First Announcement

Pisa, Italy
November 23 – 27, 2015

Models and Methods
for Advanced Reactor
Safety analysis

http://nnees.sk/mmars.html

in cooperation with

GRNSPG (Italy), NNEES (Slovakia), FPoliSolutions (USA)

NINE (Italy), University of Illinois (USA)

and with

GRS (Germany)
Objective of the Courses

The MMARS Courses will provide a transfer of experience and know-how from recognized experts for the application of Models and Methods adopted for the advanced analysis of Nuclear Power Plants. Six different courses consisting of 35 hours each are offered:

Advanced RELAP5 Training: ITF and NPP Safety Analysis
The hands-on training course is directed toward advanced RELAP5 users with system thermal-hydraulics background. The course will provide information on the nodalization techniques of components in Integral Test Facility (ITF) and on the qualification process of a system thermal-hydraulic calculation, including the qualitative and quantitative evaluation of the accuracy. The so called Kv-scaled calculation of a NPP to the selected test in a ITF is also part of the course as well as the identification of simple and complex errors in a NPP input nodalization. Finally the course provides with lectures which give an overview of the code assessment process and of the method to evaluate the uncertainty of system code calculations.

Methods and Codes for Cross Section Generations (Deterministic Methods)
The training is intended for nodal reactor physics code users who need to prepare their own cross-section for nodal LWR reactor core analysis. The course will provide an overview of the methods and codes for cross-section generation, and in depth description of requirements and procedures for nodal cross-section generation. The focus of the course is the reactor physics core analysis and the requirements of coupled codes analysis. Participants will practice hands-on cross-section generation with Serpent and HELIOS codes for LWR core modelling with reactor physics code PARCS.

Preparation and Review of Safety Related Documentation (FSAR)
The objective of the seminar is to develop practical skills required for the preparation and review of the safety related documentation. Target audiences are staff of the regulatory bodies, technical supporting organizations and plant personnel involved in the process of the preparation and review of the safety documentation. Concept of the safety assessment process is discussed including the relevant safety issues, such as defense in depth, graded approach, basic safety functions etc. The seminar primarily focuses on preparation and review of design basis deterministic safety analyses and includes practical exercises on review of selected parts of the SAR of existing PWR and/or BWR. Simplified plant simulator calculations are used during the exercises to enhance the development of review and evaluation skills. The safety assessment requirements practiced during the seminar are based on IAEA safety standards.

Statistical Methods for Uncertainty and Sensitivity Analysis
The course is directed towards beginner users of methods for statistical uncertainty and sensitivity analyses of computational results. It will provide insights into the methods generally applied in this context. The participants of the seminar will learn which types of uncertainty sources may affect a computational result and how to quantify the input uncertainties. The various options to evaluate the uncertainty of a computational result and to identify the most important uncertainty sources will be presented. A good deal of time of the seminar will be spent on hands-on training with SUSA which is based on GRS uncertainty and sensitivity analyses method.

Theoretical Training on Level-1 PSA for Internal Initiating Events
The training is directed toward beginners in probabilistic safety assessment (PSA). The training programme is developed in the way that the participants obtain sufficient knowledge on Level-1 PSA basic concepts and methodology and practical experience in the development of the PSA models using selected PSA Software. The major part of the course will be spent on hands-on training on the development of accident sequences and system models for simplified LWR. It is expected that after training participants will be able to continue PSA activity using PSA software themselves.

Thermal-Hydraulics Core Analysis – COBRA Genesis Codes
A detailed modeling of the core is becoming more important in response of the industry toward higher utilization factor. Fuel cycles increased from 12 months to 18 months and more recently to 24 months over the last three decades and average discharge burnup almost doubled in the same time period. New fuel degradation phenomena have been discovered and, as result, regulatory requirements evolved to reflect the new knowledge gathered. A detailed analysis of the core component is now typical for most scenarios, both Anticipated Operation Occurrences (AOOs) and Design Basis Accidents (DBAs). Moreover a detailed core thermal-hydraulic model is required in various disciplines associated with core engineering (core design, fuel rod design, subchannel analysis, etc.). The trend is now to develop detailed core models in the framework of multiphysics tools. The objective if this introductory course is to review the model needs with focus on the core component and the approach taken for various scenarios. The course will provide an overview of the computer codes used to perform safety analyses and address core engineering problems. Students will learn about the purpose and various uses of these methods. A version of COBRA will be used training. The syntax and the input structure of the code and plotting tools will be covered. Hands-on training on simple modeling is provided. In the last day advanced and future applications of are also presented.
The Network of Nuclear Engineering and Energy Services (NNEES) has organized Hands-on Training Courses directed toward engineers with advanced expertise in System and Core Thermal-Hydraulic Codes, Reactor Physics Codes as well as with enough background in Probabilistic Safety Analysis, Preparation and Review of Safety related documentation (i.e. Final Safety Analysis Report) and Statistical Methods for Sensitivity and Uncertainty Analysis. The Hands-on Training Courses will take place in Pisa (Italy) from 23rd to 27th November, 2015.

The seminar is open to vendors, utilities, regulatory bodies, national laboratories, consulting companies and universities. A minimum of fifteen participants is required to organize the seminar.

Further information about participation and registration as well as useful practical information can be obtained from Alessandro Petruzzi at the following email address: alessandro.petruzzi@nnees.sk. Special accommodation will be offered on a separate sheet. An internet website with the latest news is available at: http://www.nnees/mmars.html

- **Expected Products**

The Courses will provide a transfer of experience and know-how from recognized experts in the respective fields. It will thus contribute to maintaining and increasing technical competence and to ensuring the sustainable development of nuclear technology. CDs containing all lectures will be distributed to the participants.

- **Organizing Committee**

  - C. Frepoli  
  - T. Kozlowski  
  - M. Kristof  
  - A. Lyubarskiy  
  - A. Petruzzi

- **Lecturers**

  - C. Frepoli  
  - M. Kloss  
  - T. Kozlowski  
  - M. Kristof  
  - M. Lanfredini  
  - A. Lyubarskiy  
  - M. Modro  
  - A. Petruzzi  
  - S. Pogosyan

MMARS 2015, Pisa  
23 – 27 November 2015
Dr. Cesare Frepoli is the founder and principal consultant of the FPoliSolutions, LLC. Dr. Frepoli background is based on 25 years’ experience of employment in the nuclear industry. The firm specializes in the development of cutting edge evaluation models and uncertainty analyses for the nuclear industry but not limited to. Dr. Frepoli career path has been highly technical with area of specialization covering thermal-hydraulic, fluid-dynamics, reactor physics, numerical methods, physical models and methodologies for nuclear power plant safety analysis and design. Dr. Frepoli led various development programs within the industry and authored several publications in the area. Cognizant of the various licensing and regulatory aspects of safety analyses methodologies, operation and maintenance of PWRs, as well as design certification of new generation nuclear power plants (AP600/AP1000, SMRs, APWR, and APR1400). Dr. Frepoli was the main developer, and inventor of major advancement in WEC safety analysis technology, starting from ASTRUM (SER in 2004) and more recently with the Full Spectrum methodology for computer simulation and uncertainty analysis models. The development, licensing and implementation in the market of these methodologies provided the opportunity for several personal interactions with customers, electric utilities, and regulatory bodies, both in the US and internationally.

Prof. Tomasz Kozłowski has graduated with a PhD in Nuclear Engineering from Purdue University. He has worked on RELAP5/PARCS and TRACE/PARCS coupling, was one of developers of PARCS core simulator, and worked on homogenization methods for pin-by-pin neutron transport calculations. Following Purdue University, he has worked at the Royal Institute of Technology in Stockholm, Sweden on independent safety analysis of Swedish BWR power uprate. He is an expert in deterministic safety analysis, numerical methods and numerical simulation of nuclear reactors and nuclear reactor systems for analysis of reactor transients and BWR stability. He joined University of Illinois at Urbana-Champaign in October 2011. His current research includes BWR stability, numerical methods, sensitivity and uncertainty, code coupling, and reactor safety.

Dr. Martina Kloss studied Mathematics/Statistics at the technical university of Dortmund in Germany and is working as a chief expert for uncertainty and sensitivity analysis methods at the GRS - Germany’s research and expert organization in the field of nuclear safety and radioactive waste management. She has more than 25 years experiences in developing methods and mathematical models for uncertainty and sensitivity analyses and was one of the main developers of the GRS method and the tool SUSA for uncertainty and sensitivity analyses. She participated in many projects funded by the EU and the German Federal Ministries for Economic Affairs and Energy and for the Environment, Nature Conservation, Building and Nuclear Safety. She applied SUSA in combination with many codes used in deterministic (DSA) and probabilistic reactor safety analyses (PSA) and was author of many publications in these fields. In recent years, her work has been focused on developing the method and tool MCDET (Monte Carlo Dynamic Event Tree) for integrated deterministic and probabilistic safety analyses (IDPSA) where both epistemic and aleatory uncertainties are considered. She published many articles and papers on this topic and is currently working as a consultant of the IAEA for preparing a Technical Document on "Integration of Deterministic and Probabilistic Safety Analysis".

Dr. Marián Kríštof, CEO of the Network of Nuclear Engineering and Energy Services (NNEES), manages the international network of senior experts in the area of nuclear safety assessment and licensing to support various organizations ranging from regulatory authorities, TSOs and research institutes to nuclear power plant operators. He is international consultant specializing on diverse aspects of nuclear safety including TH safety calculations, uncertainty analysis, preparation and/or review of the safety related documentation, preparation of trainings and lectures and management of domestic and international projects in the area of nuclear safety. For several years he is assisting the IAEA in development and implementation of its SAET (Safety Assessment Education and Training) program. The objective of the program is to support the embarking countries in their safety assessment competence building. He worked over a decade for the Slovak regulator as an analyst performing independent safety assessment of the NPPs including code calculations, prepared and reviewed the regulatory documents such as acts, regulations or safety guides and reviewed of the safety related technical documentation to support the regulatory decision-making.
Dr. Artur Lyubarskiy has more than 30 years of experience in the PSA. From 2008 he was the nuclear safety officer of the International Atomic Energy Agency responsible among other activities for the development of various safety related publications in the area of PSA and Integrated Risk informed decision making process and the development and implementation of various training programmes for Level-1 and Level-2 PSA. Prior to joining IAEA, from 1995 till 2008 he was a head of Risk Analysis department in Scientific and Engineering centre of Russian Nuclear Regulatory authority, Moscow, Russia. In that capacity he was involved in various activities and lead different projects, including the performance of Level-1 through 3 PSA for Russian Nuclear Power plants for all type of initiators, including internal event, internal and external hazards, development of regulatory guidelines and norm documents in the area of PSA and risk analysis. In 2005 through 2008 he collaborated with ERI (Energy Research Inc., Rockville, USA), acting as a senior nuclear and reliability engineer, heading the reliability and risk assessment programs at ERI. He has performed in-depth system analysis and reliability/risk assessments as part of a regulatory evaluation of Level-1 PRAs, including external and internal hazards (fires, floods, seismic, high winds, etc.) for several NPPs in Europe. At the beginning of his career he was on the engineering staff of “Gyropress”, main design organization of VVER reactors in Russia, where he was deeply involved in the design optimization of VVER reactors based on risk insights. Dr Lyubarskiy has an extensive experience in the use of various PSA software (e.g. Risk Spectrum, SAPHIRE, FINPSA, CAFTA) and deep knowledge in all aspects of the Probabilistic Safety Assessment for NPPs and Research Reactors.

Dr. Alessandro Petruzzi is a senior engineer of the Nuclear and Industrial Engineering srl (NINE) for which he is also the President of the Board of Directors since 2011. He has more than fifteen years of experience working in the system thermal-hydraulics safety analysis with particular focus on Best Estimate Plus Uncertainty Methods and their application for the Accidents Analysis of Safety Analysis Report. He has experience with PWR, BWR, VVER and PHWR technology. From 2002 to 2004 he worked as visiting scolra at The Pennsylvania state University under contracted sponsored by USNRS (TRACE Assessment for BWR Stability) and DOE (early detection of BWR instabilities). From 2004 to 2007 he was involved as researcher of University of Pisa in the EU Tacis Project devoted to Deterministic safety analysis for VVER and RBMK. From 2007 to 2013 he was the project manager for the preparation of the Chapter 15 of FSAR for the Argentinean NPP Atucha-2. From 2010 to 2011 he worked as senior expert for the development of the RELAP model for Darlington NGS in Canada. From 2010 is a consultant of IAEA for supporting the development of nuclear competences in system thermal-hydraulics for the embarking countries (Poland, Jordan, Vietnam, Malaysia). In 2011 he was nominated European Liaison for the ASME Nuclear Engineering Division. In 2013 he was the chairman of the Organizing Committee of the ANS NURETH-15 conference. He is the founder (in 2004) of the seminar-course 3D S.UN.COP (Scaling, Uncertainty and 3D COuPled Code Calculation) and (in 2010) of the NRSHOT (Nuclear Reactor Safety Hand-On Training) for which he is the course director. He got his PhD Degree in “Nuclear and Industrial Safety” Course of the “Leonardo da Vinci” Doctoral Engineering School of the University of Pisa (I) in 2008 and he published more than 50 articles and papers covering the fields of interest mentioned above.

Dr Mike S. Modro has 37 years of experience in nuclear safety research, analyses and nuclear power system design. Recently retired from the International Atomic Energy Agency, where the last year and half he served as acting Safety Assessment Section Head and earlier as Senior Safety Assessment Officer developing and implementing programmes strengthening safety assessment capacities at IAEA Member States. The professional career includes experimental and analytical research and applications in nuclear safety; management and leadership of thermal-hydraulic research projects, safety analysis programs on advanced systems such as AP600, PRIZM, SBWR, PIUS and recently design reviews of ACR1000, AP1000, APR1400, APR1000, ATME, EPR, ESBWR, AES 2006, TOI, ACP1000. Founder and director of the U.S. Department of Energy International Center for Environmental Safety (1999-2003). U.S. System Integration Manager in charge of research and development of the supercritical water cooled reactor (SCWR), chairman of the Generation IV International Forum (GIF) System Steering Committee on SCWR and member of GIF Risk and Safety Group (2003-2005). Led development of other new reactor concepts including new small and simplified light water power reactors. Currently nuclear safety consultant supporting development of safety infrastructure in countries embarking on nuclear power and provides also support to IAEA safety assessment programmes.
Minimum number of Participants: 7

Lecturers: A. Petruzzi (NINE)
Code Instructors: A. Petruzzi (NINE), M. Kristof (NNEES), M. Lanfredini (NINE)

Day 1

- System Thermal-Hydraulics Codes: Capabilities and Limitations
- Assessment of System Codes
  - The OECD/NEA Computer Code Validation Matrix of Integral Test Facility (ITF)
  - The OECD/NEA Computer Code Validation Matrix of Separate Effect Test Facility
- Exercise: working with an Integral Effect Facility (1/4)
  - Description of the ITF
  - Completing the ITF Input Nodalization
    - Developing a Valve Component
    - Developing a Pump Component
    - Developing the Pressurizer

Day 2

- Exercise: working with an Integral Effect Facility (2/4)
  - Completing the ITF Input Nodalization
    - Developing the Steam Generator
    - Developing the Control System for the Pressurizer and the Steam Generator
- Qualification of a System Code Calculation of a ITF (1/2)
  - Demonstration of Geometrical Fidelity
  - Demonstration of Steady State Achievement
- Exercise: working with an Integral Effect Facility (3/4)
  - Running the Steady State Calculation
  - Demonstration of the Geometrical Fidelity
  - Demonstration of Steady State Achievement

Day 3

- Qualification of a System Code Calculation of a ITF (2/2)
  - Qualification at On-Transient Level
  - Application of the FFTBM
- Exercise: working with an Integral Effect Facility (4/4)
  - Description of the selected Test in ITF
  - Running the Transient Test
  - Qualification of the Transient Calculation
    - Qualitative Analysis
    - Quantitative Analysis by FFTBM
- Origin of Uncertainties in System Thermal-Hydraulics Calculations
- Approaches to perform Uncertainty Analysis
Day 4

- Description of a Generic Input Nodalization of a NPP
  - Description of the NPP Nodalization
  - Description of the relevant I&C System
  - Description of the Steady State
  - Description of the Selected Transient
- Exercise: working with a Generic NPP input Nodalization (1/2)
  - Identifying Simple Input Error –Type 1
  - Identifying Simple Input Error –Type 2
  - Identifying Simple Input Error –Type 3
  - Identifying Complex Input Error

Day 5

- Qualification of a System Code Calculation of a NPP
  - The Kv Scaled Calculation
- Exercise: working with a Generic NPP input Nodalization (2/2)
  - Developing the Kv NPP Scaled Input Nodalization respect to the selected ITF and selected Transient
  - Qualitative Analysis of the Results
- How to perform Uncertainty Analysis by CIAU (Code with the capability of Internal Assessment of Uncertainty)
- Evaluation of the Training Course
Minimum number of Participants: 7

Lecturers: T. Kozlowski (UIUC)
Code Instructors: T. Kozlowski (UIUC)

Day 1

• Opening, Introduction, Scope and Content of the training
• Features and Limitations of nodal core simulator codes
• Procedures and codes for cross-section generation
• Nodal cross-section requirements for static, transient and depletion analysis

Day 2

• HELIOS code introduction – overview of the typical structure, models; input and output
• Exercise: HELIOS single assembly modeling
• Exercise: HELIOS nodal cross-section generation for steady-state, transient and depletion calculation
• Exercise: HELIOS cross-section preparation for PARCS analysis (native format)

Day 3

• Serpent code introduction – overview of the typical structure, models; input and output
• Exercise: Serpent single assembly modeling
• Exercise: Serpent nodal cross-section generation for steady-state, transient and depletion calculation
• Exercise: Serpent cross-section preparation for PARCS analysis (native format)

Day 4

• PMAXS cross-section format
• Exercise: HELIOS cross-section preparation for PARCS analysis (PMAXS format)
• Exercise: Serpent cross-section preparation for PARCS analysis (PMAXS format)

Day 5

• Exercise: PARCS modeling of PWR with native cross-sections (steady-state and transient)
• Exercise: PARCS modeling of PWR with PMAXS cross-sections (steady-state and transient
• Exercise: PARCS modeling of PWR with PMAXS cross-sections (depletion)
• Questions, Open Issues and Individual Consultations
• Closing of the training and release of the Certificate of Attendance
Lecturers: M. Modro (former at IAEA), M. Kristof (NNEES)

**Day 1**
- Current Safety Features of Nuclear Technology (PWR, BWR)
- Process of the Safety Assessment
- Relevant IAEA Safety Standards
- Related Safety Documentation
- Training and Staffing needed for Preparation and Review of FSAR

**Day 2**
- Process of the Deterministic Safety Analysis Preparation
  - Scope
  - Postulated Initiating Events
  - Acceptance Criteria
  - Assumptions of the Analysis
  - Computer Codes

**Day 3**
- Exercise on Deterministic Safety Analysis Preparation

**Day 4**
- Structure and Content of the SAR
- Preparation and Review of the SAR
- Exercise on Selected Transient Review
  - Relevant Aspects and Phenomena associated with the Selected Transient
  - Review of the Relevant SAR Chapter

**Day 5**
- Exercise on Selected Accident Review
  - Relevant Aspects and Phenomena associated with the Selected Accident
  - Review of the Relevant SAR Chapter
- IAEA Generic Reactor Safety Review
- Evaluation of the Training Course
Minimum number of Participants: 7

Lecturers: M. Kloss (GRS)
Code Instructors: M. Kloss (GRS)

**Day 1**
- General concept of statistical uncertainty and sensitivity analyses of computational results
- Analysis input
  - Identification of uncertain input parameters
  - Techniques for input parameter uncertainty quantification
  - Using SUSA to document input parameters and to specify input uncertainties
  - Exercise on input parameter uncertainty quantification

**Day 2**
- Basics on Monte Carlo simulation
  - Sampling procedures and algorithms
  - Sampling procedures implemented in SUSA
  - Computer code runs
  - Using SUSA to perform Monte Carlo simulation
    - Implementing, compiling and running a simple computer code
    - Interfaces to complex codes used in reactor safety analyses
  - Exercise on Monte Carlo simulation

**Day 3**
- Options to quantify the uncertainty of a computational result
  - Key statistics
  - Confidence intervals for means
  - Tolerance intervals
  - Bootstrap intervals
  - Distribution goodness-of-fit tests
  - Metamodelling
  - Options implemented in SUSA
  - Exercise on uncertainty quantification for computational results
Day 4

- Sensitivity analysis methods
  - Correlation and regression based methods
  - Variance decomposition based techniques
  - Association measures from 2x2 contingency tables
  - Two sample tests
  - Graphic techniques
  - Methods implemented in SUSA
  - Exercises on sensitivity analysis methods

Day 5

- Hands-on training with SUSA
Minimum number of Participants: 7

Lecturers: A. Lyubarskiy (former at IAEA), S. Pogosyan (NRSC)
Code Instructors: A. Lyubarskiy (former at IAEA), S. Pogosyan (NRSC)

Day 1

- Brief overview of the Training Curriculum
- Overview of the Main Terms and Concepts of the Probabilistic Safety Assessment
  - Concept of Risk
  - Typical PSA Scope
  - Terms and Definitions used in PSA
  - Introduction to Boolean Algebra
  - Brief overview of Typical Level-1 PSA Tasks
- Initiating Events (IE) Analysis
  - Definition of an Initiating Event in PSA
  - Initiating Events Selection
  - Initiating Events Grouping
  - Initiating Events Frequency Assessment
  - Typical Lists of IEs for Different Reactor Types
  - IEs Frequencies Assessment
- Presentation on the Design of the Facility to be Used for the Case Study on PSA Model Development
  - Presentation of the Main Design Features of a Simplified Nuclear Installation to be used during training
    - Main Safety Systems
    - Main Support Systems
  - All needed Material will distributed in paper copies and electronic form
- Accident Sequence Models Development
  - Safety Functions and Modelling Functions in the Accident Sequence Models
  - Successful and Non-Successful End States
  - Typical definition of End States in Accident Sequence Models
  - Success Criteria Definition
  - Supporting Analyses
  - Mission Time and Cliff-Edge Effects
  - Typical Formats of Accident Sequence Models
  - Approach for Construction of Small Event Tree - Large Fault Tree PSA Models
  - Example of Event Trees Construction Process
- Introduction to PSA Software and Construction of Event Trees
  - Brief description of the PSA Software Used (SAPHIRE or RISKSPECTRUM)
  - Work in working groups: each group develops one ET (later to be integrated in one PSA model)
  - Each working group construct one ET (Large LOCA, Medium LOCA, Small LOCA, Loss of Off-Site Power)
  - Expert(s) provides advices at request and reviews the work done
Day 2

- System Models Development
  - Typical Methods for System Models Development
    - Fault Tree (FT) Method
      - Structure of FT
      - Typical Elements of FT
      - Typical Basic Events in FT
      - Types of Components, Components Failure Modes and Related Probabilistic Models
      - Components Boundaries and Link with Existing Reliability Data
  - Methods to Control the Logic of System Models in FTs (house events, boundary condition sets, etc.)
  - Typical Coding Scheme for Naming Basic Events in the PSA Model
  - Failure Modes and Effects Analysis (FMEA)
  - Example of a FT Development Process, Breaking Logic Loops
- Exercise: Construction of Fault Trees. Work in working groups:
  - Each group constructs one or two systems models - FTs (depending on the system complexity)
  - Development of the Coding Scheme to be used in the PSA Model Development Process
  - Performing and Documenting FMEA
  - Construction of System Models in the Form of FT (using PSA Software)
  - Defining Reliability Models for Basic Events and Introduction of “Dummy” Values
  - System Models Quantification
  - Analyses of Minimal Cutsets (MCSs) obtained

Day 3

- Discussion on the Intermediate Results Achieved
- PSA Data Analysis
  - Overview of Data required for PSA
  - Component Reliability Data Collection and Treatment Process
  - Initiating Events Data Collection and Treatment
  - Methods for Reliability Parameters and IEs Frequencies Estimation
    - Classical Statistic Methods
    - Bayesian Updating Process
  - Demonstration of the EXCEL Programme for Reliability Parameters and IEs Frequencies Estimation
  - Introduction to Human Error Probability (HEP) Evaluation
- Exercises on data assessment. Work in working groups:
  - Each Group Performs Assessment of the Data Needed for their part of the PSA Model
  - Treatment of the Raw Data (data to be provided by experts)
  - Estimation of Reliability Parameters for the Components and Failure Modes to be used in the PSA Model Under Development
  - Entering the Data obtained in the PSA Model
  - Quantification of MCSs with the Data obtained (replacement of “dummy” values)
Day 4

- Discussion on the Intermediate Result Achieved
- Modelling Dependencies in PSA (including Common Cause Failures [CCF])
  - Types of Dependencies to be modelled in the PSA
  - Common Cause Failures (CCF)
    - CCF Models
    - CCF Parameters and Sources of Data
    - CCF Groups
  - Inclusion of CCF Events in the PSA Model (depending on the software to be used)
- Exercise: Modelling of the CCF in the PSA model. Work in working groups:
  - Each Group perform CCF Analyses for the Equipment Modelled in their part of the Model
  - Defining CCF Groups
  - Introduction of CCF Groups in the PSA Model
  - Re-Quantification of the Model with CCF Data
- Discussion on the Intermediate Result Achieved

Day 5

- PSA Model Quantification and Analysis of the Results
  - Typical Results from Level-1 PSA
  - Analysis of MCSs and Dominant Accident Sequences
  - Importance Measures Quantification and Analysis:
    - Types of Importance Measure
    - Typical Insights from Importance Measures
  - Uncertainty Analyses and Types of Uncertainties in PSA:
    - Parametric Uncertainty
    - Modelling Uncertainty
    - Incompleteness Uncertainty
  - Sensitivity Studies:
    - Sensitivity Studies to Address Uncertainties
    - Sensitivity Studies to Assess the Impact of Modifications
    - Other Sensitivity Studies
- Exercise: PSA Model Quantification and Analysis of the Results. All working groups are working with the same model
  - Each group performs Quantification of the Model and Review MCSs obtained
  - Each group reviews the Results of Importance Measures Quantification and Uncertainty of the Results
  - Each group defines one Sensitivity Case and performs:
    - Model Adjustment for the case
    - Model Quantification for the case
- Discussion on the Final Results achieved
- High-Level overview of Internal and External Hazards Level-1 PSA, Level-2 PSA and use of PSA Results
  - High-Level overview of Internal and External Hazards Level-1 PSA
  - High-Level overview of Level-2 PSA Tasks
  - High-Level overview of Use and Application of PSA
- Evaluation of the Training Course
THERMAL-HYDRAULICS CORE ANALYSIS – COBRA GENESIS CODES

Minimum number of Participants: 7

Lecturers: C. Frepoli (FPoliSolutions)
Code Instructors: C. Frepoli (FPoliSolutions)

Day 1

- Opening, Introduction, Scope and Content of the training
- Core Thermal-Hydraulic Model Requirements within the Standard Review Plan
- Core Thermal Design – Single Channel
- Two-Phase Core Thermal Design
- Hot Channel Analysis
- Single-Channel Analysis
- Core Modeling Needs for LOCA Safety Analyses

Day 2

- Core Thermal Hydraulic Codes Available in Industry
- COBRA Code and Methods
- COBRA Applications
- COBRA Model Description
- Boundary Conditions for COBRA

Day 3

- COBRA Modeling of PWR Core
- Qualification of COBRA Model for non-LOCA Analyses
- Qualification of COBRA Model for LOCA Analyses
- Qualification of COBRA Fuel Rod Model
- Interfaces with Other Disciplines (Core Physics and Fuel Rod Thermomechanics)

Day 4

- Sample Studies and Class Exercises

Day 5

- Applications and Future Use of Detailed Core Models
- LOCA Applications
- Questions, Open Issues and Individual Consultations
- Evaluation of the Training Course
REGISTRATION FORM
To be returned by 31 July 2015

Last name: .......................................................... First name: .......................................................... Sex.............
Title:................................................ Organization:................................................................................ V.A.T #..............
Organization Address..............................................................................................................................................
City: ........................................ State: ....................... Zip Code: ............... Country .............................................
Phone: .................................................................. Fax: ...................................... Email: .....................................................
(Please type all information as you wish it to appear on your name badge)

Participants shall bring their own laptop. .................................................................

Registration Fees

Include proceedings, lunches, coffee breaks and one official dinner:
Advanced RELAP5 Training ................................................................. €2200
Methods and Codes for Cross Section Generations ........................................ €2000
Preparation and Review of Safety Related Documentation ................................ €2000
Statistical Methods for Uncertainty and Sensitivity Analysis ................................ €2000
Theoretical Training on Level-1 PSA for Internal Initiating Events ................ €2000
Thermal-Hydraulics Core Analysis – COBRA Genesis Codes ....................... €2200

* Bank charges to be added to registration fees

Payment Terms and Cancellation Policy

☐ 50% cancellation fee applies if cancellation of the registration occurs up to 4 weeks before the starting of the course
☐ 100% cancellation fee applies if cancellation of the registration occurs 4 weeks or less before the starting of the course
☐ Payment of the registration fee is by bank transfer by August 31, 2015. Information will be provided separately

The Registration Form should be sent to:
Alessandro Petruzzi :  FAX #: 0039 050 2210384  email: a.petruzzi@nnees.sk