Final Announcement

Lucca, Italy

November 13 – 17, 2017

Models and Methods for Advanced Reactor Safety analysis (3rd Course)

http://nnees.sk/mmars/

N.IN.E.

Network of Nuclear Engineering and Energy Services

in cooperation with:

AEP (Russia), APOSS (Croatia), AREVA (USA),

BWXT (USA), FPoliSolutions (USA), JRC-ITU (EC),

NCSU (USA), NRSC (Armenia)

P R O G R A M M E   O U T L I N E

Objective of the Courses

The MMARS-3 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. The seminar is open to vendors, utilities, regulatory bodies, national laboratories, consulting companies and universities. A minimum number of participants is required to organize each course. Eight 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 and 3D NK

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 Licensing 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.

Practical and 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.

Fuel Behavior Analysis by TRANSURANUS

The aim of the training is to provide both practical and theoretical insights on nuclear fuel behavior. Nuclear fuel undergoes continues changes while it is irradiated. Such changes affect the thermo-mechanical fuel characteristics and hence the fuel behavior and response both under normal and off-normal conditions. A series of relevant phenomena will be illustrated and discussed within the theoretical part and addressed into the practical sessions by hands-on training on suitable examples. The course will provide an overview of the computer code including the syntax, the preparation of the inputs and the analysis of the results.

Severe Accident Analysis: Phenomenology and Computational Tools

The severe accidents at Three Mile Island (TMI), Chernobyl, and Fukushima are a reminder that commitment to nuclear power includes a commitment to public safety. The nuclear industry recognized early the potential hazards of nuclear power. Severe accident has acquired an increasing relevance from the point of view of licensing and some severe accident are now recognized and included in the design basis accidents. Features to prevent, contain, and otherwise protect the public from reactor accidents were applied from the outset. As the industry has evolved, so has safety in the form of design features and strategies to both prevent severe accidents and mitigate consequences should they occur. This course presents both historical and technical information regarding severe accidents in the design and safety assessment of nuclear power plants. It is divided into daily morning theory and afternoon practice sessions. Theory aspects address phenomenology, accident progression, challenges to containment integrity, and radiological release and transport, computational tools. Practice aspects address licensing, computer codes applications, deterministic and probabilistic evaluation methods, and modeling.

Important Elements of Risk Quantification and PSA

The objective of the course is to give to the participants the essential knowledge of those elements and aspects of the probabilistic safety analysis (PSA) and risk quantification which are most important but are usually only briefly touched at the training courses or workshops based on particular PSA software tool and held either by a vendor or a user of particular tool. The purpose is to make the participants understand the important elements of quantitative risk modeling regardless of the software to be used and of the facility or system to be modeled, and to enable them, thus, to build the quantitative risk model by means of any software tool and for any industry or particular facility. Many of the PSA training courses begin with event trees (ET) and fault trees (FT) which are the skeleton of the risk model for any complex system. In engineering practice (and nuclear safety engineering in particular), however, no fault tree for a system or function should be developed by an analyst who does not have the essential knowledge in the topics which include reliability engineering (with underlying knowledge of probability theory), human reliability analysis (HRA) techniques, quantitative parameters estimate and quantitative treatment of uncertainty, to name some of them. Likewise, any member of a PSA / risk quantification project or group should understand the principles of the risk curve and the roles of deterministic safety margin analyses and probabilistic risk analyses in the design verification process. The course purposefully does not involve work on or presentation of any PSA software tool. However, it includes practical hands-on exercises on all relevant aspects of risk quantification by means of the elementary tools such as spreadsheets or "manual" calculations, which provide the first-hand experience. Who should come to this training course:

- PSA newcomers who want to / need to have an understanding of PSA / quantitative risk analysis as a whole;
- Specialists in specific PSA tasks such as data analysts, HRA, ET/FT model developers and others who want to / need to have an integrated understanding of PSA;
- PSA or safety analysis / assessment managers who, by definition of their positions, need to have an integrated understanding of PSA;
- Deterministic design basis assessors who need to understand the risk assessment;
- Just any analyst or assessor (either from utility or regulatory body or industry or any other organization) involved in the development, verification or review of design of systems or functions in industrial or societal facilities.

The overall course is divided into the nine main topical areas covered by nine lectures.

Organization

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 and Fuel Behavior Codes, Severe Accidents as well as with enough background in Probabilistic Safety Analysis and Preparation and Review of Safety related documentation (i.e. Final Safety Analysis Report). The Hands-on Training Courses will take place in **Lucca** (**Italy**) from **13th** to **17th November, 2017**.

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://nnees.sk/mmars/**

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

M. Kristof	NNEES, Slovakia
M. Modro	NINE, Italy
A. Petruzzi	NINE, Italy

Lecturers

M. Avramonova	NCSU, USA
G. Baiocco	NINE, Italy
S. Buznuni	NRSC, Armenia
M. Cherubini	NINE, Italy
C. Frepoli	FPoliSolutions LLC, USA
W. Giannotti	NINE, Italy
M. Kristof	NNEES, Slovakia
A. Lyubarskiy	AEP, Russia
R. Martin	BWXT, USA
M. Modro	NINE, Italy
A. Petruzzi	NINE, Italy
R. Sanders	AREVA, USA
P. Van-Uffelen	JRC-ITU, EU
I. Vrbanic	APOSS, Croatia

Dr. **Maria Avramova** is an Associate Professor in the Nuclear Engineering Department of North Carolina State University (NE NCSU). She is the Director of the Consortium for Nuclear Power (CNP) and the Director of the Reactor Dynamics and Fuel Modeling Group (RDFMG) at NCSU. Dr. Avramova earned her Ph.D. degree in nuclear engineering from the Pennsylvania State University (PSU) in 2007. Prior to joining PSU in 2001, she held a research scientist position at the Institute of Nuclear Research and Nuclear Energy, Bulgarian Academy of Science, Sofia, Bulgaria. After her graduation, Dr. Avramova has held positions at PSU of post-doctoral scholar, Assistant Professor, and Associate Professor of Nuclear Engineering. She joined NE NCSU in 2015. Her background includes development, verification, and validation of thermal-hydraulics sub-channel, porous media, and CFD models and codes for reactor design, transient and safety computational analysis. Her latest research efforts have been focused on high-fidelity multi-physics simulations and on uncertainty and sensitivity analysis of reactor design and safety calculations. Dr. Avramova has led high visibility international projects such as the OECD-NEA/U.S. NRC BFBT benchmark, the OECD-NEA/U.S.NRC PSBT benchmark, and the OECD LWR UAM Benchmark. She has advised and graduated 12 PhD students, 19 MS students, and 38 MEng students. She has published more than 100 papers in peer-reviewed journals and proceedings of international conferences.

Dr. **Giorgio Baiocco** has graduated in Nuclear Engineering from “Sapienza” University of Rome in January 2015 with a thesis concerning the transmutation of minor actinides in sodium fast reactors (SFRs). He is the winner of a European grant titled “New MA-bearing fuels: evaluation of performances and burn-up calculations into CP-ESFR neutron flux” in the framework of the PELGRIMM (PELlets versus Granulates: Irradiation, Manufacturing & Modelling) project. This internship has been carried out into ENEA research centre “Casaccia”. Since March 2015 he is working for Nuclear and INdustrial Engineering srl (NINE) in the reactor physics area. In particular he is involved in the cross section generation, core physics modeling and uncertainty quantification. Dr. Baiocco has experience with HELIOS, SCALE, PARCS, MCNP, SERPENT and GenPMAXS codes. He is involved into the OECD Benchmark for Uncertainty Analysis in Best-Estimate Modelling and member of International Serpent User Group.

Dr. **Suren Bznuni** is a senior specialist at Nuclear and Radiation Safety Center (TSO of the Armenian Nuclear Regulatory Authority). He has more than 15 years of experience in DSA in the areas of reactor physics, criticality safety, fuel safety, as well as independent safety assessment of licensee submittals for regulatory body decision making, drafting of national regulations and regulatory guides. He is experienced in PWR and WWER type technologies as well as NUHOMS technology for spent fuel storage. Dr. Bznuni graduated from Yerevan State University with a Master’s degree in Reactor Physics in 1999. He obtained his PhD in Reactor Physics and Numerical Methods and Modeling from Joint Institute for Nuclear Research (Dubna, Russia) in 2002. He was employed by Joint Institute for Nuclear Research as a scientific staff member from 1999-2002. Since 2003, Dr. Bznuni leads the Reactor Physics and Nuclear Fuel Group at NRSC. He is also Associate Professor at Yerevan State University involved in lecturing and nuclear safety research starting 2002. He is a member of the Reactor Physics Group at WWER Regulators Forum and International Technical Working Group (ITWG) on Nuclear Forensics.

Dr. **Marco Cherubini,** is a senior engineering of the Nuclear and Indutsrial Engineering srl (NINE) elected as Vice-President of the Board of Directors since 2011. He started his activities in the nuclear technology more than ten years ago dealing with system thermal-hydraulics safety assessment of nuclear installation originated by different design, such as VVER, PWR, PHWR and RBMK. 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, with a research on development of an Accident Management strategy for VVER-1000 reactor including the optimization of operator actions. He was involved in various research projects led by the University of Pisa as junior expert. Afterwards he was responsible for the thermal-hydraulic simulations in the framework of the licensing pf Atucha-2 NPP, including the preparation and review of part of the FSAR Chapter 15. In the last five years his interest moved towards nuclear fuel, acting as Italian representative in the Working Group on Fuel Safety (WGFS) under the OECD-NEA. Within WGFS is, among the other things, co-organizer of a benchmark dealing with Reactivity Initiated Accident simulation. From 2014 he is an IAEA consultant in mentoring projects for embarking countries. He coauthored tens of papers in international conferences and journals. He is reviewer of conference and journal papers and he acted as co-track leader and session chair on Topfuel-2015.

Dr. **Cesare Frepoli** is the founder and principal consultant of the FPoliSolutions, LLC. Dr. Frepoli background is based on 25years' 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 for computer simulation and uncertainty 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 LOCA product which represents a first-of-a-kind realistic small/intermediate/large break LOCA integrated Evaluation Model in the industry. Other major contributions include the analysis and the development of the technical defense with NRC/ACRS of the safety case for the AP1000 and support to the resolution of Generic Safety Issues such as GSI-191, Risk Informed Regulation initiatives such as 10 CFR 50.46a and 10 CFR 50.46c. While the main area of expertise is in evaluation models development and assessment, past contributions include participation to several test programs, starting from the SET and IET campaign for AP600, the RBHT (sponsored by NRC) in the late 1990's, and other smaller test programs within the industry throughout the years. The development, licensing and implementation in the market of these technologies provided the opportunity for several personal interactions with customers, electric utilities, and regulatory bodies, both in the US and internationally.

Dr. **Walter Giannotti** has PhD graduated in Nuclear Engineering at Pisa University. He is a senior nuclear engineering of the Nuclear and Industrial Engineering srl (NINE). He has more than twenty years of experience working in the system thermal-hydraulics safety analysis. He performed code applications for safety analysis o PWR, BWR, VVER and PHWR nuclear power plant related to thermal-hydraulic and severe accident analysis. He worked on Code Assessment (Relap, Cathare, Melcor, Smart) and Best Estimate Plus Uncertainty Methods (UMAE). From 1996 to 2012 he was a consultant of University of Pisa in the EU Tacis Project devoted to Deterministic safety analysis for VVER, RBMK and the preparation of the Chapter 15 of FSAR for the Argentinean NPP Atucha-2, SA analysis (Melcor, SCDAP, Gothic codes) of WWER (Temelin, Kozloduy), RBMK (Ignalina), PWR (Angra), PHWR (Atucha). He was senior expert in "European Project for Technical Assistance for Implementing Nuclear Safety Project - Thermo-hydraulic experiments at the PSB-RBMK integral Test facility". He performed, as tutor, courses concerning Nuclear Safety for VATESI-Lithuania (spent fuel), CNEN-Brasil (severe accident), CNPE-China (severe accident), ANRA-Armenia (safety analysis). From 2013 is a consultant of IAEA for supporting the development of nuclear competences in severe accident. He is in the Task Group WGAMA - OECD for "Long-term management guide" and "Informing SAM guidance and actions". In 2012 he worked as nuclear expert in the "Final Design for the EUREX plant in Saluggia of the Waste Management Facility (WMF)". He is qualified radioprotection expert (first level) and he is involved in radioprotection field (shielding and safety analysis) in nuclear and conventional environment.

Dr **Marián Krištof**, 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 "Gydropress", 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 **Robert Martin** (Nuc Eng Phd 1996, Penn State; Nuc Eng BS ‘87, MS ‘89, Texas A&M) is a Technical Consultant at BWX Technologies leading the safety analysis methods team for the mPower™ small modular reactor program. He has over twenty-five years of professional experience in the development and regulatory defense of nuclear-thermal-hydraulic evaluation methodologies for both design-basis coolant system faults and severe accidents. The development of these methods has been through industry- and DOE-sponsored programs advancing conventional LWRs, the SRS production reactors, AP600, SBWR, PIUS, USEPR, VHTR (gas), mPower and the TWR-P (sodium) designs. His experience includes employment at the Idaho National Laboratory supporting development of the RELAP5-3D thermal-hydraulic systems computer code. He is also a long-standing member of the American Nuclear Society (ANS), serving on executive committees for both the Thermal-Hydraulic Division and the Nuclear Installation and Safety Division.

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, ATMEA, 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.

Dr. **Alessandro Petruzzi** is a senior engineering of the Nuclear and INdutsrial 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. From 2002 to 2004 he worked as visiting scholar 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 and (in 2010) of the for which he is the course director.

Robert Sanders (Nuc Eng MS 1986, Nuc Eng BS ’85, Univ. of Tenn.) is an Advisory Engineer at AREVA NP in both the Radiological and Severe Accident Groups. He has over 30 years of professional experience in developing and performing severe accident research studies on various types of reactor designs, including the U.S. EPR, Bellefonte NP, and KERENA. As part of his career in Severe Accident Research, he spent several years at Oak Ridge National Laboratory incorporating several new models into the severe accident code, MELCOR. These models included: improved bottom head failure assessment as well as emergency condensers for BWR designs. He has also served on two expert panels for MELCOR in support of the NRC’s SOARCA and Crosswalk Projects. Currently he is engaged with DOE/NRC/EPRI/NEI on the US Efforts in Support of Examinations at Fukushima Daiichi.

Dr P. Van Uffelen graduated from the Vrije Universiteit Brussel (Belgium) with a master degree in electromechanical engineering in 1992 and from the Université Libre de Bruxelles (Belgium) with a degree in nuclear engineering in 1994. He obtained his Ph.D. in applied science from the Université de Liège (Belgium) in 2002. He was employed by SCK-CEN (Belgium) as a scientific staff member from 1994 to 2002, and during that period he was detached for 12 months to the OECD Halden Reactor Project (Norway). Since 2002 he is responsible for the modelling group in the Materials Research Unit of the JRC-ITU in Karlsruhe (Germany), and in particular for the development of the TRANSURANUS fuel performance code. He is member of the Technical Working Group for Fuel Performance and Technology (TWGFPT) from the IAEA and of the Working Party on Multi-scale Modelling of Fuels and Structural Materials for Nuclear Systems (WPMM) from the OECD-NEA.

Dr. **I. Vrbanic** is one of the founders of and a senior consultant for nuclear safety in APOSS. He has a background of 28 years in risk and safety assessments and analyses. Of those, the first 16 years he spent in the Engineering Division at Krsko NPP being involved in developing and applying PSA and risk models in support of differrent aspects of plant's operation, licensing and design. The remaining 12 years he has been spending as a full-time consultant in APOSS. He was involved, in supporting, co-ordinating or leading roles, in a number of projects concerning risk and safety analyses and their applications internationally. Those included PSA applications to support ranking and implementation of plant modifications, risk assessment from natural hazards such as earthquakes, high winds and aircraft crashes, PSA support to in-service inspection programs, risk aspects of aging of structures and equipment and a number of others. The end users of the results and/or services were utilities, regulators, technical support organizations and industry in different countries. This also included lecturing at workshops and training courses on the topics of risk assessment, PSA and safety evaluations., as well as a number of IAEA expert missions and assignments worldwide. Dr. Vrbanic is a scientific collaborator in the field of electrical engineering to the University of Zagreb and wrote or participated in writing of a number of papers published in the proceedings of international conferences and magazines.

PROGRAMME OUTLINE

ADVANCED RELAP5 TRAINING: ITF AND NPP SAFETY ANALYSIS

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 & Code Instructors:

A. Petruzzi (NINE), D. De Luca (NINE)

PROGRAMME OUTLINE

METHODS AND CODES FOR CROSS SECTION GENERATIONS (DETERMINISTIC METHODS)

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

Minimum number of Participants: 7

Lecturers & Code Instructors:

S. Bznuni (NRCS), G. Baiocco (NINE)

PROGRAMME OUTLINE

PREPARATION AND REVIEW OF LICENSING DOCUMENTATION (FSAR)

Day 1

- Fundamentals of Safety Assessment
- International Safety Standards
- Safety Features of NPP Design
- Safety Documentation

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
- Training and Staffing for Preparation and Review of SAR
- Evaluation of the Training Course

Minimum number of Participants: 7

Lecturers & Code Instructors:

M. Modro (NINE), M. Kristof (NNEES)

PROGRAMME OUTLINE

PRACTICAL AND THEORETICAL TRAINING ON LEVEL-1 PSA FOR INTERNAL INITIATING EVENTS

Day 1

- Brief overview of the Training Curriculum
- Overview of the Main Terms and Concepts of PSA
  - 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

PRACTICAL AND THEORETICAL TRAINING ON LEVEL-1 PSA
FOR INTERNAL INITIATING EVENTS

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

Minimum number of Participants: 7

Lecturers & Code Instructors:

A. Lyubarskiy (AEP), S. Pogosyan (IAEA)

PROGRAMME OUTLINE

THERMAL-HYDRAULICS CORE ANALYSIS – COBRA GENESIS CODES

Day 1

- Opening, Introduction, Scope and Content of the training
- COBRA General Overview and Formulation
- COBRA Usage
  - Subchannel analysis
  - LOCA analysis
- Analysis examples
- COBRA manuals/documentation
- Building a Model – General Overview

Day 2

- Building a model – Geometry Data
- Channel, Sections and Gaps
- Merging and splitting of subchannels
- Advection of momentum at section boundary
- Form losses setting
- Exercise 1: Start modeling a bundle (RBHT) – Two-Channel with single hot channel
- Exercise 2: Run hydro problem (single-phase adiabatic)

Day 3

- Building a model – Rods and Unheated Conductors Data
- Heat transfer package
- Exercise 3: Upgrade bundle model by adding heat structure

Day 4

- Simplified sub-channel analysis
- Exercise 4: Run and analyze simplified sub-channel analysis

Day 5

- LOCA reflood analysis
- Quench front models
- Exercise 5: Run and analyze reflood transient

Minimum number of Participants: 7

Lecturers & Code Instructors:

M. Avramova (NCSU), C. Frepoli (FPoliSolutions)

PROGRAMME OUTLINE

FUEL BEHAVIOR ANALYSIS BY TRANSURANUS

Day 1

- Brief overview of the Training Course
- Nuclear fuel behavior under normal operation
  - Fuel characteristics modification
    - Geometrical
    - Thermal
    - Mechanical
- Approach to nuclear fuel modeling
  - Set of equations
  - The TRANSURANUS approach
    - Main assumptions and limitations of the models
- TRANSURANUS application to fuel rod nominal operation

Day 2

- Nuclear fuel behavior under accident conditions: Loos of Coolant Accident (LOCA)
  - General overview of a LOCA scenario
- Relevant phenomena occurring in LOCA scenario:
  - Clad overheating, PCT and Quench
  - Zr phase transformation
  - Clad Oxidation
  - Clad deformation (ballooning)
  - Hydriding
  - (Possible) clad rupture
  - Fuel fragmentation
  - Fuel dispersal (in case of clad rupture)
  - (Transient) Fission Gas Release
- TRANSURANUS application to LOCA simulation

Day 3

- Nuclear fuel behavior under accident conditions: Reactivity Initiated Accident (RIA)
  - General overview of a RIA scenario
- Relevant phenomena occurring in RIA scenario (part 1)
  - Boiling occurrence (depending on the power peak)
  - Clad Oxidation
  - Clad overheating and PCT
  - Zr phase transformation
  - Clad Quench
  - Hydriding
  - Pellet Clad Interaction / Pellet Clad Mechanical Interaction
  - (Transient) Fission Gas Release
  - (Possible) clad rupture
- TRANSURANUS application to RIA simulation

Day 4

- Safety aspects and concerns connected with nuclear fuel
  - Safety limits understanding and justification (from nuclear fuel perspective)
- Overview of TRANSURANUS coupling
  - TH coupling for BIC derivation
- TRANSURANUS application to selected problems
  - Modeling rodlet refabrication
  - High burnup structure
  - Ramp simulation

Day 5

- Overview of experimental programs, facilities and databases related with nuclear fuel behavior
- Sensitivity analysis by statistical method in TRANSURANUS
  - Theory
  - Hands-on training
- Questions and Answers, Open Issues
- Evaluation of the Training Course

Minimum number of Participants: 7

Lecturers & Code Instructors:

M. Cherubini (NINE), P. Van-Uffelen (JRC-ITU)

PROGRAMME OUTLINE

SEVERE ACCIDENT ANALYSIS:
PHENOMENOLOGY AND COMPUTATIONAL TOOLS

Day 1

- SA History and its relevance in safety analysis
  - Short review of main historical severe accidents
  - Development of SA analysis approach in safety analysis
  - Current relevance of the SA analysis
- Phenomenology review
  - In-vessel event progression (BWR/PWR)
  - Ex vessel phenomena (PWR/BWR)
  - CANDU and fuels channel reactor
  - Special issue: SFP phenomena (loss of coolant, loss of cooling)

Day 2

- SA analysis methodologies (deterministic, probabilistic)
- Severe accident licensing
  - IAEA
  - Other organization (WENRA…)
  - Countries legislation
- Severe accident codes (description and comparison)/1
  - System codes
    - SCDAP
    - MELCOR
    - MAAP
  - Containment codes
    - Gothic
    - Cocosys
    - Smart
- Exercises:
  - Melcor application (vessel and core simulation)
  - SCDAP application (vessel and core simulation)
  - Nodalization, models set up and comparison of results

Day 3

- Severe accident codes (description and comparison)/2
- Containment integrity
  - Short term and long term actions
  - Preventive actions in modern plant
  - Mitigate actions in modern plant
- Uncertainty analysis for severe accidents
- Exercises:
  - Melcor application (containment simulation)
  - Gothic application (containment simulation)
  - Nodalization, models set up and comparison of results

Day 4

- Experimental tests for SA and code qualification
- RadioNuclide (RN)release and source term
  - Code models
- Exercises:
  - Melcor application (RN release or ex vessel phenomena simulation or Spent Fuel Pool accident)
  - MAAP application (RN release or ex vessel phenomena simulation or Spent Fuel Pool accident)
  - Nodalization, models set up and comparison of results

Day 5

- Example of codes applications results
- How to perform a SA analysis step by step: user and regulatory authorities point of view
  - Needed data
  - Code selection
  - Nodalization detail and qualification
  - Code running requirement
  - Results evaluation and verification
- Evaluation and Closure of the Training Course

Minimum number of Participants: 7

Lecturers & Code Instructors:

R. P. Martin (BWXT), W. Giannotti (NINE), R Sanders (AREVA)

PROGRAMME OUTLINE

IMPORTANT ELEMENTS OF RISK QUANTIFICATION AND PSA

Day 1

- **Boolean Algebra, Probability Theory, Random Variables**. Basic rules and properties of Boolean algebra. Laws of absorption and idempotency and why are they important for PSA. De Morgan's theorem. Definition of probability. Probability of logical sum of mutually exclusive events. Probability of logical sum of any events. Probabilities of logical product and logical sum of independent events. Conditional probability and event sequences. (Why are these things important in PSA or quantitative risk assessment in general?) Random variables, distinctive and continuous. Probability distributions. Cumulative functions and density functions. How we determine or "judge out" a probability distribution.

- **Definition of "Risk" for an Engineer and Its Role in Considerations of Safety**. "Risk curve" in the probability-consequence space. Frequency (probability) of exceedance versus frequency (probability) of occurrence. Theoretical definition of risk. (Why is risk in the PSA guidebooks presented by "complementary cumulative distribution function"?) Presentation of "risk curve" in practice: risk matrix. Simplification by means of representatives or substitutes for consequences (e.g. "reactor core damage", "large (early) releases…). Margin assessment and risk assessment. What is the real meaning of "combined use of deterministic and probabilistic safety analyses"?

Day 2

- **Definition of "Risk" for an Engineer and Its Role in Considerations of Safety (continued)**. Risk modeling: logic risk equation and quantitative (numeric) risk equation. Methods and approaches. FMEA, HAZOP and similar techniques for risk evaluation. Event trees (ETs) and fault trees as tools to develop the risk equation. Elements of ETs and FTs. "Basic events". Structure function.

- **Basics of Reliability Engineering As Used in PSA**. Basic differential reliability equations. "Repair-to-failure" cycle. "Failure-to-repair" cycle. Combined cycle. Basic terms: failure (repair) density function, failure (repair) rate, and others. (What does it really mean "reliability" and what really is "availability"?) Types of equipment ("components") modeled in PSA. Repairable versus non-repairable. Standby versus normally operating. Failure-on demand versus time-related failure. Reliability models for "basic events" representing failures of systems, structures and components (SSC) in PSA model. (Failure during mission time. Standby, periodically tested component. Frequency-type basic events associated with initiators.) Parameters required to quantify the reliability models (basic events) in PSA.

Day 3

- **Common Cause Failure (CCF) Modeling in PSA**. Independent versus dependent failure events. Unconditional versus conditional failure probability. Importance of CCF modeling in PSA model fault trees. CCF group. Basic CCF failure probability model. Beta Factor Model. Multiple Greek Letters (MGL) Model. Alpha Factors Model. Estimators for MGL and alpha factors. Availability of CCF data. Mapping ("specialization") of CCF events from source to target systems or plants. MGL vs. Alpha Factors: which one is "better" to use in a PSA?

- **Data Analysis in PSA**. Point estimate. Likelihood function. Maximum likelihood estimator (MLE). Estimate with quantitative characterization of uncertainty. Two basic concepts: confidence intervals and uncertainty distributions. Bayesian inference based on prior ("generic") knowledge and observed evidence ("plant-specific". Basic terms: Prior distribution. Likelihood function. Posterior distribution. Bayesian inference as applied to the two most basic reliability parameters in a PSA: probability of failure on demand and failure rate (Prior distributions usually found in the "generic" databases; Establishing of likelihood functions; Posterior distributions: numerical integration or analytical calculation.) Estimating or assessing other parameters for PSA model: Initiator frequencies. Unavailability due to test or maintenance (TM). Others. How to estimate failure rate or failure probability in the case of zero failures? How to establish uncertainty distribution for other types of parameters?

Day 4

- **Human Reliability Analysis (HRA)**. Main types of human errors: pre-initiator errors, initiator-inducing errors and post-initiator errors. Identification of human interfaces, characterization of human failure events (HFE), definition of success criteria and other relevant shaping factors. Overview of techniques for the assessment of human error probabilities (HEP) which are nowadays used in PSAs, including: THERP, ASEP, SLIM, ATHEANA, SPAR-H and others. Dependency among human failure events. Unconditional and conditional HEPs. Incorporation of HFEs into the PSA model. Screening. Quantification.

- **Risk Quantification**. "Risk" as a frequency (probability, likelihood) of the top event considered by the PSA. What is a "minimal cutset"? PSA-model integration principles: "master FT" versus combining system-level MCSs. Quantification of top-event based on the generated "list" of MCSs. ("Inclusion-exclusion principle" and its approximations. What is "Mincut Upper Bound" (MCUB) approximation?) Issues with "success events". Propagation of parameter uncertainty. Moment propagation. "Monte Carlo" sampling. Uncertainty distribution for top event probability (frequency). Mean of the distribution versus point estimate.

PROGRAMME OUTLINE

IMPORTANT ELEMENTS OF RISK QUANTIFICATION AND PSA

Day 5

- **Special Topics 1: Probability of Phenomena and Other Probabilities Needed for PSA and Quantitative Risk Assessments**. Concept of "Accident Progression Event Tree" (APET) or event tree for phenomena and how it compares to event trees for safety functions. Structural failure probability. Consideration of stress versus strength (load versus capacity). Convolution of stress and strength distributions. Other similar concepts. Time required versus time needed. Probability of recovering power or other supporting functions or systems ("recovery probability"). Containment fragility. Establishing probability distribution based on the results of deterministic analyses or some other concepts.

- **Special Topics 2: Quantification of Risk from External Hazards with Focus on Seismic Events**. Characterization of seismic hazard for seismic PSA. Seismic hazard frequency curve. Seismic fragility curves. Convolution of seismic hazard with seismic fragility. Quantitative screening of SSCs based on low seismic risk. Seismically-induced failures, random failures and their combinations. Seismic risk quantification.

Minimum number of Participants: 7

Lecturers:

I. Vrbanic (APOSS)

MMARS 2017

Models and Methods for Advanced Reactor Safety Analysis

Lucca, Italy

13 – 17 November 2017

REGISTRATION FORM

To be returned by 3 November 2017

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)

In case you need a laptop, please inform the organizer. No laptop is provided for the hands-on training ..…………..….. ☐

Registration Fees

Include proceedings, lunches and coffee breaks:

	Payment by Sept 22	Payment after Sept 22
Advanced RELAP5 Training …………………………….………………………..	☐ **€ 2500**	☐ **€ 2700**
Methods and Codes for Cross Section Generations and 3D NK……………………	☐ **€ 2500**	☐ **€ 2700**
Preparation and Review of Licensing Documentation …………………………	☐ **€ 2000**	☐ **€ 2200**
Theoretical Training on Level-1 PSA for Internal Initiating Events …………………	☐ **€ 2000**	☐ **€ 2200**
Thermal-Hydraulics Core Analysis – COBRA Genesis Codes ……………………	☐ **€ 2500**	☐ **€ 2700**
Fuel Behavior Analysis by TRANSURANUS ……………………………………...	☐ **€ 2500**	☐ **€ 2700**
Severe Accident Analysis: Phenomenology and Computational Tools ……………	☐ **€ 2500**	☐ **€ 2700**
Important Elements of Risk Quantification and PSA ………………………………	☐ **€ 2000**	☐ **€ 2200**

**** Bank charges to be added to registration fees***

Payment Terms and Cancellation Policy

- ☐ No cancellation fee applies if cancellation of the registration occurs before Sept 22, 2017
- ☐ 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

The Registration Form should be sent to:

Alessandro Petruzzi : **FAX #: 0039 340 4653058** **email: alessandro.petruzzi@nnees.sk**