. The spatial discretization of the neutron noise equations is based on the Box-Scheme Finite Difference Method (BSFDM) for rectangular-z, triangular-z and hexagonal-z geometries. Using the derived equations, a 3-D 2-group neutron noise simulator is developed, by which the discrete form of both the forward and adjoint reactor dynamic transfer functions (in the frequency domain) can be calculated. In addition, both types of noise sources, namely point-like and traveling perturbations, can be modeled by the developed neutron noise simulator. The results and benchmarking are then reported for the case study of a VVER-type pressurized water reactor core. Considering the boric acid fluctuations as traveling perturbations of the macroscopic thermal absorption cross sections, the induced neutron noise is calculated throughout the reactor core. Space- and frequency-dependence of the propagation noise are also investigated in this work.

PAPER 1093007

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

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NEXUS/ANC9 is the latest licensed PWR core design code system developed by Westinghouse. This system has demonstrated capabilities of modeling advanced core designs with improved accuracy in core reactivity and power distribution predictions. NEXUS/ANC9 system is being rolled out to replace the current APA system (ALPHA/PHOENIX-P/ANC) for routine core calculations. In addition to the standard core design calculations, investigations are underway to explore the possibility to expand the NEXUS/ANC9 application for safety analysis, especially at accident conditions. The main focus of the investigation is the evaluation of the NEXUS/ANC9 cross-section representation model conditions like high void and significant change of core pressure. Comparisons of the predicted parameters among ANC9, PARAGON lattice code and MCNP calculations are presented. The results show that NEXUS/ANC9 is able to model the cross-section behavior and accurately reproduce lattice code results at all simulated conditions.

PAPER 1102611

Numerical Dispersion and Dissipation Analysis of Nodal Expansion Method

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The numerical property of nodal expansion method (NEM) is studied in the paper from the perspective of numerical dispersion and dissipation, which is brand new for nodal methods and no one else has ever tried before. Besides, the more complicated transient convection diffusion equation is chosen to be the research target so as to be as general and comprehensive as possible. First, the nature and connotation of dispersion and dissipation is presented. Then, the numerical dispersion and dissipation analysis for NEM is developed with the help of Fourier analysis and solution for complex matrix generalized eigenvalue problem. Through analyzing the numerical dispersion and dissipation of NEM with different order N basis functions as well as comparing it with the central difference (CD) and first order upwind scheme (FUS), and with numerical verification, the conclusion is drawn finally: the numerical dispersion and dissipation of NEM is of an advance in that it can simulate the rather difficult problems such as steep gradients, the high frequency analytical solution, convection dominated problems even in the coarse mesh.

PAPER 1102622

The Integration of Control Rod Calculation and VSOP

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Accurate calculation of the control rod outside the active zone in pebble bed HTR requires special treatment, integration of this detailed control rod calculation with whole core calculation like VSOP package requires more efforts, which is not realized before. Accurate calculation of control rod is proposed to use the discontinuity factor-corrected diffusion method accompanied with transport calculation in advance. Appropriate coupling between the control rod calculation and whole core calculation must be developed to take into account of the buckling feedback, spectrum change, data transform and so on. In this paper, internal spectrum calculation module inside VSOP is used to model the control rod region, and the neutron leakage obtained from the whole core calculation is used as the boundary condition of the control rod region. The numerical calculation demonstrated the feasibility and accuracy of this coupling and integration method.

PAPER 1103763

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

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Normalised Generalised Equivalence Theory is used to model the ex-core reflector region of the SAFARI-1 research reactor. This method is a one-dimensional homogenisation technique based on Generalised Equivalence Theory, but with only one discontinuity factor defined per node, and divided into the nodal parameters. The SAFARI-1 reactor is modelled with the deterministic code system OSCAR-4. Cross-sections for the reflector model is generated with NEWT (part of the SCALE 6.1 package) and EQUIVA-1 (part of OSCAR-4), which calculates the NGET parameters. A period of three years in the operational history of the SAFARI-1 research reactor is modelled. Two models are used, one with traditional flux-volume weighted and the other with equivalent ex-core reflector cross-sections. The performance of the two models over the three year period is compared. Reactor parameters such as reactivity and fuel burnup are investigated. Comparisons to experimental data, in particular control rod calibrations, are also made. The model with equivalent reflector

Poster Session

parameters shows improved accuracy for control rod calibrations, a power tilt of about 10 % across the core, no noticeable change in reactivity or burnup, and significant improvement in calculational time (reduced by over 40 %) due to a reduction in the size of the core model.

PAPER 1104477
Relationship Between Computed ANSI/ANS-5.1 and ORIGEN-S Decay Heat Powers for BWR LOCA Safety Analysis

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The decay heat power fraction computed using ANSI/ANS-5.1-1979 with CASMO-4 decay heat parameters is compared with the decay heat power fraction computed using ANSI/ANS-5.1-1979 with ORIGEN-S decay heat parameters. The comparison indicates that the ORIGEN-based ANS-5.1 total decay power fraction appears very close to the CASMO-based ANS-5.1 total decay power fraction due to compensating effect between fission-product decay heat power fraction and U-239&Np-239 decay heat power fraction, although the CASMO-4 fission fractions and U-238 neutron capture ratio are considered more accurate than the ORIGEN-S fission fractions and U-238 neutron capture ratio. Therefore, it seems acceptable to calculate the total decay heat fraction using ANSI/ANS-5.1-1979 with ORIGEN-S decay heat parameters. This result is useful, since ORIGEN-S /SCALE 5.1 are easier to run than CASMO-4.
The decay heat power fraction computed using ANSI/ANS-5.1-1979 is also compared with the decay heat power fraction computed using ORIGEN-S directly. The comparison indicates that the ORIGEN-S total decay heat power fraction is much smaller than the corresponding ANS 5.1 total decay heat power fraction, which is due to the fact that the ORIGEN-S fission-product decay heat power fraction is much smaller than the corresponding ANS 5.1 fission-product decay heat power fraction. This demonstrates that the total decay heat power fraction calculated using ORIGEN-S directly is not conservative and that ANSI/ANS-5.1 must be used to calculate the total decay heat power fraction for LOCA safety analysis.

PAPER 1104519
A New Method to Measuring the α Eigenvalue of a Subcritical Reactor System

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The pulsed source method is one of the methods to confirm the time multiplication constant (α eigenvalue) for a subcritical reactor. Practically its applicability is determined by the characteristic of the neutron source. The analytic and numerical methods are both used to analyze the affect of the temporal characteristic of DPF source in the pulsed neutron method. The result indicates that the requirement of the temporal characteristic of DPF source in the pulsed neutron method is ultra fast and ultra short.

PAPER 1105008
Progress towards an Accurate Lattice-Homogenization Technique for Pressure-Tube Supercritical Water Cooled Reactor Neutronic Calculations

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The Pressure-Tube Supercritical-Water Cooled Reactor (PT-SCWR) is one of the reactor concepts investigated by the Generation-IV International Forum as a reactor concept capable of achieving an overall nuclear power plant efficiency in the vicinity of 50%, higher than the approximately 35% that characterize current nuclear power plants. The development process includes studying the neutronics methods that can provide an adequately-accurate modelling of the core. The PT-SCWR evolved from the Pressurized Heavy-Water Reactor but presents a more heterogeneous core because of the much higher discharge burnup (~40 MWd/kg vs. ~7.5MWd/kg) and greater variation of coolant density. This work presents progress towards addressing the nodehomogenization needs of the PT-SCWR core by using macroscopic cross sections and discontinuity factors calculated using quasi-exact node boundary conditions determined using global-local iterations between core and lattice calculations . Results show that cross sections and discontinuity factors obtained via the global-local iterative process are very effective at improving the accuracy of full-core diffusion calculations in the fuel region of the core and that some challenges remain in applying leakage corrections to high-leakage configurations, especially those comprising reflector nodes.

PAPER 1105869
Proposal of Subcritical PWR Core Benchmark Problems

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Subcritical PWR core benchmark problems are proposed in order to clarify the error in neutron flux distribution and the magnitude of neutron flux evaluated in fixed-source mode. Proposed benchmark problems consist of 2-stages named Stage-I and Stage-II, respectively. The difference caused by the discrepancy in theory used in simulation code will be discussed for simplified 2-D and 3-D PWR core models with fixed-source by using of same group constants in Stage-I. Then in Stage-II, we are planning to discuss how to evaluate group constants for several types of UO_2 fuel assemblies used in subcritical system analysis. This paper shows proposed benchmark problems with several results performed by Monte Carlo code as a reference.

PAPER 1105924
A New Monte Carlo-Deterministic Two-Step Method for Fast Reactor Diffusion Analysis

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This paper investigates a hybrid Monte Carlo-Deterministic hybrid method in which the cross sections are generated with the Monte Carlo method and they are used in 3-D diffusion reactor analysis. In both the new hybrid and conventional methods, the cross sections are based on a simple RZ homogeneous core model. As a computational model, a 300MWe SFR (sodium-cooled fast reactor) TRU burner core has been introduced. The generated multi-group cross sections are used by DIF3D to calculate the keff and assembly
power distribution. A whole core calculation of the heterogeneous core model was performed using MCNP5 as the reference solution for the evaluation of the analysis method. For an in-depth verification of the hybrid, it was also applied to heavily rodded core. In the case of rodded core, a new core modeling named RRZ was proposed for a better modeling of the self-shielding effect of the control assembly. Additionally, the nodal equivalence theory was successfully applied to fast reactor analysis for the first time in this work. To apply the nodal equivalence theory, a simple 1-D spectral geometry was developed to determine the flux discontinuity factors of a control assembly region. The generated group-DF values were used to correct the cross sections of the control assembly region. Moreover, the sensitivity of the DF to the 1-D spectral geometry model was also evaluated in this work. It is concluded that the hybrid method works well for the analysis of the SFR core. Particularly, the special application of the nodal equivalence theory greatly improves the accuracy of the new hybrid method for fast reactor analysis.

PAPER 1106049
Explicit Transverse Leakage Treatment Using an Analytic Basis Function Expansion

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An explicit method for calculating the transverse leakage is presented in this paper. The method is based upon the use of analytic basis functions, which represent individual eigenfunctions of the neutron diffusion equation. The intranodal flux solution is expressed as an eigenspace, and can be solved by using the already calculated surface currents and flux moments as boundary conditions. The salient feature of the method, therefore, is that no ad hoc presumptions are made with regard to the leakage shape. The individual eigenfunctions are calculated based upon already calculated parameters from the flux solution and response matrix solution, and therefore no additional parameters are introduced into the problem, which could lead to an unwanted increase in computation time. The new transverse leakage method is implemented in PSU's NEM code and is tested against the OECD/NEA 3D C5G7 rodded MOX benchmark and the C3 benchmark.

Poster Session

PAPER 1106127
Method for Calculation Capturing Reactions Contribution to Total Energy Release in Nuclear Reactors

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The growth of computers capabilities gives us the possibility to conduct computational experiments and obtain reasonable results. Computer modeling allows investigating models which are impossible for experimental investigation. This is why we can reexamine some physical phenomena at the deeper level. The energy release in nuclear reactors has been analyzed since the beginning of nuclear industry, but there are still some features that need to be investigated. In this paper, basic features of energy release in nuclear reactors and its components has been examined. The method for calculation fraction of reactions with neutron disappearance (reaction channels (n,γ), (n,α), (n,p), etc.) in total energy release has been developed. Calculations and comparative analysis of this fraction for three models of Water-Water Power Reactor-1000 (WWER-1000) have been made. The fraction of capturing energy in total energy release for WWER-1000 is about 3.5%. It is shown that depending on fuel assembly's type the total energy release could vary by 0.5%.

PAPER 1106156
A Preliminary Analysis of the Accuracy of Homogenized 2D Cross Section in 3D Nodal Calculations for BWRs

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Current methods in reactor analysis employ a 3D nodal diffusion code which utilizes homogenized cross sections generated from a 2D lattice physics calculation. This approach has been used successfully for a wide range of LWR core design and safety analysis. With the advent of more advanced axial heterogeneous fuel assemblies, particularly for BWRs, it is worthwhile to assess the accuracy of the 2D cross section generation method for advanced fuel designs. The goal of the work reported here was to provide an estimate of the accuracy of 2D cross section in this situation, specifically for control blade insertion. Results indicated that while the power shape can be estimated properly in some cases, significant differences in k-eff were observed. Work continues in the areas of void distribution, histories, and burnup.

PAPER 1106178
Deterministic Lattice Code Development at UNIST

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A new method of characteristics code, STREAM, has been developed at Ulsan National Institute of Science and Technology (UNIST). With regard to a resonance treatment in STREAM, the gray resonance treatment method and doubly heterogeneous self-shielding method were implemented based on the equivalence theory. STREAM can calculate eigenvalue and pin power distribution of a two-dimensional multi-assembly with CMFD acceleration. In this paper, designed pin cell and two-dimensional C5G7 benchmark problems were solved to verify the STREAM library and solver, respectively. As a practical problem, the VENUS-2 benchmark from research reactor experiments was calculated up to assembly level by the STREAM code. To check applicability of STREAM to new fuel analysis for a boron free reactor, innovative fuel problems were designed and calculated. STREAM solutions of these problems were compared with the Monte Carlo reference solutions, and showed a good agreement with them on a similar level.

PAPER 1106220
Implications of Mesh Refinement in Lattice Physics on BWR Core Analysis and Nuclear Design

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A BWR axial power tilt and reactivity bias with increasing coolant void content has been observed due to the application of a spatially flat source approximation in lattice transport calculations. In order to address such a reactivity bias using the CCCP method of HELIOS one needs to employ a sufficiently fine spatial and angular mesh in the lattice calculations to limit any undesired axial power oscillation to occur in real core operation and to reduce the uncertainties in important safety parameters such as the SDM. The challenge in this regard is to find a sensible balance between the use of current coupling and collision probabilities without a too large penalty in computational cost rendering such lattice calculations impractical. Based on both 2D lattice transport and 3D nodal core analyses, it was recognized that the most important mesh improvement component was to add azimuthal coolant regions in the fuel pin cells coming with a rather low computational cost. The use of collision probabilities over larger sub-domains of non-boiling moderator regions was considered appropriate mainly based on its theoretical merits as neither significant differences in results nor mesh convergence was obtained in full-lattice geometry for this modeling choice.

PAPER 1106272
Generating Multigroup Data Stochastically for a Highly Heterogeneous VHTR Problem

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A necessary step in any reactor physics solution process is the compilation of nuclear data. While the use of continuous energy cross sections in a whole core Monte Carlo calculation will always yield accurate results if enough particle histories are run, such an approach is so computationally expensive as to be considered not feasible for practical core eigenvalue calculations. It is therefore necessary to produce accurate multigroup cross sections for homogenized regions of the core, and let a faster method determine the core solution. A particular challenge, due to its more complicated spectrum and higher heterogeneity, is the generation of data for the VHTR, a gas-cooled reactor whose fuel is in particle form. In this paper, the stochastic cross section generation method MoCS-Gen is used to generate 26 group data for the VHTR. Problems in a pin cell and a fuel block are solved, and compared to the original fully heterogeneous problem in continuous energy and also to the same problems solved in continuous energy but utilizing a simple volume homogenization strategy. Results are presented which clearly show the benefit of MoCS-Gen as an accurate multigroup cross section generation method.

PAPER 1106273
Reference Solution for Core Physics

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Core calculations of nuclear reactors are usually performed by core physics codes (e.g. with NEM or FDM solvers) in diffusion or SP3 approximation of the transport equation. For each fuel type parameterized data libraries are prepared by means of a lattice code. The data libraries are burnup dependent, and the parameterization covers the hyperspace of admissible values of all operational parameters (fuel temperature, moderator density, boron concentration etc.)
This approach has two weak spots. The first is, that it is difficult to make perfect parameterization of the data library because of relatively broad range of the parameter values and the fact that the parameters' effect on the macroscopic cross-sections are not mutually independent. The second is that even for perfect parameterizations with precise approximations of the data changes with respect to the feedback parameters the so-called history effects are neglected. It is generally difficult to assess the cumulative errors arising due to the approximative parameterization of the data libraries and due to the history effects. It is as well difficult to assess the efficiency of techniques developed in order to incorporate the history effect in the data library (such as time integration). In this paper we present a tool for reference core calculations in which the above stated approximations are eliminated.
This paper presents the solution method, its implementation, as well as the results of a demonstration calculation showing the improvement of the calculation results over the traditional approach, assessing the magnitude of history and parameterization effects importance.
The most important feature of the presented method is that it provides the perfect parameterization of macroscopic data, allowing the core physics code developers to understand sources of modeling uncertainties by completely removing the parameterization error (including, unlike other approaches, a complete representation of the history effects). It is therefore a highly important tool for improving the performance of core physics calculations with to an extent which was not previously available.

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PAPER 1106738
Development of a Generalized Cross Section Library Applicable to Various Reactor Types

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A method for generating a cross section library applicable to various reactor types is presented. In this method, an ultrafine group (2158 groups) cross section library is first prepared using NJOY and MC2-3, which includes absorption, nu-fission, and scattering resonance cross section tables as a function of the background cross section and temperature. Subsequently, for a specific reactor or reactor type of interest, this base ultrafine group library is condensed to broad-group cross section libraries using the group condensation optimization algorithm that minimizes the change of cross section and eigenvalue over different compositions and geometry. Based on equivalence theory, the escape cross sections representing the local heterogeneity effect are calculated by iteratively solving the fixed source problems in which resonance cross sections are updated during the iteration. Preliminary verification tests indicate that the base ultrafine group cross section library is able to accurately estimate eigenvalues and cross sections of various compositions from different reactor types including LWR, HTR, and SFR. The broad group cross section libraries condensed from the base ultrafine group cross section library for a specific reactor or reactor type show good agreements in eigenvalue with corresponding Monte Carlo solutions.

PAPER 1119837
Monte Carlo Analysis of Doppler Reactivity Coefficient for UO_2 Pin Cell Geometry

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Monte Carlo analysis has been performed to investigate the impact of the exact resonance elastic scattering model on the Doppler reactivity coefficient for the UO_2 pin cell geometry with the parabolic temperature profile. As a result, the exact scattering model affects the coefficient similarly for both the flat and parabolic temperature profiles; it increases the contribution of uranium-238 resonance capture in the energy region from ～ 16 eV to ～ 150 eV and does uniformly in the radial direction. Then the following conclusions hold for both the exact and asymptotic resonance scattering models. The Doppler reactivity coefficient is well reproduced with the definition of the effective fuel temperature (equivalent flat temperature) proposed by Grandi et al. In addition, the effective fuel temperature volume-averaged over the entire fuel region negatively overestimates the reference Doppler reactivity coefficient but the calculated one can be significantly improved by dividing the fuel region into a few equi-volumes.

PAPER 1125254
Assessment of the Depletion Capability in MPACT

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The objective of the paper is to develop and demonstrate the depletion capability with pin resolved transport using the MPACT code. The first section of the paper provides a description of the depletion methodology and the algorithm used to solve the depletion equations in MPACT. A separate depletion library for MPACT is used based on the ORIGEN-S library to provide the basic decay constants and fission yields, as well as the 3-group cross-sections which are used for the isotopes not contained in the MPACT multi-group library. The cross sections for the depletion transmutation matrix were collapsed using the transport flux solution in MPACT based on either the 47 group HELIOS library based on ENDF-VI or a 56 group ORNL library based on ENDF-VII. The second section of this paper then describes the numerical verification of the depletion algorithm using two sets of benchmarks. The first is the JAERI LWR lattice benchmark which had participants from most of the lattice depletion codes currently used in the international nuclear community and the second benchmark is based on data from spent fuel of the Takahama-3 reactor. The results show that MPACT is generally in good agreement with the results of the other benchmark participants as well as the experimental data. Finally, a full core 2D model of CASL AMA benchmark was depleted based on the central plane of the Watts Bar reactor core which demonstrates the whole core depletion capability of MPACT.

PAPER 1125817
Monte Carlo Modelling of VR-1 Reactor Core

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The possibilities of reactor core analysis by precise Monte Carlo codes are gradually increasing along with the accessibility of computing power. In the case of zero power research reactors, where temperature and burn-up effects remain negligible, model can approximate the reality to a very high degree. In such a case, most of calculation uncertainty can be caused by uncertainties in technical specifications of fuel and reactor internals. Thus performance of the modelling and its predictive power can be significantly improved via comparison with a large set of experimental data that can be acquired during reactor operation and via subtle tuning and improving the calculation model. The paper describes the case for neutronics calculations of VR-1 zero power reactor core.

PAPER 1126966
Assessment of the 2D/1D Implementation in MPACT

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The 2D/1D method is used in the MPACT code to obtain 3D solutions of the Boltzmann transport equation for practical reactor geometries. The OECD C5G7 transport benchmark problem is used first to assess the accuracy of the method with a fixed set of cross-sections. The VERA Core Physics Progression Problems are then used to compare the accuracy of the transport solver using a 56-group library based on ENDFB-VII.0. Single assembly PWR designs are simulated, and the eigenvalue and pin powers are compared to continuous-energy Monte Carlo results. A 3×3 assembly cluster with a control rod inserted into the center assembly is then compared to Monte Carlo to assess the ability of MPACT to predict a control rod worth curve. Finally, MPACT is used to simulate the initial critical states of a full 3D initial core of a PWR at zero power conditions.

PAPER 1127524
Coupled Neutronics and Thermal-Hydraulic Solution of a Full-Core PWR Using VERA-CS

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The Consortium for Advanced Simulation of Light Water Reactors (CASL) is developing a core simulator called VERA-CS to model operating pressurized water reactors (PWRs) with high resolution. This paper describes how the development of VERA-CS is being driven by a set of progression benchmark problems that specify the delivery of useful capability in discrete steps. As part of this development, this paper will describe the current capability of VERA-CS to perform a multiphysics simulation of an operating PWR at Hot Full Power (HFP) conditions using a set of existing computer codes coupled together in a novel method. Results for several single-assembly cases are shown that demonstrate coupling for different boron concentrations and power levels. Finally, high-resolution results are shown for a full-core PWR reactor modeled in quarter-symmetry.

PAPER 1127710
Assessment of the WIMS9A/PARCS/TRACE Code System for Power Density Calculations of the Westinghouse AP1000^{TM} Reactor

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The objective of this paper is to assess the accuracy of the WIMS9A/PARCS/TRACE code system for power density calculations of the Westinghouse AP1000^{TM} nuclear reactor, as a representative of modern pressurized water reactors (Gen III+). The cross section libraries were generated using the lattice physics code WIMS9A (the commercial version of the legacy lattice code WIMSD). Nine different fuel assembly types were analyzed in WIMS9A to generate the two-group cross sections required by the PARCS core simulator. The nine

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assemblies were identified based on the distribution of the discrete burnable absorbers (Borosilicate glass) and the integral fuel burnable absorbers (IFBA) in each fuel assembly. The generated cross sections were passed to the coupled core simulator PARCS/TRACE which performed 3-D, full-core diffusion calculations from within the US NRC Symbolic Nuclear Analysis Package (SNAP) interface. The results which included: assembly power distribution, effective multiplication factor (k_{eff}), radial and axial power density, and whole core depletion were compared to reference Monte Carlo results and to a published reactor data available in the AP1000 Design Control Document (DCD). The results of the study show acceptable accuracy of the WIMS9A/PARCS/TRACE code in predicting the power density of the AP1000 core and, hence, establish its adequacy in the evaluation of the neutronics parameters of modern PWR designs. The work reported here is new in that it uses, for the first time, the combination of WIMS9A/PARCS/TRACE codes to perform neutronics calculation for the Westinghouse' AP1000^{TM} reactor, as a representative of modern PWRs, with its challenging core configuration.

PAPER 1127773
Feasibility of Nodal Equivalence Theory Using Functionalized Discontinuity Factors

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The conventional Simplified Equivalence Theory (SET) is based on the single assembly lattice calculation with net-zero current boundary condition. However, SET has a problem that it can't reflect the actual node interface current on the Discontinuity Factor (DF) so that it shows discrepancy with the reference value. For more accurate reactor core analysis, the conventional SET was modified by introducing Functionalized Discontinuity Factors (FDFs). DFs are functionalized using lattice calculation result from several boundary conditions. The FDFs are updated during iterative whole-core calculation without compromising the computing time. The test calculation for one-dimensional slab geometry core showed that FDFs provided much accurate keff value that error reduced by up to 63%.

PAPER 1127819
The Multigroup Neutronics Model of NuStar's 3D Core Code EGRET

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As a key component of NuStar's core analysis system for PWR application, EGRET is designed to perform steady-state coupled neutronic/hydraulic analysis of PWRs. This paper presents EGRET's unique 3D nodal diffusion model and 2D pin power reconstruction (PPR) model. Unlike the practice in most of today's production codes that iteratively solves the global 3D coarse-mesh problem and the local axially 1D fine-mesh problem to handle the axial heterogeneity within a node caused by fuel grid and partially-inserted control rod, EGRET resolves the issue by inventing a new nodal technology and introducing the adaptive meshing technique to follow the movement of control rod tip. The new nodal method employs fine-mesh heterogeneous calculation with coarse-mesh transverse coupling such that the axial heterogeneous nodes can be explicitly modeled in exact geometry and directly incorporated into the scheme of transversely coupled coarse-mesh nodal methods. Each axial channel can have its own fine-mesh division without the need of dividing the whole core into radially coupled fine-meshes. There is no need to do 1D fine-mesh and 3D coarse-mesh iteration either. While for the PPR model, EGRET adopts a group-decoupled direct fitting method, which avoids both the complication of constructing 2D analytic multigroup flux solution and any group-coupled iteration. Another unique feature of the PPR model is that it fully utilizes all the information available from 3D core calculation into the downstream PPR process. Particularly, for the first time, the 1D profiles of transversely-integrated fluxes are utilized as the additional conditions to reconstruct pin power. Numerical results of series of benchmark problems verify the good performance of EGRET's unique multi-group neutronics model.

PAPER 1128004
Simulation of Watts Bar Initial Startup Tests with Continuous Energy Monte Carlo Methods

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The Consortium for Advanced Simulation of Light Water Reactors is developing a collection of methods and software products known as VERA, the Virtual Environment for Reactor Applications. One component of the testing and validation plan for VERA is comparison of neutronics results to a set of continuous energy Monte Carlo solutions for a range of pressurized water reactor geometries using the SCALE component KENO-VI developed by Oak Ridge National Laboratory. Recent improvements in data, methods, and parallelism have enabled KENO, previously utilized predominately as a criticality safety code, to demonstrate excellent capability and performance for reactor physics applications. The highly detailed and rigorous KENO solutions provide a reliable numeric reference for VERA neutronics and also demonstrate the most accurate predictions achievable by modeling and simulations tools for comparison to operating plant data. This paper demonstrates the performance of KENO-VI for the Watts Bar Unit 1 Cycle 1 zero power physics tests, including reactor criticality, control rod worths, and isothermal temperature coefficients.

Track2: Deterministic Transport Theory

PAPER 1039082
The Application and Performance of ACMFD Acceleration in 2D/3D Full Core MOC Transport Fuse Method

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It has been shown that the Analytic Coarse Mesh Finite Difference (ACMFD) method is very robust in nodal diffusion acceleration because of its rigorous derivation, and also should be applicable in core transport methods. In the past decade, the Method of Characteristics (MOC) was widely studied either in 2D or 2D/1D fuse cases with Coarse Mesh Finite Difference (CMFD). In this paper, the application of ACMFD in 2D or 2D/1D fuse MOC transport theory as an acceleration method is presented. Numerical result indicates that the performance of ACMFD is similar to CMFD.

PAPER 1083797
A Coupling Method of Subgroup and Wavelet Expansion for the Resonance Parameter Calculation

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Owing to their geometric flexibility, subgroup method and wavelet expansion method have become attractive approaches to obtain effective self-shielding microscopic cross sections within resonance energy groups for geometrically complex problems. However, the subgroup method is good in the dense resonance range, while the wavelet expansion method is good in the sparse resonance range. In order to get the resonance parameter in the whole resonance energy range more accurately and effectively, this paper developed a new coupling resonance calculation model based on subgroup method and wavelet expansion method. In this coupling model, the subgroup method is employed to handle the higher resonance energy groups, and the wavelet expansion method is employed to handle the lower resonance energy groups. At the coupling interface, they are coupled by transferring scattering source. In order to verify the coupling model, a series of benchmark problems are calculated in this paper. It is demonstrated that compared with subgroup method and wavelet expansion method respectively, this coupling resonance model has the ability to provide more exactly self-shielding microscopic cross sections in the whole resonance energy range while keeping enough efficiency.

PAPER 1104084
Verification of Ray Effect Elimination Module in the Transport Code ARES

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ARES is a multi-group of anisotropic scattering transport shielding code based on discrete ordinates method. The 3D radiation transport benchmark problems proposed by Kobayashi were calculated by ARES with sub-module ARES_RayEffect which using first collision method for ray effect mitigation. ARES_RayEffect calculates uncollided

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flux and first collision source moments for ARES. The uncollided flux is obtained by a ray tracing calculation between a source point and a target mesh center. In addition, ARES_RayEffect has a modifying factor function to improve the quality of uncollided flux calculation. For verification, the results of MCNP code are used as reference solution and the results of TORT with FNSUNCL3 are compared. ARES_RayEffect introduced the modifying factor to reduce the relative errors of meshes near the source region. For example, at the position (15,15,15) of Problem 1 case i, the relative difference of the result of ARES with ARES_RayEffect is -2.37%, while relative difference of the result of TORT with FNSUNCL3 is -11.95%. The calculated total neutron fluxes show good agreement with the MCNP solutions. For the pure absorber cases, the maximum differences are less than 3%. For the half scattering cases, the maximum differences are less than 11%. Numerical results demonstrate that ray effect can be effectively mitigated.

Track3: Monte Carlo Methods

PAPER 1084078
3-D Monte Carlo Neutron-Photon Transport Code JMCT and Its Algorithms

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JMCT Monte Carlo neutron and photon transport code has been developed which is based on the JCOGIN toolbox. JCOGIN includes the geometry operation, tally, the domain decomposition and the parallel computation about particle (MPI) and spatial domain (OpenMP) etc. The viewdata of CAD is equipped in JMCT preprocessor. The full-core pin-mode, which is from Chinese Qinshan-II nuclear power station, is design and simulated by JMCT. The detail pin-power distribution and keff results are shown in this paper.

PAPER 1099429
Monte Carlo Calculation of Neutron Generation Time in Critical Reactor and Subcritical Reactor with an External Source

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The neutron generation time Λ plays an important role in the reactor kinetics. However, it is not straightforward nor standard in most continuous energy Monte Carlo codes which are able to calculate the prompt neutron lifetime l_p directly. The difference between Λ and l_p are sometimes very apparent. As very few delayed neutrons are produced in the reactor, they have little influence on Λ. Thus on the assumption that no delayed neutrons are produced in the system, the prompt kinetics equations for critical system and subcritical system with an external source are proposed. And then the equations are applied to calculating Λ with pulsed neutron technique using Monte Carlo. Only one fission neutron source is simulated with Monte Carlo in critical system while two neutron sources, including a fission source and an external source, are simulated for subcritical system. Calculations are performed on both critical benchmarks and subcritical system with an external source and the results are consistent with the reference values.

PAPER 1099694
Neutron Channels Shield Design Analyses of KIPT Neutron Source Facility

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Argonne National Laboratory of the United States and the Kharkov Institute of Physics and Technology in Ukraine have been collaborating on design and construction of an experimental neutron source facility. It is an electron accelerator driven system (ADS) that utilizes a subcritical assembly driven by an electron accelerator. This facility will be used to conduct basic and applied nuclear research, produce medical isotopes, and train young nuclear specialists. The shield is designed to reduce the biological dose to < 0.5 mrem/hr outside its boundary during normal operation. Monte Carlo computer code MCNPX was utilized as the main design tool. In the shield design analyses, the radial heavy concrete shield was deemed to have a cylindrically symmetric geometry, to reduce the computation time of MCNPX. The required thickness of heavy concrete was determined to be 140 cm for this cylindrical shield geometry. Recently the radial shield design was finalized, with built-in neutron channels, beam shutters and plug-in shield, and the outer shield boundary is not cylindrical. MCNPX calculation were carried out to verify the performance of this finalized radial shield, with the neutron and photon doses calculated at selected locations outside the shield boundary. The previously developed neutron source file procedure was utilized for the neutron dose calculation, while the photon dose calculations started with the electron source particles. Due to the asymmetric geometry of the finalized radial shield, three dimensional weight windows were generated and utilized for both neutron and photon dose calculations. The calculated results show that the total radiation dose at the selected locations outside the radial shield is < 0.5 mrem/hr.

PAPER 1101864
Development of a New Convergence Criterion for Monte Carlo Simulation with Thermal-Hydraulics Feedback

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Coupled multi-physics approach plays an important role in improving computational accuracy. Compared with deterministic neutronics codes, Monte Carlo codes have the advantage of higher resolution level. In the present paper, a three-dimensional continuous-energy Monte Carlo reactor physics burnup calculation code, Serpent, is coupled with thermal-hydraulics safety analysis code, RELAP5. A new convergence criterion for the coupled simulation is developed based on the statistical uncertainty in power distribution in Monte Carlo code, rather than an arbitrarily chosen criterion in previous research.
The coupled simulation is based on the OECD-NEA/NRC PWR MOX-UO_2 Core Transient Benchmark. The convergence criterion of normalized axial power distribution is tested on both UO_2 and MOX single assembly models. Compared with previously implemented convergence criteria based on temperature, eigenvalue or flux (or power), it takes into account both the local and global convergence. It does not use a pre-set tolerance limit and is decided by the statistical accuracy of the Monte Carlo code itself. This new convergence criterion is shown to be stable, more stringent and direct, equally convenient to use but may need a few more steps to converge.

PAPER 1102407
JCOGIN: A Parallel Programming Infrastructure for Monte Carlo Particle Transport

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The advantages of the Monte Carlo method for reactor analysis are well known, but the full-core reactor analysis challenges the computational time and computer memory. Meanwhile, the exponential growth of computer power in the last 10 years is now creating a great opportunity for large scale parallel computing on the Monte Carlo full-core reactor analysis. In this paper, a parallel programming infrastructure is introduced for Monte Carlo particle transport, named JCOGIN, which aims at accelerating the development of Monte Carlo codes for the large scale parallelism simulations of the full-core reactor. Now, JCOGIN implements the hybrid parallelism of the spatial decomposition and the traditional particle parallelism on MPI and OpenMP. Finally, JMCT code is developed on JCOGIN, which reaches the parallel efficiency of 70% on 20480 cores for fixed source problem. By the hybrid parallelism, the full-core pin-by-pin simulation of the Dayawan reactor was implemented, with the number of the cells up to 10 million and the tallies of the fluxes utilizing over 40GB of memory.

PAPER 1104378
Void Transit Time Calculations by Neutron Noise of Propagating Perturbation Using Complex-Valued Weight Monte Carlo

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This paper focuses on the propagation of neutron noise induced

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by void formation in coolant/moderator moving upward in a reactor core within the limit of the first order perturbation. A new Monte Carlo method to solve the frequency domain transport equation of neutron noise observed by in-core neutron detectors has been proposed by adopting the complex-valued weight Monte Carlo technique. The technique has already been established by the author of the present paper to implement the B_1 approximation method into the Monte Carlo method. The newly-developed Monte Carlo method is compared with the conventional method based on the diffusion theory for neutron noise analyses. The Monte Carlo method makes a significant difference in neutron noise distribution as compared to the diffusion theory, suggesting that the transport theory should be introduced for accurate estimation of neutron noise in a reactor core. A numerical test is conducted to simulate the measurement of void transit time or void velocity in a reactor core by calculating a cross power spectral density between two in-core detectors.

PAPER 1105897
Simulating Fast Transients with Fuel Behavior Feedback Using the Serpent 2 Monte Carlo Code

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Simulating transients with reactivity feedback effects using Monte Carlo neutron transport codes can be used for validating deterministic transient codes or estimating for example the total deposited energy in a fuel rod following a known reactivity insertion in the system. Recent increases in computational power as well as developments in calculation methodology makes it possible to obtain a coupled solution for several aspects of the multi-physics problem in a single calculation. This paper describes the different methods implemented in Serpent 2 Monte Carlo code that enable it to model fast transients with fuel behavior feedback. The capability is demonstrated in a prompt critical pin-cell case, where the transient is shut down by the negative reactivity from rising fuel temperature.

PAPER 1105995
Status of Monte Carlo Code Development at UNIST

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This paper presents the status of Monte Carlo (MC) code development at Ulsan National Institute of Science and Technology (UNIST). UNIST Monte Carlo code named “MCS” has 3D whole core modeling capability. ENDF/B-VII.0 and ENDF/B-VII.1 nuclear cross section libraries are used as continuous energy cross section data. MCS can estimate not only the fast spectrum reactor but also the thermal spectrum reactor which requires thermal scattering capability. Two types of calculations are possible: criticality calculation and real time fixed source calculation. The criticality calculation ability is verified with selected cases of ICBEP benchmark problem, C5G7 benchmark, VENUS-2 benchmark, and Hoogenboom benchmark problems. The real time fixed source calculation ability is tested with noise analysis. Furthermore, Hybrid form of neutron transport solver and modified power iteration method are implemented in the MCS.

PAPER 1106262
Criticality Benchmarking of ANET Monte Carlo Code

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In this work the new Monte Carlo code ANET is tested on criticality calculations. ANET is developed based on the high energy physics code GEANT of CERN and aims at progressively satisfying several requirements regarding both simulations of GEN II/III reactors, as well as of innovative nuclear reactor designs such as the Accelerator Driven Systems (ADSs). Here ANET is applied on three different nuclear configurations, including a subcritical assembly, a Material Testing Reactor and the conceptual configuration of an ADS. In the first case, calculations of the effective multiplication factor (K_{eff}) are performed for the Training Nuclear Reactor of the Aristotle University of Thessaloniki, while in the second case K_{eff} is computed for the fresh fueled core of the Portuguese research reactor (RPI) just after its conversion to Low Enriched Uranium, considering the control rods at the position that renders the reactor critical. In both cases ANET computations are compared with corresponding results obtained by three different well established codes, including both deterministic (XSDRNPM/CITATION) and Monte Carlo (TRIPOLI, MCNP). In the RPI case, K_{eff} computations are also compared with observations during the reactor core commissioning since the control rods are considered at criticality position. The above verification studies show ANET to produce reasonable results since they are satisfactorily compared with other models as well as with observations. For the third case (ADS), preliminary ANET computations of K_{eff} for various intensities of the proton beam are presented, showing also a reasonable code performance concerning both the order of magnitude and the relative variation of the computed parameter.

PAPER 1123217
Effective Diffusion Homogenization of Cross Sections with the Monte Carlo Method

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Homogenizations of heterogeneous regions using Monte Carlo methods is one of the possible intermediate steps to transfer from the deterministic codes to a full scale Monte Carlo calculations of a large complex reactor systems. With the use of capable computer the generation of neutron homogenized multigroup cross sections using Monte Carlo method could be performed, which are later used in deterministic codes to provide neutron solution on a coarse mesh. In the Monte Carlo code Serpent two methods are used to obtain the homogenized cross sections. First method is the simplest homogenization method based on a flux and volume weighting (FVH) of cross sections. The second method is based on the B1 method. While the first method suffers in the presence of the strong absorbers the second method is applicable only for the cases with fissile materials. The implementation of an effective diffusion homogenization (EDH) method is proposed in this paper. With the use of the Serpent code and the EDH method coarse group cross sections for a sample problem are calculated. Obtained cross sections are tested against the determinist results obtained with the WIMSD code and against the B1 method within the Serpent code. Accuracy of the calculated cross sections is tested by calculating multiplication factor using diffusion approximation in GNOMER code and comparing the results to the reference values obtained with the MCNP code.

PAPER 1228305
Estimating Local In- and Ex-Core Responses within Monte Carlo Source Iteration Eigenvalue Calculations

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Local in- and ex-core responses are calculated by employing variance reduction within the Monte Carlo source-iteration scheme. This is done by employing the Direct Statistical Approach to search for an optimum trade-off between sampling the local response and sampling the fundamental mode. Superhistories are employed to improve the trade-off point. Realistic test problems are run that show a good agreement between the predicted and actual calculational figure-of-merits. For the sample problems treated, gains in efficiency over analog (i.e. without variance reduction) range from 1 - 2 orders of magnitude for in-core responses to many orders of magnitude for ex-core responses. An alternative way of finding the trade-off point using the classic adjoint flux formalism showed substantial differences for one of the problems.

Track4: Verification, Validation and Uncertainty Analysis

PAPER 1067489
Review of Neutronic Assessments Applied to Small Reactor Core Physics

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In its design division for material test reactors and research reactors, AREVA TA has to characterize these manufactured cores. This step is sequential with neutronics benchmarks associated with validation (standard Verification & Validation approach). The previous two points are embedded in core projects and can be run separately

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especially when experimental tests are foreseen for validation database enrichment. Methodological standard is given in order to match validation and benchmark process illustrated alongside with two specific items on critical research reactors (AZUR – JHR) and subcritical mock up (AZUR).

PAPER 1084893
A New Neutronics Analysis Code System for Fast Reactors and Validation

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A new neutronics analysis code system has been developed for detailed analysis of fast reactor cores. The code system is composed of a calculation code of effective cross sections, an assembly calculation code based on the method of characteristics, and a full core transport/diffusion calculation code. The validity of the code system is investigated by applying it to the prototype fast reactor Monju, and by comparing the calculation results with measured ones.

PAPER 1100864
Uncertainty Analysis of Delayed Neutron Fissile Material Assay Using a Genetic Algorithm

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An uncertainty analysis of the delayed neutron non-destructive assay method was conducted to explicitly define the accuracy with which plutonium content in an uncharacterized sample can be assessed. Perturbing various parameters allowed for an investigation of the sensitivity of this method to various nuclear data, and it was determined that the relative delayed neutron group abundances had the largest effect on the genetic algorithm. Specifically, for a sample containing ^{235}U, ^{238}U, and ^{239}Pu, irradiations in the thermal spectrum were shown to be more sensitive to ^{235}U and ^{239}Pu data, while irradiations in a fast spectrum were shown to be more sensitive to the ^{238}U data. The overall uncertainties of the mass estimates were 15%, 5%, and 30% for ^{235}U, ^{238}U, and ^{239}Pu, respectively. Finally, reducing the first delayed neutron group abundances by a factor of three as suggested by recent research reduced the overall uncertainties to 10%, 3%, and 20%.

PAPER 1102809
IR Approximation for Calculating Sensitivity and Uncertainty of PWR Cells by Taking Account of Self-Shielding Effect

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A new improved method has been developed for calculating sensitivity coefficients of neutronics parameters in pressurized water reactor cells relative to infinite dilution cross-sections by taking account of resonance self-shielding effect. In our paper, the IR approximation is used in order to get accurate results in both high and low energy groups. This method is applied to UO_2 and MOX fueled PWR cells to calculate sensitivity coefficients and uncertainties of eigenvalue responses. We have verified the improved method by comparing the sensitivities with MCNP code and good agreement is found. For uncertainty, the improved results are compared with TSUNAMI-1D, and demonstrate that the differences are caused by the use of different covariance matrix.

PAPER 1102810
Uncertainty Quantification of Reactor Kinetics Parameters Using JENDL-4.0 Covariance Data

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Nuclear data-induced uncertainties of the neutron generation time Λ, which is one of the important reactor kinetics parameters, and related quantities β_{eff} and β_{eff}/Λ are quantified with deterministic calculations. A simple calculation method, which is based on the ordinary perturbation theory, for Λ sensitivities is proposed. The covariance data given in the JENDL-4.0 library is used. Numerical calculations reveal that the uncertainties of Λ are not negligible in comparison with the β_{eff} uncertainties. In some fast neutron systems, uncertainties of Λ are larger than uncertainties of β_{eff} . We conclude that uncertainties of Λ should be properly considered when one discusses uncertainties of β_{eff}/Λ, which is one of measurable parameters during reactor physics experiments.

PAPER 1103703
Survey on Effect of Crystal Texture of Beryllium on Total Cross-Section to Improve Neutronic Evaluation in JMTR

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Neutronic evaluations in JMTR have been performed for irradiation tests by Monte Carlo method with thermal neutron scattering law, S(α, β), data for beryllium metal, etc, and the calculation accuracy of fast and thermal neutron fluxes are ± 10% and ± 30%, respectively. Analytical and experimental investigations to achieve higher calculation accuracy, especially for the thermal neutron flux up to the fast neutron flux level, have been performed to offer higher value data technically to the JMTR users. In order to investigate an effect of fabrication method of beryllium material on the calculation accuracy, total cross-sections of beryllium specimens were measured using KURRI-LINAC, and it was found that the total cross-section was different from the evaluated one, and depended on the crystal texture, etc. The S(α, β) data for beryllium metal was adjusted based on the measured data, and the applicability to the neutronic evaluation in the JMTR was verified.

PAPER 1103924
Uncertainty and Sensitivity Analysis for an OECD/NEA HTGR Benchmark with XSUSA

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Sensitivity and uncertainty (S/U) analyses with respect to uncertainties in microscopic neutron cross sections and fission yields are performed for HTGR fuel, of both pebble bed and prismatic types, with UO2 as initial fuel. This is done with the sampling-based XSUSA program, with criticality and depletion calculation sequences from the SCALE 6.1 code system. One specific goal is to provide a tool for systematic HTGR S/U analysis with respect to core physics in the framework of the "IAEA Coordinated Research Program on the High-Temperature Gas cooled Reactor Uncertainty Analysis in Modelling". As a basis, an HTGR fuel depletion benchmark initiated by U.S. NRC and OECD/NEA is chosen. Criticality and depletion calculations are performed for pebble bed and prismatic fuel arrangements in infinite lattice geometry. As sample size, a sufficiently high number is chosen to be able to determine not only uncertainties, but also the main contributors to these uncertainties. Uncertainties and importance indicators are determined for relevant quantities, such as multiplication factors, isotopic inventories, and few-group cross sections. For most of the output quantities, a very similar behavior is observed for pebble bed and prismatic fuel concerning uncertainties and sensitivities. Also, in many respects similarity to results for LWR fuel is obtained. The few-group cross sections obtained from the calculations with varied nuclear data can be used later on in steady-state and transient HTGR core simulations, as planned within the IAEA Coordinated Research Program.

PAPER 1104067
First Verification and Validation Steps of MENDEL Release 1.0 Cycle Code System

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For each new code system, verification and validation process is a need to prove the efficiency and accuracy of the calculated physical quantities.
MENDEL is the new CEA depletion code system, whose first release was done at the end of 2013. It offers iso-capacity with the already

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well-established DARWIN. MENDEL is the successor of DARWIN, and can be used as a stand-alone code system for reactor cycle studies to compute interest output quantities. MENDEL also provides its depletion solvers to both Monte Carlo TRIPOLI-4® and deterministic APOLLO3® transport code systems.
The purpose of this paper is to present the first contributions to MENDEL release 1.0 verification and validation process. This first release has been used with nuclear data coming from both JEFF-3.1.1 and ENDF/B-VII.1 nuclear data evaluations, and its results are compared either with experimental data, either with DARWIN results.

PAPER 1104316
Application of the GRS Method for Estimation of Uncertainties of LMFBR Type Reactor Physics Parameters with Taking into Account Macroscopic Experiments

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A number of recent studies have been devoted to estimation of errors of fast reactor nuclear-physics parameters by the GRS (Generation Random Sampled) method. The method is based on direct sampling input data resulting in formation of random sets of input parameters which are used for multiple neutronics calculations. Once these calculations are performed, a statistical processing of calculation results is carried out to determine the mean value and the variance of each calculation parameter of interest.
In this study the GRS full scale method was used for estimation of errors of calculation of nuclear-physics parameters (k-eff, power density, dose rate)for a perspective LMFBR type sodium-cooled fast reactor with taking into account different sets of integral and macroscopic experiments. The multiple neutronics calculations are performed by the nodal diffusion three-dimensional code TRIGEX and CONSYST/ABBN nuclear data system. The sampling of input data, nuclide nuclear cross-sections, is based on the ABBN-93 covariance matrices.

PAPER 1104670
Benchmarking of Photon and Coupled Neutron and Photon Process of SuperMC 2.0

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Super Monte Carlo Calculation Program for Nuclear and Radiation Process (SuperMC), developed by FDS Team in China, is a multi-functional simulation program mainly based on Monte Carlo (MC) method and advanced computer technology. This paper focuses on the benchmarking of physical process of photon and coupled neutron-photon of SuperMC2.0. Integral leakage rate of photon in the spherical and hemispherical shell experiment was tested to verify the physical process of photon and coupled neutron and photon transport. Vanadium assembly experiment and ADS benchmark were given as comprehensive benchmarks. The correctness was preliminarily verified by comparing calculation results of SuperMC with experimental results and MCNP calculation results.

PAPER 1104787
In Depth Uncertainty Estimation of the Neutron Computational Tools

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Verification & validation processes are required to demonstrate the ability of simulation tools to accomplish the task they have been designed for. The values of the characteristics of both 3^{rd+} generation PWR cores and 4th generation fast neutron reactor cores should be associated with a validation dossier.
The new APOLLO3® code, under development at CEA, has to undergo such validation process in particular when treating reactor cores with advanced features. In order to prepare this task, the way that validation has been done in the past for a code like ECCO is being revisited. The aim of this work is to illustrate the change in paradigm between what was done 25 years ago and what is currently done.
Two ways to quantify the bias introduced in deterministic calculations exist:

1. Performing deterministic calculations in which some approximations are eliminated or reduced.
2. Comparing deterministic calculations with stochastic ones (TRIPOLI4).

The first approach was the one being used for ECCO when programming it since it was accessible with computers available at the time but is not sufficiently independent from the tool being tested. The method could identify the origin of discrepancies with the use of the perturbation method so as to upgrade the method and/or track some possible compensating errors.
The second approach is often done by comparing integral results such as reactivity or sodium void worth obtained with deterministic codes with the one of Monte Carlo codes using continuous energy libraries. This approach is somehow lacking in depth understanding of the origin of discrepancies, hence is not very instructive for finding a way to improve the computational scheme. The use of the perturbation theory is applied in this paper between ECCO/ERANOS deterministic results and TRIPOLI4 Monte Carlo results for reactivity or sodium void worth of Sodium Fast Reactors (SFR) so as to illustrate this innovative approach.

PAPER 1105709
Updated Validation of the PSI Criticality Safety Evaluation Methodology Using MCNPX2.7 and ENDF/B-VII.1

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A validation of the PSI Criticality Safety Evaluation (CSE) methodology using the most recent evaluated nuclear data library ENDF/B-VII.1 is presented in this paper. The basis for the methodology is a well-defined benchmark suite of 149 low-enriched thermal compound uranium critical experiments contained in the International Handbook of Evaluated Criticality Safety Benchmark Experiments. The CSE validation of code/libraries is then focused on the evaluation of the lower tolerance bound along with trend analyses for both design parameters as well as nuclear spectrum related parameters. Here, the overall performance of the new ENDF/B-VII.1 library for these selected thermal systems is evaluated with MCNPX-2.7.0 and is found to be very similar to that obtained with ENDF/B-VII.0, confirming the existing literature. Moreover, no specific trends as function of design based parameters and/or spectral parameters are observed with the new library. However, due to some specific library changes, some distinct differences between the libraries are shown for certain groups of experiments the specific library changes. For completeness, the changes compared to MCNPX-2.5.0 are also assessed in order to address code effects and including in that framework, an evaluation of the impact from the upstream NJOY processing. Finally, a comparison of ENDF/B-VII.1 results against those obtained with other libraries, including JEFF-3.1.1, is also presented.

PAPER 1105825
Validation of Two Monte Carlo Codes for LWR Burnup Calculations

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This paper presents an assessment conducted at the Paul Scherrer Institut (PSI) of two Monte Carlo-based codes, namely Monteburns and Serpent, for LWR depletion calculations. First, a validation of predicted spent fuel compositions is presented on the basis of a selected Swiss spent fuel sample from the LWR-PROTEUS Phase II program. Second, an additional assessment is performed for an OECD/NEA burnup credit benchmark which is also related to isotopic predictions. In both cases, equivalent pin-cell models are applied and the results are presented in terms of relative isotopic concentrations between measured/benchmark and predicted values. Additional comparisons are also made to previously published PSI results using the MCNPX/CINDER and CASMO-4E codes. The results show that all codes provide an overall good agreement with experimental data although a few specific trends are observed for some nuclides which are pointed out and briefly discussed.

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PAPER 1106101
Benchmarking of DeCART2D against Critical Experiments

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The benchmark calculation against various critical experiments has been performed to validate the 2-dimensional whole core transport code DeCART2D and its multi-group cross section library. In addition, several multi-group libraries provided by our own library processing system are employed to investigate the dependency on both the version of the nuclear data library and the number of neutron energy groups. Although these experiments are 3-dimensional, DeCART2D is able to simulate these experiments directly with a consideration of the axial leakage, which can be defined as the multiplication of the diffusion coefficient with the measured geometrical buckling. The calculated solutions for a critical condition are compared with the measured data, which include the pin-by-pin fission rate distribution in the central assembly as well as the eigenvalue. The comparison results show that a discrepancy in the eigenvalue is highly different from the multi-group library, whereas there is little difference in the fission rate. In conclusion, it can be seen that the ENDF/B-VII.1 based multi-group library gives the most reasonable solutions.

PAPER 1106250
Automated Reactor Records Evaluation Framework

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The only truly reliable method for core physics code validation is comparison against experimental data – and for power nuclear reactors, the only reasonably acquirable kind of experimental data are the reactor records. However, the amount of the data coming from the reactor operation is often so vast that it can be discouraging for the code developers to use it properly. Thus, the validation package is further reduced because the data is hard to use.
This paper presents an elaborate, fully automated framework, which was designed and implemented in our institute, for reactor records processing and its use for core physics code validation. The workflow, implemented as a Web 2.0 application, provides a practical and painless solution for use of reactor records data for code development and validation.

PAPER 1109432
Monte Carlo Based Equilibrium Cycle Analysis of One-Dimensional Breed and Burn Benchmark Problem

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As part of initial neutronics codes verification and validation (V&V) efforts for Traveling Wave Reactors (TWR) at TerraPower, the equilibrium cycle analysis of the one-dimensional breed and burn benchmark problem was performed using Monte Carlo codes (MCNP-5.1.60 and MCNP-XT), the depletion code, ORIGEN-2.2, and a set of driver Python scripts. Several goals of this work are: 1) to establish the reference solution for the benchmark problem by series of sensitivity calculations verifying both the equilibrium cycle modeling methodology and base modeling input parameters; 2) to verify the MCNP-XT transport solver against MCNP-5 for depletion problems; 3) to demonstrate the reactor shutdown length effect. The computational benchmark problem consists of 100 one-dimensional fuel slabs of 5 cm thickness each. Two fuel shuffling sequences examined are the inward convergent shuffling and the convergent divergent shuffling. The equilibrium cycle modeling methodology was developed by iterating flux calculations and depletion calculations. Investigated sensitivities include the number of depletion slices in a fuel slab, the number of depletion steps per cycle, and the depletion step setting, i.e., fluxes and reaction rates at beginning of step (BOS), end of step (EOS), or average of BOS and EOS. Furthermore, the reactivity effect of the reactor shutdown length is found to be about +100 pcm affecting only beginning of equilibrium cycle (BOEC), which is caused by the holdup of Np239 inventory in the core.

PAPER 1126078
Verification of the Monte Carlo Code RMC with a Whole PWR MOX/UO_2 Core Benchmark

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Several types of V&V work are being carried out for the Reactor Monte Carlo code RMC, including the heterogeneous whole core configurations. In this paper, a whole PWR MOX/UO2 core benchmark which contains both UO2 and MOX assemblies with different enrichments and various burn-up points is chosen to verify RMC's criticality calculation capability, and the results of RMC and other codes are discussed and compared, such as eigenvalues, assembly power distributions, pin power distributions and so on. The discrepancies in eigenvalues and power distributions are satisfactory, which proves the accuracy of RMC's criticality calculation. Also, the influences of different cross-section libraries are discussed upon the results of RMC. Besides these results, the detailed comparisons between RMC and MCNP with the same ENDF/B-VII.0 cross-section library are carried out in this paper, including the comparisons of control rod worths calculated by both RMC and MCNP. According to the results, RMC and MCNP agree quite well in eigenvalues, power distributions and other results. The discrepancies of eigenvalues and control rod worth are fairly small and the relative differences of assembly and pin power distributions are acceptable. All these results contribute to the conclusion that the criticality calculation performance of RMC is accurate and excellent.

PAPER 1126219
An Improved Method for Inverse Uncertainty Quantification for Nuclear Data Assessment

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Safety analysis and design calculations depend on the precise prediction of various reactor attributes, e.g., multiplication factor and pin powers. Focusing on neutronics calculations, where nuclear data are known to constitute the major source of calculational uncertainty, the attributes prediction can be improved by reducing the uncertainty associated with the nuclear data. An inverse problem in tandem with a cost function that accounts for the cost of nuclear data measurements can be employed to measure and prioritize the contribution of the various nuclear data to the total propagated attributes uncertainties. The results of this study can help guide the experimental program focused on obtaining improved measurements of nuclear data.
This approach has already been proposed by others and appeared in the literature under the name of 'target accuracy assessment'; it has been primarily applied in the context of critical benchmark experiments used in support of fast reactor technology. The overarching objective of our work is to generalize the application of this approach to core-wide reactor models, expected to be high dimensional. As a first step towards this objective, this manuscript proposes an efficient inversion technique for solving the associated optimization problem, expected to be infeasible with the current direct approach due to the explosion in the dimensionality of nuclear data. In particular, we employ a subspace-based reduced order modeling algorithm for solving the optimization problem which constrains the search to a small subspace in order to render the search computationally feasible. A quarter PWR fuel assembly is employed to assess the performance of the proposed algorithm, and compare it to conventional optimization techniques that search the full space of nuclear data.

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PAPER 1127444
On How Sensitive the Cross-Section Sensitivity Calculations Are to P_N Order Approximations

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In the mathematical expression of the nuclear data sensitivity function as derived from the first order perturbation theory both the angular fluxes and their expansion in the Legendre flux moments are required. Both terms are calculated by the traditional discrete ordinates transport codes and stored in the files. This leaves the developers of the nuclear data sensitivity and uncertainty codes several possible choices of the method to account for the neutron flux anisotropy. Three possibilities available in the SUSD3D code were studied here and compared using the FLATTOP-Pu benchmark exercise. This benchmark was particularly suitable due to its high flux anisotropy.

PAPER 1137713
Validation of NuStar's PWR Core Analysis System

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A new PWR core analysis system has recently been developed in NuStar. It adopts not only conventional theoretical framework that widely used in today's production code, but also latest progress that achieved internationally and within the company. To validate the system, a total of 44 operation cycles of Qinshan Phase 1 and 2 reactors are evaluated and results are compared against the measurement data. Statistics analyses of prediction errors for both low-power startup physics test states and high-power normal operation conditions demonstrate that the development of the system is quite a success. It not only has a generic applicability for reactors with different design, but also has a very good accuracy for parameters that are measurable at the site. Among all the validated cases, only for very rare cases that the acceptance tolerance is violated, and the error distributions for all the validated parameters present to be good ones that are close to the normal distribution. Based on the validation results, the system is considered to be qualified for practical applications.

Track6: Reactor Physics Experiments

PAPER 1085573
Determination of the ^{58}Ni (n,p) ^{58}Co Reaction Cross Section for both Ground and Isomeric States

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A nickel (Ni) activation detector is used to determine fast neutron fluence in nuclear reactors via the ^{58}Ni (n, p) ^{58}Co reaction. Proper evaluation of ^{58}Co-induced activity requires the use of correction factors, especially the burn-up of product atoms during irradiation. The irradiation of ^{58}Ni with fast neutrons produces ground and excited isomeric states of ^{58}Co. This article discusses how to determine the k-ratio between the cross section of ground and exited states. The ^{58m}Co isomer has a very high thermal neutron absorption cross section and a short half-life compared to the ground state. The theoretical solution of differential equations for changes in the concentration of a particular nuclide and experiments performed at the LVR-15 research reactor served to determine the production of ^{58}Co in its ground and excited states. The experiments were designed so that the Ni activation detectors were irradiated without shielding and under the same conditions as with cadmium (Cd) shielding. Cd shielding captures thermal neutrons and thus prevents the burn-up of ^{58m}Co and ^{58}Co products. The correct k-ratio between the cross section of ground and exited states will provide the burn-up correction factor for ^{58m}Co and ^{58}Co products. The burn-up factor allows more accurate determination of fast neutron fluence.

PAPER 1086633
Investigation on Subcriticality Measurement Using Inherent Neutron Source in Nuclear Fuel

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The subcriticality measurement techniques using inherent neutron source are investigated in this study. In facilities where nuclear fuel is treated, it is considered to install a real-time subcriticality monitoring system. However, it is often unfavorable or impractical to bring an external neutron source into a nuclear facility to carry out subcriticality measurement. Thus, we focus on the inherent neutron source in nuclear fuel such as spontaneous fission and (α, n) reaction. By utilizing the inherent neutron source, the subcriticality measurement technique, which does not introduce an external neutron source, can be realized. In this study, the Feynman-α method and the neutron source multiplication (NSM) method are chosen as techniques which can be carried out only with the inherent neutron source, and have affinities with a real-time monitoring system. In the Kyoto University Critical Assembly (KUCA), subcriticality measurement experiments of these two techniques were carried out. From the experimental result, it is confirmed that both techniques can be carried out only with the inherent neutron source. It is noted that, however, measured values of subcriticality by both methods show different trends. While the subcriticality measured by the Feynman-α method is fairly constant among detectors, the measured value does not agree with that obtained by the pulsed neutron source (PNS) method, in deep subcritical systems. On the other hand, the subcriticality measured by the NSM method show significant dependence on detectors. However, the measurement values by the NSM method using detectors at appropriate position agree with values by the PNS method even in deep subcritical systems.

PAPER 1094641
Benchmark Calculations of Sodium Fast Critical Experiments

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The high expectations from fast critical experiments impose the additional requirements on reliability of final reconstructed values, obtained in experiments at critical facility. Benchmark calculations of critical experiments are characterized by impossibility of complete experiment reconstruction, the large amounts of input data (dependent and independent) with very different reliability. It should also take into account different sensitivity of the measured and appropriate calculated characteristics to the identical changes of geometry parameters, temperature, and isotopic composition of individual materials. The calculations of critical facility experiments are produced for the benchmark models, generated by the specific reconstructing codes with its features when adjusting model parameters, and using the nuclear data library. The generated benchmark model, providing the agreed calculated and experimental values for one or more neutronic characteristics can lead to considerable differences for other key characteristics. The sensitivity of key neutronic characteristics to the extra steel allocation in the core, and ENDF/B nuclear data sources is performed using a few calculated models of BFS-62-3A and BFS1-97 critical assemblies. The comparative analysis of the calculated effective multiplication factor, spectral indices, sodium void reactivity, and radial fission-rate distributions leads to quite different models, providing the best agreement the calculated and experimental neutronic characteristics. This fact should be considered during the refinement of computational models and code-verification purpose.

PAPER 1104329
Critical Experiments for BWR Fuel Assemblies with Cluster of Gadolinia Rods

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Gadolinia-bearing fuel rods are needed for high-burnup fuels. Strong neutron absorption of gadolinia makes an assembly heterogeneous from the viewpoint of reactor physics. The cluster of gadolinia-bearing fuel rods is useful for higher-burnup fuels than current fuels.

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Few critical experiments have been reported for fuel assemblies with the cluster of gadolinia-bearing fuel rods. We conducted critical experiments for BWR fuel assemblies with the cluster of gadolinia-bearing fuel rods in the Toshiba Nuclear Critical Assembly (NCA). Critical water level and power distribution were measured. Measurements were compared with analyses by a continuous-energy Monte Carlo code, MCNP, with the JENDL-3.3 nuclear data library.

PAPER 1104817
The Calculation and Measurement of Fast Neutron Reflection in the VVER-1000 Mock-Up Model Placed in the LR-0 Reactor

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The paper attempts to address the issue of the effect of biological shielding simulator on neutron transport in its vicinity. Neutron reflections from the biological shielding model may worsen the radiation conditions in the end part of the pressure vessel. A similar problem emerges in connection with in the design of transport experiments, where the reflections may distort the effect of transmitted material on transport.
The experiments were conducted in the VVER-1000 mock-up placed in the LR-0 reactor whose modular character allows extraction of its central part. The spectra were observed in conditions of full biological shielding as well as of its modification (in which case the central part of the biological shielding on was removed). The neutron spectra were measured with the use of proton recoil method. In neutron energy range below 1MeV the hydrogen spherical proportional counters (diameter of 40 mm) with various pressures of hydrogen were used, while in the case of over 1MeV scintillation detector with 10 x 10 mm cylindrical stilbene detector was employed. The results of the experiment were compared with calculations. The calculations were performed with ENDF/B-VII.0, JEFF-3.1. JENDL-3.3, JENDL-4.0, ROSFOND 2009, and CENDL-3.1 nuclear data libraries.
On the basis of the resulting calculations and experiments, we may conclude that neutron reflections from the biological shielding increase the neutron flux density in the vessel. Generally, the calculation overvalues the effect of reflections. In the case of the back concrete reflector addition, the increase in neutron flux density in 0.1-0.2 MeV was measured as 18 % while calculational result is 29 %.

PAPER 1105052
Development of Reactivity Meter with Novelty Neutron Source Intensity Evaluation Model for BWR Application

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The Ohma nuclear power plant has a plan to be the world's first fully MOX-operated plant. Therefore, it is important to evaluate some key parameters representing core characteristics (e.g. control rod worth (CRW), moderator temperature coefficient (MTC), etc.) through some nuclear reactor physics experiments from the view point of keeping nuclear design reliability. With respect of BWRs (Boiling Water Reactors), Positive Period Method which forces operators to take complicated manipulation for core operation and to take long time has been widely used for the evaluation of above mentioned parameters. On the other hand, a reactivity meter which can evaluate those parameters in short time without complicated manipulation has a fewer experiences in comparison with PWRs (Pressurized Water Reactors) due to the difficulty of input parameters setting (e.g. neutron source intensity). In this paper, a new concept of reactivity meter equipping a novelty neutron source intensity evaluation model is proposed. Applying proposed reactivity meters to BWRs helps operators to obtain high accuracy of core key parameters with much simpler test steps and data analyses.

PAPER 1106265
Boron Carbide Neutron Screen for GRR-1 Neutron Spectrum Tailoring

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The presence of fast neutron spectra in new reactor concepts (such as Gas Cooled Fast Reactor, new generation Sodium Cooled Fast Reactor, Lead Fast Reactor, Accelerator Driven System and nuclear Fusion Reactors) is expected to induce a strong impact on the contained materials, including structural materials (e.g. steels), nuclear fuels, neutron reflecting materials (e.g. beryllium) and tritium breeding materials (for fusion reactors). Therefore, effective operation of these reactors will require extensive testing of their components, which must be performed under neutronic conditions representative of those expected to prevail inside the reactor cores when in operation. Depending on the material, the requirements of a test irradiation can vary. In this work preliminary studies were performed to observe the behavior of the neutron spectrum within a boron carbide neutron screen inserted in a hypothetical reflector test hole of the Greek Research Reactor. Four different screen configurations were simulated with Monte Carlo code TRIPOLI-4. The obtained data showed that the insertion of boron carbide caused not only elimination of the thermal ($E < 1$eV) component of the neutron energy spectrum but also absorption of a considerable proportion of the intermediate energy neutrons ($1 \cdot 10^{-6}$ MeV $< E < 1$ MeV).

PAPER 1106850
Research and Development Activities for Transmutation Physics Experimental Facility in J-PARC

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The Japan Atomic Energy Agency (JAEA) has the plan to construct Transmutation Physic Experimental Facility (TEF-P) under a framework of J-PARC (Japan Proton Accelerator Research Complex) project. TEF-P is a critical assembly which can load Minor Actinide (MA) fuels to perform reactor physics experiments for transmutation systems such as Accelerator-Driven System (ADS) or Fast Reactor (FR). The facility can also use proton beam from the J-PARC accelerator to investigate the controllability of ADS. Current status and activities for TEF-P are described.

PAPER 1108237
Analysis of TCA Criticality, βeff and βeff/l Using CASMO-4 and CASMO-5

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MNF and AREVA have been developing the BWR core calculation system CASMO-4/MICROBURN-B2. This code system has been modified to treat the ATRIUM 10XM bundle characteristics and satisfy Japanese regulatory requirements. In this system, the cross section data and the kinetics parameter are supplied by CASMO-4 code. The accuracy of the treatments for nuclear criticality and transient events are very important for the reload core design and the verification and the validation on these items are requested by the regulatory agencies. MNF and AREVA calculated the neutron multiplication and the βeff transient parameter using CASMO-4 and CASMO-5. MNF compares the difference between CASMO-4 and CASMO-5 for both of these parameters. The results of these calculations demonstrate excellent agreement with differences of 0.13% in CASMO-4 and 0.19% in CASMO-5. Using the importance I value of the CASMO-5 calculation for the β eff measurement geometry, MNF confirmed that the βeff and βeff/l which were estimated by CASMO-4 β value and CASMO-5 importance I were in good agreement.

PAPER 1126731
Measurement of Subcriticality Using Delayed Neutron Source Combined with Pulsed Neutron Accelerator

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A new experimental method for subcriticality measurement was developed by using delayed neutron source which is produced by

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external pulsed neutron source to increase accuracy of measured results by overcoming the space dependency problem which means difference of measured results in different detector position and often appeared in almost all other subcriticality measurement techniques. Experiments were performed at Kyoto University Critical Assembly (KUCA) combined with a DT accelerator to produce pulsed neutron in outside of the core repeatedly. In this method, neutron detection counts in the prompt neutron time region which are appeared just after injection of pulsed neutron are omitted, whereas neutron counts in the delayed neutron time region which are appeared after disappearance of exponential decay of the prompt neutron are adopted in analysis based on neutron source multiplication method or neutron noise analysis method; the variance to mean ratio method. In the delayed neutron time region, neutron sources to initiate fission chain reactions in subcritical state are delayed neutrons from delayed neutron precursors which are mainly produced by fission chain reactions in the prompt neutron time region, and delayed neutron precursors exist only in the fuel region, which makes possible to decrease the space dependency problem. The obtained results were compared with conventional pulsed neutron method, and it was found that the space dependency problem in subcriticality measurement can be fairly decreased by using the present new method compared with conventional one.

PAPER 1127355
Rossi-α Parameter Measurement of Dalat Nuclear Reactor by Analysis of Cross Power Spectral Density Obtained from 2 Ion Chambers

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Measurement and analysis of reactor power fluctuations from the compensated ion chambers (CIC) placed in the reactor are a powerful method in experimental study of nuclear reactor. This method allows us to determine the important kinetic parameters such as Rossi-α parameter (prompt neutron decay constant), reactivity, square module of reactor transfer function, effective delayed neutron fractions and neutron generation time, etc.
In this report, the authors present basis on theory, system of equipment used to measure power spectral density (PSD) and cross power spectral density (CPSD) of signals obtained from 2 KHK-56 ion chambers placed in the Dalat nuclear reactor. By fitting of the theoretical curves with the measured curves, Rossi-α parameter was determined.

PAPER 1127688
Analysis of Integral Experiment for Thorium Fuel Cycle at Kyoto University Critical Assembly

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To measure integral neutronics characteristics of thorium loaded core, critical experiments had been carried out at Kyoto University Critical Assembly (KUCA). The critical experiments were performed with various neutron spectra and thorium inventories. The thorium loaded core has two regions which are a test zone and a driver fuel zone. The test zone consists of thorium plates and graphite plates. In order to change the neutron spectrum of the experimental neutron field systematically, the graphite/Th-232 ratio at the test zone had been systematically varied by changing the combination of the thorium plates and the graphite plates in a unit cell. In this study, the criticalities of thorium loaded core were analyzed by MVP2.0 with JENDL-4.0, JENDL-3.3. In addition, sensitivity analyses were performed by SAGEP code and uncertainties of the numerical results were evaluated by using cross section covariance matrix.

PAPER 1127907
Studies of Potential for Conversion of the Idaho National Laboratory TREAT Transient Test Reactor to Low-Enrichment Fuel

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In light of a potential restart of the Transient Reactor Test Facility at the Idaho National Laboratory, the conversion of the existing high-enrichment uranium (HEU) fuel to a low-enrichment uranium (LEU) equivalent is being studied. This paper describes the development and validation of the methods to simulate the performance o f the existing HEU core. Experimental data from critical configurations and transient measurements have been used to develop both steady state and transient models and corresponding simulation approaches. In this process, the uncertainties of the core parameters have been identified and potential biases established. Modeling of the HEU core has largely been completed and is now being applied to the assessment of candidate LEU fuel and core designs.

Track7: Reactor Concepts and Designs

PAPER 1068764
Axially Heterogeneous Thorium Fuel Designs for Transuranic Burning in Reduced-Moderation BWRs

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Reduced-moderation Boiling Water Reactors (RBWRs) can allow sustained burning of transuranics (TRU), such that full actinide recycle can be achieved. However, the void coefficient (VC) tends to become positive with reduced moderation and high TRU loading, which can severely limit the design feasibility and performance. This motivates use of thorium (Th) as an alternative feed to uranium (U), as this tends to result in a more negative VC, leading to much improved neutronic performance. While axially homogeneous fuel design is preferable for ease of fuel fabrication, it is valuable to evaluate axially heterogeneous fuel designs to see if improved neutronic performance can be realised, which could lead to a reduction in fuel reprocessing and fabrication throughput. Multi-recycled Th-TRU fuel contains Th, U, Pu and MAs, leading to a wide range of possible fuel designs. Axially heterogeneous designs are considered using 3D pincell calculations using the Monte Carlo code Serpent. Radially heterogeneous assembly designs are considered in a companion paper. Spatial separation of Th-TRU and Th-U3 into regions of the order of a few thermal neutron diffusion lengths greatly improves neutronic performance. This can be accomplished radially or axially, but radial separation results in significantly easier fuel fabrication. Axial seed-blanket heterogeneity improves neutron economy at the expense of high power peaking, such that the radially heterogeneous assembly design is preferred. Separation of Th-TRU and Th-U3 into larger regions is not effective, due to increased power density in the Th-TRU region with voiding and with burn-up, leading to a more positive VC. It is therefore concluded that there is no motivation to pursue axially heterogeneous Th-TRU RBWR burner designs.

PAPER 1090085
Method Development and Reactor Physics Data Evaluation for Improving Prediction Accuracy of Fast Reactors' Minor Actinides Transmutation Performance

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The Ministry of Education, Culture, Sports, Science and Technology in Japan has launched a national project entitled "technology development for the environmental burden reduction" in 2013. The present study is one of the studies adopted as the national project. The objective of the study is the efficient and safe transmutation and volume reduction of MAs with long-lived radioactivity and high decay heat contained in HLW in sodium cooled fast reactors. We are aiming to develop MA transmutation core concepts harmonizing MA transmutation performance with core safety and to improve design accuracy related to MA transmutation performance. To validate and improve design accuracy of the high safety and high MA transmutation performance of SFR cores, we develop methods for calculating the transmutation rate of individual MA nuclides and estimating uncertainty of MA transmutation by using burnup sensitivity. Preliminary results are shown for the methods. Also we show various measured reactor physics data to be used to reduce the uncertainty of MA transmutation calculations. In the present study the overall consistency of the measured data is investigated by evaluating the usefulness of conventional static data as well as those related to MA transmutation obtained from various facilities like Monju, Joyo, FCA, BFS and PFR.

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PAPER 1094719
A Long Life Sodium Cooled Fast Reactor Concept with Radial Shuffling

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This paper proposes an idea of designing a long-life sodium cooled fast reactor. The reactor aims to be with commercial scale of 1000MWe and without too much challenge on the materials and operational conditions. The long-life is realized based on the breeding and burning strategy, in which the periodical radial shuffling is adopted. The one-through fuel cycle is adopted for nuclear non-proliferation, therefore, only the uranium fuel is loaded as the seed fuel and no plutonium is recycled.The design of seed assemblies is similar to the current sodium cooled fast reactor using metallic fuel. The enriched uranium-zirconium alloy with enrichment of 11.5% is used as the fuel. The depleted uranium from the tails in enriching processes is used in the blanket. An inward-convergent shuffling scheme is proposed. In the scheme, the assemblies in the outer blanket region are first moved to the inner blanket region and then shuffled to the burning region while the assemblies in the burning region are shuffled inward ring by ring and then skip to the outer blanket region. Preliminary calculations had proved that the shuffling scheme is feasible and the discharged blanket assemblies can be used as driver assemblies in another reactor with similar core design. The core life can reach over 38 years with a shuffling period of 2 years. The key parameters of the core are within the acceptable range in current fast reactor designs.

PAPER 1099566
Critical Boron Concentration Reduction Method in a Core Design

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A systematic core design method is developed to determine Gd-bearing fuel assembly having two types of Gd rods, low wt% of Gd rod and high wt% of Gd rod. The purpose of the method is to lower the critical boron concentration (CBC) of the preliminary core loading pattern (PLP), and consequently to achieve more negative or less positive moderator temperature coefficient (MTC). In this method, both the ratio of the number of low-Gd rod to the number of high-Gd rod (r) and assembly average Gd wt% (w) are the decision variables. The target function is the amount of soluble boron concentration reduction, which can be converted to Δk_{TARGET}. Δk-tables parameterizes k-infinite difference (Δk) between assembly in PLP and new designed assembly with the decision variables. The Δk-tables relating (r,w) to Δk is generated prior to the determination of Gd-bearing fuel assembly pattern using the least square method. The constraints required to determine a set of Δk are physically realizable pair, (r,w), and the sum of Δk of new designed assemblies as close to Δk_{TARGET} as possible. A set of new designed assembly Δk is determined to satisfy the target function expressed as Δk_{TARGET}. New Gd-bearing assemblies selected based on valid pairs of (r,w) are replaced with existing assemblies in a PLP. This design methodology is applied to Shin-Kori Unit 3 Cycle 1 used as a reference model. Shin-Kori Unit 3 is APR 1400. CASMO-3 is used to formulate Δk-tables and MASTER for depletion calculation. CASMO-3/MASTER calculations with new designed assemblies produce lower CBC than the expected CBC, proving that the proposed method works successful.

PAPER 1104423
BigT - A New Burnable Absorber Concept for PWR

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A new burnable absorber concept for Pressurized Water Reactor (PWR) is presented in this paper. The proposed concept, named “Burnable absorber-Integrated control rod Guide Thimble” (BigT), requires minor modifications on standard control rod guide thimble in existing PWR fuel assemblies. The BigT offers various design flexibilities such that any loading pattern and core management objective may potentially be met. Few technical challenges are expected with the BigT, but none is a show-stopper. Preliminary lattice analyses of the BigT absorbers installed in Westinghouse 17x17 and Combustion Engineering 16x16 fuel assemblies show that the proposed concepts perform as well as the commercial technologies in term of lattice reactivity and power peaking management. In addition, sufficiently high worth of control rods are obtained with the BigT in place. Since only slight modifications are required to the existing PWR lattice, the BigT can easily and readily be retrofitted into commercial fuel assemblies. The capability of replacing the BigT during refueling enables installation of fresh burnable absorbers in any assembly at beginning-of-cycle. This in turn offers a promising solution to realizing a low critical boron concentration core, and potentially even a soluble boron-free PWR. All neutronic simulations in this study were performed using Monte Carlo SERPENT code with ENDF/B-VII.0 library.

PAPER 1104448
Preliminary Design of a Spherical Breed/Burn Reactor

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A breed/burn reactor (BBR) is a reactor that through transmutation process can burn depleted uranium (or other fertile material as thorium) with only a small amount of fissile material. The advantages of BBR over other type of nuclear reactors are wide, taking into account its long life operation without refueling. In this paper we present the simulation results done with MCNPX code and JEFF-3.1 cross sections library at 1200 K. The purpose of this study is to compare two different types of BBR: a spherical core and a cylindrical core, with the same nuclear fuel in both reactors. We found that both reactors maintained a supercritical state during 4500 days (12.3 years) without refueling. The plutonium production was achieved along the reactor in both cases. We also find that the spherical reactor operated as a stationary wave reactor, where the neutron flux uniformly increases in the sphere along the breeding zone with time, meanwhile the cylindrical reactor operated as a traveling wave reactor where neutron flux propagates as a wave in the cylinder along burnup. Further analysis will include Th-U fuel cycles and a more detailed model.

PAPER 1104495
Optimization of Ultra-Long Cycle Fast Reactor Core

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An optimization of an ultra-long cycle fast reactor (UCFR) design with a power rate of 1000 MW (electric), UCFR-1000, has been performed to increase the safety of UCFR. Firstly, geometric optimization has been performed to decrease its peaking factors so that the peak temperatures measured by thermal hydraulic feedback are within the limit of design basis event (DBE). Secondly, fuel composition optimization has been performed by adopting Pressurized Water Reactor (PWR) spent fuel as a blanket material instead of natural uranium. Lastly, a small-size UCFR with a power rate of 100 MWe, UCFR-100, has been proposed for developing a short term deployable nuclear reactor. The major optimization process for UCFR-100 is decreasing maximum neutron flux and fast neutron fluence.
The optimized UCFR-1000 has been enlarged radially and shortened axially from the initial UCFR design and this modification makes the burning speed of active core movement slower. It has been confirmed that a full-power operation of 60 years without refueling is feasible for both UCFR-1000 and UCFR-100 core designs by a breed-and-burn strategy. By the design optimization study, the reductions of maximum neutron flux, fast neutron fluence, and axial power peaking have been achieved, which are favorable for the safety of the UCFR.

PAPER 1106189
The Main Characteristics of the Evolution Project VVER-S with Spectrum Shift Regulation

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The researches on the increase of achievable fuel burn-up in light water reactors (LWR) with so-called «spectrum shift» have been carried out in the world for a long time. The main idea of the «spectrum shift» is based on neutron spectrum shifting from the resonance energy region at the beginning of the cycle to the thermal region at the end of the cycle. There are many different ways of such regulation in the core – starting from coolant density variation during reactor cycle, to changing water-uranium ratio with some mechanical equipment. Spectrum shift reactivity regulation with zirconium displacers is presented in this paper for uranium and mixed uranium and plutonium oxide (MOX) fuel loadings for modernized reactor core VVER-S with new fuel assemblies with special guide tubes for displacers, developed for these purposes.

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PAPER 1127032

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

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The conceptual design of a Multipurpose Experimental Sodium cooled Fast Reactor (MESOF) based on U-Zr driver fuel was proposed at the previous study. Core performance characteristics and safety features are evaluated for alternative cores with U-TRU-Zr fuel and UO_2 fuel. All nuclear design calculations were done with TRANSX / DANTSYS / REBUS3(with DIF3D module) system, except effective delayed neutron fraction (β_{eff}) which was evaluated with MCNPX.
The inherent safety feature is proven by negative sodium void worth for all cores. The shutdown margin of U-TRU-Zr core is enough to control large excess reactivity at BOC, but the safety feature for UO_2 core will be evaluated in later study.

PAPER 1127409

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

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The Pressure-Tube Supercritical-Water Cooled Reactor (PT-SCWR) is one of the reactor concepts investigated by the Generation-IV International Forum as a reactor concept capable of achieving an overall nuclear power plant efficiency in the vicinity of 50%, higher than the approximately 35% that characterizes current nuclear power plants. The development process includes studying the neutronic methods that can provide an adequately-accurate modelling of the core. The PT-SCWR evolved from the conventional pressurized heavy-water reactor but presents a more heterogeneous core because of the much higher discharge burnup (~40 MWd/kg vs. ~7.5 MWd/kg) and greater variation of coolant density. The latest PT-SCWR concept features re-entrant fuel channels, each consisting of a pressure tube lined by a ceramic thermal insulator, and a central inner flow-tube. The fuel assembly is located in the space between the ceramic insulator and the central flow-tube. The re-entrant channel configuration presents unique coolant voiding characteristics because of the separate central and outer coolant regions. There may be the possibility, under certain postulated accident conditions, that the coolant in the region between the flow-tube and insulator can void before the coolant in the flow-tube. This work presents preliminary calculations of the resulting coolant-void reactivity for the case when only the outer coolant voids as well as for the case when the entire coolant in the channel voids. Lattice as well as full-core calculations are performed. Results show that voiding of only the outer coolant region yields a small, ~2 mk, positive reactivity, while the subsequent complete coolant voiding induces a large (>20 mk) negative reactivity.

Track8: Reactor Operation and Safety

PAPER 1092230

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

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This paper presents an introduction to the development and preliminary verification of a new on-line core monitoring software system (CMSS), named SOPHORA, for fixed in-core detector (FID) system of PWR. Developed at China General Nuclear Power Corporation (CGN), SOPHORA integrates CGN's advanced PWR core simulator COCO and thermal-hydraulic sub-channel code LINDEN to manage the real-time core calculation and analysis. Currents measured by the FID are re-evaluated and used as bases to reconstruct the 3-D core power distribution. The key parameters such as peak local power margin and minimum DNBR margin are obtained by comparing with operation limits. Pseudo FID signals generated by data from movable in-core detector (MID) are used to verify the SOPHORA system. Comparison between predicted power peak and the responding MID in-core flux map results shows that the SOPHORA results are reasonable and satisfying. Further verification and validation of SOPHORA is undergoing and will be reported later.

PAPER 1105990

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

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This paper presents the development of risk monitor software RiskAngel at FDS Team and its applications as a plant specific risk monitor, which supports risk-informed configuration risk management for the two CANDU 6 units at the Third Qinshan Nuclear Power Plant in China. It also describes the regulatory prospective on risk-informed PSA applications and the use of risk monitor at operating nuclear power plants, high level technical and functional requirements for the development of CANDU specific risk monitor software, and future development trends.

PAPER 1127006

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

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Additional instrumentation tubes (neutron detectors, thermocouples) are scheduled for deployment in Texas A&M University's TRIGA core in order to collect measurements for multiphysics code validations. This instrumentation will displace water and reduce coolant flow in some areas between fuel bundles. Thermal-hydraulic analyses are carried out using COBRA-TF and STAR-CCM+ to analyze the change in cooling capabilities. Results show that the coolant exit temperature increases by ~9°C and that DNBR margins are not affected. Additionally, since the instrumentation tubes act as flow straighteners, an overall reduction in hot spots is observed. The study also provided insight into the comparison of COBRA-TF with the higher fidelity code; STAR-CCM+.

Track9: Transient and Safety Analysis

PAPER 1084717

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

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Efforts are going on both at the CEA Cadarache and at the Karlsruhe Institute of Technology (KIT) to develop an efficient and robust three-dimensional neutron kinetics solver to be employed as a new neutronic tool for simulating severe accident transients with the SIMMER code. To this aim, the SNATCH space-time kinetics has been recently extended and coupled with the SIMMER code in a multiple-channel approach. On the other hand, the PARTISN code was coupled to KIN3D, a time-dependent model for neutron kinetics of the ERANOS code system and applied to transient analyses. In this paper, the recent SNATCH extensions are introduced and a comparison to the KIN3D/PARTISN code is provided. SNATCH implementation of the Improved Quasi-Static Method (IQM) relies on a special treatment in which Point Kinetics (PK) parameters are computed for each micro time step by assuming a linear variation of the flux shape over a shape time step. This procedure provides a better coupling between the PK parameters and material perturbations. A new formulation for the reactivity computation is implemented in SNATCH, as the original formulation does not account for reactivity variation for source-induced transients. Comparative results show that, overall, SNATCH neutron kinetics results are coherent with KIN3D/PARTISN for a fast transient induced by a source-jerk in a three-dimensional subcritical system

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driven by an external source. It is also found that reactivity variations are more correctly assessed by the new formulation on reactivity.

PAPER 1094483
Different Methods to Model the MSLB without Primary Cooling Pumps Using HEMERA V1 System Codes

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HEMERA.V1 (Highly Evolutionary Methods for Extensive Reactor Analyses), is a fully coupled 3D computational chain developed jointly by IRSN and CEA. It is composed of CRONOS2 (core neutronics), FLICA4 (core thermal-hydraulics) and the system code CATHARE2. Multi-level and multi-dimensional models are developed to account for neutronics, core thermal-hydraulics, fuel thermal analysis and system thermal-hydraulics, dedicated to best-estimate and conservative simulations and sensitivity analysis. In the IRSN, the HEMERA.V1 chain is adopted to investigate several types of reactivity accidents. This paper presents an example of HEMERA.V1 utilization for the simulation of the Main Steam Line Break (MSLB) without primary cooling pumps. For this transient, two different methods to model the water flow at the core entrance are compared. The model using a matrix flow which takes into account the flow in the loops leads to the loss of natural circulation which may occur several DNBR (and the minimum one may not be the first one) and the model using an uniform distribution of flow rates leads to an unique DNBR and the intact loops have an asymptotic flow rate (there is no loss of natural circulation).

PAPER 1101709
Some Results of Studying of Spatial Kinetics in Fast Reactors

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The paper presents an analysis of the solution to the spatial non-stationary equation of neutron transport the example of a fast reactor. To solve the spatial kinetics equation in three-dimensional geometry in diffusion approximation a calculated code TIME was created. In a direct problem the neutron flux density and its derivatives (for example, reactor power) are determinate at each time step. Solution of a direct problem allows to investigate a deformation of a steady neutron field form which is one of the main reasons of spatial effects occurrence in a reactor. Such deformations affect spatial distribution of delayed-neutrons that is reflected in behavior of reactivity and in related with it results of experiments. The paper also considers some questions of reducing of spatial effects through a choice of a point of reactivity input or detector location. In the inverse problem the reactor reactivity is calculated using the known dependence of reactor power on time. The paper describes some problems of solving the inverse problem taking into account spatial effects.

PAPER 1103760
Atucha-2 Obliquely Inserted Control Rods RELAP5-3D/NESTLE Model

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Atucha-2 is a Siemens-designed PHWR reactor in phase of commissioning in the Republic of Argentina. Its geometrical complexity and peculiarity (e.g., oblique control rods, positive void coefficient) required a developed and validated complex three dimensional (3D) neutron kinetics (NK) model. In the framework of the agreement between NA-SA and University of Pisa a detailed NES-TLE (three-dimensional neutron kinetics code) model of the Atucha-2 NPP was developed. This document summarizes the procedures for the implementation of the oblique control rods into the RELAP5/NESTLE model: a particular arrangement of RELAP5/NESTLE control rods insertion mode for such kind of oblique control rods and an implementation into the homogenized two group cross sections of ad-hoc calculated correction factors (these parameters were obtained by previously executed Monte Carlo calculations) was developed. Some applications, among the scenarios selected to perform safety analysis of the Atucha-2 NPP (CNA-II), are also reported: preliminary Scram Rod Worth, analysis of a Control Rod Ejection Accidents and a CR faulty withdrawal.

PAPER 1104619
The SIMMER/PARTISN Capability for Transient Analysis

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SIMMER-III is extensively used in many institutions for reactor transient and safety analyses. It is a two dimensional fluid-dynamics code coupled with a spatial neutron kinetics model. It features a modular structure. A three dimensional version, SIMMER-IV, is also available. The recent trend to parallel computations has prompted the parallelization of the fluid-dynamics module, yet both versions of SIMMER still rely on non-parallel neutron transport solvers, the TWODANT and THREEDANT codes, respectively.
In this paper, we consider substituting these two solvers with their parallel evolution: PARTISN. PARTISN stands for PArallel TIme-dependent SN and it is a 3D neutron transport code. Our aim is to establish a SIMMER version with a parallelized neutronic module. We describe here the coupling procedure and the modifications needed in both SIMMER and PARTISN to make the coupling possible. We then test our SIMMER-III/PARTISN and SIMMER-IV/PARTISN coupled codes for several cases. These first tests provide an initial validation of the coupled codes. Comparison with the standard SIMMER-III and SIMMER-IV codes results proves that the coupling is correct. The advantage, in term of computational time of using a parallel neutron transport solver is found to be quite substantial. In one case the speedup of the total computational time is ≈ 40% with only 8 processors, with a reduction in the cumulative time spent into the neutron transport solver of a factor of 3 or more. This may be useful especially for three-dimensional cases that feature a very large number of meshes and many energy groups.

PAPER 1105969
Development of a High-Fidelity Monte Carlo Thermal-Hydraulics Coupled Code System Serpent/SUBCHANFLOW - First Results

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Worldwide efforts to develop high-fidelity or ab-initio, high performance multi-physics tools are on-going due to the availability of relatively cheap, large-scale parallel computers. In order to arrive at a coupled neutronics and thermal-hydraulics tool with a higher spatial resolution and accuracy than currently used Best-Estimate tools, an external coupling between the Monte Carlo Reactor Physics code Serpent and the sub-channel code SUBCHANFLOW has been developed. The coupled code system is intended to serve as reference for deterministic reactor dynamics code developments in future exploiting the fact that Serpent was conceived as a lattice code for such deterministic tools. The achieved coupling is based on Serpent's recently introduced universal multi-physics interface. Enabling the interface, Serpent treats temperature dependence of nuclear data using target motion sampling. Furthermore, variance reduction techniques and under-relaxation schemes were utilized. The development is verified by code-to-code comparison with the deterministic tool DYNSUB. Simulation results of both code systems for a pressurized water reactor fuel assembly under hot full power conditions were found in good agreement. Finally, the impact of the thermal-hydraulic feedback on Serpent's numerical performance was assessed. For a hot full power simulation, Serpent ran roughly ten times slower than with pre-broadened continuous energy cross sections for homogeneous hot zero power thermal-hydraulic conditions. However, the developments presented here simply represent a first step towards Serpent/SUBCHANFLOW as a comprehensive reference for deterministic safety analysis tool development as many issues such as combining target motion sampling with bound atom scattering have yet to be resolved.

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Track10: Nuclear Data

PAPER 1068671
Analysis of Radioactivity Ratios of Fission Product Nuclides Deposited to Soil in Fukushima Dai-Ichi Nuclear Power Plant Accident

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The present study focuses on radioactivity ratios between different fission product isotopes being the same element in nuclear fuel loaded in the Fukushima Dai-Ichi nuclear power plant. Using the measured radioactivity data for the soil of the power plant site, we estimate the radioactivity ratios at the reactor shutdown and compare them with numerically calculated values. Detail irradiation histories for each unit and axial profiles of void fraction and power distribution are approximately considered.
Calculation values of the following radioactivity ratios over a whole core, Cs-134/Cs-137, Cs-136/Cs-137, Sr-89/Sr-90 and Te-132/Te-129m, agree with the measurement-based values within 50% difference. Especially the measurement-based radioactivity ratio of Cs-134 to -137 and that of Cs-136 to -137 are well reproduced by numerical calculations within 25% difference. On the radioactivity ratios of Te-132 to -129m, good agreement between calculation and measurement-based values is obtained when the JEFF-based nuclear data library is used.

PAPER 1081806
Experimental Uncertainty Estimation on the Effective Capture Cross Sections Measured in the PROFIL Experiments in Phenix

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A desire of increasing nuclear system safety and fuel depletion is directly translated by a better knowledge on nuclear data. PROFIL and PROFIL-2 experiments give integral information on capture and (n,2n) cross sections and cumulative fission yields for several isotopes (^{95}Mo, ^{97}Mo, ^{101}Pd, ^{105}Pd, ^{133}Cs, ^{143}Nd, ^{144}Nd, ^{145}Nd, ^{147}Sm, ^{149}Sm, ^{151}Eu, ^{233}U, ^{234}U, ^{235}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{244}Cm ...). Interpretation have been done many times in the past but without experimental uncertainty estimation. The cross section library JEFF-3.1.1, the covariance data base COMAC and the code system ERANOS-2.2 are used for this updated interpretation. This study is focusing on the uncertainty estimation on experimental values sensitive to capture cross sections. Three steps are required: the fluence scaling, the uncertainty propagation on the fluence and finally the uncertainty estimation on ratio variation of interest. This work is done with CONRAD using Bayesian adjustment and marginalization method. Mean C/E results and conclusions are identical to the previous interpretation. A fluence uncertainty of 1.4% is found for the two experimental pins of PROFIL-2 and 1.9% for PROFIL. Propagating this new information on the fluence to ratio variation of interest gives experimental uncertainties between 1% to 2.5% for the isotopes present in the experimental pins. One of the main results are for ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu capture cross sections: C/E are respectively equal to 1.03, 0.98, 0.97, 1.08 and 1.14 with an uncertainty lower than 2.5%. All the results will provide feedback on variance-covariance matrices for further works.

PAPER 1105105
Updated Multi-Group Cross Sections of Minor Actinides with Improved Resonance Treatment

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The study of minor actinide in transmutation reactors and other future applications makes resonance self-shielding treatment a significant issue for criticality and isotope depletion. Resonance treatment for minor actinides has been carried out by subgroup method with improved interference effect through interference correction. Subgroup data was generated using RMET21 and GENP codes along with multi-group cross section data by NJOY nuclear data processing system. Updated multi-group cross section data library for a neutron transport code nTRACER was compared with solutions from MCNPX. The resonance interaction of uranium with minor actinides has been included by modified interference treatment of interference correction in subgroup methodology. The comparison of cross sections and multiplication factor in pin and assembly problems showed significant improvement from systematic resonance treatment especially for ^{237}Np and ^{243}Am.

PAPER 1106244
Parameterized Representation of Macroscopic Cross Section for PWR Reactor Considering with 12 Burnable Absorber Fuel Rods in the Fuel Element

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The purpose of this work is to describe, by means of Tchebychev polynomial, a parameterized representation of the homogenized macroscopic cross section for PWR fuel element as a function of soluble boron concentration, moderator temperature, fuel temperature, moderator density and $^{235}_{92}U$ enrichment, considering burnable absorber rods in the fuel element. The cross-section data nalyzed are fission, scattering, total, transport, absorption and capture. The parameterization enables a quick and easy determination of problem-dependent cross-sections to be used in few groups calculations. The methodology presented in this paper will allow generating group cross-section data to perform PWR core calculations without the need to generate them based on computer code calculations using standard steps. The results obtained by the proposed methodology when compared with results from either the SCALE code calculations or with the MCNPX code calculations show very good agreement.

Track11: Research Reactors and Spallation Sources

PAPER 1087763
Jules Horowitz Reactor.France Experimental Loop Development According Optimized Irradiation Process.

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Experimental reactors in the world enable researchers to meet the needs of industry and institutions not only by providing support to the existing nuclear infrastructure (Gen.2) but also by preparing the future generation (Gen.3, Gen.4) or even by responding to other needs as well (supports with fusion, medical applications). It is for this specific purpose that the Jules Horowitz Polyvalent Irradiation Reactor is now being built at the CEA Cadarache Research Center (located in the south of France). This Material Testing Reactor (MTR type) is designed to irradiate materials or fuel samples for various experimental tests. The reactor will also produce Mo99 radioelements that will supply 25% to 50% of current European needs.
The goal of this paper is to describe a fuel irradiation loop, now under study, that will be designed to carry out power ramps tests to provide technical support to the Generation 2 and 3 nuclear power plants. In order to increase its irradiation capacity (2 to 3 per cycle), this loop takes into account the requirements that will lead to the optimization of all experimental processes in the facility (such as non-destructive examinations before and after the test, specific loading tools). All these considerations are being taken into account in order to offer to the customer’s complete and optimized conditions in terms of experimental irradiations processes.

PAPER 1106121
Preliminary Neutronic Design for the Conceptual Fluid Granular Spallation Target

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A new concept of spallation target design for Accelerator-Driven Subcritical system (ADS) was performed. The main material of the target was tungsten, Helium was used as coolant, proton beam current is 4mA, and energy is 250MeV. Some physical parameters such as the filling mode, neutron leakage spectrum, the average leakage neutron yield, leakage neutron distribution along the axial of spallation target, the stability of spallation target, and differences in spallation target energy deposition distribution have been analyzed, based on the differences between granular particle size and the form of arrangement. From the analyzed result, when the filling mode

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is face-center cubic arrangement, and the affection by diameter of granular is minimal, the perturbation is no more than 16%. And this perturbation will be lower with the smaller size of granular. The radius of spallation target should be a little larger than the beam radius, it is appropriate to choose 10cm for spallation target radius and 20cm for its length, respectively.

PAPER 1126142
Accumulation of Tritium in Beryllium Material under Neutron Irradiation

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In the present work the programming code is created on the basis of which the accumulation kinetics of tritium and isotope of He^4 in the Be^9 sample is analyzed depending on the time. The program is written in C++ programming language and for the calculations Monte Carlo method was applied. This program scoped on the calculation of concentration of helium and tritium in beryllium samples depending on the spectrum of the neutron flux in different experimental reactors such as JMTR, JOYO and IPEN/MB. The processes of accumulation of helium and tritium for each neutron energy spectrum of these reactors were analyzed.

PAPER 1127738
A Preliminary Study of an Improved Area Method, Adapted to Short Time Transients in Sub-Critical Systems

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Since the '90s interest has grown in the characterization of sub-critical systems. Because they should not have control/shutdown devices, it is necessary to prevent by design neutron flux divergence in all conditions. Most of the available theoretical models on neutron kinetic behavior in multiplying systems are tailored on the description of critical systems; then they have decreasing validity as well as the system is far and far from criticality; as a consequence many of the available monitoring methods are of questionable use for sub-critical systems and give contradictory informations: this is true, among the others, for the Area Method. One of the difficulties in understanding the kinetic behavior of a sub-critical system is the scarcity of experimental measurements available. Then we decided to develop a full time-dependent MCNP6 simulation of a sub-critical core to obtain some "phenomenological-like" data whose analysis leads to conclude that geometrical inhomogeneity of the system plays a key role in determining its time behavior over short time scales. Starting from an original observation by Weinberg and Wigner, we conclude that, for the short time scale considered (~ μs), we cannot use the "classical" diffusion approximation but a little bit more complicated formulation, similar to what is known as telegrapher's equation: along this line we find a reasonable justification for the idea that, before using any analysis method grounded on the assumption that the system has relaxed into its fundamental mode, we could have to wait for a longer time than the prompt neutrons lifetime. We conclude that from an experimentalist's point of view it should be at lest necessary to cut from the analysis the lowest part of the time interval before using the Area Method, but more refined analyses are needed.

PAPER 1128078
Preliminary Neutronics Analysis of a Spallation Target for Transmutation

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Accelerator Driven subcritical System (ADS) was recognized as an effective nuclear waste transmutation device. Target in liquid or solid in an independent loop bombarded by the charged particle beam was considered as the neutron source. Heavy metal was chosen as target material or coolant. The present work was to discuss the possibility of taking Minor Actinides as part of spallation target material, for a better transmutation performance of entire ADS. According to the thermal cooling and irradiation time limitation, a conceptual design of target for transmutation was proposed. And preliminary neutronics analysis for target performance assessment including neutron flux, neutron yield as well neutron spectrum is shown in this work.

PAPER 1128332
Preliminary Optimization Analysis of the Radiation Shielding of the China Lead-Based Research Reactor

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Accelerator Driven subcritical System (ADS) is recognized as an efficient nuclear waste transmutation device. Supported by the Strategic Priority Research Program of "the Future Advanced Nuclear Fission Energy-ADS transmutation system", the China LEAd-based Research Reactor (CLEAR-I) is proposed. Along with the approaching of the CLEAR-I design, the radiation shielding for CLEAR-I is updated and optimized step by step to meet with new shielding requirements. Employing the modeling program MCAM and calculation system VisualBUS developed by FDS Team, the shielding capability was verified using Monte Carlo method. As shown from the results, the fast neutron flux for components in reactor vessel is under the limitation and the neutron radiation for mechanism in containing room has been as low as possible. After shutdown for 7 days, the dose rate in most area of containing room is lower than 100 μSv/hr, allowing hands on operation. Replacement of components such as the spallation target in containing room is possible.

PAPER 1233696
Preliminary Analysis of Radioactive Source Term for Normal Operation of China Lead-Based Research Reactor (CLEAR-I)

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China LEAd-based Research Reactor (CLEAR-I) is a 10 MW lead-bismuth cooled research reactor, which serve as ADS and Lead cooled Fast Reactor technology verification platform. The evaluation of radioactive source term is needed for reactor decommissioning, worker and public dose rate. In this contribution, radioactive source term of CLEAR-I was calculated for normal operation in reactor core, primary coolant, cover gas and secondary coolant by CAD-based Multi-Functional 4D Neutronics & Radiation Simulation System (VisualBUS) developed by FDS Team. This calculation is a guidance for reactor decommissioning, worker and public dose and radioactive waste disposal and transportation.

Track12: Fuel Cycle and Actinide Management

PAPER 1092834
Fusion Hybrids for Generation of Advanced ($^{231}Pa+^{232}U+^{233}U+^{234}U$)-Fuel in Closed (U-Pu-Th)-Fuel Cycle

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Technology of controlled thermonuclear fusion (CTF) is traditionally regarded as a practically inexhaustible energy source. However, development, mastering, broad deployment of fast breeder reactors and closure of nuclear fuel cycle (NFC) can also extend fuel base of nuclear power industry (NPI) up to practically unlimited scales. Under these conditions, it seems reasonable to introduce into a circle of the CTF-related studies the works directed towards solving some principal problems which can appear in a large-scale NPI in closed NFC.

The first challenge is a large scale of operations in NFC back-end that should be reduced by achieving substantially higher fuel burn-up in power nuclear reactors. The use of ^{231}Pa-^{232}Th-^{232}U-^{233}U fuel in light-water reactor (LWR) opens a possibility of principle to reach very high (about 30% HM) or even ultra-high fuel burn-up.

The second challenge is a potential unauthorized proliferation of fissionable materials. As is known, a certain remarkable quantity of ^{232}U being introduced into uranium fraction of nuclear fuel can produce a serious barrier against switching the fuel over to non-energy purposes.

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Involvement of hybrid thermonuclear reactors (HTR) into NPI structure can substantially facilitate resolving these problems. If HTR will be involved into NPI structure, then main HTR mission consists not in energy generation but in production of nuclear fuel with a certain isotope composition.
The present paper analyzes some neutron-physical features in production of advanced nuclear fuels in thorium HTR blankets. The obtained results demonstrated that such a nuclear fuel may be characterized by very stable neutron-multiplying properties during full LWR operation cycle and by enhanced proliferation resistance too. The paper evaluates potential benefits from involvement of HTR with thorium blanket into the international closed NFC.

PAPER 1100837
^{233}U Fuel Production and 30-Year Utilization without Reprocessing and Refuelling Using Heavy Water Coolant

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This study examines the physics of a thorium fuel cycle based on generating the initial fissile (^{233}U) fuel inventory in a Deuterium-Tritium fusion device and on operating a 600 MW_{th} fission reactor. For both phases of the fuel cycle, the fuel form is an aqueous slurry consisting of thorium oxide micro-particles dispersed into heavy water. The slurry is the fuel carrier and the coolant. After 180 full power days in the fusion driven device, the fuel enrichment is 1.4%. The enrichments is defined as the ratio between the fissile actinides mass and the total actinides mass. After the removal of fission products, the 1.4% enriched slurry thorium-uranium fuel can be used for longer than 30 full power years in a 600 MW_{th} critical reactor core, without adding any fissile material. The critical reactor has three zones: inner fissile, central fertile, and outer reflector.

PAPER 1101727
Uncertainty Analysis for Fuel Flux Calculations of Fast Reactors with External Fuel Cycle

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The paper focuses on the results of uncertainty analysis when calculating nuclide composition in fuel of fast reactors and on uncertainties of determining nuclide composition in the external fuel cycle. As demonstrated, the main contributions to the uncertainty of nuclide composition are due to:

- uncertainties in operation of the reactor and in the fuel-cycle time;
- uncertainties in nuclide clean-up factors at the Closed Nuclear Fuel Cycle (CNFC) stages when reprocessing spent nuclear fuel;
- uncertainties in isotopic-kinetics cross-sections;
- uncertainties in nuclide decay data.

PAPER 1105292
Study on Transmutation and Storage of LLFP Using a High-Temperature Gas-Cooled Reactor

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There is a need to temporally store high-level radioactive waste (HLW) until the location of final disposal is decided. HLW contains several types of long-lived fission product (LLFP) which stay radioactive for hundreds of thousands of years. In addition, they tend to be chemically mobile when dissolved into ground water thus may not be suited for geological disposal. A facility that is able to store and incinerate LLFP simultaneously is desirable. The high-temperature gas-cooled reactor (HTGR) is one of the fourth generation nuclear reactors currently under research and it has some favorable characteristics that allow the reactor to destroy LLFP through nuclear transmutation. In this study, the capability of HTGR as LLFP transmuter was evaluated in terms of neutron economy. Considering gas turbine high-temperature reactor with 300 MWe nominal capacity (GTHTR300) as HTGR, transmutations of four types of LLFP nuclide were estimated using Monte Carlo transport code MVP and ORIGEN. In addition, burn-up simulations for whole-core region were carried out using MVP-BURN. It was numerically shown that the neutron fluxes change significantly depending on the arrangement of LLFP in the core. When 15 t of LLFP is placed in an ideal manner, the GTHTR300 can sustain sufficient reactivity for one year while transmuting up to 30 kg per year. Additionally, there are more space available for storing larger amount of LLFP without affecting the reactivity. These results suggest that there is a possibility of using GTHTR300 as both LLFP storage and transmuter.

PAPER 1105802
Core Library for Advanced Scenario Simulation, C.L.A.S.S. : Principle & Application.

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The global warming, the increase of world population and the depletion of fossil resources have lead us in a major energy crisis. Using electronuclear energy could be one of the means to solve a part of these issues. The way out of this crisis may be enlightened by the study of transitional scenarios, guiding the political decisions. The reliability of those studies passes through the wide variety of the simulation tools and the comparison between them.
From this perspective and in order to perform complex electronuclear scenario simulation, the open source Core Library for Advance Scenario Simulation (CLASS) is being developed. CLASS main asset is its ability to include any kind of reactor, whether the system is innovative or standard. A reactor is fully described by its evolution database that must contain a set of different fuel compositions in order to simulate transitional scenarios. CLASS aims at being a useful tool to study scenarios involving Generation IV reactors as well as innovative fuel cycles, like the Thorium cycle.
The following contribution will present in detail the CLASS software. Starting with the working principle of this tool, one will explain the working process of the different modules such as the evolution module. It will be followed by an exhaustive presentation of the UOX-MOX bases generation procedure. Finally a brief analysis of the error made by the CLASS evolution module will be presented.

PAPER 1106324
Fuel Composition Generation Techniques of Nuclear Fuel Cycle Simulators

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Nuclear fuel cycle simulators track the flow of materials through the facilities that comprise a nuclear energy system. The composition of these materials, which simulators specify at the elemental or isotopic level, is driven by the neutronic characteristics of the reactors in the system. Therefore, all simulators include a method for generating input and output compositions for the reactor fuel they track, widely known as recipes. This paper surveys the recipe generation approaches taken by five simulators, which range from pre-computed reactor physics modeling to on-the-fly calculations. It concludes with an illustrative example of the canonical parametric recipe generation problem simulators are called upon to solve.

PAPER 1106382
Core Burnup Calculation of Uranium Rock-Like Oxide Fuel PWR for Spent Fuel Composition Estimation

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Burnup calculations of uranium rock-like oxide (U-ROX) fuel were carried out using a PWR core model, in order to obtain the spent U-ROX fuel composition, which is to be used for the analysis of the LWRs "replace once" scenario in Japan. Fuel assemblies loading and shuffling were simulated for the power peaking reduction, and U-ROX fuels were burnt for 3 times of 400 EFPD cycle. As a result, plutonium production amount in U-ROX is much less than that in UO_2, and the fraction of ^{238}Pu in total plutonium becomes large. This is due to less ^{238}U conversion to ^{239}Pu in U-ROX than UO_2, while the burnup of ^{235}U and the conversion to ^{238}Pu is as much as that in UO_2. The power peaking factor of the U-ROX PWR core tends to be large, and the reduction of the peaking factor seems important.

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Track14: Education in Reactor Physics

PAPER 1085651
Multi-Collision Theory for Educated Pedestrians

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The aim of the paper is to present the linear transport theory in a form which can be easily accessible also to person with a basic graduate knowledge in mathematics and physics. The time dependent linear Boltzmann equation is approached using a multi-collision methodology and the solution for each propagation direction is sought via classical methods used for linear ODE, such as Laplace transform and Green's function method. Some results that can be fully managed analytically are presented and a general strategy for the numerical implementation proposed.

PAPER 1073641
Virtual Labs on Unique Experimental Equipment

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A description of a database on the support of laboratory work on a unique experimental equipment, concept development and use of virtual labs in the learning process are present.

PAPER 1104802
PINSPEC: A Monte Carlo Code for Pin Cell Spectral Calculations for Educational Applications

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Students in many reactor physics courses are exposed to canonical reactor physics concepts through theoretical problems simplified to allow for tractable analytical solutions. Such problems typically require tedious mathematical derivation which is often not the most effective approach to teaching basic reactor physics concepts. A new complementary methodology to introduce these concepts is made possible with PINSPEC, a pin cell Monte Carlo code for educational use. PINSPEC enables students to simulate pin cell models for various reactor types with a simple-to-use Python interface. PINSPEC uses point-wise cross section data and includes a module for Single-Level Breit-Wigner cross-section generation and Doppler broadening. The PINSPEC code supports a variety of tallies which students may use to compute resonance integrals, multi-group cross sections, and more for various materials and pin configurations. PINSPEC is undergoing review for open source release in the near future such that it will be a free and accessible tool for instructors developing reactor physics curricula with an applied and interactive approach to learning.

PAPER 1127873
Unique Approaches in Emphasizing the Role of Reactor Laboratories and Facilities for Training and Education of Future Nuclear Engineers 'without the Borders'

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The 21st century brings numerous challenges into daily lives of nuclear engineers world-wide, such as at nuclear power plants (facing new safety paradigm influenced by 2011-unfortunate Japanese Fukushima incident), at the universities in educating and training new generations of nuclear engineers in capturing new expectations of safety standards, and in reaching out to countries that are interested to develop new or revitalize their decades-old education and training programs as pertaining to nuclear engineering and science. In this paper we present two novel aspects we have developed: (a) the establishment of novel educational approaches pertaining to training and practices of nuclear safety culture in university curricula, and (b) development of novel digital-type class focused on the basic aspects of nuclear science and engineering shared between a class in the State of Utah (USA) and Uruguay.

Track15: Research Related to Fukushima Accident

PAPER 1102869
Spatial Correlation Modeling of Macroscopic Cross Section with Weierstrass Function

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Material media formed via sedimentation are under the random fluctuation of constituent atoms/molecules with spatial correlation. Extreme physical disorder is known to lead to the 1/f fluctuation. In order to model these characteristics inherent in the formation of fuel debris, a randomized Weierstrass function (RWF) in fractal geometry is examined in this work. Here, RWF is a version of Weierstrass function, is able to confine the influence of correlation within a certain range of coordinate increments, and is globally under a fixed variance. Therefore, RWF is capable of modeling the macroscopic cross section with some unknown spatial distribution of constituent isotopes. The potential impact of RWF modeling on neutron transport simulation is assessed using Woodcock tracking. Results obtained indicate that the distance-to-collision sampling over realizations of a continuously varying random medium will be necessary for the Monte Carlo calculation if the influence range of spatial correlation is larger than the mean free path of the corresponding non-random homogeneous medium.

SpecialSession2: Reactor Physics and Criticality Safety Activities in OECD/NEA Working Party

PAPER 1109848
Effects of Nuclear Data Library and Ultra-Fine Group Calculation for Large Size Sodium-Cooled Fast Reactor OECD Benchmarks

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The present paper summarizes calculation results for an international benchmark proposed by the Sodium-cooled Fast Reactor core Feed-back and Transient response (SFR-FT) under the framework of the Working Party on scientific issues of Reactor Systems (WPRS) of the Nuclear Energy Agency of the OECD. It focuses on the large size oxide-fueled SFR. Library effect for core performance characteristics and reactivity feedback coefficients is analyzed using sensitivity analysis. The effect of ultra-fine energy group calculation in effective cross section generation is also analyzed. The discrepancy is about 0.4% for a neutron multiplication factor by changing JENDL-4.0 with JEFF-3.1. That is about -0.1% by changing JENDL-4.0 with ENDF/B-VII.1. The main contributions to the discrepancy between JENDL-4.0 and ENDF/B-VII.1 are ^{240}Pu capture, ^{238}U inelastic scattering and ^{239}Pu fission. Those to the discrepancy between JENDL-4.0 and JEFF-3.1 are ^{23}Na inelastic scattering, ^{56}Fe inelastic scattering, ^{238}Pu fission, ^{240}Pu capture, ^{240}Pu fission, ^{238}U inelastic scattering, ^{239}Pu fission and ^{239}Pu nu-value. As for the sodium void reactivity, JEFF-3.1 and ENDF/B-VII.1 underestimate by about 8% compared with JENDL-4.0. The main contributions to the discrepancy between JENDL-4.0 and ENDF/B-VII.1 are ^{23}Na elastic scattering, ^{23}Na inelastic scattering and ^{239}Pu fission. That to the discrepancy between JENDL-4.0 and JEFF-3.1 is ^{23}Na inelastic scattering. The ultra-fine energy group calculation increases the sodium void reactivity by 2%.

PAPER 1121153
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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In the frame work of the working party on reactor and system (WPRS) of the OECD nuclear Energy Agency (NEA), the benchmark on sodium cooled fast reactor (SFR) was conducted. Within the OECD/NEA/WPRS benchmark, study on medium size metallic fuel core was

Poster Session

performed using code system for fast reactor core calculation with deterministic method MARBLE and with Monte Carlo method MVP. The latest nuclear library JENDL-4.0 is used for calculation of eigenvalues (keff) and reactivity (sodium void, Doppler and control rod worth). The differences of calculation results between the analysis methods are summarized in this paper. Sensitivity studies of eigenvalue and sodium void reactivity for the medium size metallic fuel benchmark core are conducted to determine the main reactions contributing to the difference between JENDL-4.0 and other libraries JEFF-3.1 and ENDF/B-VII.1.

PAPER 1195260
Quantifying the Effect of Undersampling in Monte Carlo Simulations Using SCALE

C.M.Perfetti, B.T.Rearden

Oak Ridge National Laboratory, Oak Ridge, USA

This study explores the effect of undersampling in Monte Carlo calculations on tally estimates and tally variance estimates for burnup credit applications. Steady-state Monte Carlo simulations were performed for models of several critical systems with varying degrees of spatial and isotopic complexity, and the impact of undersampling on eigenvalue and flux estimates was examined. Using an inadequate number of particle histories in each generation was found to produce a bias of about 100 pcm in eigenvalue estimates and biases that exceeded 10% in fuel pin flux estimates.

SpecialSession5: Multiscale, Multiphysics Approaches in Nuclear Science and Engineering Applications

PAPER 1102894
Research on SCWR Core Characteristics Utilizing Pin-Wise Neutronics Thermal-Hydraulic Coupling Method

Q.Hong, G.Niu, C.Tang, L.Lu, G.Shi

Shanghai Nuclear Engineering Research & Design Institute, Shanghai, China

According to the intense flow heterogeneity and strong Neutronics / Thermal-Hydraulic feedback of SCWR, in present work, a pin-wise neutronics and thermal-hydraulic coupling method is developed and a core analysis code package SCAP is established. The package utilizes sub-channel program NCED-SCWR, lattice program PARAGON, nodal diffusion program MRAPS, and multi-functional code COUPLE. Furthermore, 3D core power distribution & flow distribution, reactivity parameters and isotope density are studied for a typical core with 121 fuel assemblies. The results show that both the coupling method and code package are applicable to the SCWR core analysis. Based on the analysis, it is pointed out that the fuel rod with the highest cladding temperature may be not the one with peaking power in SCWR, which is different from PWRs. The obtained conclusion will be benefit for SCWR core design.

SpecialSession6: Nuclear Criticality Safety of Fuel Debris

PAPER 1087714
Post-Accident Defueling Procedure and Its Criticality Safety Evaluation of the Fukushima-Daiichi Nuclear Power Plants

N.Takaki

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Post-accident defueling procedure for the Fukushima-Daiichi power plants is discussed. The authors propose a rotating plug defueling system that enables access to deep vessel bottom and a water cleanup system to avoid loss of visibility in the water during defueling works. Criticality of fuel debris accumulated at the bottom of reactor vessel was evaluated based on conservative assumptions for wide range of water volume ratio. Required amount of neutron absorbing materials to prevent re-criticality accident throughout defueling processes was also evaluated. Boron pellet and boron gel are developed and some fundamental experiments were performed to obtain the properties.

Track1-10 Reactor Analysis Methods

Session Chair: Aldo Dall'Osso(AREVA NP), Christophe Demaziere(Chalmers Univ. of Tech.)

8:00 AM

An Asymptotic Homogenized SP_2 Approximation to the Boltzmann Equation. I. Derivation

T.G.Saller, E.W.Larsen, T.Downar

University of Michigan, Ann Arbor, USA

Because many of the current-generation reactor analysis codes provide neutron fluxes based on diffusion theory, there is considerable interest in methods that provide a more accurate diffusion solution without significantly increasing computational costs. In this work, an asymptotic analysis, previously used to derive a homogenized diffusion equation for lattice systems, is generalized to derive a one-dimensional, one-group homogenized SP_2 equation as a more accurate alternative to the standard homogenized diffusion equation. This analysis results in new diffusion coefficients and an improved formula for flux reconstruction. Five fixed-source lattice functions are required to calculate the diffusion coefficients, three of which are also used in the flux reconstruction. In a homogeneous medium, these lattice functions reduce to simple polynomials in angle, and the homogenized SP_2 equation reduces to its standard form. The new flux reconstruction formula includes three “correction” terms that account for spatial derivatives of the scalar flux and differences between the infinite-lattice eigenvalues and the core eigenvalue.

8:20 AM

An Asymptotic Homogenized SP_2 Approximation to the Boltzmann Equation. II. Discontinuity Factors and Numerical Testing

T.G.Saller, E.W.Larsen, T.Downar

University of Michigan, Ann Arbor, USA

The current generation of LWR reactor analysis codes based on homogenized diffusion theory has provided sufficient accuracy for a wide range of reactor application. However, there is interest in methods that can improve the accuracy of these codes by using a more rigorous diffusion coefficient and a more robust flux reconstruction method. In a companion paper, a one dimensional, one-group homogenized SP_2 equation is asymptotically derived, and an improved formula for flux reconstruction is obtained. In the present paper, the SP_2 equation is tested, and the flux reconstruction formula is used to obtain flux and current discontinuity factors for the SP_2 equation. For a test problem, the asymptotic SP_2 formulation is compared to standard $SP_{2,}$ asymptotic diffusion, and standard diffusion. The eigenvalue and reconstructed fluxes are both examined. In general, the asymptotic equations are more accurate than the standard equations, and SP_2 is more accurate than diffusion theory, especially for optically small systems.

8:40 PM

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

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Universitat Politecnica de Valencia, Valencia, Spain

Most of the neutron diffusion codes use numerical methods giving accurate results in structured meshes. However, the application of these methods in unstructured meshes to deal with complex geometries is not straightforward and it may cause problems of stability and convergence of the solution. By contrast, the Finite Volume Method (FVM) is easily applied to unstructured meshes and is typically used in the transport equations due to the conservation of the transported quantity within the volume. In this paper, the FVM algorithm implemented in the ARB Partial Differential Equations Solver has been used to discretize the multigroup neutron diffusion equation to obtain the matrices of the generalized eigenvalue problem, which has been solved by means of the SLEPc library. Nevertheless, these matrices could be large for fine meshes and the eigenvalue problem resolution could require a high calculation time. Therefore, a transformation of the generalized eigenvalue problem into a standard one is performed in order to reduce the calculation time.

9:00 AM

A Generalized Multigroup Method Based on Finite Elements

A.T.Till, J.E.Morel, M.L.Adams

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The standard multigroup (MG) method for energy discretization of the transport equation can be sensitive to approximations in the weighting spectrum chosen for cross-section averaging. As a result, MG often inaccurately treats important phenomena such as self-shielding variations across a fuel pin. From a finite-element viewpoint, MG uses a single fixed basis function (the pre-selected spectrum) within each group, with no mechanism to adapt to local solution behavior. To address these issues, we introduce a Petrov-Galerkin finite-element multi-group (PG-FEMG) method, a generalization of the MG method that is related to the family of multiband (MB) methods. The only approximation in PG-FEMG is that the angular flux is a linear combination of basis functions. The coefficients in this combination are the unknowns. A basis function is non-zero only in the discontiguous set of energy intervals associated with its band. We implement the PG-FEMG method for several realistic pin-cell problems and find it to be significantly more accurate per degree of freedom than MG for the quantities of interest that we tested, including criticality eigenvalue and power profile shape. We find PG-FEMG is much less sensitive to errors in weighting spectra compared to standard MG. We discuss straightforward generalizations to multi-dimensional problems of practical interest.

9:20 AM

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

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

Due to its relatively low computational cost, the neutron diffusion approximation remains one of the most commonly-used computational tools for reactor analysis. Although diffusion requires more approximations than higher-order “transport” methods such as SN and Monte Carlo, the computational cost of these methods prohibits their use for routine reactor analysis. It has long been known that diffusion is anisotropic in heterogeneous reactors, i.e., neutrons diffuse more rapidly in some directions than others. While in many reactors the anisotropic diffusion effects are negligible, in lattices containing voided or optically-thin channels these effects are significant. It is therefore desirable that the homogenized diffusion coefficient be a tensor. In the past, monoenergetic diffusion tensors have been asymptotically derived from the exact, continuous energy Boltzmann transport equation for large systems with a periodic lattice structure. We have now derived multigroup diffusion tensors using asymptotic analysis of the multigroup, homogenized diffusion equations. The multigroup diffusion tensors are defined such that the asymptotic limits of the continuous energy transport equation and the multigroup diffusion equation are the same. Our numerical results indicate that the asymptotic, multigroup diffusion tensors are more accurate than the standard scalar multigroup diffusion coefficients for certain problems. In particular, the asymptotic diffusion tensor is more accurate for large lattice systems, and systems containing optically-thin channels.

9:40 AM

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

C.N.Mcgraw, M.L.Adams, W.D.Hawkins, M.P.Adams, T.Smith

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Significant literature exists on the accuracy of the Method of Characteristics (MOC) for solving the transport equation for reactors with realistic representations of geometries. The same is not true for Discontinuous Finite Element Methods. We present a resolution study and error analysis detailing how the Linear DFEM (LD) spatial discretization method performs on the well-known two-dimensional C5G7 benchmark problem as a function of spatial and angular resolution, for spatial meshes that conform to the pin geometries. We compare pin powers and k-eigenvalues against reference MCNP results, as a function of spatial and angular resolution. We use “product” Gauss-Chebyshev quadrature sets that range from 2 to 24 polar levels

Track1-10 Reactor Analysis Methods

Session Chair: Aldo Dall'Osso(AREVA NP), Christophe Demaziere(Chalmers Univ. of Tech.)

and 32 to 512 azimuthal angles. Our spatial resolution ranges from 64 to 900 quadrilateral cells per pincell. We find that the LD method performs well, with k and pin-power results consistent with third-order truncation error. The LD error on our quadrilateral cells appears to be between the errors of “lumped” and standard (un-lumped) versions of PieceWise Linear DFEM (PWLD). (LD has 3 unknowns per polygonal cell; the PWL methods have four per quadrilateral cell, five per pentagonal cell, etc.) On a coarse spatial mesh with three rings in fuel and one in water per pin cell, the LD spatial truncation error is <10 pcm in k and well below 0.5% in each of the 1056 pin powers. Our error analysis indicates that our high-resolution solution (900 cells per pincell, 24 polar angles, and 256 azimuthal angles) is accurate to < 1 pcm in k and to <0.004% in each pin power. We provide these pin powers. These high-resolution solution errors are significantly smaller than the statistical precision reported for the reference Monte Carlo results.

Track4-6 Verification, Validation and Uncertainty Analysis

Session Chair: Laurent Chabert(AREVA TA), Kenji Nishihara(JAEA)

8:00 AM
Sensitivity and Uncertainty Calculations Methods of Neutronics Parameters in PWR Cores Part I: Theory and Sensitivity Calculations

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Sensitivity and uncertainty calculations methods of neutronics parameters in pressurized light water reactors have been developed. The sensitivity is composed of three terms; the first is the sensitivity of cell-averaged multi-group cross-sections relative to multi-group infinite dilution cross-sections, the second is the sensitivity of assembly averaged few-group macroscopic cross-sections relative to cell-averaged multi-group cross-sections, and the third is the sensitivity of neutronics parameters in PWR cores relative to few-group macroscopic cross-sections. Combining the three sensitivities, the sensitivity of neutronics parameters in PWR cores relative to multi-group infinite dilution cross-sections is obtained. The discussion of this method will be presented in two papers; the present paper is part I, where the theory and some numerical results for typical pin cells, fuel assemblies and a simple PWR core are shown. The present method gives us multi-group sensitivities for individual nuclides in each reaction type, and wide ranges of applications are possible to the fields such as cross-section adjustment and uncertainty reduction.

8:20 AM
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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The nuclear community relies heavily on computer codes and numerical tools. The results of such computations can only be trusted if they are augmented by proper sensitivity and uncertainty (S&U) studies. This paper presents some aspects of S&U analysis when constrained quantities are involved, such as the fission spectrum or the isotopic distribution of elements.
A consistent theory is given for the derivation and interpretation of constrained sensitivities as well as the corresponding covariance matrix normalization procedures. It is shown that if the covariance matrix violates the “generic zero column and row sum” condition, normalizing it is equivalent to constraining the sensitivities, but since both can be done in many ways different sensitivity coefficients and uncertainties can be derived. This makes results ambiguous, underlining the need for proper covariance data. It is also highlighted that the use of constrained sensitivity coefficients derived with a constraining procedure that is not idempotent can lead to biased results in uncertainty propagation. The presented theory is demonstrated on an analytical case and a numerical example involving the fission spectrum, both confirming the main conclusions of this research.

8:40 PM
Adjoint-Based Sensitivity and Uncertainty Analysis of Lattice Physics Calculations with CASMO-4

M.Pusa

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The topic of this paper is the development of sensitivity and uncertainty analysis capability to the reactor physics code CASMO-4 in the UAM (Uncertainty Analysis in Best-Estimate Modelling for Design, Operation and Safety Analysis of LWRs) benchmark. The developed calculation system enables the uncertainty analysis of homogenized multi-group cross-sections, diffusion coefficients and pin powers with respect to nuclear data. The uncertainty analysis methodology is deterministic, meaning that the sensitivity profiles of the responses are computed first, after which uncertainty is propagated by combining the sensitivity profiles with the covariance matrices of the uncertain nuclear data. The sensitivity analysis is based on perturbation theory which enables computing the sensitivity profiles efficiently by solving one generalized adjoint system for each response. The mathematical background of this work is reviewed and the main conclusions related to the implementation are summarized. Special emphasis is placed on the sensitivity analysis of two-group homogenized diffusion coefficients which require some modifications to the standard equations of generalized perturbation theory. Numerical results are presented and analyzed for a PWR fuel assembly with control rods out and inserted. The computational efficiency of the calculations is discussed.

9:00 AM
Nuclear Data Uncertainty Propagation on Power Maps in Large LWR Cores

A.Santamarina

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Nuclear data uncertainty propagation is more and more required in safety calculations of large NPP cores. In this study, nuclear data uncertainties have been propagated to LWR cores using accurate transport calculations. PWR benchmarks representative of the French NPP fleet were selected: one 900MWe (Fessenheim), one 1500MWe (CHOOZ), as two complementary GEN-III benchmarks belonging to the OECD/UAM benchmark, representative of large PWR UOX and –partially loaded - MOX cores surrounded by thick stainless steel reflector. Reactivity and assembly power map sensitivity coefficients to multigroup cross sections have been calculated using perturbation theory tools implemented in ERANOS2/SNATCH transport code.
Different sources of nuclear data uncertainty were compared: covariance matrices from BOLNA, and French COMAC associated to JEFF3.1.1 evaluations, were used for uncertainty propagation. The total uncertainty on the multiplication factor keff is not sensitive to the size and the loading of the PWR cores; this uncertainty amounts to 560pcm (1σ) with COMAC, compared to 780 pcm with BOLNA covariances. The main component of k_{eff} uncertainty is ν^{U235}, followed by ^{238}U capture, σ_c^{U235} and σ_f^{U235}.
The uncertainty on the radial power map increases with the core size, due to higher Eigen Value Separation factor (EVS>20). For example, using COMAC covariances, the uncertainty in central assemblies increases from 1.5% for PWR-900MWe to 3.4% for large N4 and GEN-III PWRs. In the peripheral assemblies which can host the power peak, the uncertainty increases from 1.1% for 900MWe to 2% for large PWRs. Therefore, the potential power swing bias in JEFF3.1.1 core calculations varies from 2.6% for PWR-900MWe to 5-6% for N4 and large GEN-III PWRs. Using BOLNA covariance files, the uncertainty on the P_{centre}/P_{periph} power ratio amounts to 10% (5% with COMAC) for GEN-III PWR core, and up to 24% in the Gen-III MOX benchmark (12% with COMAC). Our powerful deterministic calculation scheme allowed the decomposition of the LWR parameter uncertainty in isotopic partial cross-section components. The main contributor to the power uncertainty is the ^{238}U inelastic scattering: 3% and 5%, respectively for COMAC and BOLNA, in GEN-III central assemblies. However, ^{16}O and ^{1}H elastic scattering cross-sections should be also validated and improved in order to reduce their individual 1%-2% current contribution to the central power uncertainty.

9:20 AM
Are Modeling Uncertainties Properly Considered in Neutronics Data Assimilation Analysis?

P.Athe, H.Abdel-Khalik

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Data assimilation employs the body of available experimental data to reduce nuclear data uncertainties. The assimilation algorithm is based on the assumption that nuclear data constitute the major source of simulation uncertainty, implying that modeling uncertainties are negligible. This manuscript questions this basic assumption and proposes a further extension of the assimilation algorithm to allow one to differentiate between modeling and nuclear data-introduced uncertainties. This is particularly important when the objective of the analysis is to improve predictions at reactor conditions that are not covered by the available body of experimental data, e.g., prediction of hot reactor conditions based on the assimilation of cold benchmark experiments. We employ representative BWR pin cell models to demonstrate the challenges of the current assimilation algorithm, and compare its performance to the proposed algorithm.

Track4-6 Verification, Validation and Uncertainty Analysis

Session Chair: Laurent Chabert(AREVA TA), Kenji Nishihara(JAEA)

9:40 AM
Sensitivity Analysis via Reduced Order Adjoint Method

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Notwithstanding the voluminous literature on adjoint sensitivity analysis, it has been generally dismissed by practitioners as cumbersome with limited value in realistic engineering models. This perception reflects two limitations about adjoint sensitivity analysis: a) its most effective application is limited to calculation of first-order variations; when higher order derivatives are required, it quickly becomes computationally inefficient; and b) the number of adjoint model evaluations depends on the number of responses, which renders it ineffective for multi-physics model where entire distributions, such as flux and power distribution, are often transferred between the various physics models. To overcome these challenges, this manuscript employs recent advances in reduced order modeling to re-cast the adjoint model equations into a form that renders its application to real reactor models practical. Past work applied reduced order modeling techniques to render reduction for general nonlinear high dimensional models by identifying mathematical subspaces, called active subspaces, that capture all dominant features of the model, including both linear and nonlinear variations. We demonstrate the application of these techniques to the calculation of first-order derivatives, or as commonly known sensitivity coefficients, for a fuel assembly model with many responses. We show that the computational cost becomes dependent on the physics model itself, via the so-called rank of the active subspace, rather than the number of responses or parameters.

Track7-2 Reactor Concepts and Designs

Session Chair: Jan L. Kloosterman(Delft University of Technology), Deokjung Lee(UNIST)

8:00 AM
SmAHTR-CTC Neutronic Design

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Building on prior experience for the 2010 initial Small Advanced High-Temperature Reactor (SmAHTR) neutronic design and the 2012 neutronic design for the Advanced High-Temperature Reactor (AHTR), this paper presents the main results of the neutronic design effort for the newly repurposed SmAHTR-CTC reactor concept. The results are obtained based on full-core simulations performed with SCALE 6.1. The dimensionality of the SmAHTR design space is reduced by using constraints originating in material fabricability, fuel licensing, molten salt chemistry, and thermal-hydraulic and mechanical considerations. The new design represents in many regards a substantial improvement from the neutronic performance standpoint over the 2010 SmAHTR concept. Among other results, it is shown that a fuel cycle length of over two years or discharged fuel burnup of 40 giga-watt-day per metric ton of initial heavy metal are possible with an 8% fuel enrichment in a once-through fuel cycle, while eight-year once-through fuel cycle lengths are possible at higher fuel enrichments.

8:20 AM
Suppression of Excess Reactivity of Small Long-Life Prismatic HTGR with Passive Decay-Heat Removal

O.Sambuu, T.Obara
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Since the accident at Fukushima Daiichi Nuclear Power Plant in 2011, design concepts for nuclear reactors have been reconsidered with much greater emphasis placed upon passive systems for decay-heat removal. The present study focuses on the passive safety features of decay-heat removal in HTGRs. Generally, the feasibility of such features of HTGRs depends on reactor design parameters such as reactor power, maximum power density, initial temperature of core and core size. In a previous work, conditions for design parameters for small, prismatic HTGRs with passive decay-heat removal to be operated continuously for only one-year without temporarily shutting down for regular safety inspection were obtained by residual-heat transfer calculation using equations for fundamental heat transfer mechanisms. In the present study, analogue conditions for long-term-operation reactors were obtained, and comparisons of the results with those previous results were discussed. The appropriate size of reactor core for a 100 MW_t reactor having a maximum power density of 3 W/cm^3 and initial core temperature of 1123 K were obtained. Consequently, criticality and burnup analyses for the proposed reactor with 20 wt% uranium enrichment were performed to confirm the possibility of designing a long-life core with a core size and reactor power which meets the condition of removing decay heat successfully. Optimizations for core burn-up with uniformly distributed burnable poison (BP) particles with various diameters and concentrations were carried out; the objective in these optimizations was to suppress the excess reactivity at the beginning of the cycle and thus flatten the burnup reactivity.

8:40 PM
Conceptual Design of a Self-Sustanable Pressurized Water Reactor with Boiling Channels

M.Margulis(1), E.Shwageraus(2)
1)Ben Gurion University, Beer Sheva, Israel, 2)Cambridge University, Cambridge, UK

Parametric studies have been performed on a seed-blanket Th-U^{233} fuel configuration in a pressurized water reactor (PWR) with boiling channels to achieve high conversion ratio. Previous studies on seed-blanket concepts required substantial reduction of the core power density in order to operate under nominal PWR system conditions. Boiling flow regime in the seed area allows better heat removal, which in turn, may potentially allow increasing the power density of the core. In addition, the reduced moderation improves the breeding performance. A 2-dimensional design optimization study was carried out with BOXER and SERPENT codes in order to determine the most attractive fuel assembly configuration that would ensure breeding. Effects of various parameters, such as void fraction, blanket fuel form, number of seed pins and their dimensions, on the conversion ratio were examined. The obtained results, for which the power density was set to 104 W/cc, created a map of designs with their corresponding fissile inventory ratio (FIR) values. It was found that several options have the potential to achieve the main objective a self-sustainable Thorium fuel cycle in PWRs without significant reduction in the core power density.

9:00 AM
Application of the BigT Burnable Absorber to a Soluble Boron-Free PWR Core

H.Yu, M.Yahya, Y.Kim
KAIST, Daejeon, Korea

A new burnable absorber concept named “Burnable absorber-Integrated control rod Guide Thimble” (BigT) was recently proposed for the Pressurized Water Reactor (PWR). This paper presents the application of the BigT burnable absorber to a soluble boron-free (SBF) Korean PWR core design. Preliminary lattice calculations based on the PLUS7 fuel assembly installed with three types of BigT burnable absorbers were performed to characterize the BigT with gadolinium and boron carbide burnable absorbers. A 3D SBF OPR1000 core loaded with the BigT absorbers was subsequently modeled and depleted to find its equilibrium cycle corresponding to a 3-batch fuel management. In the current configurations, both the BigT and conventional erbia-urania burnable absorbers were simultaneously used to control the core excess reactivity. Preliminary results indicate the promising potentials of the BigT absorbers to realize a SBF PWR as the core reactivity was shown to be properly managed. All neutronic calculations were completed by using the continuous energy Monte Carlo SERPENT code with ENDF/B-VII.0 library.

9:20 AM
Axially Homogeneous Thorium Fuel Designs for Transuranic Burning in Reduced-Moderation BWRs

B.A.Lindley, G.T.Parks
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Reduced-moderation Boiling Water Reactors (RBWRs) can allow sustained burning of transuranics (TRU), such that full actinide recycle can be achieved. However, the void coefficient (VC) tends to become positive with reduced moderation and high TRU loading, which can severely limit the design feasibility and performance. This motivates use of thorium (Th) as an alternative feed to uranium (U), as this tends to result in a more negative VC. Using coupled neutronic-thermal-hydraulic core calculations, Th-fuelled RBWRs are shown to be capable of achieving high discharge burn-ups, while incinerating an external supply of TRU. This gives them some flexibility to incinerate TRU with different isotope vectors. Spatial separation of Th-TRU and Th-U3 into regions of the order of a few thermal neutron diffusion lengths greatly improves neutronic performance. This can be accomplished radially or axially, but radial separation results in significantly easier fuel fabrication. A companion paper considers axially heterogeneous fuel designs, which are not found to deliver improved performance. A radially heterogeneous RBWR core design, can achieve 135 kg/GWthyr waste incineration rate, and at least 70–80 GWd/t discharge burn-up, depending on the TRU cooling time, and whether a single pass through a conventional PWR is used as an initial step before full recycle in an RBWR. Alternatively, a micro-heterogeneous assembly design with Th-Pu and Th-U3-MA pins can reduce fuel fabrication costs, although this limits the TRU incineration rate to ~100 kg/GWthyr. ~200 cm and ~120 cm core heights are considered, with the former being generally preferred due to the possibility to match both the rating and pressure vessel size of an ABWR.

9:40 AM
Neutronic Analysis of a Micro Modular Reactor

F.Venneri(1), C.Jo, J.Chang(2), A.Hawari(3)
1)USNC, Los Alamos, USA, 2)KAERI, Daejeon, Korea, 3)NC State University, Raleigh, USA

Small Modular Reactor of few mega watts range meets power need at remote area when it can be “factory manufactured” and extremely safe. We have developed a concept of Micro Modular Reactor based on the FCM compact fuel which has superior safety characteristics. We have made a preliminary neutronic analysis to confirm the long life time which is essential for remote application, and to check the controllability based on the criticality analysis. A core of 200 cm diameter and 200 cm or 300 cm long cylinder shape can achieve the core life of 5 ~ 10 years with 20 ~ 10% LEU. When the rated power is reduced by design, the service period can be extended to 20 ~ 40 years.

Track6-2 Reactor Physics Experiments

Session Chair: Jacques Di Salvo(CEA), Kenichi Yoshioka(Toshiba)

8:00 AM

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

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The bunching technique has been widely utilized in Feynman-α neutron correlation analysis to synthesize neutron counts of longer gate widths by bunching neutron counts stored in adjacent MCS channels. An alternative technique referred to as "moving-bunching technique" was proposed to reduce a statistical scatter of variance-to-mean ratio of neutron counts. The conventional bunching technique has no overlap of adjacent bunches, while the present technique has a much longer overlap to filter white noise similarly to the moving-average technique. A Feynman-α experiment was performed in the UTR-KINKI, to confirm the applicability and advantage of the proposed bunching technique. When a neutron detector was placed far from the core, a Feynman-α analysis with the conventional bunching technique led to a scattered variance-to-mean ratio from which prompt-neutron decay constant was never determinable. However, another analysis with the proposed moving-bunching technique gave a successful result even for such a remote detector. For a neutron detector close to the core, the proposed technique resulted in a slight reduction of statistical error of the decay constant.

8:40 PM

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

A.Kochetkov, P.Baeten, W.Uyttenhove, G.Vittiglio, J.Wagemans(1), A.Billebaud, S.Chabod(2), J.-L.Lecouey, G.Ban, F.-R.Lecolley, N.Marie, G.Lehaut(3), X.Doligez(4), F.Mellier(5)

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As a prolongation of the FP-6 GUINEVERE (Generation of Uninterrupted Intense NEutron pulses at the lead VEnus REactor) project at the VENUS-F facility at SCK• CEN, the FP-7 FREYA (Fast Reactor Experiments for hYbrid Applications) project was launched in 2011. The FREYA project aims at validating the methodology for sub-criticality monitoring of a subcritical core as proposed in the MUSE experiments conclusions. This is needed for the reactivity monitoring of a powerful ADS (Accelerator Driven System) like MYRRHA. Experiments are also needed for sub-critical and critical ADS and LFR (Lead Fast Reactor) core characterisation for benchmarking and licensing purposes. Several levels of sub-criticality of the first sub-critical core were measured applying the integral Source Jerk method and compared to the reference results obtained with the MSM (Modified Source Multiplication) method.

8:20 AM

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

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The FREYA project (Fast Reactor Experiment for hYbrid Applications) investigates various aspects connected to the physics and operation of Accelerator Driven Systems (ADS). The experimental facility consists of the GENEPI-3C deuteron accelerator coupled to the VENUS-F fast reactor. The study of robust and convenient methods for online reactivity monitoring in different operating conditions is one of the primary goals of the FREYA project. A study is presented of two such methods applied to a deep-subcritical configuration ($k_{eff} \approx 0.90$) of the VENUS-F reactor: the area method, which can be performed in Pulsed Neutron Source (PNS) experiments, and the source jerk technique, suitable in case of a continuous source. Both strategies require, to some extent, input from models or simulation. Theoretical and experimental uncertainties are discussed.

9:00 AM

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

T.M.Chevret, J.-L.Lecouey, N.Marie, F.-R.Lecolley, G.Lehaut, G.Ban(1), A.Billebaud, S.Chabod(2), X.Doligez(3), A.Kochetkov, P.Baeten, W.Uyttenhove, G.Vittiglio, J.Wagemans(4), F.Mellier(5), V.Bécares, D.Villamarin(6)

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In order to incinerate minor actinides and thus to reduce the issues linked to nuclear waste management, Accelerator-Driven Systems (ADS) are being under study. An ADS consists in the coupling of a particle accelerator with a sub-critical fast reactor. The on-line reactivity monitoring is a serious issue regarding safety, therefore several methods to estimate the reactivity of such sub-critical systems have to be investigated. Here, we present one method based on the study of the neutron population evolution during beam interruption experiments carried out in the framework of the FREYA FP7 program [1,2] at the GUINEVERE facility, which couples the fast lead sub-critical reactor VENUS-F with the deuteron accelerator GENEPI-3C at SCK-CEN in Mol, Belgium. After describing the facility, the analysis based on point kinetics theory and preliminary results of the reactivity measurements will be presented. Then, spatial effects that are not taken into account by point kinetics theory will be highlighted using MCNP simulations, and correction factors to raw results will be calculated. In the end, final results will be compared to reference reactivity values obtained with the Modified Source Multiplication (MSM) method.

9:20 AM

Reactivity Measurements at GUINEVERE Facility Using the Integral k_p Method

S.Chabod, A.Billebaud(1), F.-R.Lecolley, J.-L.Lecouey, G.Lehaut, N.Marie, G.Ban(2), X.Doligez(3), A.Kochetkov, P.Baeten, A.Krása, W.Uyttenhove, G.Vittiglio, J.Wageman(4), F.Mellier(5), D.Villamarin(6)

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In the framework of the GUINEVERE (EUROTRANS-IP FP6) and FREYA (FP7) projects, a two-level procedure is developed to measure the time variations of an ADS reactivity ρ(t). First, the time variations of the reactor power and neutron source intensity are monitored online, allowing to extract the fast fluctuations of ρ(t) around a reference value. Then, absolute measurements of ρ are periodically performed to readjust the relative measurements. These calibration measurements require the analysis of the decay of the neutron population triggered by programmed beam interruptions. In this paper, we describe an innovative method developed to measure the prompt multiplication factor, k_p, of a subcritical reactor (and therefore its absolute ρ). Unlike conventional methods used for reactivity measurements, our approach only relies on the prompt decay of the neutron population, therefore enabling the use of shorter beam interruptions (< 50 μs) than usual (~ 1 ms). After a review of the method's mechanics, we apply it to measurements performed at the GUINEVERE facility, with the VENUS-F reactor, a zero power lead subcritical fast reactor. During these measurements, the neutron source was operated in pulse mode. Our results, promising, are compared with reference reactivity values provided by a MSM analysis.

Track6-2 Reactor Physics Experiments

Session Chair: Jacques Di Salvo(CEA), Kenichi Yoshioka(Toshiba)

9:40 AM

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

E.Gilad, O.Rivin, H.Ettegui, I.Yaar(1), B.Geslot, A.Pepino, J.Di Salvo, A.Gruel, P.Blaise(2)

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A method for determining the effective delayed neutron fraction β_{eff} using in-pile reactivity oscillations and Fourier analysis is presented. This method is based on measurements of the reactor's power response to small periodic in-pile reactivity perturbations and utilizes Fourier analysis for reconstruction of the reactor zero power transfer function.This approach enables the estimation of β_{eff} using multi-parameter nonlinear weighted least-squares fit. The method extends previous work (Yedvab et al. 2006) by accounting for higher harmonics excitation in the frequency domain by the trapezoidal reactivity signal, both in the reactivity perturbation and in the reactor power response. We show that by using this new approach it is possible to obtain the reactor transfer function in a wide range of frequencies, using only a single oscillation frequency. This method is applied to a set of measurements of the MAESTRO core configuration in the MINERVE Zero Power Reactor located at the Cadarache center (CEA). The derived value of β_{eff}, using this method, is 711±17 (1σ) pcm, which is in agreement with the calculated value of 716 pcm.

Track10-1 Nuclear Data

Session Chair: Pierre Leconte(CEA), Junichi Hori(KURRI)

8:00 AM

COMAC: Nuclear Data Covariance Matrices Library for Reactor Applications

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CEA Cadarache, Saint Paul-lez-Durance, France

Uncertainty quantification for new reactor concepts such as the ASTRID project (sodium-cooled FBR studies at CEA) requires a precise evaluation of nuclear data covariances. Since 2005, the SPRC at CEA-Cadarache has developed a covariance matrices library, called COMAC (COvariance MAtrices from Cadarache, version 0.1), for the most important nuclear data: cross sections, multiplicities (prompt and delayed neutrons), prompt fission neutron spectrum (PFNS), fission yields. This paper presents the methodology used to produce those covariance matrices, illustrated with representative cases (^{23}Na, ^{238}U, ^{239}Pu, ^{240}Pu). We have used the CONRAD evaluation code to analyse simultaneously differential and specific integral experiments and to generate a coherent and realistic set of covariances. The quality of those covariances are illustrated with uncertainty propagation calculations on key parameters for ASTRID.

8:20 AM

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

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

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Model-based fission product yields from the fission of various important target nuclides have been calculated for incident neutron energies up to 20 MeV, divided into a 77 energy group structure. The calculations have been performed with two versions of the GEF code, which have been externally coupled to TALYS-1.4. In this application, the TALYS-1.4 code calculates any pre-fission nucleon or gamma emission from the compound nucleus as well as the probabilities of excitation states at the time it undergoes fission. The obtained quantities, fully characterizing the fissioning nucleus, are then passed to GEF, which generates the corresponding primary fission product yields in a Monte Carlo calculation. Cumulative fission product yields have been calculated using these primary fission product yields together with evaluated radioactive decay data as input. The interim and final results from the modelling, i. e. cross-sections, independent and cumulative fission yields, have been compared to experimental data. Important results from this, as well as sensitivities and reliabilities of the models, are discussed in this paper. The objective of this work was to generate energy dependent fission product yields data to serve as a basis for further investigations on potential improvements of evaluated data for nuclear reactor applications, which are beyond the scope of this publication.

8:40 PM

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

The delayed neutron nuclear data are limited by its short lifetime, limited yield fraction, difficulty to obtain a well-defined sample, an inefficient (uncharacterized) experimental system, and numerical instabilities of fitting methods. The uncertainties on the relative abundances of the longer-lived delayed neutron groups are up to 13% for thermal fissions of ^{235}U and 16% for fast fissions of ^{238}U. To evaluate more accurate delayed neutron nuclear data, a novel approach to experimentally measure the relative uncertainties on the delayed neutron group abundances has been proposed by Jordan and Perret. This approach combines delayed gamma and delayed neutron fission rate measurement techniques; with two independent estimates of the same fission rate, higher uncertainty delayed neutron parameters can be linked to lower uncertainty delayed gamma parameters. The present authors have designed, optimized, and built an experimental apparatus capable of producing high accuracy measurements for delayed gamma-rays and neutrons, with the goal of implementing the above approach. Finally, sensitivity studies of the parameters that have non-negligible contributions to the overall uncertainty were carried out. With regards to the designed setup and the combined technique, the relative uncertainty of the longest-lived delayed neutron group is calculated. The calculation shows that the uncertainty of the longest-lived delayed neutron group of ^{238}U can be reduced by a factor of two to 8.81%.

9:00 AM

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

A.I.Hawari, B.W.Wehring

North Carolina State University, NC, USA

Experimental measurements and computational simulations were performed to study the slowing down and thermalization of neutrons in graphite. The experiment was conducted at the ORELA facility of Oak Ridge National Laboratory and utilized the slowing-down-time technique to observe the behavior of ORELA neutron pulses that were injected into a rectangular reactor-grade graphite pile with dimensions of 70×70×70 cm3. The time dependent reaction rate of neutrons leaking from the pile was recorded in two Li-6 glass scintillation detectors that were placed on top and in the back of the pile. Simulations of the experiment were performed using the MCNP5 Monte Carlo code to perform basic design and to reproduce the experimental results.To examine neutron thermalization effects, the simulation invoked thermal neutron scattering, i.e., S(α,β), treatment for graphite by using the thermal scattering data libraries as given in the ENDF/B-VII database. The comparison of experimental and computational data showed good agreement at early times that correspond to leaking neutron energies in the epithermal and fast range. However, noticeable differences were found between the measured and calculated time spectra as the neutrons reached the thermal energy range. These findings motivate examining the validity of using the ENDF/B-VII graphite S(α,β) libraries to capture accurately the phenomenon of neutron thermalization in reactor-grade graphite, which is used in various nuclear systems including advanced and Generation IV nuclear reactors.

9:20 AM

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

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

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We have measured the neutron capture cross section of ^{232}Th by the neutron time-of-flight (TOF) method using a 12.7 m flight path and a 46-MeV electron linear accelerator (linac) at the Research Reactor Institute, Kyoto University (KURRI). For the capture cross section measurement, an assembly of 12 pieces of $Bi_4Ge_3O_{12}$ (BGO) scintillators was used as a total energy absorption detector. The incident neutron spectrum on the sample was determined with the ^{10}B(n,αγ) standard reaction and the relative capture cross section of ^{232}Th was normalized to the evaluated values of JENDL-4.0 at the 21.8-eV resonance. The present results were compared with the existing experimental values and the evaluated data in JENDL-4.0.

9:40 AM

New Revisions of Reactor Physics Standards

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This paper presents an overview of two of the most basic reactor physics standards, the newly revised “Steady State Neutronics Methods for Power Reactor Analysis” and its companion standard, “Nuclear Data for Reactor Design”. These two popular state-of-the-art standards provide important guidance for developing the necessary input data needed to calculate reactor lattice and core parameters such as reaction rates spatial distributions, reactivity and flux distributions in power reactors for all currently used reactor types, from fast to thermal reactors. The standards provide guidance for the selection of cross section data and libraries, the development of nuclear data sets suitable for specific applications, energy group structures and group collapsing. Key elements in the complex sequence of power reactor calculations are outlined. The effects of simplifications and approximations made in the treatment of the neutronic and geometric models and the biases and uncertainties resulting from such simplifications and assumptions are discussed. In the newly-revised standard on neutronics methods, clear distinction is made between the two important tools for assessing the reliability of

Track10-1 Nuclear Data

Session Chair: Pierre Leconte(CEA), Junichi Hori(KURRI)

the results of the calculations, verification and validation. To provide an auditable path in the verification and validation processes, the standard requires detailed documentation including methods used, selection of calculation models and experimental data and results of higher order calculations.

Track1-11 Reactor Analysis Methods

Session Chair: Alain Hebert(Ecole Polytechnique de Montreal), Ren-Tai Chiang(Energy Engineering Service)

10:20 AM

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

A.Hébert

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The Clio perturbative approach is a powerful tool available in the APOLLO2 lattice code for performing a detailed reactivity analysis between two heterogeneous systems in fundamental mode conditions. It is possible to compute group wise and isotope wise reactivity effects due to a modification in one of the two systems. This modification can be related to isotopic composition, temperature, cross-section uncertainties, or even a swap of solution algorithm. The Clio perturbative approach is an exact method based on the knowledge of the adjoint flux of the unperturbed system. Our contribution consists in a new normalization scheme for distributing the group wise reactivity effects due to the leakage term of the transport equation in a way that is consistent to both the direct and adjoint equations.

10:40 AM

A Generalization of λ-Mode Xenon Stability Analysis

J.M.Pounders, J.D.Densmore

Bettis Atomic Power Laboratory, PA, USA

A new method for analyzing xenon transients in nuclear reactors based on λ-mode stability analysis is developed. Like previous work, the new analysis method is based on a linearization of the coupled diffusion, iodine, and xenon equations, but the approach is generalized here to permit flux perturbations that include a component in the fundamental spatial mode and thus a complete expansion of any perturbation. This extension also allows power to be explicitly constrained. The typical constant-reactivity approach to λ-mode analysis appears as a special case of the new generalized theory.

11:00 AM

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

J.Kim, Y.Kim

KAIST, Daejeon, Korea

In the standard analysis model of the Xenon worth in thermal reactors, Xe-135m has been neglected due to its short half-life (15.29 min). The impact of Xe-135m during a transient was briefly analyzed using point reactor model with the TENDL-2011 and TENDL-2013 data that it will lead the Xenon worth to be increased. These results are then applied to the PCR measurement based on the measured data from Wolseong2 reactor in order to analyze the potential impacts of Xe-135m on the PCR of CANDU reactors. When the Xenon worth is increased, the measured PCR is generally decreased by 0.54pcm/%P with TENDL-2011, and by 0.20pcm/%P with TENDL-2013 on average.

11:20 AM

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

Y.Bilodid, E.Fridman(1), M.Margulis(2), D.Kotlyar, E.Shwageraus(3)

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Recently, a new method for accounting for burnup history effects on few-group cross sections was developed and implemented in the reactor dynamic code DYN3D. The method relies on the tracking of the local Pu-239 density which serves as an indicator of burnup spectral history. The validity of the method was demonstrated in PWR and VVER applications. However, the spectrum variation in BWR core is more pronounced due to the stronger coolant density change. Therefore, the purpose of the current work is to further investigate the applicability of the method to BWR analysis. The proposed methodology was verified against recently developed BGCore system, which couples Monte Carlo neutron transport with depletion and thermal hydraulic solvers and thus capable of providing a reference solution for 3D simulations. The results clearly show that neglecting the spectral history effects leads to a very large deviation (e.g. 2000 pcm in reactivity) from the reference solution. However, a very good agreement between DYN3D and BGCore is observed (on the order of 200 pcm in reactivity), when the Pu-correction method is applied.

11:40 AM

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

A.Calloo, H.Leroyer, M.Fliscounakis, D.Couyras

EDF R&D/SINETICS, Clamart, France

EDF R&D is presently developing a new, state-of-the-art calculation chain called ANDROMEDE including the APOLLO2/JEFF3-based CEA multigroup library/REL2005 scheme package for assembly computations and COCAGNE 3D code for core computations. The goal of this paper is to validate the calculation chain and its methodologies on a numerical benchmark of a small PWR which has been loaded with mixed fuel, KAIST 1A. The latter is challenging, being highly heterogeneous as it has assemblies with burnable poison, offers a rodded configuration and includes both UOX-MOX and core-reflector interfaces. Thus, we will test the capabilities of the models used in ANDROMEDE to compute such cores. The validation methodology employed is as follows: stochastic calculations are used to validate the ability of assembly schemes SHEM-MOC and REL2005 for the computation of 2D full cores. Afterwards, industrial two-group diffusion calculations were set up. Reactivity coefficients and pin-by-pin power distributions were compared with those obtained from REL2005. Finally, the last section gives the prospects of the use of multigroup SPn for industrial calculations. They raise several questions such as the energy meshes to be used as well the 2D reflector model to be applied. A reflector model is set up to test the SPn solver on full-core calculations with results compared to those of the REL2005 scheme.

Track4-7 Verification, Validation and Uncertainty Analysis

Session Chair: Maria Pusa(VTT), Tadafumi Sano(KURRI)

10:20 AM

Sensitivity and Uncertainty Analysis on Reactor Kinetic Parameters Using Perturbation Theory

C.Bouret, L.Buiron, G.Rimpault

CEA Cadarache, France

The analysis of nuclear reactors behavior in transient depends on the neutronic parameters including the effective delayed neutron fraction (βeff) and its family dependent components (βeff), the neutron generation lifetime (l_p), and the decay constants (λ_i) for each delayed neutron family. With the aim of an accurate assessment of neutron parameters used to simulate transient, uncertainties on these parameters must be estimated by considering the contributions of different nuclear data, including those from the delayed data (fission yield v_d and fission spectrum χ_d). This paper presents a methodology based on the generalized perturbation theory which allows the assessment of kinetic parameters sensitivities to nuclear data. The methodology is applied to the heterogeneous core design "CFV" considered for sodium-cooled reactor ASTRID. The contributions of indirect term and direct terms ($\sigma_{fission}$ v and $v_{d,}$ χ and χ_d) are decomposed by isotopes and reactions and uncertainties on the kinetic parameters are calculated using the most recent data covariance suggested in COMAC (COvariance Matrices from Cadarache, version 0.1).

10:40 AM

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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Criticality, reactor physics, fusion and shielding benchmarks are expected to play important roles in GENIV design, safety analysis and in the validation of analytical tools used to design these reactors. For existing reactor technology, benchmarks are used to validate computer codes and test nuclear data libraries. However the selection of these benchmarks are usually done by visual inspection which is dependent on the expertise and the experience of the user and thereby resulting in a user bias in the process. In this paper we present a method for the selection of these benchmarks for reactor applications and uncertainty reduction based on Total Monte Carlo (TMC) method. Similarities between an application case and one or several benchmarks are quantified using the correlation coefficient. Based on the method, we also propose two approaches for reducing nuclear data uncertainty using integral benchmark experiments as an additional constraint in the TMC method: a binary accept/reject method and a method of uncertainty reduction using weights. Finally, the methods were applied to a full Lead Fast Reactor core and a set of criticality benchmarks.

11:00 AM

DICE 2013: New Capabilites and Data

I.Hill, J.Gulliford, N.Soppera, M.Bossant

OECD Nuclear Energy Agency, Issy-les-Moulineaux, France

The validation of the codes and nuclear data libraries for reactor applications relies widely on the International Handbook of Evaluated Criticality Safety Benchmark Experiments. DICE, the Database for ICSBEP, is distributed with the Handbook to assist users to identify relevant validation benchmarks. This paper describes and provides examples of updates made to the DICE software in 2013 that significantly improve the tool. Data available to handbook users vastly increased between the 2012 and the 2013 edition, as the number of sensitivity data files distributed grew from approximately 700 to 3500, or from 15% to 75% coverage of cases. In conjunction, a new searching feature for three group sensitivity data was implemented allowing users to identify the set of benchmarks with the most sensitivity to a particular nuclear reaction. The new sensitivity data has been compared to existing DICE data, illustrating the agreement of sensitivity data and neutron balance data within DICE. This is shown to be an effective tool for both searching evaluations and detecting data entry errors. Additionally, DICE has implemented a trend plot feature to create plots similar to those appearing in validation publications using ICSBEP data. Users can incorporate their own data into these plots, to compare different codes or nuclear data libraries. Finally, DICE is now publically available as a web-start via the NEA website.

11:20 AM

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

I.C.Gauld(1), N.Sly(2), F.Michel-Sendis(3)

1)Oak Ridge National Laboratory, Tennessee, USA, 2)University of Tennessee, Tennessee, USA, 3)OECD NEA Data Bank, Issy-Les-Moulineaux, France

Experimental data on the isotopic concentrations in irradiated nuclear fuel represent one of the primary methods for validating computational methods and nuclear data used for reactor and spent fuel depletion simulations that support nuclear fuel cycle safety and safeguards programs. Measurement data have previously not been available to users in a centralized or searchable format, and the majority of accessible information has been, for the most part, limited to light-water-reactor designs. This paper describes a recent initiative to compile spent fuel benchmark data for additional reactor designs used throughout the world that can be used to validate computer model simulations that support nuclear energy and nuclear safeguards missions. Experimental benchmark data have been expanded to include VVER-440, VVER-1000, RBMK, graphite-moderated MAGNOX, gas-cooled AGR, and several heavy-water moderated CANDU reactor designs. Additional experimental data for pressurized light water and boiling water reactor fuels have also been compiled for modern assembly designs and more extensive isotopic measurements. These data are being compiled and uploaded to a recently revised structured and searchable database, SFCOMPO, to provide the nuclear analysis community with a centrally accessible resource of spent fuel compositions that can be used to benchmark computer codes, models, and nuclear data. The current version of SFCOMPO contains data for eight reactor designs, 20 fuel assembly designs, more than 550 spent fuel samples, and measured isotopic data for about 80 nuclides.

11:40 AM

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

W.Zheng, R.G.McClarren

Texas A&M University, Texas, USA

The thermal neutron scattering cross sections of ZrHx are heavily affected by the solid frequency distributions, also called “phonon spectra”, of Zr and H in ZrH_x. Although the phonon spectra vary for different x in ZrH_x, current reference data, e.g. ENDF, are based on ZrH_2. This may introduce non-negligible errors in the simulations for TRIGA reactors. In the previous work, we have proposed parameterized phonon spectrum (PPS) models to explore the effects of changing the spectra by varying the parameters and investigated the effects on reactivity and fuel fission rate density on TRIGA lattice model.
In this work, we extend the analyses to quantities of interest (QOIs) on the realistic full-core geometry of the TRIGA reactor at Texas A&M University. In this work, we sampled the parameters with Latin Hypercube sampling designs (LHS) in a novel way and generated corresponding phonon spectra. NJOY and MCNP were used to carry out the calculations. We investigated reactivity (ρ), neutron mean generation time (Λ), fuel temperature feedback coefficient α_T^{Fuel}(293.6 - 600 K) and ex-core/in-core detector reaction rates. Statistical analyses indicate ρ, Λ and α_T^{Fuel} are sensitive to the parameters while other QOIs are not.
We calibrated the parameters for ENDF-VII as a surrogate for experimental data. Results show the feasibility of the parameter calibrations. Future work will perform experiments to archive QOIs and to calibrate the parameters in the PPS model to generate thermal scattering cross sections used in future simulations.

Track8-2 Reactor Operation and Safety

Session Chair: Fausto Franceschini(WH), Naoyuki Nakadozono(Hitachi)

10:20 AM

Advanced Surveillance of Resistance Temperature Detectors in Nuclear Power Plants

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The dynamic response of several RTDs located at the cold leg of a PWR has been studied. A theoretical model for the heat transfer between the RTDs and the surrounding fluid is derived. It proposes a two real poles transfer function. By means of noise analysis techniques in the time domain (autoregressive models) and the Dynamic Data System methodology, the two time constants of the system can be found. A Monte Carlo simulation is performed in order to choose the proper sampling time to obtain both constants. The two poles are found and they permit an advance in situ surveillance of the sensor response time and the sensor dynamics performance. One of the poles is related to the inner dynamics whereas the other one is linked to the process and the inner dynamics. So surveillance on the process and on the inner dynamics can be distinguished.

10:40 AM

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

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1)Chalmers University of Technology, Gothenburg, Sweden, 2)Ringhals AB, Väröbacka, Sweden, 3)Technical University of Madrid (UPM), Madrid, Spain

Surveillance and diagnostics of core barrel vibrations has been performed in the Swedish Ringhals PWRs for several years, with main focus on the pendular motion (beam mode). The monitoring of the beam mode showed that the amplitude of the corresponding peak in the ex-core neutron spectra increases along the cycle, and decreases after refueling. Previous investigations on the reason of this behaviour, i.e. whether it is due to the increase of the core barrel vibration amplitude or to the increase of the neutron physics coupling between vibrations and neutron noise, were not decisive. The objective of the work reported here is to clarify this question. From frequency analysis, two modes of vibration have been identified in the frequency range of the beam mode. Several results coming from the trend analysis performed during recent years indicate that one of the modes is due to the core barrel motion itself and remains constant during cycle, and the other is due to the individual flow induced vibrations of the fuel elements, showing an increasing trend during the cycle. In this work, the method to separate the contributions from the two modes has been refined, and the results of this approach to the latest measurements are presented. The results confirm the origin of the two vibration modes and show constant amplitude of the core barrel motion throughout the cycle.

11:00 AM

Assessment of Flow Induced Vibration Limits in Preliminary I^2S-LWR Fuel Designs

G.E.Sjoden, M.Chin, C.Yi, B.Petrovic

Georgia Institute of Technology, Atlanta, USA

The Integral Inherently Safe Light Water Reactor (I^2S-LWR) is a novel PWR concept being developed by a multi-institutional team, led by Georgia Tech, under the auspices of the Department of Energy's Nuclear Energy University Programs Integrated Research Projects (DOE NEUP IRP). The I^2S-LWR aims at delivering an electric power of ~1 GW while, simultaneously, achieving an overall level of safety that is enhanced with respect to GW-class Generation III+ LWRs, including considerations for "accident tolerant fuels". The adoption of inherent safety features and unconventional materials for the main core components are key design features intended to permit the I^2S-LWR to achieve the design objectives. This work summarizes a preliminary approach to identify Flow Induced Vibration (FIV) limits for new fuel designs proposed for the I^2S-LWR. While not a substitute for a detailed, system-level Computational Fluid Dynamics (CFD) analysis, the approach presented here provides a methodology assembled from "best practices" documented in the literature to establish design thresholds and enable designs that resist FIV damage. This is an essential task to evaluate early in the design process, simply because the best designs considering other perspectives may inherently fail due to FIV related causes. Because different fuels behave differently, we identify the essential design considerations and focus on the main fuel candidates for the I^2S-LWR, i.e. UO_2 and U_3Si_2, augmenting the fuel material selection process and the fuel rod design in general.

11:20 AM

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

H.Y.Jeong, M.G.Park(1), S.H.Seong, J.H.Jeong(2)

1)Sejong University, Seoul, Korea, 2)Korea Atomic Energy Research Institute, Daejeon, Korea

An early detection of flow blockage in a subassembly is one of the important concerns for the safety of a Sodium-cooled Fast Reactor. We propose a concept of blockage index based on the temperature change at measurement point for each subassembly and combining it with a kernel function for the relative position of subassemblies. The effectiveness of the proposed method as a reactor protection parameter is evaluated and compared with the existing methods. We see the possibility of increasing the reliability of blockage detection and also of detecting the occurrence of blockage at an early stage if we use the blockage index.

11:40 AM

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

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In order to increase the protection of the reactor pressure vessel beltline weld against fast neutron flux a second generation shielding assemblies are planed to be designed and installed in the nearest future at Ringhals 3 and 4. These assemblies might be designed with three axial zones. The top and bottom zones contain low enriched ^{235}U while the central shielding part contains only steel rods. Different designs with varying ^{235}U enrichment and length of the central zone is studied in order to find a balance between the desired shielding factor at the beltline weld and core design feasibility. The most favorable design was found to have 7 steel nodes (106.68 cm), 5 bottom nodes (76.20 cm) and 12 top nodes with standard fuel rods containing 1.5 w/o ^{235}U. This design results in an estimated shielding factor of 7.8 ± 0.7 and has small impact on cycle length and core design. The reduced number of fuel pins due to the shielding steel zone will result in a $F_{\Delta H}$ penalty of 2.2 %.

Track6-3 Reactor Physics Experiments

Session Chair: Nicholas Brown(BNL), Hironobu Unesaki(KURRI)

10:20 AM

BNL Metal Fuel Lattice Experiments: Candidates for Reactor Physics Benchmark Evaluation

N.R.Brown, A.Aronson, M.Todosow

Brookhaven National Laboratory, NY, USA

The paper presents an argument that some legacy experiments conducted by Brookhaven National Laboratory and Westinghouse Atomic Power Division may be interesting candidates for evaluation as part of the OECD/NEA International Reactor Physics Experiment Evaluation Project. Material buckling measurements from several experiments with water-moderated lattices of uranium metal rods (clad in Al) are compared with scoping calculations from two well-known lattice tools: the TRITON/NEWT deterministic code from the SCALE package and the Serpent Monte Carlo code. In the case of Serpent the measured material buckling agrees well with the calculations, with the exception of cases where both the moderator-to-fuel ratio is high (4.0) and the fuel rod diameter is large (0.600"). The SCALE results show a small systematic bias when compared with the data from the experiments. Measured quantities that could be considered as part of a future benchmark evaluation include: criticality, material buckling, migration area, four factor formula parameters, intercell and intracell flux traverses, and kinetics parameters.

10:40 AM

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

J.Bess

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Benchmark models were developed to evaluate the cold-critical start-up measurements performed during the fresh core reload of the Neutron Radiography (NRAD) reactor with Low Enriched Uranium (LEU) fuel. The final upgraded core configuration with 64 fuel elements has been completed. Evaluated benchmark measurement data include criticality, control-rod worth measurements, shutdown margin, and excess reactivity. Dominant uncertainties in keff include the manganese content and impurities contained within the stainless steel cladding of the fuel and the 236U and erbium poison content in the fuel matrix. Calculations with MCNP5 and ENDF/B-VII.0 nuclear data are approximately 1.4% greater than the benchmark model eigenvalue, supporting contemporary research regarding errors in the cross section data necessary to simulate TRIGA-type reactors. Uncertainties in reactivity effects measurements are estimated to be ~10% with calculations in agreement with benchmark experiment values within 2σ. The completed benchmark evaluation details are available in the 2014 edition of the International Handbook of Evaluated Reactor Physics Experiments (IRPhEP Handbook). Evaluation of the NRAD LEU cores containing 56, 60, and 62 fuel elements have also been completed, including analysis of their respective reactivity effects measurements; they are also available in the IRPhEP Handbook but will not be included in this summary paper.

11:00 AM

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)

1)LANL, NM, USA, 2)CEA, France

CALIBAN and Godiva-IV are both fast burst reactors with HEU cores. A description and comparison of the two assemblies is presented. Joint LANL and CEA measurements were conducted to measure configurations on the CALIBAN and Godiva assemblies using He-3 detector systems. Measured configurations included many subcritical configurations at various reactivity states as well as configurations at and above delayed critical. The subcritical and delayed critical measurements were analyzed using the Hage-Cifarelli formalism of the Feynman Variance-to-Mean method. This analysis can be used to calculate the multiplication of each configuration, from which the multiplication factor and reactivity are determined. These results are compared to the reactivity states based upon control rod worth curves and MCNP simulations. For each delayed critical configuration an estimate of the delayed neutron fraction is also calculated. Traditionally subcritical neutron noise methods have not been used on systems at, near, or above delayed critical. This work discusses the use and problems encountered of such methods in this regime. Count rate measurements from the positive period configurations were also very accurate but correlated neutron analysis for these configurations was not successful. Very few data sets which can be analyzed using neutron multiplicity analysis exist over such a large range of reactivity values (-$20 to +$0.14).

11:20 AM

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

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1)Interdisciplinary Graduate School of Science and Engineering, Kinki University, Osaka, Japan, 2)Nuclear Engineering Science Division, Research Reactor Institute, Kyoto University, Osaka, Japan

A series of power spectral analyses for a thermal subcritical reactor system driven by a pulsed spallation neutron source was carried out at Kyoto University Critical Assembly (KUCA), to determine the prompt-neutron decay constant of the Accelerator-Driven System (ADS). High-energy protons (100MeV) obtained from the fixed field alternating gradient accelerator were injected onto a lead-bismuth target, whereby the spallation neutrons were generated. In the cross-power spectral density between time-sequence signal data of two neutron detectors, many delta-function-like peaks at the integral multiple of pulse repetition frequency could be observed. However, no continuous reactor-noise component could be measured. This is because these detectors have too high count-rate to be placed closely to the core. From the point data of these delta-function-like peaks, the prompt-neutron decay constant could be determined. At a slightly subcritical state, the decay constant was consistent with that obtained by a previous power spectral analysis for a pulsed 14MeV neutron source and by a pulsed neutron experiment. At another deeply subcritical state, however, the present analysis leads to an underestimate of the decay constant.

11:40 AM

Equivalency of Open Loop and Closed Loop Reactivity Measurement Techniques

B.A.Baker, G.R.Imel

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There is a need for integral physics data on reactivity worth of minor actinides (transmutation studies) and fission products (burn-up credit) for validation of differential data. In France, the MINERVE facility has been used to determine low-worth reactivity measurements using a closed-loop oscillator technique. However, it has been deemed unfeasible to perform such measurements in a simulated fast reactor spectrum in MINERVE as was done in the past. We propose that reactivity worth measurements could be directly performed in the fast reactor MASURCA using an open loop oscillator technique. Theoretically, these two methods should be equal in how well a measurement can be known. This paper compares open loop techniques (pile oscillator method analyzed with harmonic analysis and inverse kinetics) with the closed loop technique (reactivity oscillator method). Open and closed loop techniques are defined in the classic control sense – closed loop utilizes feedback while open loop has no imposed feedback. Experiments were performed to show variations based on frequency and power as well as the determination of a small worth sample on the order of 0.04 cents with standard deviations on the order of 0.003 cents. All techniques were shown to produce results that were limited only by reactor noise.

SS4 Advanced Geometry Processing in Deterministic and Monte Carlo Methods

Session Chair: Hyung Jin Shim(Seoul National Univ.), Shinya Kosaka(MHI)

10:20 AM

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

S.G.Hong(1), J.W.Kim, D.H.Kim, Y.O.Lee(2)

1)Kyung Hee University, Gyeonggi-do, Korea, 2)Korea Atomic Energy Research Institute, Daejon, Korea

This paper reviews the computational methods used in the MUST (Multi-group Unstructured geometry S_N Transport) code for solving the multi-group S_N transport equation in general geometries and describes the status of development of MUST. MUST solves the multi-group transport equation with unstructured tetrahedral meshes for modeling complicated geometrical problems. For tetrahedral mesh generation, input generation, and output visualization, we developed a management program where the mesh generation is based on Gmsh and TetGen that are open softwares. The geometrical modeling is done with the commercial CAD softwares such as CATIA. MUST uses the discontinuous finite element method (DFEM) and two-sub cell balance methods with linear discontinuous expansion (LDEM-SCB) to spatially discretize the transport equation. We applied MUST to three neutron and gamma coupled test problems for testing MUST.

10:40 AM

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

J.Leppanen(1), M.Aufiero(2)

1)VTT Technicl Research Centre of Finland, VTT, Finland, 2)Politecnico di Milano, Milano, Italy

This paper presents a new unstructured mesh based geometry type, developed in the Serpent 2 Monte Carlo code as a by-product of another study related to multi-physics applications and coupling to CFD codes. The new geometry type is intended for the modeling of complicated and irregular objects, which are not easily constructed using the conventional CSG based approach. The capability is put to test by modeling the “Stanford Critical Bunny” – a variation of a well-known 3D test case for methods used in the world of computer graphics. The results show that the geometry routine in Serpent 2 can handle the unstructured mesh, and that the use of delta-tracking results in a considerable reduction in the overall calculation time as the geometry is refined. The methodology is still very much under development, with the final goal of implementing a geometry routine capable of reading standardized geometry formats used by 3D design and imaging tools in industry and medical physics.

11:00 AM

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

D.Wang(1), S.Yu(2), Q.Gan(1), B.Wu(2), J.Song(3), M.Cheng(2), Y.Yuan(4), P.Long, L.Hu, Y.Wu(1)

1)Institute of Nuclear Energy Safety Technology, CAS FDS Team, China, 2)Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences, Hefei, Anhui, China, 3)University of Science and Technology of China, Hefei, Anhui, China, 4)Shenzhen University, Shenzhen, Guangdong, China

The Multi-Physics Coupling Analysis Modeling Program (MCAM), developed by FDS Team, China, is an advanced modeling tool aiming to solve the modeling challenges for multi-physics coupling simulation. The automatic modeling method for SuperMC, the Super Monte Carlo Calculation Program for Nuclear and Radiation Process, was recently developed and integrated in MCAM5.2. This method could bi-convert between CAD model and SuperMC input file. While converting from CAD model to SuperMC model, the CAD model was decomposed into several convex solids set, and then corresponding SuperMC convex basic solids were generated and output. While inverting from SuperMC model to CAD model, the basic primitive solids was created and related operation was done to according the SuperMC model. This method was benchmarked with ITER Benchmark model. The results showed that the method was correct and effective.

11:20 AM

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

L.Lu, U.Fischer, Y.Qiu

Karlsruhe Institute for Technology (KIT)., Karlsruhe, Germany

McCad is a three-dimension (3D) geometry conversion tool that has been developed at KIT to enable the bi-directional conversions between CAD models and the semialgebraic geometry representation utilized by most Monte Carlo particle transport codes. This paper introduces the recent improvements of core conversion algorithms, as well as the developments of new interfaces. The use of McCad for neutronics applications is illustrated on the examples of some fusion facilities with complicated geometries. The current status of McCad, including its capabilities and limitations, and the future development plans are discussed as well.

11:40 AM

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

Monte Carlo simulations continue to grow in size and fidelity. Accordingly, with millions of core regions and materials and billions of tallies the need arises for efficient and scalable indexing. This paper presents a novel method for geometry and material management based on hierarchical geometries within the Constructive Solid Geometry (CSG) paradigm. The required algorithms are presented in detail with a series of performance evaluations. Finally, the memory demands are compared between theoretical and empirical benchmarks. The presented indexing scheme will be at the core of future efforts to overcome the obstacles preventing the routine use of fully detailed Monte Carlo calculations for nuclear reactor analysis.

Track1-12 Reactor Analysis Methods

Session Chair: Dimitrios Cokinos(BNL), Rong-Jiun Sheu(Institute of Nuclear Engineering and Science)

13:30 PM
The Role of the Eigenvalue Separation in Reactor Dynamics and Neutron Noise Theory

I.Pàzsit, V.Dykin

Chalmers University of Technology, Göteborg, Sweden

The eigenvalue separation has been used in the past for characterisation of the space-time kinetics of reactor transients, and the stability properties of large loosely coupled cores. In this paper we explore the role of the eigenvalue separation on the neutronic response of a critical core to small stochastic perturbations, in particular the spatial characteristics of the arising neutron noise.

13:50 PM
Investigation of Conditional Transport Update in Method of Characteristics Based Coarse Mesh Finite Difference Transient Calculation

Y.S.Jung, H.G.Joo

Seoul National University, Seoul, Korea

As an effort to achieve efficient yet accurate transport transient calculations for power reactors, the conditional transport update scheme in method of characteristics (MOC) based coarse mesh finite difference (CMFD) transient calculation is developed. In this scheme, the transport calculations serves as an online group constant generator for the 3-D CMFD transient calculation and the solution of 3-D transient problem is mainly obtained from the 3-D CMFD transient calculation. In order to reduce the computational burden of the intensive transport calculation, the transport updates is conditionally performed by monitoring change of composition and core condition. This efficient transient transport method is applied to 3x3 assembly rod ejection problem to examine the effectiveness and accuracy of the conditional transport calculation scheme.

14:10 PM
Implementation of an a Priori Time Step Estimator for the Multigroup Neutron Diffusion Equation in Asynchronously Coupled RELAP5-3D

M.W.Hackemack(1), J.M.Pounders(2)

1)Texas A&M University, TX, USA, 2)Bettis Atomic Laboratory, PA, USA

In this work, an a priori time step estimation scheme based on local truncation error analysis is developed for the multigroup neutron diffusion equations. The estimator is used to optimize the time step size of the neutron diffusion solution, which is asynchronously coupled to the thermal-hydraulic solution. Numerical results using RELAP5-3D are shown.

14:30 PM
Development of a Three-Dimensional Kinetics Code for Commercial-Scale FBR Full Core Analysis

Y.Shimazu, T.Takeda, W.F.G.van Rooijen

University of Fukui, Tsuruga, Japan

A transient analysis code has been developed for the analysis of future, commercial-size Fast Breeder Reactors. The code uses diffusion theory, a nodal expansion method for the spatial discretization, while the temporal discretization uses either a direct numerical solution, the Improved Quasi-Static method, or the adiabatic method. Numerical acceleration is used to obtain practical calculation times on large problems. Trial calculations have shown satisfactory code performance.

14:50 PM
Use of Adjoint Functions for Comparing Measured and Calculated Parameters in the Subcritical Systems

A.Popykin, S.Shevchenko, R.Shevchenko, N.Zhylmaganbetov(1), V.Kulikov(2)

1)SEC NRS, Moscow, Russia, 2)VNIPIET, Saint Petersburg, Russia

The report examines the use of adjoint functions for comparing the measured and calculated values in subcritical systems. We will consider the quantities expressed in terms of fractional-linear functionals of solutions of the corresponding tasks: the effective multiplication factor and reactivity. The paper describes the direct and adjoint stationary problems and the problems with coefficients depending on time. As examples in first part of the paper we considered the comparing of measured and calculated values in the subcritical system consisting of plutonium, and in the second part of the paper – the calculated and measured reactivity of VVER-1000 reactor.

15:10 PM
Development of the Neutron Source Evaluation Method and Predictor of SRM/SRNM Count Rate in BWR Simulator

M.Tojo, H.Suzuki, H.Sato, T.Iwamoto

Global Nuclear Fuel-Japan, Yokosuka, Japan

The source range monitors (SRMs) and the start-up range neutron monitors (SRNMs) are important instruments from the BWR criticality safety viewpoints. There is a limitation of the minimum count rate (3cps) to guarantee the normality of the SRMs/SRNMs. After the long outage, this limitation is critical for the fuel shuffling due to the decay of the neutron sources in the fuel. The neutron source intensity evaluation method based on a micro burn-up model and the predictor function of the SRM/SRNM count rate are developed in AETNA01, GNF's three dimensional neutronic-thermal hydraulic BWR core simulator. These new functions are validated through the comparisons between operating BWR's measured data after shut-down and during shuffling. Through these comparisons, high accuracy of the SRM/SRNM count rate predictor of AETNA01 was presented.

Track14 Education in Reactor Physics

Session Chair: Ben Forget(MIT), Takanori Kameyama(Tokai Univ.)

13:30 PM

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

Guest Speaker: K.Hiraiwa

TOSHIBA Corporation, Japan

13:50 PM

Past, Present and Future of MIT Reactor Physics

B.Forget, K.Smith

Massachusetts Institute of Technology, MA, USA

The reactor physics curriculum at MIT has been undergoing lots of changes in recent years. This paper describes the course sequence and philosophy that we have implemented and continue to develop. Additionally, thoughts on future direction with respect to modularity and web-based learning are presented.

14:10 PM

New Practical Exercises at the JSI TRIGA Mark II Reactor

L.Snoj, S.Rupnik, A.Jazbec

Jožef Stefan Institute, Ljubljana, Slovenia

Since the 1990s the Jožef Stefan Institute (JSI) TRIGA reactor has been extensively used for performing training in experimental reactor physics. In 2012 we upgraded some of the existing and introduced some new exercises. The pulse mode operation exercise was upgraded by installation of new data acquisition system. The critical experiment exercise was improved by adding a new detector inside the reactor core and changing the data acquisition system. Now we monitor neutron population with two independent fission chambers on different locations. In the past the void reactivity coefficient exercise was performed by inserting Al tube into various positions in the reactor core and measuring the corresponding reactivity changes. In order to make the exercise more realistic, we installed a pneumatic system for generating air bubbles just below the core. The aim of the exercise is to measure reactivity changes versus flow rate and air bubble position. The second new exercise was measurement of water activation. In this exercise we installed special system which pumps the water through the core at a constant flow rate to the reactor platform, where the water activity is measured. The purpose of the exercise is to measure the ^{16}N and ^{19}O gamma line intensity and dose rate versus reactor power. The third new exercise, named in core flux mapping, was performed by measuring the axial fission rate distribution at various radial positions in the core. We used CEA – developed mini fission chambers and a special home developed system for moving the fission chamber in axial direction and measuring the count rate versus fission chamber position. In the paper the experiments are presented together with results.

14:30 PM

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

This paper presents the development of a course in nuclear reactor modelling at Chalmers University of Technology in Gothenburg, Sweden. This course, part of an international master program in nuclear engineering, deals with the modelling of deterministic neutron transport, fluid dynamics, and heat transfer in nuclear cores. The objective of the course is to present the methods applied in the simulation tools used by the industry for both steady-state and transient calculations. The originality of the course lies with the fact that all the above fields are tackled in a single course and that all the fundamental algorithms, together with their approximations and limitations, are detailed. The paper also reports on a web-based teaching framework lately adopted for the course. In this configuration, the course was entirely based on using the web for all lectures, tutorials, examination, and communication with the teaching staff. All course materials were accessible on the web to all students at any time, with the lectures recorded in advance and the tutorials live broadcasted and also available for on-demand viewing. The on-site students also had the possibility to attend the tutorials in a classical campus-based teaching set-up. An evalua-tion of the web-based teaching methods was made possible by the fact that the course was earlier run in a campus-based format. Despite a limited number of students making difficult to draw any systematic conclusions, a deeper learning to the course concepts was perceived.

14:50 PM

Introduction to the Status of Reactor Physics Education in Tsinghua University

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

Tsinghua University, Beijing, China

This paper introduces the general situation of reactor physics education in Tsinghua university, including its history, the course series of the undergraduates and graduates and the reactor physics experiment education. This paper also simply introduces the students studying in nuclear engineering field and other nuclear engineering related courses.

15:10 PM

Reactor Physics Education at Seoul National University

H.G.Joo, H.J.Shim

Seoul National University, Seoul, Korea

The reactor physics education and research programs of Seoul National University (SNU), which focus on high fidelity and efficient reactor simulation and uncertainty evaluation, are presented. In order to foster the students to have proper knowledge and experience in both deterministic and probabilistic reactor analysis methods with clear understanding of the physical behaviors of nuclear reactors, the undergraduate and graduate courses cover various mathematical and numerical methods as well as the principles of nuclear characteristics and physical behaviors. The research areas span from the development of the methods and computer programs for direct whole core calculation involving the method of characteristics transport calculation to the Monte Carlo uncertainty analysis. Those covers cross section generation, resonance treatment, depletion method, advanced nodal methods, space-time kinetics method, Monte Carlo whole core calculation with thermal feedback and et cetera. The reactor physics curriculum, the contents of the relevant courses, and the cutting edge research topics and the achievements of SNU reactor physics education are detailed.

Track7-3 Reactor Concepts and Designs

Session Chair: Liangzhi Cao(Xi'an Jiaotong Univ.), Naoto Aizawa(Tohoku Univ.)

13:30 PM
Development of the 900 Second Specific Impulse Carbide Low Enriched Uranium Nuclear Thermal Rocket

P.F.Venneri, Y.Kim(1), P.Husemeyerr, S.Howe(2)
1)KAIST, Daejeon, Korea, 2)CSNR, Idaho, USA

This study presents the Carbide Low Enriched Uranium Nuclear Thermal Rocket (LEU-NTR) in its latest form. First, a background on the design methodology is given along with a summary of previous work on the mass optimization of the core. The issue of the rather large power peaking is then raised and addressed through the implementation of radial enrichment zoning and increasing the radial reflector thickness. The advantages and disadvantages of each variation are then addressed before the issue of how to raise the coolant exit temperature is raised. Two methods to do so are then discussed and combined to produce three configurations of the Carbide LEU-NTR that operate with a specific impulse of 900 sec. Finally, the new performance characteristics are presented and compared with previous iterations of the design.

13:50 PM
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
Oregon State University, Oregon, USA

The most widely used and versatile medical radioisotope today is technetium-99m. Roughly 30 million people depend on this radioisotope for diagnostic imaging procedures each year, and this demand is expected to grow. Although there are numerous ways of procuring this isotope, the most common is from fission product molybdenum-99. Mo-99 is produced in all nuclear reactors fueled with U-235 as a fission fragment with a yield of around 6.1%. Mo-99 has a half-life of just over 2.5 days, and it will decay to Tc-99m 87% of the time. The Reduced Enrichment for Research Test Reactors (RERTR) program was established at Argonne National Laboratory in 1978 to investigate technology that would aid in converting High Enriched Uranium (HEU) facilities to Low Enriched Uranium (LEU) fuel. Since the majority of all Mo-99 produced currently comes from the irradiation of HEU fuel targets, there has been a growing effort to design LEU targets that can yield comparable quantities of high Specific Activity (SA) Mo-99. Recently, a novel LEU target design has been developed for use in TRIGA reactors for production of Mo-99, and this work examines the reactor behavior of three different core configurations fueled solely with this new target. MCNP5 was the simulation tool used to perform this analysis.

14:10 PM
Preliminary Design of the Delft Isotope Production Reactor (DIPR)

J.L.Kloosterman, M.V.Huisman, M.Rohde
Delft University of Technology, Delft, The Netherlands

The abundant and strongly growing use of Technetium-99m in medical diagnostics depends on just a few producers and processors of Molybdenum-99 world-wide, making the molybdenum supply chain very sensitive to interruptions. New production routes are needed, especially because some of the reactors used for the production of this isotope will soon reach the end of their economic lifetime. In this paper a preliminary design of a special purpose isotope production reactor, named DIPR, is presented, which could produce about 8% of the world-wide demand.
The reactor design is based on aqueous homogeneous reactors studied in the past and has been evaluated using coupled neutronics and CFD calculations. Both steady state and transient analyses have been carried out, showing the mild behavior of the reactor in various situations. The consequences of an operation error leading to an increase of the uranium concentration in the fuel solution has to be investigated in greater detail, taking into account a more complete physics model. Until now, no show stopper has been identified and the DIPR seems a promising reactor for securing the isotope supply chain.

14:30 PM
A Study of Safety Core Design on Beam Transient for Accelerator Driven System

N.Aizawa, T.Iwasaki
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Beam transient is the inherent transient on accelerator driven system, and safety analyses on beam transient are performed in the present study. The safety of the core is evaluated by the cladding temperature and the stress state on a cladding from the aspects of short term and long term. Beam shape change and beam incident position change are analyzed. As the result of safety analyses, the rapid increase of the load for a cladding is seen especially in beam incident position change, and the original core design does not fulfill the criteria specified in the present study. The safety core design is investigated in response to the results of safety analyses. The modification of the fuel composition is adopted as the core design approach, and safety analyses are performed for the designed cores. By the introduction of various fuel compositions, the cladding temperature is reduced by about 110 degC at most on the normal beam state, and the load for a cladding is relieved to a great degree on the transient conditions. The core design with the modification of the fuel composition is found to be able to satisfy all the criteria. However, the requirements for an accelerator specification are increased, and further development is needed in the accelerator to assure the safety of the core only by the modification of core design.

14:50 PM
New Inverted Hydride Fuel Design Concept for Pressure Tube Type Super Critical Water Reactors

A.Ahmad, L.Cao, H.Wu
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In this study, an innovative core design having inverted configuration has been proposed for pressure tube type supercritical water reactors. In this design the relative positions of fuel and coolant have been inverted and U-Th-Zr-hydride fuel has been used. A coupled neutronics and thermal hydraulics analysis was done for the proposed Inverted Pressure Tube Type (IPTT) SCWR. The neutronics analysis was carried out by using a 3D fine mesh diffusion theory code and thermal hydraulics calculations were done by using single channel model. These two codes were coupled with each other by a link code. The average outlet temperature for the proposed IPTT-SCWR was found to be 625 degC with maximum clad surface temperature (MCST) under the design limits i.e. below 850 degC. Moreover a core loading pattern has also been proposed to achieve uniform radial power distribution and lower cladding surface temperature.

15:10 PM
Preliminary Safety Analysis of a Thorium High-Conversion Pebble Bed Reactor

F.J.Wols, J.L.Kloosterman, D.Lathouwers, T.H.J.J.van der Hagen
Delft University of Technology, Delft, The Netherlands

An inherently safe thorium High-Conversion Pebble Bed Reactor would combine the inherent safety characteristics of the Pebble Bed Reactor with the favourable waste characteristics and resource availability of the thorium fuel cycle. Previous work by the authors showed that high conversion ratio's can be achieved within a thorium Pebble Bed Reactor (PBR) at a practical operating regime. The thorium PBR core design consists of a cylindrical core with a central driver zone surrounded by a breeder zone. The breeder pebbles have a 30 g heavy metal (HM) loading to enhance conversion of Th-232 into U-233, while the driver pebbles (10 w% U-233) contain a lower metal loading to enhance fission.
In previous studies, thorium PBR designs were presented for three core diameters, using a 7.5 g heavy metal (HM) loading for the driver pebbles. The current paper investigates the safety of these thorium PBR designs in terms of reactivity coefficients and possible reactivity insertion due to water ingress. Early results indicated that the values of the reactivity coefficients for the three designs with 7.5 g HM loading per driver pebble were rather small and the possible reactivity insertion due to water ingress was very large. Therefore, also a lower HM loading per driver pebble (4 g) was investigated to reduce the impact of water ingress, since the core becomes less under-moderated.
For the three core diameters investigated, it is shown that reducing the metal loading in the driver pebbles to 4 g is indeed advantageous in

Track7-3 Reactor Concepts and Designs

Session Chair: Liangzhi Cao(Xi'an Jiaotong Univ.), Naoto Aizawa(Tohoku Univ.)

terms of safety, water ingress leads to a smaller reactivity increase but also the reactivity coefficients become stronger negative. Secondly, the breeding performance of the cores with a 4 g driver pebble HM loading improves. On the downside, the driver pebble residence times become shorter, which could increase fuel reprocessing costs. Fuel pebbles would have to be recycled at an increased rate, which might be more challenging from a practical perspective.

Track6-4 Reactor Physics Experiments

Session Chair: Mohamed Ouisloumen(WH), Toru Yamamoto(NRA)

13:30 PM
Overview of the 2014 Edition of the International Handbook of Evaluated Reactor Physics Benchmark Experiments (IRPhEP Handbook)

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The International Reactor Physics Experiment Evaluation Project (IRPhEP) is a widely recognized world class program. The work of the IRPhEP is documented in the Interna-tional Handbook of Evaluated Reactor Physics Benchmark Experiments (IRPhEP Hand-book). Integral data from the IRPhEP Handbook is used by reactor safety and design, nuclear data, criticality safety, and analytical methods development specialists, worldwide, to perform necessary validations of their calculational techniques. The IRPhEP Hand-book is among the most frequently quoted reference in the nuclear industry and is expected to be a valuable resource for future decades.

13:50 PM
Analysis of Tungsten Gray Rods Critical Experiments Using PARAGON with Ultra-Fine Energy Mesh Methodology

M.Ouisloumen, H.Huria(1), K.Yoshioka, T.Umano, T.Kikuchi, S.Gunji, H.Matsumiya, S.Sugahara(2)

1)Westinghouse Electric Company LLC, PA, USA, 2)Toshiba Corporation Power & Industrial Systems R&D Center, Kawasaki, Japan

New critical experiments using gray control rods with tungsten were recently performed at the Toshiba NCA critical facility. This paper presents analyses of these experiments using both stochastic and deterministic codes. We used the continuous energy Monte Carlo code MCNP and the Westinghouse lattice physics code PARAGON. The basic nuclear data source for the cross-sections is ENDF/B-VII.1. First, the tungsten data is validated against Monte Carlo calculations. The ultra-fine energy mesh with 6064 group cross-sections library was used in PARAGON to extend the validation of the methodology to the cold temperature conditions. Comparisons focused on reactivity and the measured fission rate distributions. The results show that the ENDF/B-VII:1 data adequately reproduces the measured tungsten gray rod reactivity worth. Also, the energy mesh and the methodology used in PARAGON are seen to be adequate in predicting the reactivity and fission rate distributions for these challenging and highly heterogeneous experiments.

14:10 PM
Monte Carlo Assessment of Spatial and Energy Effects in the VENUS-F Subcritical Configurations and Application for Reactivity Determination

V.Bécares, D.Villamarín, E.M.González-Romero(1), A.Kochetkov, P.Baeten, W.Uyttenhove, G.Vittiglio, J.Wagemans(2), A.Billebaud, S.Chabod(3), J.-L.Lecouey, F.-R.Lecolley, G.Lehaut, N.Marie(4), X.Doliguez(5), F.Mellier(6)

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Experiments with a subcritical core and an external pulsed neutron source (PNS) have been performed at the VENUS-F reactor of the SCK-CEN during the year 2012 within the frame of the FREYA project of the 7th European Framework Programme. Different reactivity levels, obtained by varying the position of the control rods and the safety rods, have been investigated, with values of kef f ranging between 0.9 and 0.96. The area-ratio and the prompt decay constant techniques are well-known techniques to determine the reactivity of the system from the results of PNS experiments. However, both techniques are based on the point kinetics model and hence they are affected by the spatial and energy effects present in real systems, which will cause them to produce biased values of the reactivity. In this work, we apply a methodology based on Monte Carlo simulations to assess the spatial and energy effects present in the VENUS-F core and to determine relationships between the reactivity and the measured parameters (area-ratio and prompt decay constant) that take into account these effects and allow the application of the reactivity determination techniques to obtain accurate reactivity values beyond the limits of the point kinetics model. The results of the validation of this methodology against the experimental results obtained in the VENUS-F reactor will also be discussed.

14:30 PM
Validation of ORIGEN2 Coupled with JENDL-4.0 Base Libraries for Isotopic Compositions of Irradiated Light Water Reactor Fuels

T.Yamamoto

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A set of cross section libraries for ORIGEN2, ORLIBJ40, have been developed based on JENDL-4.0 by the Japan Atomic Energy Agency. It has been applied to the analysis of measured isotopic composition data of PWR and BWR irradiated fuels. The number of fuel samples referred in the present study was 33 both for PWR and BWR UO_2 fuels. The fuel samples of the PWR fuels were taken from 15x15, 17x17 and 18x18 fuel assemblies, and those of the BWR fuels from 8x8 and 9x9 fuels. The enrichments of the PWR sample were from 1.7 to 4.5 wt%, those of the BWR fuels from 2.1 to 4.9 wt%. The ranges of fuel burnups were 14 to 78 GWd/t for the PWR fuels and 17 to 69 GWd/t for the BWR fuels. Nuclides studied were 19 actinides and 42 fission product nuclides. The calculations were performed taking into account initial fuel compositions and irradiation histories which were consistent with the sample burnups determined based on measured isotopic compositions. The calculation results were compared with the measured results and the biases (C/E - 1's) were obtained. The C/E-1's were plotted against sample burrnups and checked about anomaly and trend with burnups. It was noted that the compositions of ^{235}U of samples taken from corner rods of a BWR 9x9 fuel assemblies and those of ^{237}Np of part of samples from a PWR 17x17 assembly were considerably overestimated. After screening of the data, the biases were statistically analyzed to obtain the average and standard deviation for each nuclide.

14:50 PM
The AMMON Experiment in EOLE Facility: A Challenging Program Dedicated to the Experimental Validation of JHR Neutronic and Photonic Calculation Tools

J.Di Salvo, A.Gruel, B.Geslot, P.Blaise, F.Mellier, A.Pepino, A.Roche, S.Arnaud, K.Blandin, A.Foucras, D.Garnier, J.-F.Ledoux, C.Morel, J.Bonora, J.-M.Girard, C.Jammes, H.Philibert, N.Thiollay, J.-F.Villard(1), C.Vaglio-Gaudard, A.-C.Colombier(2)

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The new AMMON integral experiment has been dedicated to the analysis of the Jules Horowitz Reactor (JHR) neutron and photon physics. The last shutdown of this 3-years program, in the EOLE zero power experimental reactor at CEA Cadarache, occurred at the end of March 2013. The core configurations consisted of an experimental zone of 6 or 7 JHR assemblies with U_3Si_2-Al 27% ^{235}U enriched curved fuel plates, surrounded by a driver zone with enough standard PWR UOx fuel pins to reach criticality. The purpose of this paper is to synthesize the main features of this program. Experimental methods and their associated uncertainties will be presented, with a focus on some new results obtained for nuclear heating measurements. It emphasizes the importance of the measurements performed in critical mock-ups for the calibration of biases and associated uncertainties for neutron and photon calculation tools.

15:10 PM
Monte Carlo Analysis of Reactivity Effect Measurements in the AMMON Experimental Program Dedicated to JHR Neutron Studies

C.Vaglio-Gaudard, A.C.Colombier, M.Lemaire(1), J.Di Salvo, A.Gruel(2)

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The AMMON integral experiment was loaded in the EOLE critical mock-up at CEA Cadarache: it aims at providing experimental

Track6-4 Reactor Physics Experiments

Session Chair: Mohamed Ouisloumen(WH), Toru Yamamoto(NRA)

data for assessing the calculation biases due to nuclear data on the Jules Horowitz Reactor (JHR) neutron and photon parameters. The experiment has a very specific design in comparison with the previous cores (MISTRAL, EPICURE, FUBILA, PERLE, etc.) consisting in the loading of UOX or MOX fuel pins in EOLE. The AMMON core contains an experimental zone composed of JHR U_3Si_2-Al circular assemblies and a driver zone of several hundreds of UOX 3.7% ^{235}U enriched fuel pins. This experimental program was very ambitious, since five main configurations – a reference configuration and 4 perturbed configurations - had to be studied for 2 years, with a lot of different measurement types. The experimental program was finished last year, in early 2013; all the main foreseen measurements were achieved, with a very good control of all the experimental (measurement+technological) uncertainties. Almost all the measurements have been today analyzed with TRIPOLI-4® Monte Carlo reference calculations, using the JEFF3.1.1 nuclear data library. This paper gives a general overview of the analysis of the critical states studied in the AMMON experiment: the comparison between Calculation and Experiment concerns the core excess reactivity and the reactivity effect induced by the introduction of perturbations in the reference configuration. The analysis shows a good prediction of the reactivity excess +120 pcm ±300 pcm (at one standard deviation). The reactivity effects, resulting from perturbations introduced in the AMMON core – hafnium, beryllium, water hole, isothermal temperature - are also well calculated; in particular, the bias on hafnium reactivity worth, supposed to be only due to nuclear data, amounts to +1.0% ± 1.7% (at one standard deviation). The calculation bias on the isothermal temperature coefficient in the AMMON experimental zone reaches -0.4 ± 0.8 pcm/°C on the 20°C-35°C range, and -1.5 ± 1.0 pcm/°C on the 35°C-50°C range. These results will be used in the Verification & globale Validation – Uncertainty Quantification (V&V&UQ) process implemented for the JHR neutron and photon calculation tools.

Track10-2 Nuclear Data

Session Chair: Nicholas Stauff(ANL), Pascal Archier(CEA)

13:30 PM
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

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The ENDF-6 format used for evaluation files allows different interpolation modes between distributions related to discrete incident energies. It is the case for angular distributions, energy distributions, energy-angle distributions and continuous photon energy spectra. The linear-linear interpolation is very often used particularly in the JEFF and ENDF/B libraries. This has no major impact when the distribution covers all the range (e.g. for angular distributions and for prompt fission neutron spectra), but, in all the other cases it could induce a violation of kinematic energy conservation. Certain codes for nuclear data processing (e.g. NJOY) and particle transport (e.g. MCNP) reinterpret data of evaluation files by changing the interpolation mode, in order to improve compliance with the kinematics of the physical reactions, even if this approach does not allow the evaluators to subsequently test the quality of their work. A new option for the TRIPOLI-4 ® code offers the users the opportunity to choose between the raw evaluation and a more physical treatment of these data for shielding and criticality benchmarks using neutron or coupled neutron-photon calculations. Thus, it is now possible to quantify the impact of such an interpretation, moreover this allows the evaluators to really test the relevance of their evaluation. The most sensitive configurations appear to be high energy neutron and coupled neutron-photon calculations.

13:50 PM
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

Neutronic analyses rely on accurate cross-sections from nuclear data libraries. The differences in the cross-sections from the JEFF-3.1 and ENDF/B-VII.0 nuclear data libraries are evaluated in this paper by estimating their impact on the eigenvalue of Sodium-cooled Fast Reactors (SFR). This analysis is performed with three SFR cores characterized by a variety of power ranges, fuel types (uranium and thorium) and fuel forms (oxide and metal). It was observed that the reactivity value is consistently larger when JEFF-3.1 is used instead of ENDF/B-VII.0 and the discrepancy can be as high as 1,900 pcm. A perturbation analysis was performed with the ERANOS2.2 code system to evaluate the contributions by isotope, reaction and energy range to the observed reactivity changes. It was found that the capture cross sections of Th-232 and U-233 are both responsible for as high as 700 pcm of the discrepancy in the case of the thorium-fueled core. Plutonium and sodium are the two main contributors to the observed reactivity change in the case of uranium-fueled cores. The reactivity impact due to the differences in the cross-sections from ENDF/B-VII.0 and JEFF3.1 is also compared to the uncertainties calculated respectively with the COMMARA-2.0 and the COMAC.V0 covariance matrixes.

14:10 PM
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

In burn-up calculations, accuracy of isotopic generation prediction strongly depends on the nuclear data, such as neutron cross sections, fission yields and decay constants. In this study, we perform burn-up calculations using JENDL-4.0, ENDF/B-VII.1 and JEFF-3.1.1, and compare the calculation results with PIE (Post Irradiation Examination) data. We also perform burn-up sensitivity analysis based on the generalized perturbation theory to clarify the cause of difference on isotopic generation between the libraries. As a result, there are large discrepancies between JEFF-3.1.1 and the others generally. Furthermore, we clarify their causes for each nuclide and energy group. For neutron cross sections, some nuclides have large discrepancies between JEFF-3.1.1 and the others, and they give large impacts on specific isotopic generation predictions. On fission yields, ones from Pu-239 and Pu-241 have large discrepancies between JEFF-3.1.1 and the others, and they give large impacts on specific isotopic generation predictions, especially Gd-160. Decay constant discrepancies do not give any large impacts on isotopic generation predictions.

14:30 PM
Validation of a Pointwise Energy Neutron Cross Section Library Generated by RXSP-BETA2.0 Using ENDF/B-VII.0

J.Yu, W.Li, K.Wang(1), J.Vujic(2)

1)Tsinghua University, Beijing, China, 2)University of California at Berkeley, California, USA

A new pointwise energy neutron cross section library named ENDFb7_r in ACE format for Reactor Monte Carlo code RMC has been generated by Reactor Cross Section Processing code RXSP using ENDF/B-VII.0. The pointwise energy cross section library called ENDFb7_n generated by NJOY has also been constructed for inter-comparison of results. Benchmark tests for series of criticality reactor cores and assemblies including both uranium and plutonium fuels with thermal, intermediate and fast neutron spectrum have been performed with the code RMC using these two libraries ENDFb7_r and ENDFb7_n. The k-effective and neutron flux calculated with two libraries show very good agreement with each other. Moreover, another practical PWR fuel assembly depletion model is further constructed and simulated by RMC. The calculated results of k-effective and isotopic concentration swings with burnup agree very well with each other. It has been proved that the self-processed library named ENDFb7_r is accurate enough to be used for both criticality and depletion calculations.

14:50 PM
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, I.S.Holubeva(2)

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MCNP version 6.1 was used to model the Giacint facility of Belarus with uranium zirconium carbon nitride low enriched uranium (LEU) fuel. The fuel rods have either stainless steel or niobium clad. Niobium absorbs less neutrons than stainless steel. The fuel rods are immerged into a water tank. In the experiments, the water level is changed according to the fuel loading pattern to obtain critical condition. The MCNP numerical simulations used nuclear data libraries based on ENDF/B-VII.1, JEFF-3.1.2, or JENDL-4 and the obtained results were compared. The analyses included: the effective multiplication factor, the delayed neutron fraction, the neutron generation time, and the sensitivity coefficients. The ACE format of the JENDL-4 nuclear data library has been generated at Argonne National Laboratory using NJOY version 2012.

15:10 PM
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)

1)CEA Cadarache, Saint Paul-Lez-Durance, France, 2)OECD/NEA, Issy-les-Moulineaux, France, 3)AMEC, Dorset, UK

Benchmark measurements of irradiated and un-irradiated fuel samples were performed in the framework of the CERES collaborative programme between AEA and CEA. These experiments provide relevant data for the validation of fuel burn-up and criticality-safety calculations for the whole fuel cycle. As part of this programme, pile-oscillation measurements were carried out on a range of mixed oxide samples with Plutonium of various mass and isotopic contents, both in the MINERVE and DIMPLE reactors. Four core configurations, two over-thermalized situations and two PWR-type situations, were constituted with different forward and adjoint flux spectra, emphasizing fission and/or capture contributions. The experiments were analyzed using reference TRIPOLI4 calculations with the JEFF-3.1.1 library, using exact 3-dimensional models of the core configurations. In a

Track10-2 Nuclear Data

Session Chair: Nicholas Stauff(ANL), Pascal Archier(CEA)

first step, calculations of each DIMPLE configuration were performed and compared with the experiment, showing very good agreements with a maximum C-E of -230 pcm. In the second step, reactivity worth experiments were analyzed, using recently developed exact-perturbation capabilities in TRIPOLI4. A consistent assimilation of Calculation over Experiment discrepancies was performed with the CONRAD code, using the Integral Data Assimilation method. Covariance matrix on multigroup neutron cross sections and multiplicities were generated and significant trends were identified, especially on the ^{239}Pu and ^{240}Pu capture cross sections in the thermal energy range (E < 0.1 eV). Further investigations should be required to confirm these conclusions, due to the strong dependence of these trends and of posterior covariances to prior covariances.

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PHYSOR 2014 Program

September 28 (Sun.)	**7:30**	**Registration Open**				
	8:00-12:00	Workshop 5 (Mizuho_A)	Workshop 3 (Mizuho_B)	Workshop 6 (Mizuho_C)	Workshop 2 (Hiei)	
	12:00-13:30	Lunch (on own)				
	13:30-17:35	Workshop 1 (Mizuho_A)	Workshop 4 (Mizuho_B)	Workshop 7 (Mizuho_C)	Workshop 8 (Hiei)	
	16:00-20:00	Welcome Cocktail (Mizuho_D)				
September 29 (Mon.)	**7:30**	**Registration Open**				
	8:30-8:50	Opening Session (Mizuho_A, B) **Reiko Fujita,** President, Atomic Energy Society of Japan **Michaele C. Brady Raap,** President, American Nuclear Society				
	8:50-10:00	Plenary Session 1 (Mizuho_A, B)				
	10:20-11:30	Plenary Session 2 (Mizuho_A, B)				
	11:30-13:00	Hosted Lunch (Mizuho_C, D) **Etsuro Saji,** Mitsubishi Heavy Industries				
	13:00-15:30	Track1-1 Reactor Analysis Methods (Mizuho_A)	SS2-1 Reactor Physics and Criticality Safety Activities in OECD/NEA Working Party (Mizuho_B)	SS1-1 Molten Salt Reactors (Hiei)	SS5 Multiscale, Multiphysics Approaches in Nuclear Science and Engineering Applications (Atago)	Track11-1 Research Reactors and Spallation Sources (Cosmos)
	15:45-18:15	Track1-2 Reactor Analysis Methods (Mizuho_A)	SS2-2 Reactor Physics and Criticality Safety Activities in OECD/NEA Working Party (Mizuho_B)	SS1-2 Molten Salt Reactors (Hiei)	SS7 Control Rod Withdrawal Tests Performed During the PHENIX End-of-Life Experiments (Atago)	Track11-2 Research Reactors and Spallation Sources (Cosmos)
September 30 (Tue.)	**7:30**	**Registration Open**				
	8:00-10:05	Track1-3 Reactor Analysis Methods (Mizuho_A)	Track5-1 Nuclear Criticality Safety (Mizuho_B)	Track2-1 Deterministic Transport Theory (Hiei)	Track3-1 Monte Carlo Methods (Atago)	SS8 Reactor Physics of Non-Traditional LWR Fuel Design (Cosmos)
	10:20-12:00	Track1-4 Reactor Analysis Methods (Mizuho_A)	Track5-2 Nuclear Criticality Safety (Mizuho_B)	Track2-2 Deterministic Transport Theory (Hiei)	Track13 Radiation Applications and Nuclear Safeguards (Atago)	Track9-1 Transient and Safety Analysis (Cosmos)
	12:00-13:30	Hosted Lunch (Mizuho_C, D) **Sun-Doo Kim,** KEPCO Nuclear Fuel				
	13:30-15:40	Track1-5 Reactor Analysis Methods (Mizuho_A)	Track4-1 Verification, Validation and Uncertainty Analysis (Mizuho_B)	Track2-3 Deterministic Transport Theory (Hiei)	Track3-2 Monte Carlo Methods (Atago)	Track12-1 Fuel Cycle and Actinide Management (Cosmos)
	15:55-18:05	Track1-6 Reactor Analysis Methods (Mizuho_A)	Track4-2 Verification, Validation and Uncertainty Analysis (Mizuho_B)	Track2-4 Deterministic Transport Theory (Hiei)	Track3-3 Monte Carlo Methods (Atago)	Track12-2 Fuel Cycle and Actinide Management (Cosmos)
	18:30-21:00	Conference Banquet (Mizuho_C, D) **Ronald J. Ellis,** Chair, ANS Reactor Physics Division **Toshikazu Takeda,** Honorary Chair, Univ. of Fukui				
October 1 (Wed.)	**7:30**	**Registration Open**				
	8:00-10:05	Track1-7 Reactor Analysis Methods (Mizuho_A)	Track4-3 Verification, Validation and Uncertainty Analysis (Mizuho_B)	SS3 Hybrid Particle Transport Methods for Solving Complex Problems in Real-Time (Hiei)	Track3-4 Monte Carlo Methods (Atago)	Track9-2 Transient and Safety Analysis (Cosmos)
	10:20-12:00	Track1-8 Reactor Analysis Methods (Mizuho_A)	Track4-4 Verification, Validation and Uncertainty Analysis (Mizuho_B)	Track8-1 Reactor Operation and Safety (Hiei)	Track6-1 Reactor Physics Experiments (Atago)	Track9-3 Transient and Safety Analysis (Cosmos)
	12:00-13:30	Lunch (on own)				
	13:30-15:40	Track1-9 Reactor Analysis Methods (Mizuho_A)	Track4-5 Verification, Validation and Uncertainty Analysis (Mizuho_B)	Track7-1 Reactor Concepts and Designs (Hiei)	Track3-5 Monte Carlo Methods (Atago)	Track15 Research Related to Fukushima Accident (Cosmos)
	16:15-18:15	Poster Session (Mizuho_C, D)				
October 2 (Thu.)	**7:30**	**Registration Open**				
	8:00-10:05	Track1-10 Reactor Analysis Methods (Mizuho_A)	Track4-6 Verification, Validation and Uncertainty Analysis (Mizuho_B)	Track7-2 Reactor Concepts and Designs (Hiei)	Track6-2 Reactor Physics Experiments (Atago)	Track10-1 Nuclear Data (Cosmos)
	10:20-12:00	Track1-11 Reactor Analysis Methods (Mizuho_A)	Track4-7 Verification, Validation and Uncertainty Analysis (Mizuho_B)	Track8-2 Reactor Operation and Safety (Hiei)	Track6-3 Reactor Physics Experiments (Atago)	SS4 Advanced Geometry Processing in Deterministic and Monte Carlo Methods (Cosmos)
	12:00-13:30	Lunch (on own)				
	13:30-15:40	Track1-12 Reactor Analysis Methods (Mizuho_A)	Track14 Education in Reactor Physics (Mizuho_B)	Track7-3 Reactor Concepts and Designs (Hiei)	Track6-4 Reactor Physics Experiments (Atago)	Track10-2 Nuclear Data (Cosmos)
	16:00-17:00	**Closing (Mizuho_C, D)**				