

# INRNE – JRC ROUND TABLE

18 NOVEMBER, 2008, SHERATON HOTEL BALKAN, SOFIA



## INRNE capabilities in the field of Nuclear Safety

Presented by

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## Abstract

The purpose of this presentation is to present some results and activities performed in the Institute for Nuclear Research and Nuclear Energy in the field of nuclear safety research. The work is performed in the several laboratories: Laboratory for “NPP Safety Analyses”, “Nuclear Fuel Performing Analyses”, “Nuclear Fuel Modeling Laboratory“, “Reactor Physics” and etc.

The other goal of this presentation is to demonstrate the capability of the staff working in this field and to specify the possibility for future work and co-operation with EU partners in JRC projects in the field of nuclear power plant safety.

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## *NPP Safety Analyses Laboratory*

In summary the research activities in “NPP Safety Analyses” Laboratory have been defined by performance of signed contracts with DOE/USA, KNPP, FP 5, 6 and 7, PHARE, bilateral contracts with IRSN, FZK, GRS and etc. in the field of nuclear safety. Generally, performed work could be splited in several parts:

1. Analytical Code Validation and Validation of input models for VVER 440 and VVER 1000 for RELAP5, ASTEC, MELCOR.
2. Analytical validation of Emergency Operating Procedures (EOPs) for Units 3&4 of KNPP (VVER 440) and Units 5&6 (VVER 1000) of KNPP.
3. Severe accident analyses with ASTEC, MELCOR and other computer codes. Investigation of phenomena at different EU facilities.

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## Analytical Code Validation and Validation of Inputs

In field of analytical code validation the use of plant data in the code assessment process is a valuable addition to the validation process. The overall objectives are to assess computer codes used in the safety analysis of VVER power plants from one side and to validate developed models for VVER reactors against plant data from the other side.

In this connection, the following steps have been performed.

It has been developed a Quality Assurance (QA) program for developing and using thermal hydraulic models.

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It has been created a database for thermal hydraulic modeling of VVER 440 and VVER 1000 reactors working at KNPP.

In the Laboratory has been developed a thermal hydraulic model for RELAP5 Mod 3.2 of KNPP Units 5 and 6 with VVER-1000 reactors.

A thermal hydraulic model has been validated based on transients and abnormal events, provided from KNPP Units 5&6 (VVER 1000) such as: trip off/on MCP, loss of feed water, switching off all MCPs at 5%, loss of in-house supply power etc.

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In the Laboratory has also been developed a thermal hydraulic model for KNPP Units 3 & 4 with VVER-440 reactors.

The Thermal hydraulic model has been validated based on transients and abnormal events, provided from KNPP Units 3&4, VVER–440 such as trip off one and more MCPs, Drop of Control Rod Assembly (CRA) down to the fully inserted position, Shutdown of turbine generator #7 at Kozloduy NPP Unit #4, Actuation of ARM etc.

The specialists of the Laboratory performed an independent analysis of VVER 440 Rivne SG header rupture.

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## Analytical Validation of EOPs

In the field of analytical validation of Emergency Operating Procedures (EOPs) more than 150 thermal-hydraulic analyses for Kozloduy NPP, Unit 5&6 (VVER–1000) and Units 3&4 (VVER–440) have been performed using RELAP5/MOD3.2 computer code.

The investigated scenarios could be split in the following groups: LOCA analyses (Small, Medium and Large Break); Blackout analyses; Steam Generator Collector Rupture (Primary to Secondary large leak analyses); Total loss of feed water; Steam Line Break; Single SG pipe break (Primary to Secondary small leak analyses); ATWS transients.

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Symptomatic Emergency Operating Instruction (EOI) analyses have been performed using a realistic, best-estimated approach so as not to distort the depiction of the situation confronting the operator.

Symptom-based EOPs attempt to maintain Critical Safety Functions (CSFs) within acceptable limits. These CSFs are intended to preserve the integrity of fission product barriers (FPBs).

Symptomatic EOP analyses attempt to validate the efficacy of the operator actions in maintaining the CSFs and preserving the integrity of the FPBs.

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## Severe accident analyses

With support of US Laboratory ANL it has been developed a baseline input deck for VVER 1000 for MELCOR code. The model have been used for providing of deterministic support for PSA 2 for KNPP units 5&6 as well as for prediction of hydrogen generation for old and new fuel assemblies.

A specialist from “NPP safety analyses” participated in the experiment conducted in CEA Cadarache at Plinius platform COLIMA facility (support test facility for Vulcano facility). Facility was dedicated to studying the behavior of real corium materials. The COLIMA test have been used for investigation the aerosols produced by a corium pool including non-volatile fission products in the COLIMA facility.

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The specialists also took participation in a LACOMERA program in 6-th framework program in QUENCH tests.

The purpose of QUENCH experiments are to investigate the hydrogen source term resulting from the water or steam injection into an uncovered degraded core of a Light-Water Reactor (LWR), to examine the behavior of overheated fuel elements under different flooding conditions, and to create a data base for model development and improvement of Severe Fuel Damage code packages.

The other project that the laboratory took participation is the SARNET program. The purpose of this program is developing and validating of European computer code for severe accident analyses ASTEC.

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In the laboratory have been developed a model for VVER 1000 reactor based on ASTEC integral computer code. It have been performed more than 10 analyses for ex-vessel investigation.

## The INRNE interest in co-operation with JRC

The “NPP Safety Analyses” Laboratory have interest in participation in the following topics:

- Assessment of computer codes ASTEC, CATHARE, RELAP, MELCOR which are used currently in the Lab based on plant data as well as data from European facilities. Validation of input models and so improvement of confidence in VVER and PWR safety analyses.
- Filling gaps in the OECD cross-reference Matrix for Transients.

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## Nuclear Fuel Performance Analysis

The NFPA lab was established in 1993; it is managed by Guner Passage, Svetla Stefanova and Ivatz Mandev.

Activity focused on investigations of fuel operational safety and reliability in the Kozloduy NPP Units with WWER-440 and WWER-1000 reactors.

The thermomechanical codes PINw99, TRANSURANUS and FEMAXI-6 are applied to analyse quantitatively the phenomena occurring in the fuel rod and their impact on the operation characteristics at diverse conditions.

Eight consequent research contracts with the Kozloduy NPP, summarized in 37 reports, on investigations, calculations and analyses of the WWER-440 and WWER-1000 fuel rod thermomechanical behavior for selected highest loaded and highest burnt fuel rods, to justify the operation and safety margins at steady-state, start-up and operation transient regimes

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Participation and management of the EC PHARE research project FERONIA - Fuel Modelling and Performance.

Participation in the IAEA Co-ordinated Research Programmes FUMEX-I (Fuel Modelling at Extended Burnup) (1993–1996), and FUMEX-II (2003–2005).

In 1994, 1999, 2001, 2003, 2005, 2007: organization and hosting, in co-operation with the IAEA, of the 1<sup>st</sup>, 3<sup>rd</sup>, 4<sup>th</sup>, 5<sup>th</sup>, 6<sup>th</sup> and 7<sup>th</sup> International Seminars / Conferences on WWER Fuel Performance, Modelling and Experimental Support; the 8th Conference, 30.08.–05.09.2009, is under preparation.

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## Nuclear Fuel Modelling Laboratory

The INRNE Laboratory of Nuclear Fuel Modelling worked on studying fuel behaviour, especially in the field of:

- fuel behaviour modelling as a part of reactor material research;
- fuel performance modelling as a component of fuel design;
- fuel modelling as a component of commercial fuel licensing;
- fuel modelling for NPP's operational needs and decision making;
- training of research and operational NPP staff for fuel licensing by using TRANSURANUS code.

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- The TRANSURANUS code is one of the computer codes for modelling processes on operating fuel of commercial Low Water Reactors that cover different fuel subtypes and irradiation conditions. It is developed in European Commission Institute for Transuranium Elements in Karlsruhe, Germany;
- More than 10 years collaboration work on WWER version of the TRANSURANUS code was verified against Russian experimental data for steady state reactor operation;

## Achievements:

- Verification of WWER version of TRANSURANUS code;
- Prove the applicability of the TU-WWER version, to the safety analyses, performed at NPP – Kozloduy;
- Initiation of the TU-WWER at NPP-Kozloduy, including staff training and following analyses of spent fuel;

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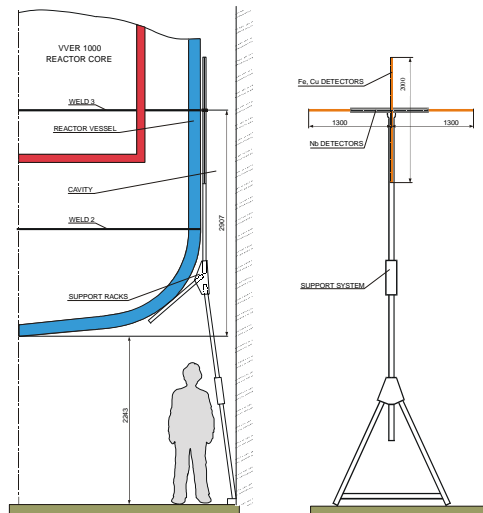
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## Reactor Physics Laboratory

### Development of Neutron Fluence Evaluation Methodology

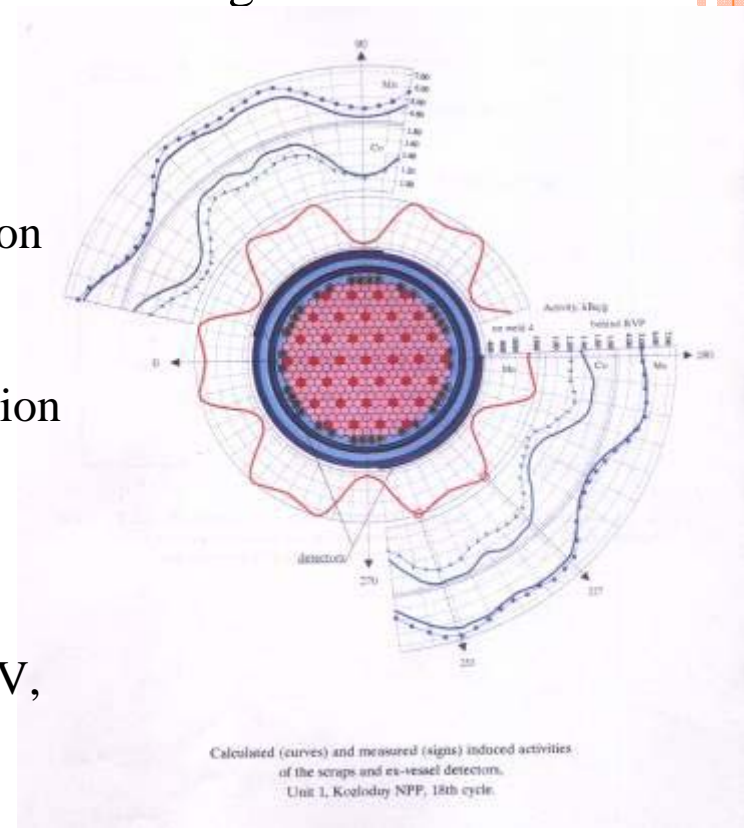
#### Validation/Verification on NPP

- Calculation/validation of neutron fluence for RPV (reactor pressure vessel) radiation damage assessment and NPP LT (lifetime) prediction are being carried out from 1992 till now.



The validation of neutron fluence is a main requirement of the QA policy for safety operation of NPP

Device for positioning of activation detectors behind RPV, used at Kozloduy NPP for neutron fluence verification





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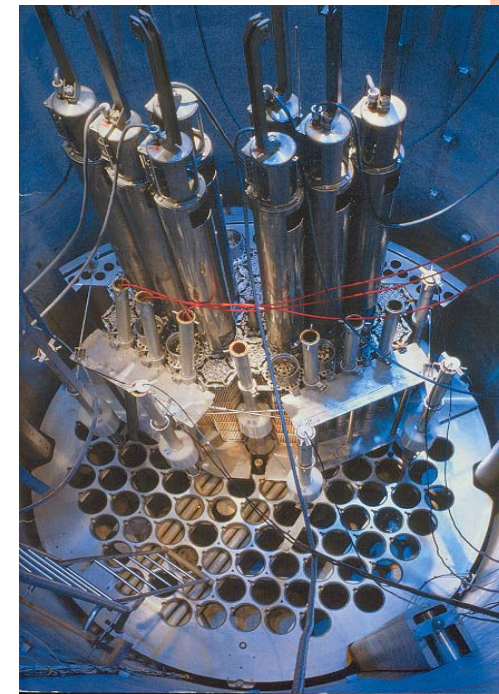
## **Development of Neutron Fluence Evaluation Methodology** Validation/Verification on Benchmarks

**WWER-1000 and WWER-440 RPVs' benchmarks** developed at Czech LR-0 reactor, Rez, near Prague as a base for fluence calculation verification.

### **The LR-0 Reactor**

Within common projects REDOS of FP-5, RADE of FP-6 of EC, RER/4/017 of IAEA of countries operating WWER reactors, Russia, Ukraine, Czech Republic, Finland, Hungary, Slovakia and Bulgaria, together with specialists from Western European countries, France, Spain, Germany, Belgium, the Netherlands, and UK, operating PWR and BWR type reactors **together JRC-IE**

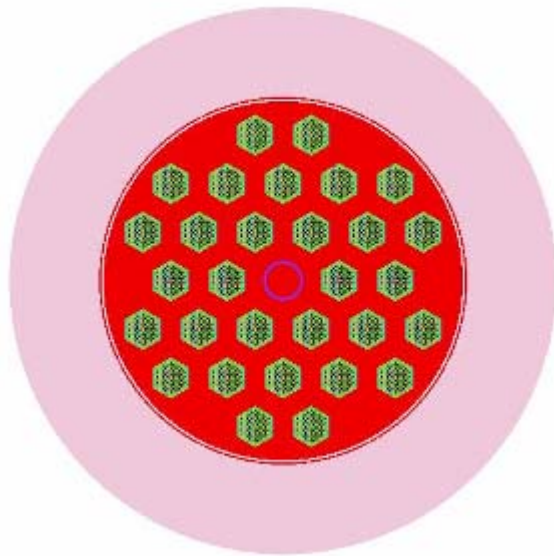
Participation in the *Nuclear Knowledge Preservation & Consolidation Workshops organized by the JRC-IE* for review of the *WWERs* radiation damage papers of the nuclear database ODIN.JRC.NL collected papers



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## CRITICALITY SAFETY ANALYSES OF SPENT FUEL FACILITIES



***Model of transport cask for 30 WWER-440 SFAs***

***Safety analyses of SNF facilities are carried out for NPP Kozloduy and BNRA.***

A methodology is developed for criticality safety analyses with burnup credit implementation, especially for the VVER spent fuel management facilities

Methodology is based on the SCALE and NESSEL-NUKO code systems and it is in process of extensive validation for VVER applications.

### **Problems:**

Isotope inventory  
Spent fuel casks Storage facility design

***FP6 Project COVERS***