Monte Carlo Code Poster Session
5:30pm – 7:30pm
Tuesday, April 21st
Plantation Lobby

This poster session will provide a forum for Monte Carlo code development teams to showcase their recent code developments and discuss their newest code features. This session will feature an introductory keynote speech by Dr. John Wagner, Director of the Reactor and Nuclear Systems Division at Oak Ridge National Laboratory. All conference attendees are welcome to attend.

Brief descriptions of the codes that are participating in this session are given below, as well as the layout for the posters. This layout was determined randomly using, of course, the Monte Carlo method.

1 – MC21

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<th>Organization:</th>
<th>Bechtel Marine Propulsion Corporation</th>
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<td>Presenters:</td>
<td>David Griesheimer</td>
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<tr>
<td>Abstract:</td>
<td>MC21 is a continuous-energy Monte Carlo radiation transport code for the calculation of the steady-state spatial distributions of reaction rates in three-dimensional models. The code supports neutron and photon transport in fixed source problems, as well as iterated-fission-source (eigenvalue) neutron transport problems. MC21 has been designed and optimized to support large-scale problems in reactor physics, shielding, and criticality analysis.</td>
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applications. The code supports many in-line reactor feedback effects, including depletion, thermal feedback, xenon feedback, eigenvalue search, and neutron and photon heating. MC21 uses continuous-energy neutron/nucleus interaction physics over the range from 10E-5 eV to 20 MeV. The code treats all common neutron scattering mechanisms, including fast-range elastic and non-elastic scattering, and thermal- and epithermal-range scattering from molecules and crystalline materials. For photon transport, MC21 uses continuous-energy interaction physics over the energy range from 1 keV to 100 GeV. The code treats all common photon interaction mechanisms, including Compton scattering, pair production, and photoelectric interactions. For geometry representation, MC21 employs a flexible constructive solid geometry system that allows users to create spatial cells from first- and second-order surfaces. Models can also be built as hierarchical collections of previously defined spatial cells, with interior detail provided by grids and template overlays. Results are collected by a generalized tally capability, which allows users to edit integral flux and reaction rate information. Results can be collected over the entire problem or within specific regions of interest through the use of phase filters that control which particles are allowed to score each tally.

Keywords:
MC21, Monte Carlo, Reactor Calculations, Feedback, High-Performance Computing

2 – Monte Carlo Application Toolkit (MCATK)

Organization:
Los Alamos National Laboratory

Partner Organizations:

Presenters:
Jeremy Sweezy
Steve Nolen
Travis Trahan

Abstract:
The Monte Carlo Application ToolKit (MCATK) is a modern C++ component-based software library for Monte Carlo particle transport that has been in development at Los Alamos National Laboratory (LANL) since 2008. It is designed to provide new component-based functionality for existing software as well as provide the building blocks for specialized applications. Over the last year a number of new capabilities have been developed including: including probability of initiation (POI), multi-temperature cross-sections, surface source read and write, and 3-D computational solid body geometry.

Keywords:
Monte Carlo Particle Transport, Probability of Extinction, Computational Solid Body Geometry, MCATK

3 – SCALE Code System

Organization:
Oak Ridge National Laboratory
### Abstract:

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 selected 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.

### Keywords:

SCALE, Eigenvalue, Radiation Shielding, Depletion, Sensitivity/Uncertainty Analysis

### 4 – Serpent

**Organization:**
VTT Technical Research Centre of Finland

**Presenters:**
Jaakko Leppanen  
Ville Valtavirta

**Abstract:**

The recent development in the Serpent 2 Monte Carlo code is described. The work is focused on two major topics: 1) spatial homogenization and group constant generation for deterministic reactor simulator codes, and 2) Coupled multi-physics applications involving neutronics, thermal hydraulics and fuel behavior modeling. Entirely new applications for Serpent include fusion neutronics and radiation shielding, which are briefly introduced.

**Keywords:**

Serpent, Monte Carlo, Spatial Homogenization, Multi-physics

### 5 – MCNP
Abstract:
The latest version of MCNP (version 6.1.1) was released by RSICC in September 2014. This beta release followed the production release of MCNP 6.1 in June 2013. There have been nearly 8000 copies of MCNP6 distributed both domestically and internationally to users from academia, industry, and the US government. MCNP6.1.1 has all of the features of previous versions but makes advances in charged particle and light ion transport including correlated source emissions. Significant improvements in our capabilities for transport and variance reduction on unstructured meshes have also been included. MCNP6.1.1 is significantly faster than MCNP6.1. We will review the current features of MCNP and discuss future direction.

Keywords:
Monte Carlo, Radiation, Particle Transport, MCNP

6 – Shift

Abstract:
This poster presents the massively parallel Monte Carlo radiation transport package Shift, developed at Oak Ridge National Laboratory and it gives the capabilities, implementation, and parallel performance of this code package. This code package is designed to scale well on high performance architectures. Scaling results demonstrate very good strong and weak scaling as applied to LWR analysis problems. Also, benchmark results from various reactor problems show that Shift results compare well to other contemporary Monte Carlo codes and experimental results.

Keywords:
Monte Carlo, Radiation Transport, Massively Parallel

7 – ADVANTG
ADVANTG is a software package for generating variance reduction parameters for fixed-source, continuous-energy neutron and photon Monte Carlo transport simulations using MCNP5. ADVANTG automates the process of generating three-dimensional (3-D) space- and energy-dependent weight-window bounds and consistent biased source distributions based on approximate multigroup transport solutions that are efficiently generated by the Denovo 3-D, parallel discrete ordinates package. The code implements the Consistent Adjoint Driven Importance Sampling (CADIS) method for accelerating individual tallies and the Forward-Weighted CADIS method for obtaining relatively uniform uncertainties across tallies over multiple regions and/or energy bins, including mesh tallies. Variance reduction parameters are output in a format directly usable by unmodified versions of MCNP. ADVANTG can also be used as a front-end for Denovo and is capable of driving parallel SN calculations.

Keywords: Variance Reduction, MCNP, Denovo, Fixed-Source Transport

8 – MVP

A general-purpose Monte Carlo code MVP has been developed for continuous-energy neutron and photon transport calculations since the late 1980s at Japan Atomic Energy Agency. The MVP code is designed for nuclear reactor applications such as reactor core design/analysis, criticality safety and reactor shielding. The code has been widely in domestic use since the first release in 1994 and the second release in 2005. Modifications and enhancements have been made with advanced Monte Carlo methodology for reactor physics applications. Featured capabilities for version 3 are the perturbation calculation for the k-effective value, treatment of delayed neutrons, group constant generation, exact resonance elastic scattering model, reactor kinetics parameter calculation. The perturbation calculation is based on the correlated sampling and differential operator sampling methods. The impact of the perturbed fission-source distribution can be also taken into account. Delayed neutrons can be explicitly treated in eigenvalue and time-dependent fixed-source problems. The group constants can be generated with the newly implemented tally capability.
of group-to-group scattering reaction rates. The isotropic diffusion coefficient can be also calculated with the average cosine of the scattering angle. An exact resonance elastic scattering model based on the weight correction method can improve the calculation accuracy of the Doppler reactivity worth. The reactor kinetics parameters of the effective delayed neutron fraction and the generation time can be calculated with the differential operator sampling method. The above-mentioned capabilities are integrated into the code and MVP version 3 is planned to be released domestically in the near future.

**Keywords:**
MVP, Monte Carlo, Neutron/Photon Transport, Reactor Physics/Design

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**9 – Meteor**

**Organization:**
Sellafield Ltd

**Presenters:**
Keith Searson
Fabrice Fleurot

**Abstract:**
Meteor is a new criticality code developed by Sellafield Ltd., which supports fast direct tracking through CAD models, including those with NURBS surfaces, without relying on model simplifications or faceting. Meteor is currently in the testing phase and this poster presents the current k-effective and speed comparisons against the MONK criticality code and k-effective comparisons to experimental cases. The tests show very good agreement, with k-effective typically within 0.2% of MONK’s. Meteor also shows higher calculation speed, being on average about 2.5 times faster on the MONK validation set. With the CAD models currently tested, the speeds are either comparable or not significantly slower (1.5 times slower for the model presented here) than CSG models. This last result is encouraging as traditionally direct CAD tracking is believed to be orders of magnitude slower.

**Keywords:**
Criticality, Meteor, CAD, OiNC2, Tracking

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**10 – Light and Individual Computer-Oriented Neutron Transport Code based on Monte Carlo Method (LIONMC)**

**Organization:**
Hanyang University and Institute for Basic Science

**Presenters:**
Song Hyun Kim
Do Hyun KIM
Sangjin Lee
Abstract:
In this presentation, the current status of the development of Monte Carlo simulation code in HANYANG University is described for reactor analyses. The MC code has been developed to offer some specific features to analyze reactor core characteristics with user-friendly functions. The functions are spherical particle modeling function in stochastic medium, automatic decision function on active/inactive cycle, on-the-fly sampling-based sensitivity and uncertainty analyses, fission matrix-based MC simulation module, automatic applications of variance reduction techniques, volume calculation function of cells, and the others. The code is being developed based on the C++ program language, and it is planned that a test version of the code will be distributed at the end of this year.

Keywords:
Monte Carlo Transport, Nuclear Reactor, User-Friendly, Automatic Decision

11 – FLUKA

Organization:
CERN

Partner Organizations:
INFN

Presenters:
Vasilis Vlachoudis

Abstract:
FLUKA is a fully integrated particle physics Monte Carlo simulation package. It has many applications in high-energy experimental physics and engineering, shielding, detector and telescope design, cosmic ray studies, dosimetry, medical physics and radiobiology. The capabilities and physics models implemented inside the FLUKA code are briefly described, with emphasis on hadronic interaction. Examples of the performances of the code are presented including basic (thin target) and complex benchmarks, and radiation detector specific applications. The advanced user-friendly graphical interface "flair" greatly steepened the learning curve of FLUKA for beginners, as well easing the use of the code for experienced users.

Keywords:
FLUKA, flair

12 – TRIPOLI

Organization:
CEA

Presenters:
### Abstract:
TRIPOLI is the generic name of a Monte Carlo radiation transport codes family dedicated to radiation protection and shielding, core physics with depletion, criticality safety and nuclear instrumentation analyses. It has been continuously developed at CEA since the mid 60s, at Fontenay-aux-Roses then at Saclay. TRIPOLI-4, the fourth generation of the family, is the corner stone of the CEA Radiation Transport Software Suite, which also includes the APOLLO codes, deterministic solvers dedicated to reactor physics analyses (at both lattice- and core-level), the depletion code MENDEL, the photon point-kernel code NARMER, and CONRAD and GALILEE for nuclear evaluation and data processing. TRIPOLI-4 is the reference industrial code for CEA (labs and reactors), EDF (58 PWRs), and branches of AREVA. It is also the reference code of the CRISTAL Criticality Safety package developed with IRSN and AREVA.

### Keywords:
TRIPOLI, Monte Carlo, CEA, Neutron, Photon

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### 13 – ARCHER

**Organization:**
Rensselaer Polytechnic Institute (RPI)

**Presenters:**
Tianyu Liu  
Noah Wolfe  
X. George Xu

**Abstract:**
ARCHER (Accelerated Radiation-transport Computations in Heterogeneous EnviRonments) is a Monte Carlo simulation code for coupled photon-electron transport that is run on CPUs, GPUs and MICs. The central part of the code is a photon-electron transport kernel. To accommodate to different needs, several application-specific modules are being developed, including CT dosimetry, radiotherapy dosimetry, radiation shielding design and nuclear medicine dosimetry modules. Each module is uniquely designed and optimized for that application. For example, nuclear medicine dosimetry module can simulate radioactive decay and account for biokinetic factors in order to accurately quantify internal dose for a patient. ARCHER also has a simplistic neutron transport module for one-group criticality calculation used to deepen our understanding of GPU/MIC performance tuning. In addition, several powerful utility modules are being developed, for instance, to programmatically evaluate the energy consumption of the code, to allow different computing devices CPU/GPU/MIC work concurrently, etc. The capability of ARCHER has been continuously expanded.

**Keywords:**
ARCHER, Monte Carlo, GPU, MIC, Xeon Phi
### 14 – PHITS

**Organization:**
JAEA

**Partner Organizations:**
CEA, Chalmers University, JAXA, KEK, Kyushu University, RIKEN, RIST

**Presenters:**
Tatsuhiko Ogawa

**Abstract:**
Particle and Heavy Ion Transport code System, PHITS is a general-purpose Monte Carlo particle transport simulation code developed under collaboration of several institutes in Japan and Europe. The Japan Atomic Energy Agency (JAEA) is responsible for managing the entire project. PHITS can deal with the transport of nearly all particles, including neutrons, protons, heavy ions, photons, and electrons, over wide energy ranges using various nuclear reaction models and data libraries. PHITS has several important features, such as an event-generator mode for low-energy neutron interaction, beam transport functions, a function for calculating the displacement per atom (DPA), and a microdosimetric tally function. Due to these features, it has been widely used for various applications. For example, PHITS was extensively used in the design of the shielding, target, and neutron beam lines for the J-PARC project. Calculation of the dose and dose equivalents in human bodies irradiated by various particles was carried out using PHITS in order to determine radiological protection needs and medical physics issues. The event-generator mode is useful in the estimate of the soft error rates of semi-conductor devices. The microdosimetric function was used in the development of a new computational model for calculating the relative biological effectiveness (RBE)-weighted dose for charged particle therapy. This PHITS package is distributed to many countries via the Research Organization for Information Science and Technology, the Data Bank of the Organization for Economic Co-operation and Development's Nuclear Energy Agency, and the Radiation Safety Information Computational Center.

**Keywords:**
General-purpose Transport Simulation Code, DPA, Microdosimetry, Event Generators

### 15 – McCARD

**Organization:**
Seoul National University

**Presenters:**
Hyung Jin Shim
Chnag Hyo Kim
Ho Jin Park

**Abstract:**
McCARD is a Monte Carlo (MC) neutron-photon transport simulation code designed
exclusively for neutronics analyses of various nuclear reactor and fuel systems. McCARD estimates neutronics design parameters such as effective multiplication factor, neutron flux and current, fission power, etc. by using continuous-energy cross section libraries and detailed geometrical data of the system. Since its predecessor MCNAP was first introduced in 1999 as a MC burnup analysis tool with an ORIGEN2-type fuel depletion equation solver, it has evolved to a versatile MC tool which is capable of performing the whole-core neutronics calculations, the reactor fuel burnup analysis, the few group diffusion theory constant generation, sensitivity and uncertainty (S/U) analysis, and uncertainty propagation analysis. It has some special features such as the anterior convergence diagnostics, real variance estimation, neutronics analysis with temperature feedback, B1 theory-augmented few group constants generation, kinetics parameter generation and MC S/U analysis based on the use of adjoint flux. In the course of its evolution, a wide range of nuclear systems such as SMART, PMR-200, KALIMER-600, fusion blankets, subcritical systems like HYPER, YALINA, and IPEN/MB-01 have been subjected to the neutronics analyses. The R&D efforts to meet both functional and non-functional requirements for these analyses have played crucial roles in developing McCARD into its current status.

Keywords:
McCARD, Whole Core Transport Calculation, Few Group Constant Generation, Uncertainty Propagation Analysis, Sensitivity/Uncertainty Analysis

16 – FinMCool

Organization:
Texas A&M University

Presenters:
Ryan McClaren
Jacob Landman
Alex Long

Abstract:
FinMCool is a Monte Carlo code for high-energy density physics radiative transfer that is based on the Fleck and Cummings implicit Monte Carlo method. The current capabilities of the code include Cartesian and cylindrical geometries, domain-replicated parallelism, and several variance reduction techniques. The code's design and philosophy make it a useful testbed for new methods development and we have been actively developing new weight windows, implicit capture, and gradient estimation techniques in the code.

Keywords:
Monte Carlo Methods, Radiative Transfer, High-Energy Density Physics, Variance Reduction

17 – OpenMC

Organization:
Computational Reactor Physics Group at the Massachusetts Institute of Technology
Abstract:
OpenMC was developed by the Computational Reactor Physics Group (CRPG) at the Massachusetts Institute of Technology as a tool for nuclear reactor simulation on high-performance computing platforms. Given that many legacy codes do not scale well on existing and future parallel computer architectures, OpenMC was developed from scratch with a focus on high-performance scalable algorithms as well as modern software design practices. OpenMC is ideal for developing, testing, and optimizing numerical methods that increase simulation accuracy and reduce memory and computational requirements. The windowed multipole Doppler broadening method has been developed and implemented in OpenMC providing cross sections on-the-fly at any temperature in the resolved resonance region with performance similar to single temperature ACE file lookup. Additionally, reductions in nuclear data memory requirements are achieved with high-fidelity, on-the-fly methods for calculating Doppler-broadened unresolved resonance region cross sections, and similar methods are being developed for thermal scattering data (\(S(\alpha, \beta)\)). Spatial domain and data decomposition algorithms are being investigated to address per-node memory limitations that are often encountered due to the large number of particles, tallies, and materials needed for a full-core Monte Carlo simulation. Finally, the inclusion of multiphysics feedback in Monte Carlo simulations has been investigated in OpenMC using a low-order CMFD operator and alternatively the Multiphysics Object-Oriented Simulation Environment (MOOSE). In this latter investigation Functional Expansion Tallies are used to minimize the data transfer between multiphysics applications while maintaining a high level of accuracy when mapping to an unstructured finite element mesh.

Keywords:
Monte Carlo, OpenMC, Windowed Multipole

18 – Geant4

Organization:
The Geant4 Collaboration

Partner Organizations:
SLAC (USA), IN2P3/LLR (France)

Presenters:
Makoto Asai
Marc Verderi

Abstract:
Geant4 is a general purpose Monte Carlo simulation toolkit for elementary particles and nuclides passing through and interacting with matter. Geant4 covers all relevant physics processes, including electromagnetic and hadronic physics for energy range spanning from eV to TeV scale, decay and optical processes. The transport of low energy neutrons down to thermal energies is also handled. The software can also simulate remnants of hadronic
interactions, including atomic de-excitation and provides extension to low energies down to the DNA scale for biological modeling. Geant4 offers many types of geometrical descriptions to describe most complicated and realistic geometries. Geant4 also offers several variance reduction options, scorers, visualization and graphical user interfaces. Its areas of application include high energy, nuclear and accelerator physics, studies in medical and space science, shielding and radiation protection, and newly arising material science. The recent major release (Geant4 version 10.0 released in December 2013) delivered event-level parallelism via multithreading. It has already demonstrated excellent scalability up to hundreds of threads with good memory footprint reduction on various computing architectures including Xeon, Xeon Phi and AMD. For example, all LHC experiments have started their projects to migrate their simulation codes to multithread with Geant4 version 10.

### Keywords:
- Geant4
- Radiation transport
- Toolkit
- C++
- Multithread

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### 19 – NTS

**Organization:**
International Business Nuclear Energy Software Development Corporation (IBNESD)

**Presenters:**
Changyuan Liu

**Abstract:**
NTS (Neutron Transport System) NTS code features automatic topologies and volume calculations for selected geometry, a cross section generation engine using ENDF files directly with lightening fast doppler broadening, full 64-bit supports, high-quality graphic rendering.

**Keywords:**
- NTS
- Doppler Broadening
- ENDF

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### 20 – SuperMC: Super Monte Simulation Program for Nuclear and Radiation Process

**Organization:**
Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences - FDS Team

**Presenters:**
Jing Song
Tao He
Bin Wu

**Abstract:**
SuperMC is a general purpose, intelligent and multi-functional program for the design and safety analysis of nuclear systems. It is designed to perform the comprehensive neutronics
calculation, taking the radiation transport as the core and including the depletion, radiation source term/dose/biohazard, material activation and transmutation, etc. It supports the multi-physics coupling calculation including thermo-hydraulics, structural mechanics, biology, chemistry, etc. The main technical features are hybrid MC-deterministic methods and the adoption of advanced information technologies. The main usability features are automatic modeling of geometry and physics, visualization and virtual simulation and cloud computing service. The latest version of SuperMC can accomplish the transport calculation of neutrons, gamma rays, and can be applied for criticality and shielding design of reactors, medical physics analysis, etc. SuperMC has been verified by more than 2000 benchmark models and experiments. The handbook of International Criticality Safety Benchmark Evaluation Project (ICSBEP) and the Shielding Integral Benchmark Archive Database (SINBAD) were used to verify the correctness of SuperMC. The fusion reactor (ITER benchmark model, FDS-II), fast reactor (BN600, IAEA-ADS), PWR (BEAVRS, HM, TCA) and cases from the International Reactor Physics handbook Evaluation Program (IRPhEP) were employed for validating the comprehensive capability for reactor applications. The benchmarking results have been compared with MCNP5, demonstrating higher accuracy and calculation efficiency of SuperMC, and also significant enhancement of work efficiency due to its functions of automatic modeling and visualized analysis. SuperMC has been applied in the nuclear design and analysis of ITER and the China Lead-based Reactor (CLEAR).

Keywords:
Monte Carlo, Particle Transport, Fusion, Fission