Sensitivity Analysis and Uncertainty Propagation from Basic Nuclear Data to Reactor Physics and Safety Relevant Parameters

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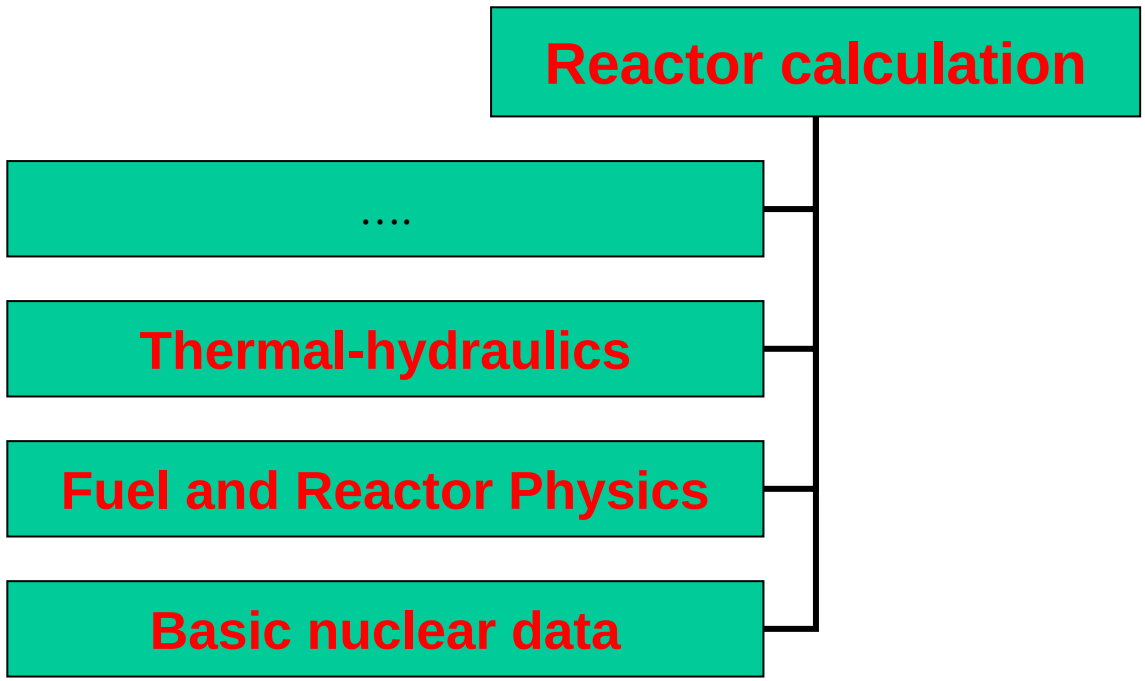
Calculational overlay



Particle transport methods

- ***Monte Carlo (→ continuus energy or multigroup XS):*** MCNP, KENO, McBEND, TRIPOLI, MORSE, EGS4, PENELOPE, MONK, ITS, FLUKA, LAHET
- ***Deterministic transport or diffusion codes (→ multigroup XS):*** ANISN, DOORS, DANTSYS, PARTISN, TWOTRAN, CEPXS/ONELD, WIMS, APOLLO, CASMO

Deterministic vs. Monte Carlo Methods

DETERMINISTIC (e.g. SN):

- **Discretisation of independent variables, i.e. space, energy, direction;**
- **Relatively low CPU requirements**
- **Suitable for sensitivity/uncertainty analysis**
- **Multigroup nuclear XS data: self-shielding, weighting spectra.**

MONTE CARLO:

- **Arbitrary geometry;**
- **Continuous energy cross section description**
- **Longer computer times**
- **Statistical uncertainty**

Sources of Uncertainty in Reactor Calculations

- **Mathematical methods and simplifications:** e.g. M/C statistics, SN space/energy/angular discretization, anisotropic scattering order, convergence criteria, diffusion equation
- **Nuclear data uncertainties**: nuclear cross-sections, fission spectra, standards
- **Radiation source description** (space, energy distribution)
- **Geometry modelling, material compositions, dimensions, conditions**
- **“Human factor”**

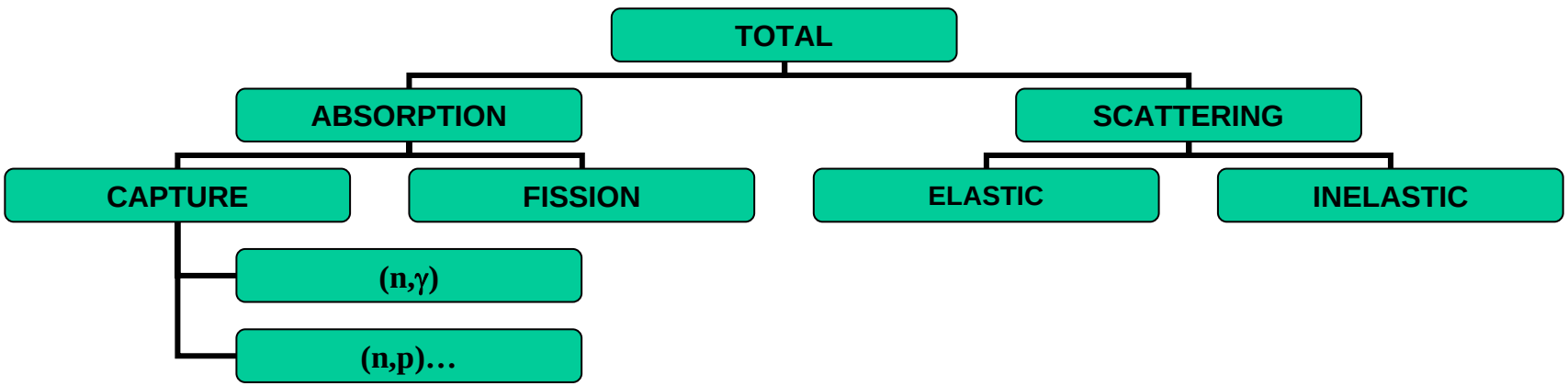
NEA-DB activities related to sensitivity/uncertainty analysis

- ***Cross section covariance matrix libraries***
- ***Codes for cross section sensitivity and uncertainty analysis***
- ***Reactor pressure vessel surveillance project***, VENUS-1 and VENUS-3 benchmark interlaboratory comparison
- ***Sensitivity and uncertainty analysis of criticality benchmark*** experiments - IRPhE project
- ***Databases of internationally verified benchmarks*** (SINBAD, ICSBEP)
- ***Fusion benchmark analysis*** (EFF project)

Cross Section Evaluation

Nuclear models:
multi-particle interactions, nuclear forces (optical potential or other approximate models); model input parameters are deduced by the comparison with the experimental data.

Measurements:
least square fitting of measured data sets using Bayesian analysis

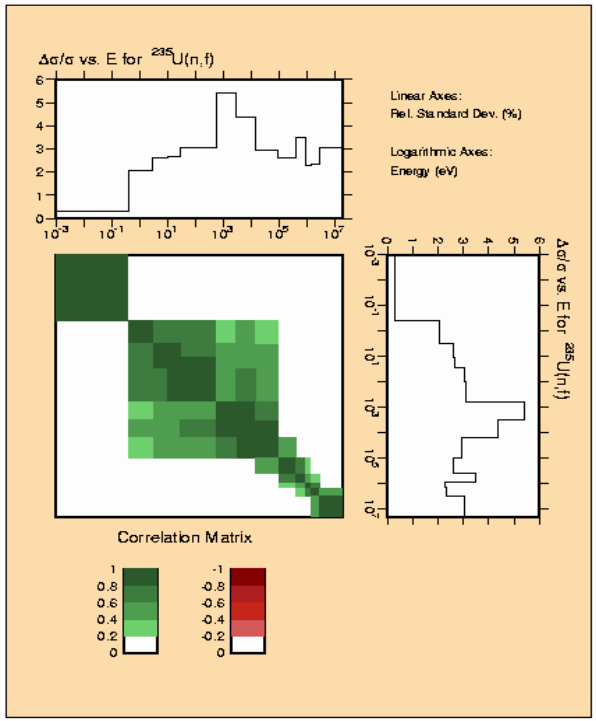


Cross Section Covariance Matrix Evaluation

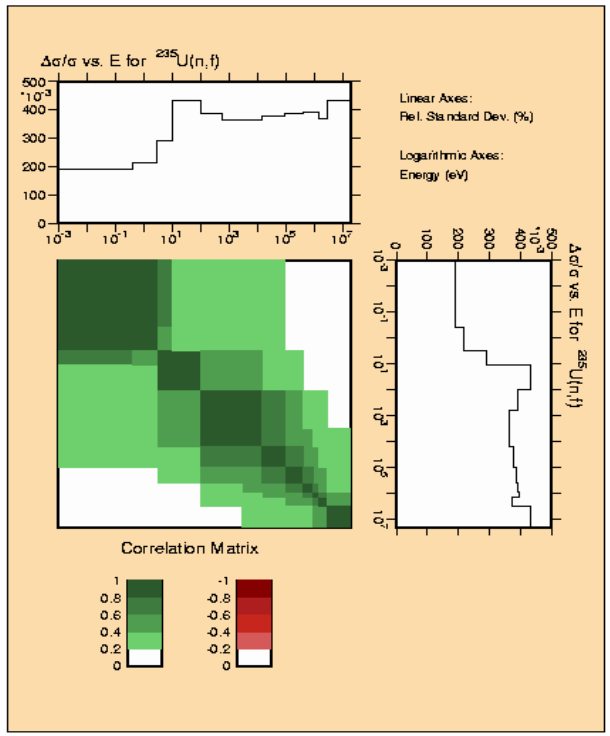
Nuclear models:
model approximations and deficiencies; expressed in terms of covariances of input parameters and sensitivities (uncertainty propagation law), uncertainty in input parameters is deduced by the comparison with the experimental data using Bayesian analysis.

Measurements:
cross section error consists of statistical uncertainty (representing scatter among data) and systematic error: instrument resolution, personal reading bias, inexact values for standards, constants; incomplete knowledge of measurement conditions, geometry and composition approximations (dosimeter positioning); unphysical adjustment errors

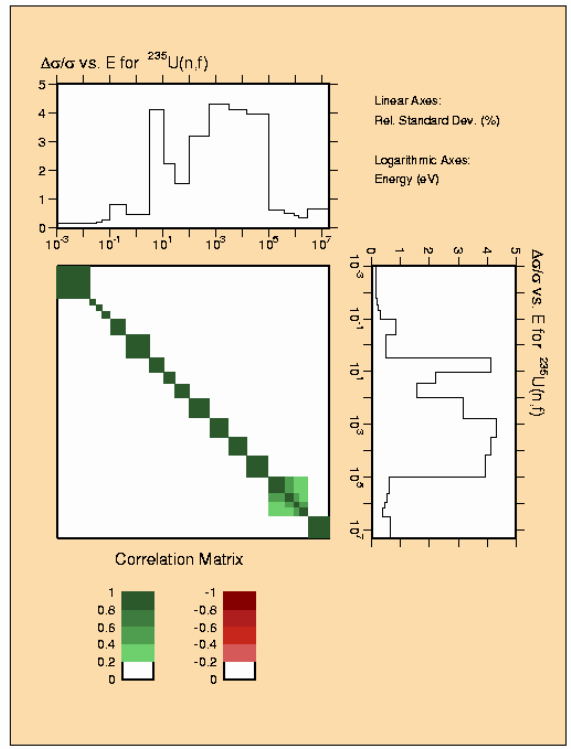
U-235(n,f)



JEF-2.2 (ENDF/B-V)

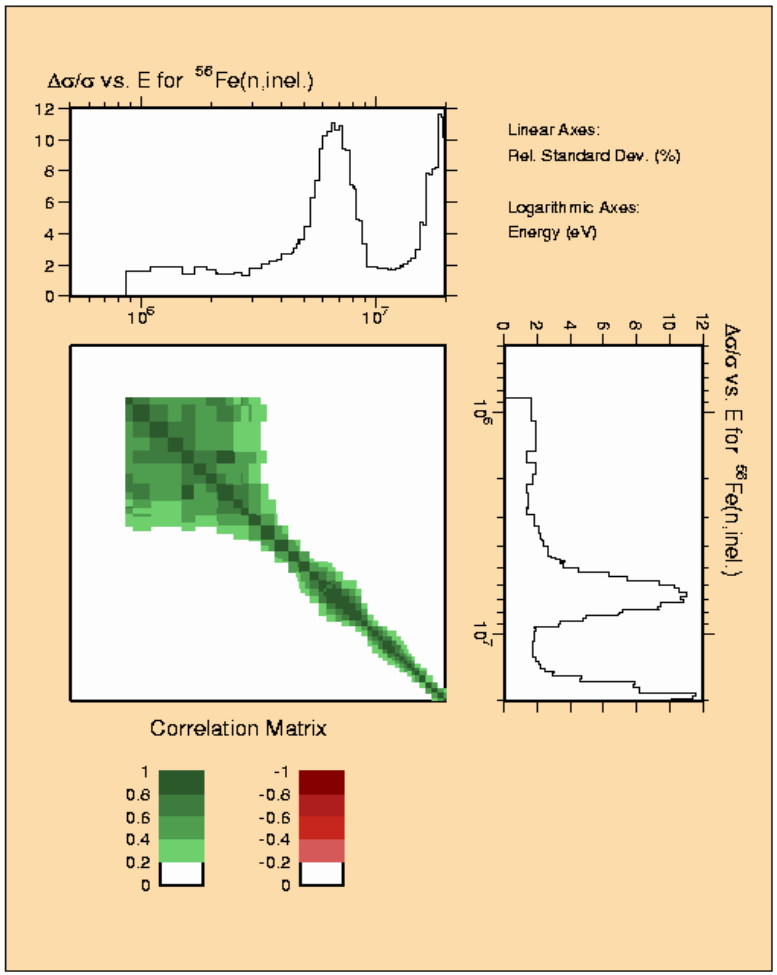


IRDF-90

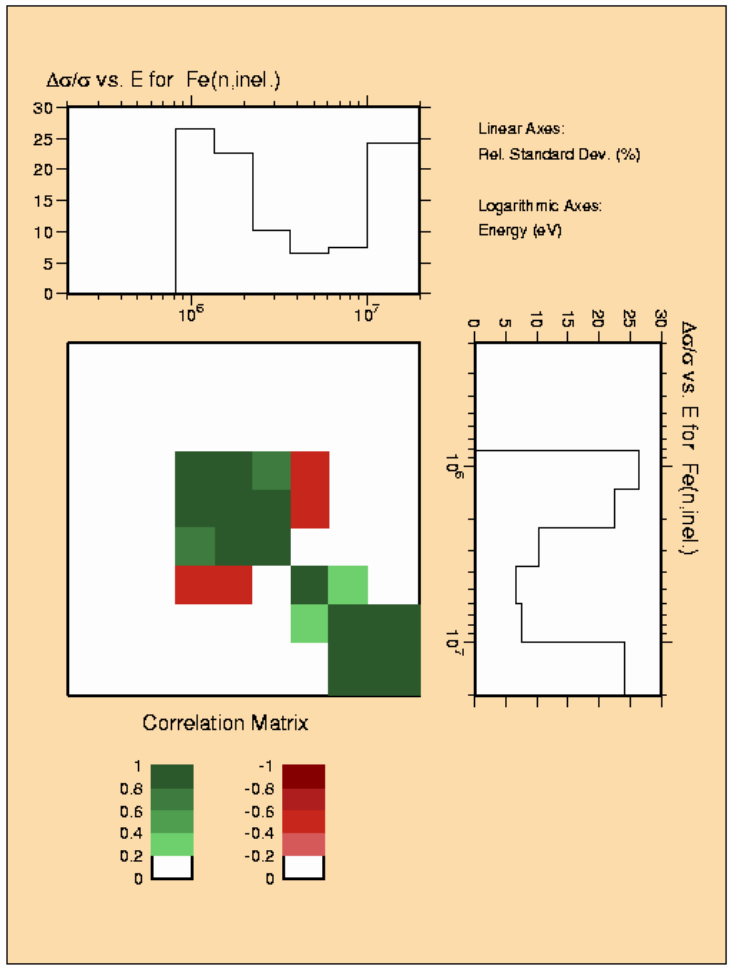


JENDL-3.2

Fe-56(n,inel)



EFF-3.1



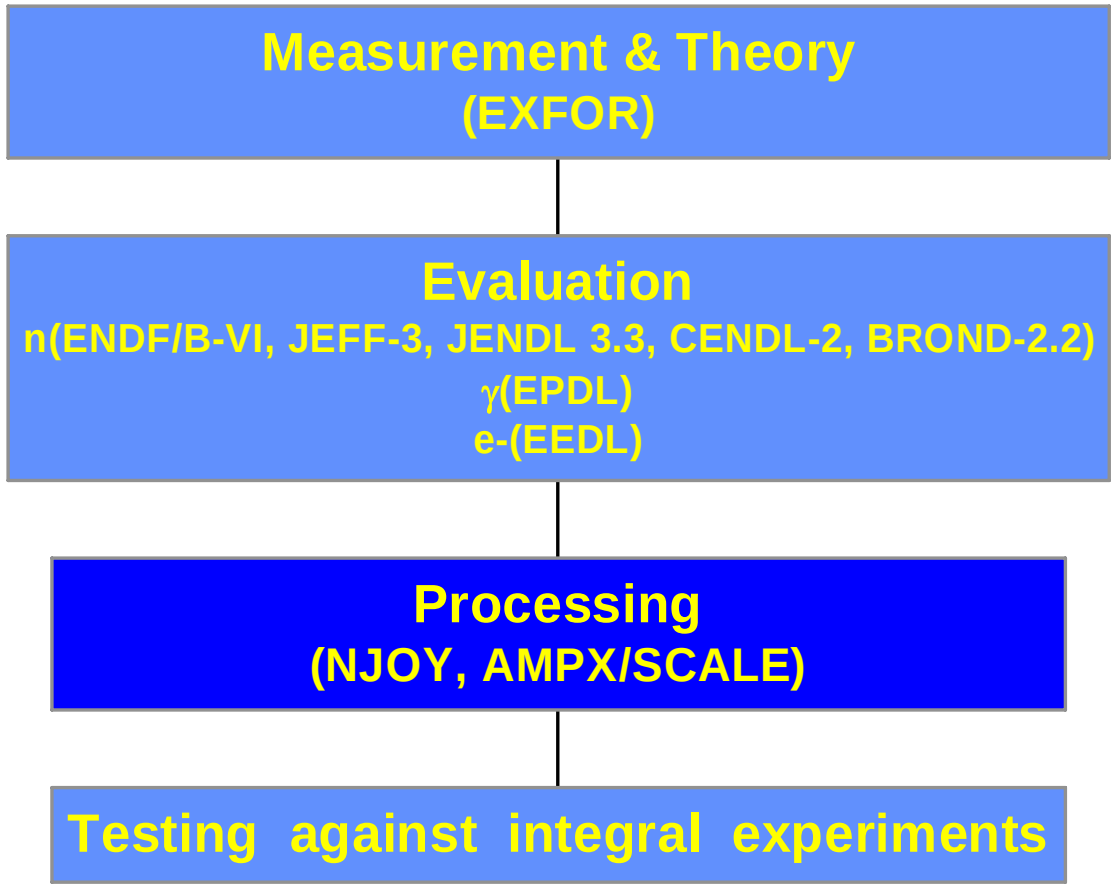
JENDL-3.2

Processed Multigroup Covariance Data Libraries

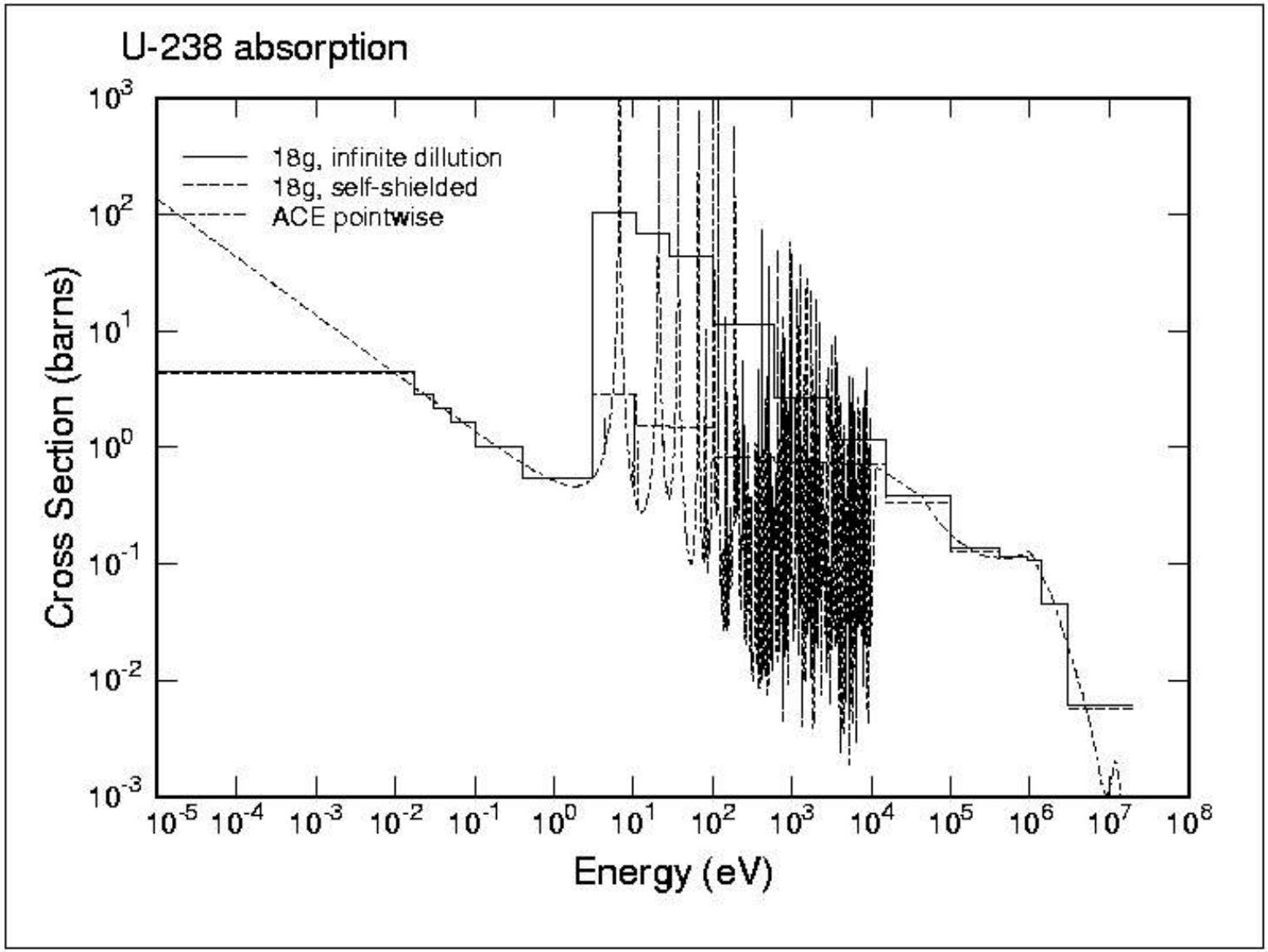
- ***ZZ-COVFILS***: 30-Group Neutron Cross-Section Covariance Library from **ENDF/B-V** (in BOXER format)
- ***ZZ-COVFILS-2***: 74-Group Covariances for Fusion Reactors (**ENDF/B-V**)
- ***PUFF-2***: Multigroup Covariances from **ENDF/B-V** & processing code (COVERX for.)
- ***ZZ-DOSCOV***: 24-Group Covariance Library from **ENDF/B-V** for Dosimetry Calcul.
- ***ZZ-COVERV***: Multigroup Cross-Section Covariance Matrices from **ENDF/B-V**
- ***ZZ-VITAMIN-J/COVA***: Covariance Matrix Library based on **JEF-1, ENDF/B-IV** & **-V** data; processing & verification codes
- ***ZZ-VITAMIN-J/COVA/EFF2***: **EFF-2.3** covariance matrices for 18 materials, detector response function covariances from **IRDF 90.2**
- ***ZZ-VITAMIN-J/COVA/EFF3***: **EFF-3** covariance matrices for Be-9, Si-28, Fe-56, Ni-58, Ni-60; processing & verification utilities
- ***ERRORJ***: processing code & **JENDL-3.2** covariance matrices
- ***ZZ-COV-15GROUP-2005:*** overwiev of the available covariance data (under preparation)

Cross-section Generation

- **Basic nuclear data**
- **Evaluated data libraries**
- **Application libraries**
  - **pointwise data**
  - **multigroup data (fine- / few-group)**



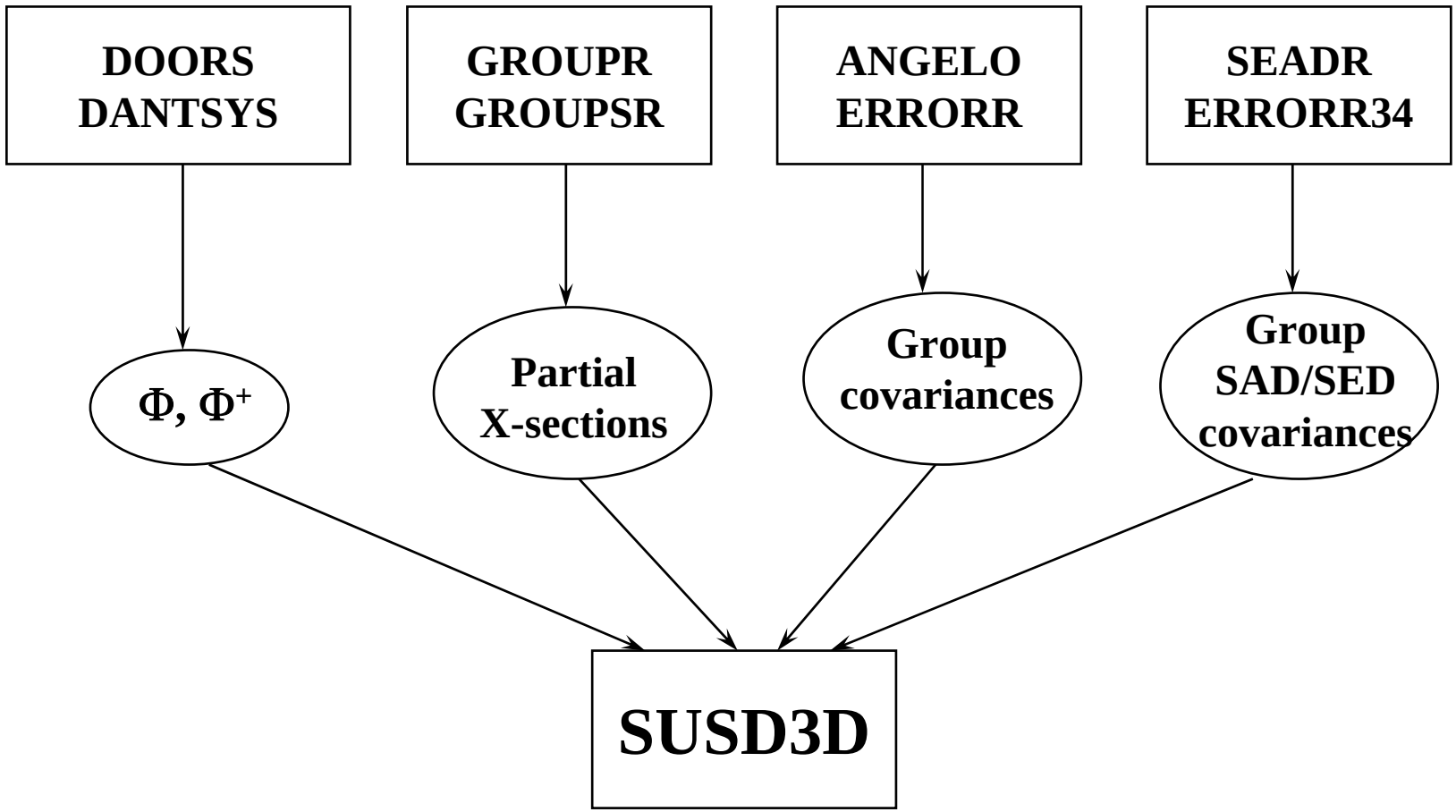
U-238 absorption cross-sections in point-wise and 18 group structure



Cross Section Sensitivity Analysis

- *Several independent calculations (brute force) - unpractical*
- ***Perturbation method based on forward and adjoint flux***: first order perturbations (deterministic & M/C methods):
  - **SN: SWANLAKE** (1D), **SENSIT&SUSD**(1D, 2D, SED/SAD)
  - **McBEND** (**M/C** 1st order perturbations)
  - **SUSD3D** (1D, 2D, 3D SN uncertainty including SED/SAD);
  - **TSUNAMI** (SCALE-5): 1D SN, 3D M/C (KENO5)
- ***Monte Carlo methods:*** (correlated sampling, first and second order perturbations)
  - **MCNP4C** differential operator perturbation method (material density, composition, cross sections)

Sensitivity / uncertainty code system



Examples of the Use of Sensitivity/Uncertainties Analysis

- ***Reactor pressure vessel surveillance:*** uncertainty in predicted dosimeter reaction rates and PV exposition, determination of safety margins --> reactor lifetime predictions
- ***New project design studies or improved design*****:** design and safety margins: parameter studies for fusion shielding blanket (tritium breeding ratio, heating, dose rates), ADS
- ***Pre- and post-analysis of benchmark experiments*****:** optimisation of experimental configuration, explain eventual discrepancies, representativity studies, data consistency: fusion benchmarks (FNG), PV benchmarks (ASPIS, VENUS), Criticality benchmarks **(**VENUS-2, KRITZ, SNEAK)
- ***Criticality safety***
- ***Nuclear data evaluations***

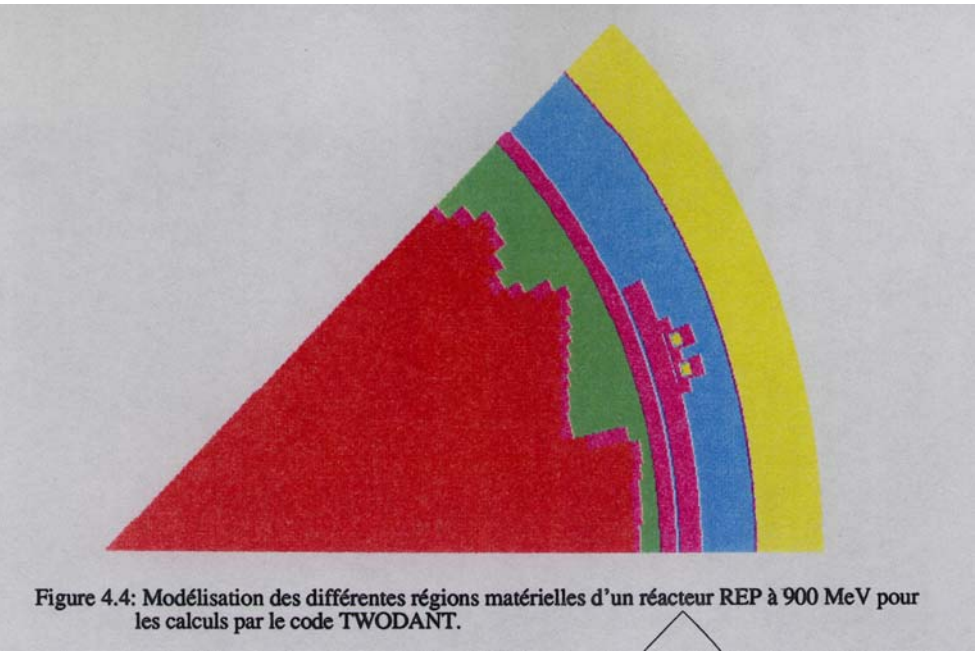


Figure 4.4: Modélisation des différentes régions matérielles d'un réacteur REP à 900 MeV pour les calculs par le code TWODANT.

Pressure vessel surveillance dosimetry

OECD/NEA NSC Task Force on Computing Radiation Dose and Modeling of Radiation-induced Degradation of Reactor Components

Insufficient information about the accuracy of the neutron fluence of the neutron field and spectrum (and therefore of the radiation damage) would require large safety margins, and consequently affect the operating conditions, the life of the nuclear installations, and their cost.

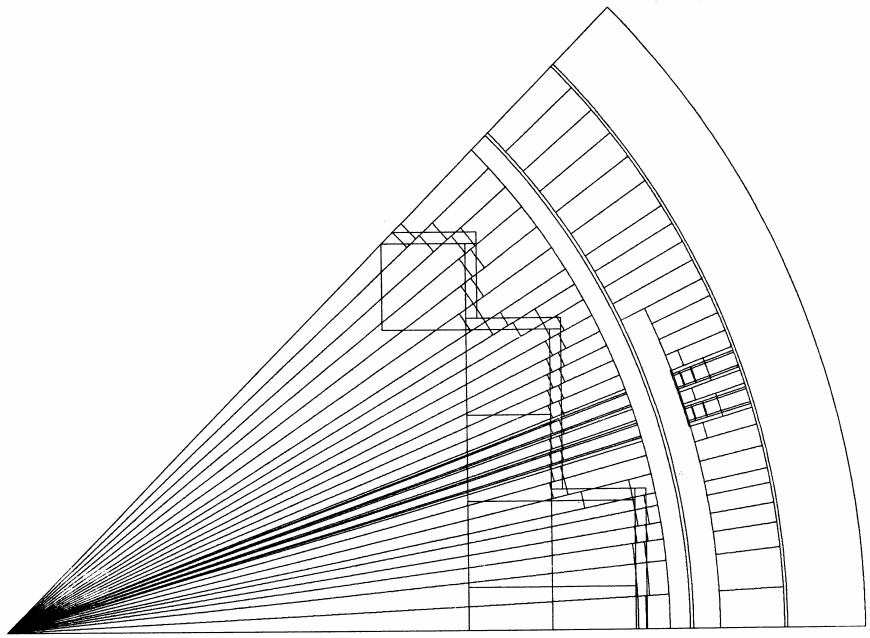
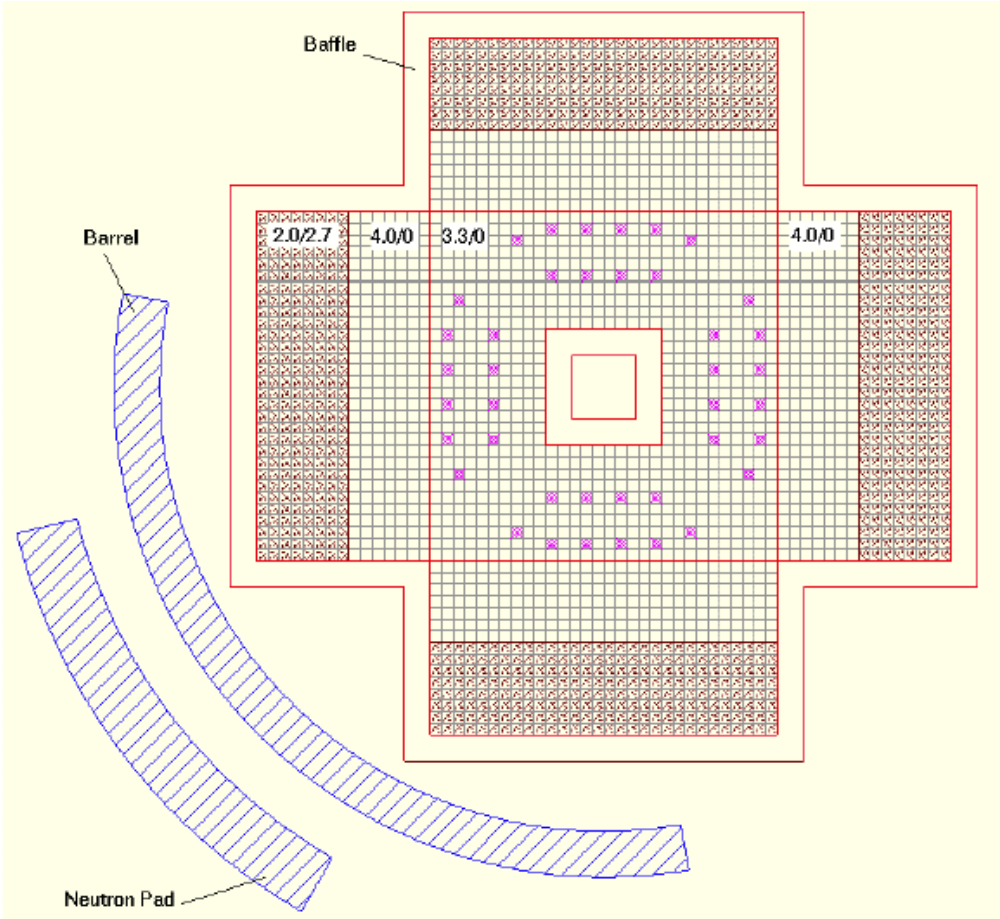


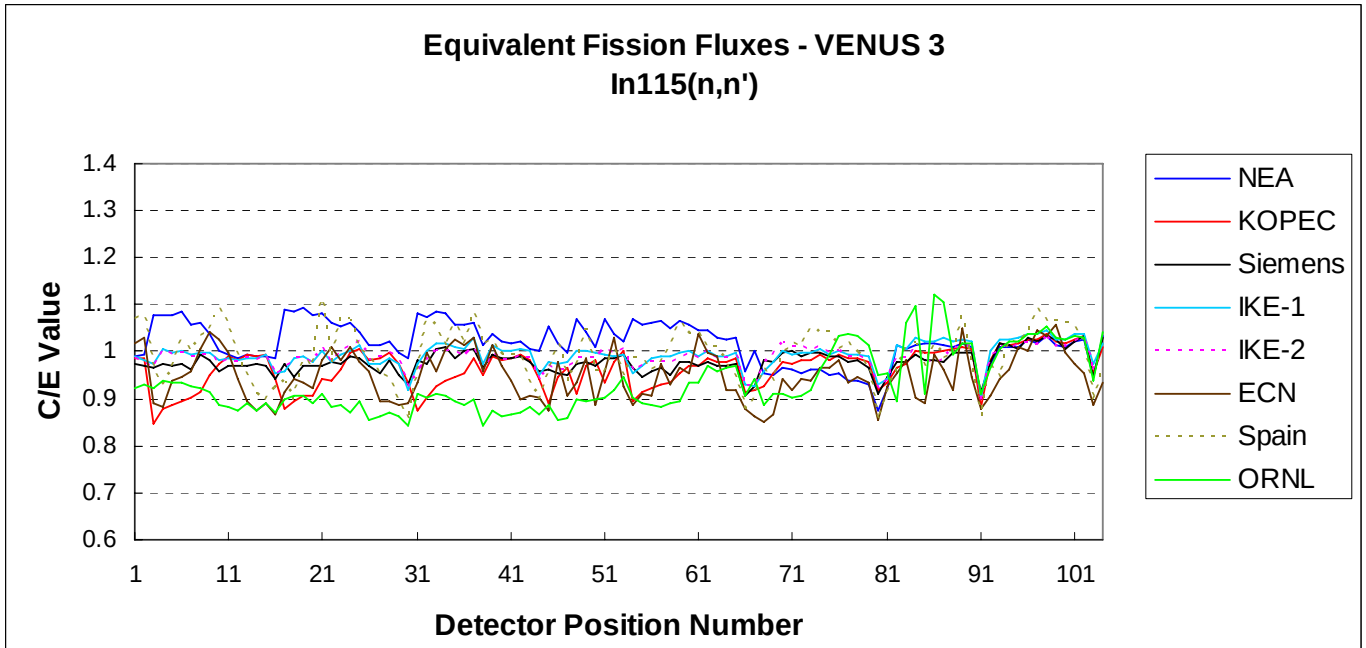
Figure 4.5: Modélisation du maillage utilisé dans les calculs d'un réacteur CPY par TWODANT. En particulier la description des frontières entres le coeur, le cloisonnement du coeur et l'eau, ainsi que l'environnement des capsules est présentée.

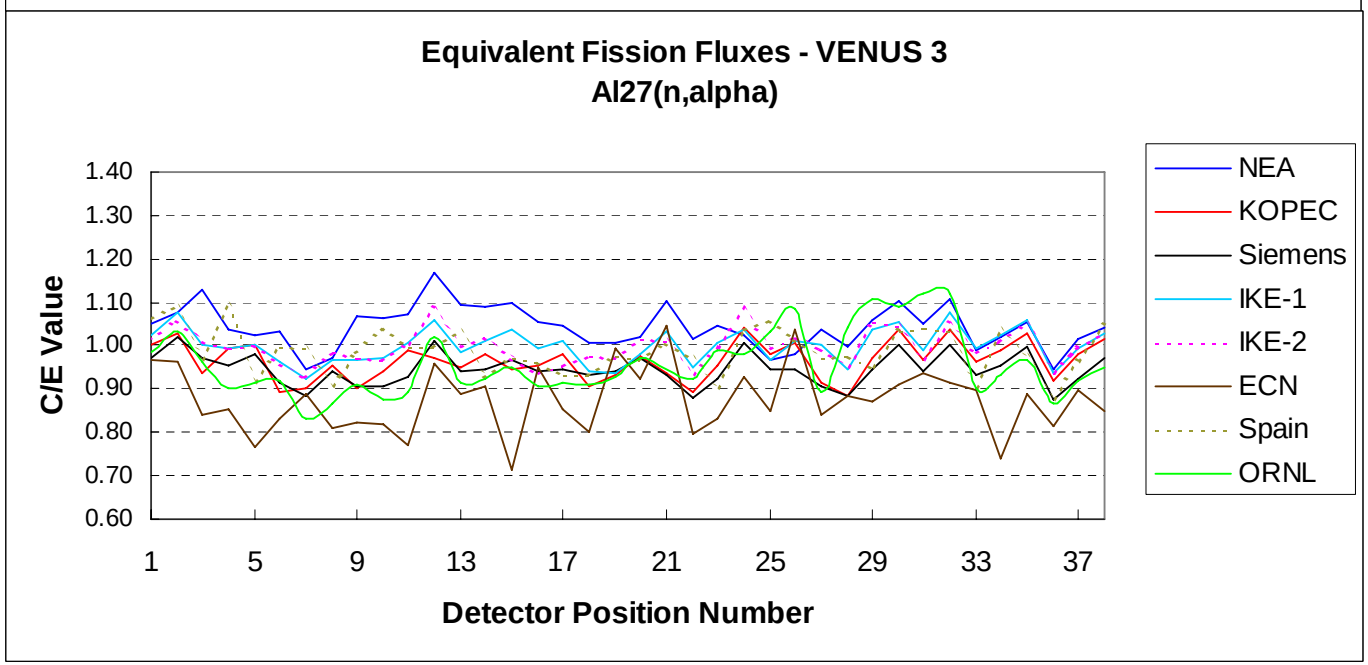
Venus-2 Configuration (B.C. Na)

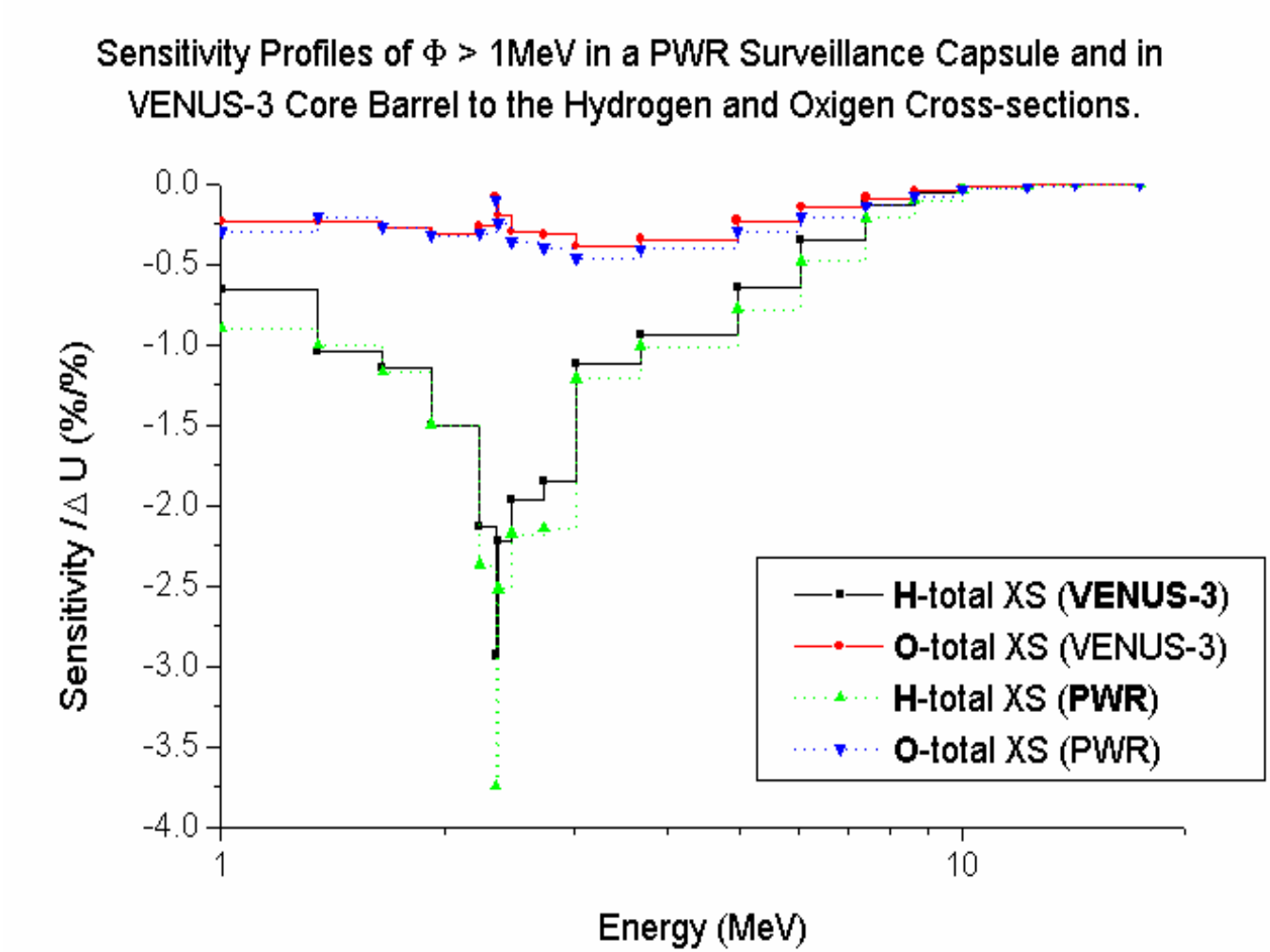


VENUS-3 UNCERTAINTIES

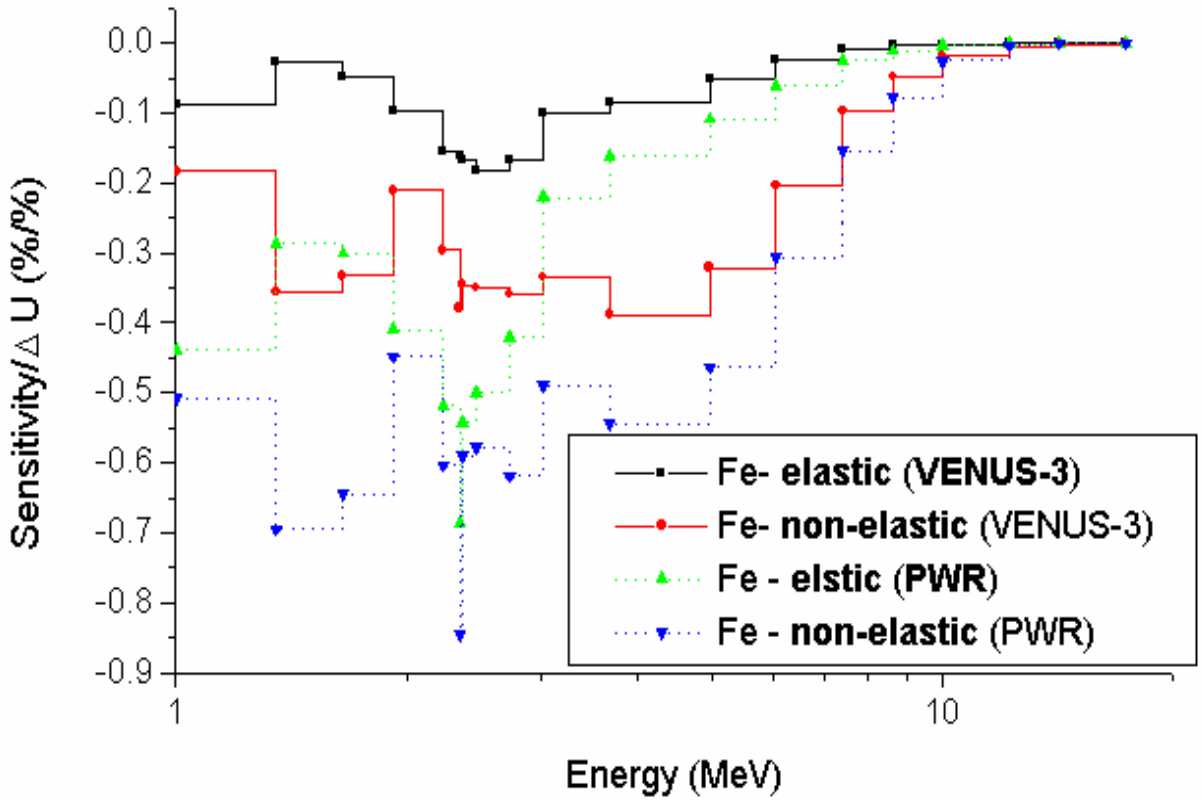
Source of Uncertainty		Uncertainty (%)			
		$\Phi > 1$ MeV	$^{27}Al(n,\alpha)$	$^{58}Ni(n,p)$	$^{115}In(n,n')$
Fission spectrum		4.4	12	6.5	4.5
Source space distribution		1.5 - 4			
Absolute power		4			
Response funct.		0	1.4	2.5	2.2
Cross-sections	H	1.9	1.1	1.4	1.6
	O	0.6	1.6	0.7	0.5
	Fe	2	5	2.5	2.1
Total		~7	~14	~9	~8


Equivalent Fission Fluxes - VENUS 3
In115(n,n')
C/E Value
1.4
1.3
1.2
1.1
1
0.9
0.8
0.7
0.6
1
11
21
31
41
51
61
71
81
91
101
Detector Position Number
NEA
KOPEC
Siemens
IKE-1
IKE-2
ECN
Spain
ORNL

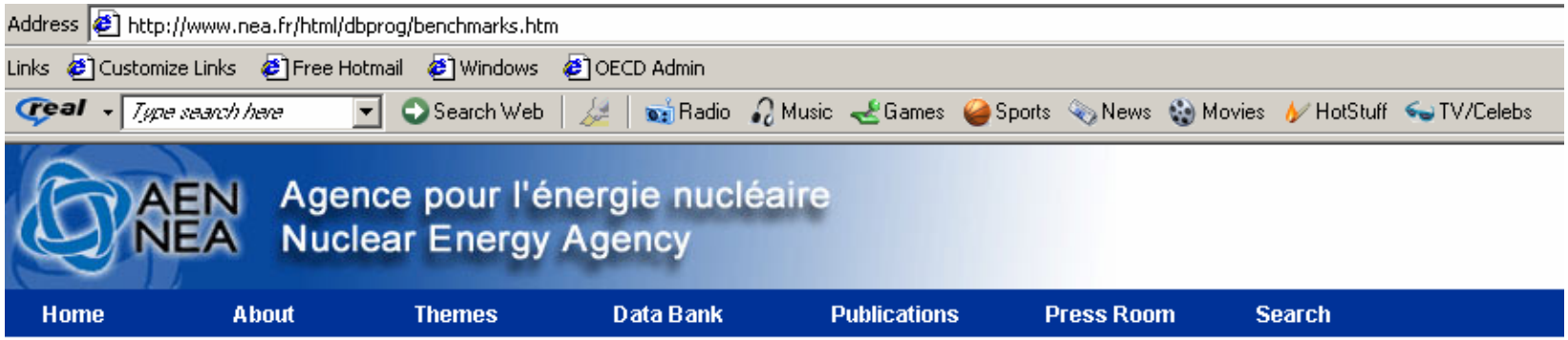

Equivalent Fission Fluxes - VENUS 3
Al27(n,alpha)
C/E Value
1.40
1.30
1.20
1.10
1.00
0.90
0.80
0.70
0.60
1
5
9
13
17
21
25
29
33
37
Detector Position Number
NEA
KOPEC
Siemens
IKE-1
IKE-2
ECN
Spain
ORNL


Sensitivity Profiles of Φ > 1MeV in a PWR Surveillance Capsule and in VENUS-3 Core Barrel to the Hydrogen and Oxigen Cross-sections.
Sensitivity / Δ U (%/%)
0.0
-0.5
-1.0
-1.5
-2.0
-2.5
-3.0
-3.5
-4.0
1
10
Energy (MeV)
H-total XS (VENUS-3)
O-total XS (VENUS-3)
H-total XS (PWR)
O-total XS (PWR)

Sensitivity Profiles of Φ > 1MeV in a PWR Surveillance Capsule and in VENUS-3 Core Barrel to the Iron Cross-sections.



/dbprog/benchmarks.html



NEA Data Bank Computer Program Services

Benchmark Experiments for Data and Code Validation

Benchmark Validation Databases (NSC):

- ICSBEP : International Criticality Safety Benchmark Evaluation Project
- IFPE: International Fuel Performance Experiments
- SINBAD : Shielding Integral Benchmark and Database (RSICC)
- IRPhE : International Reactor Physics Experiments Project

Data and reports from joint research projects related to Nuclear Safety (CSNI):

- CCVM: CSNI Code Validation Matrix Integral Test Data
- CCVM: CSNI Code Validation Matrix Separate Effects Test Data
- STRESA: CSNI Code Validation Matrix on-line (requires password)
- PSB-VVER Project: Experiments for transient analysis of VVER-1000 reactors project (available to participants)
- SETH Project: PKL and PANDA experiments relevant for accidents management (available to participants)
- IRSN Cabri Water Loop Project: High burnup fuel behaviour in RIA conditions (available to participants)
- RASPLAV Project: High burnup fuel behaviour in RIA conditions (available to OECD countries and Russia)

Evaluated Nuclear Data Processing and Visualisation:

- NJOY General Nuclear Data Processing System for Files in ENDF Format

International integral experiments databases

- SINBAD - Radiation Shielding Experiments (>70)
- ICSBEP - International Handbook of Evaluated Criticality Safety Benchmark Experiments (>300 evaluations)
- IRPhE - Reactor Physics Experiments
- IFPE - International Fuel Performance Experiments (~500 rods/samples)
- CCVM - CSNI Validation Matrix for thermal-hydraulic system codes for reactor transient and LOCA (>70 experiments)

SINBAD - an International Database for Integral Shielding Experiments - List of Experiments and Links to Abstracts

Objective: Validation and Benchmarking of Computer Codes and Nuclear Data used for Radiation Transport and Shielding Problems Preservation of a unique set of experiments for the needs of today and tomorrow.

Co-ordinators: OECD Nuclear Energy Agency (NEA) and Radiation Safety Information Computational Center (RSICC)

Contributors:

- AEA Technology (AEAT),
- Comissariat a l'Energie Atomique (CEA),
- EC Joint Research Centre (ISPRA),
- Ente per le Nuove Technologie, L'Energia e l'Ambiente (ENEA),
- Forschungszentrum Karlsruhe (FZK),
- Georgia Institute of Technology (GIT),
- Institute of Nuclear Techniques, Technical University of Budapest (TUB),
- Institute of Physics and Power Engineering,
- Interfaculty Reactor Institute (IRI), Delft University of Technology,
- Japan Atomic Energy Institute (JAERI),
- Jozef Stefan Institute (IJS),
- Los Alamos National Laboratory (LANL)
- National Institute of Standards and Technology, Gaithersburg (NIST)
- Oak Ridge National Laboratory (ORNL),
- Paul Scherrer Institute (PSI),
- Research Centre Mol (SCK-CEN)
- Technische Universitaet Dresden (TUD),
- University of Illinois,
- University of Osaka,
- University of Pavia,
- University of Tokyo,

and many experts who have contibuted to the compilation, validation and review of the data.

SINBAD Shielding Experiments

- **Reactor shielding, reactor pressure vessel dosimetry (35)**
- **Fusion Neutronics Shielding (26)**
- **Accelerator Shielding (13)**

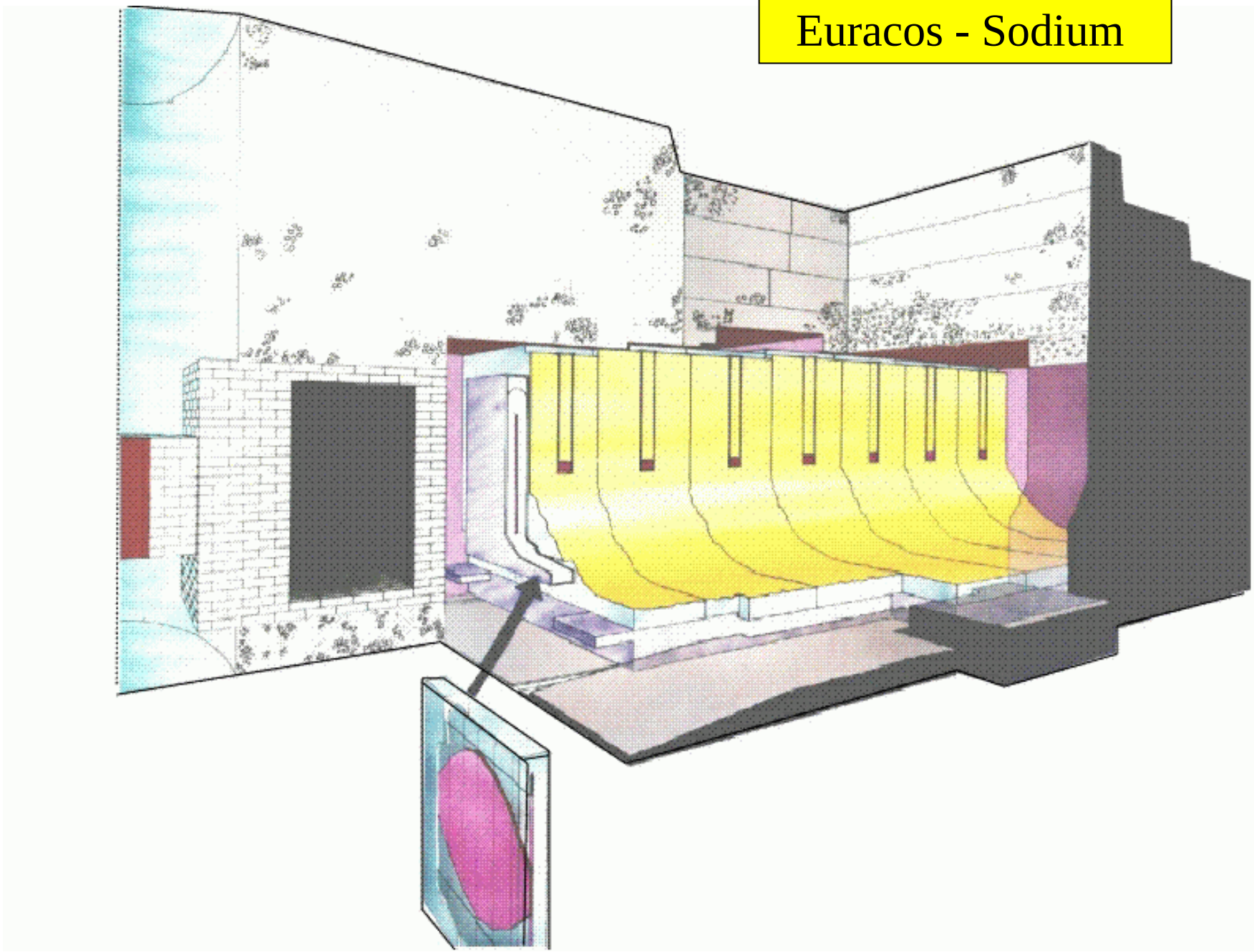
- **RSICC**: http://www-rsicc.ornl.gov/BENCHMARKS.html
- **OECD/NEA:** http://www.nea.fr/html/science/shielding/sinbad/sinbadis.htm

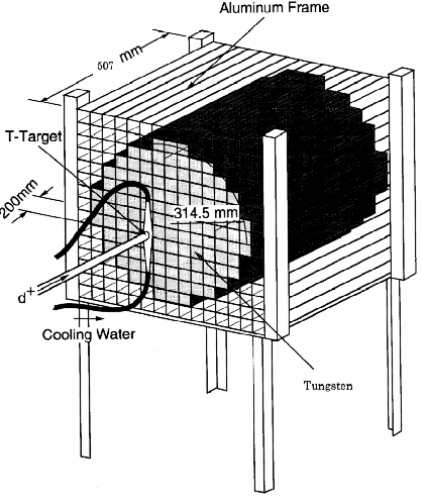
SINBAD - Radiation Shielding Experiments

Materials

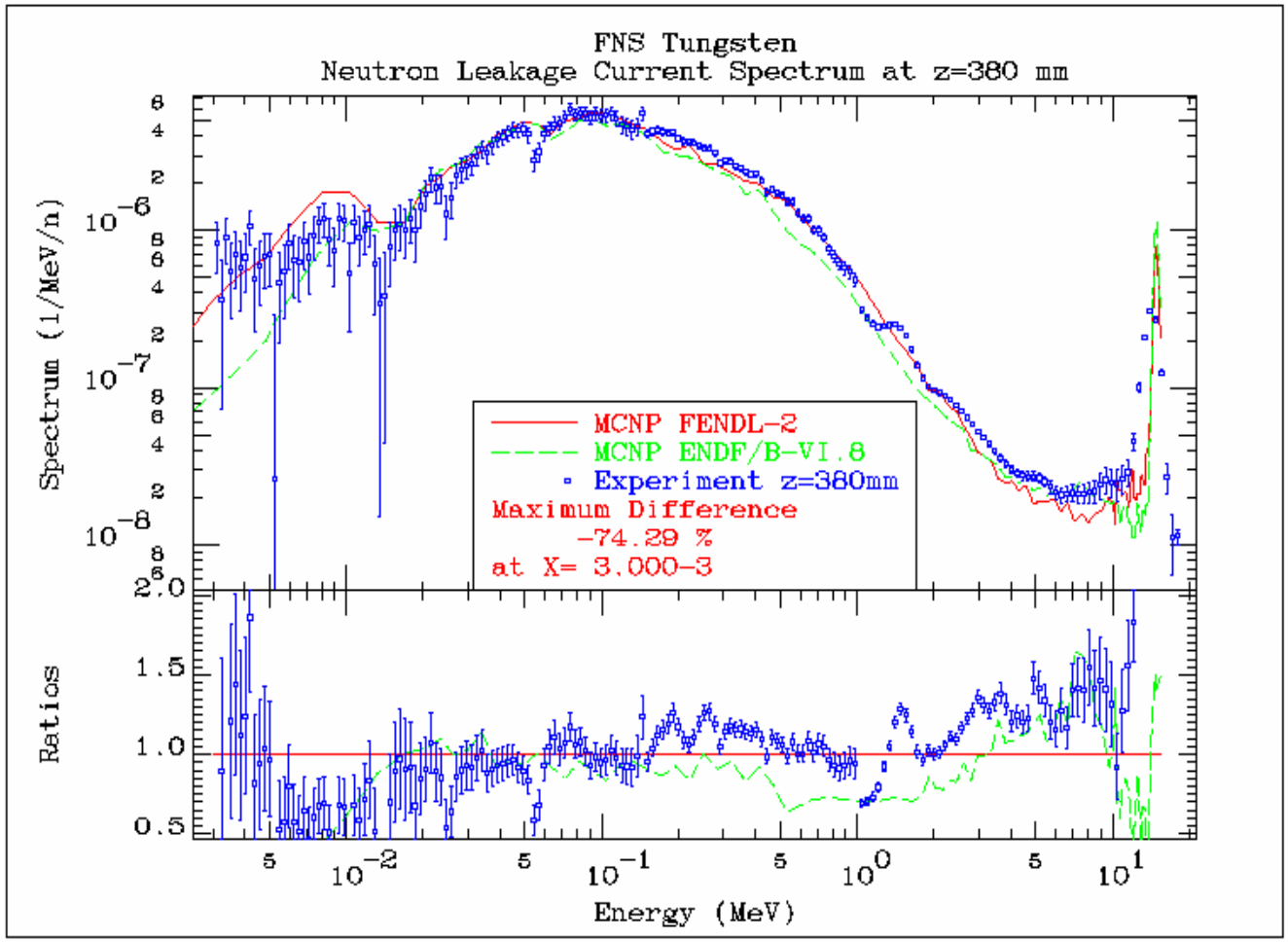
- B, Ti, H (1)
- C (graphite) (2)
- N (1)
- O (2)
- Na (4)
- H_2O (2)
- H_2O, C, Fe (1)
- H_2O, C, Pb (1)
- H_2O, Fe (2)
- H_2O, Steel (2)
- H_2O, Steel, Al (2)
- Concrete (1)
- Al (2)
- Al, Nb (1)
- Be (1)
- Fe (11)
- Fe, Pb (1)
- Fe, Concrete, $(CH_2)_{2n}$ (3)
- Ni (1)
- Steel (2)
- SS (2)
- Fe & SS (1)
- SS & $(CH_2)_{2n}$ (1)
- SS, $(CH_2)_{2n}$ & Cu (1)
- Pb (1)
- Si, SiC (2)
- V (2)
- W (3)
- Air (4)
- Multiple materials (8)

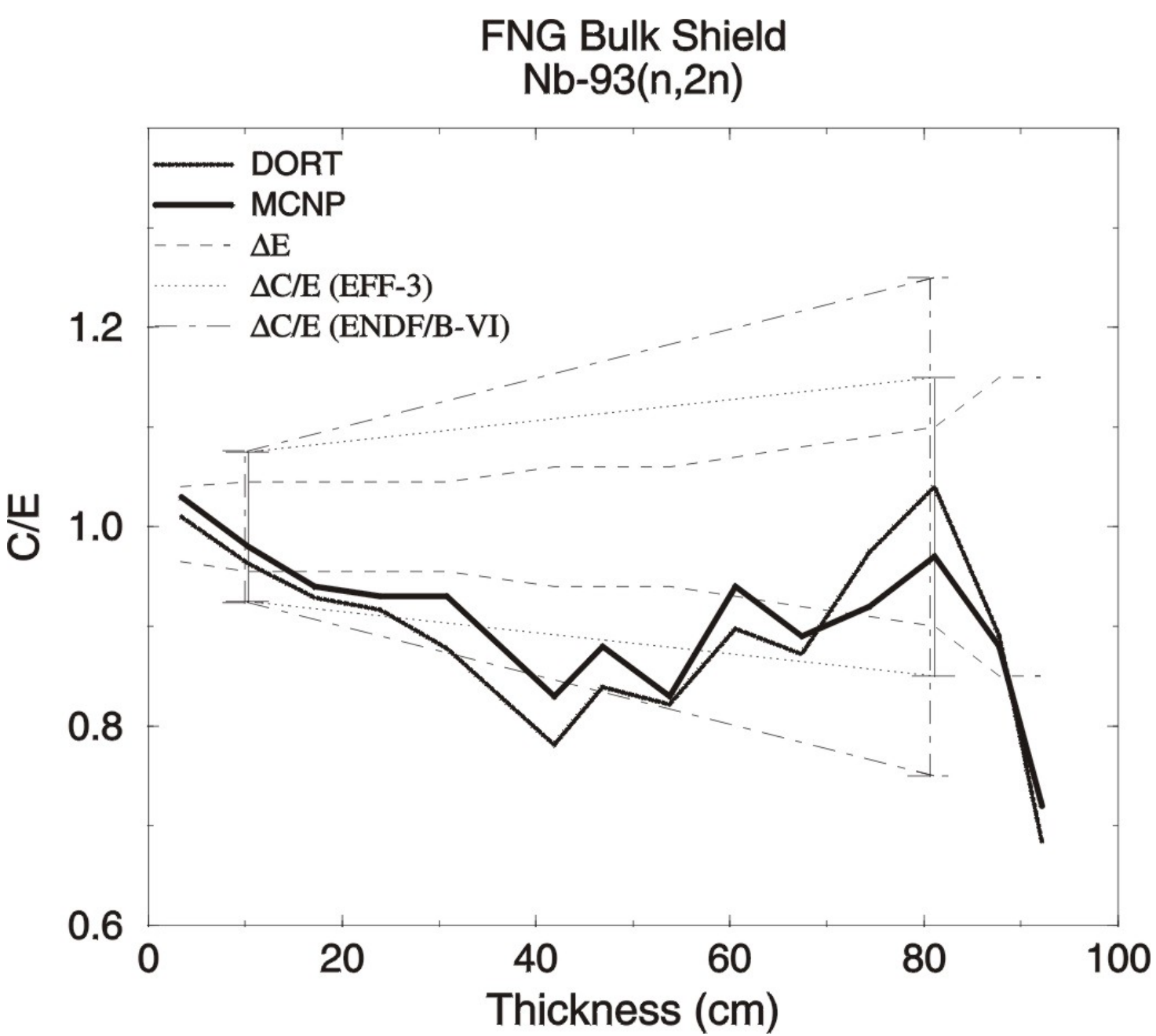
Euracos - Sodium





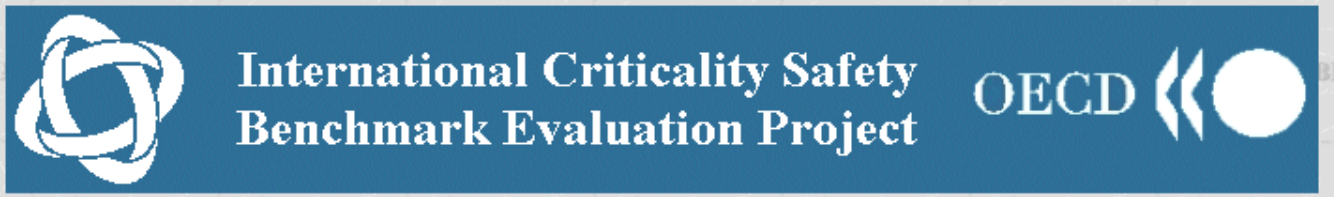
FNS Tungsten




FNG Bulk Shield
Nb-93(n,2n)
DORT
MCNP
ΔE
ΔC/E (EFF-3)
ΔC/E (ENDF/B-VI)
C/E
1.2
1.0
0.8
0.6
0
20
40
60
80
100
Thickness (cm)

Back

ICSBEP



Security/Privacy

- About the International Criticality Safety Benchmark Evaluation Project
- International Handbook of Evaluated Criticality Safety Benchmark Experiments (Request a copy of the handbook on CD-ROM)
- Database for the "International Handbook of Evaluated Criticality Safety Benchmark Experiments" (DICE)
- U.S. Department of Energy Nuclear Criticality Safety Program
- Request for Evaluation
- Evaluations in Progress

The International Criticality Safety Benchmark Evaluation Project is an official activity of the Organization for Economic Cooperation and Development - Nuclear Energy Agency (OECD-NEA). This web site is maintained at the Idaho National Engineering and Environmental Laboratory, a facility operated for the U.S. Department of Energy.

ICSBEP - International Handbook of Evaluated Criticality Safety Benchmark Experiments

- **Experiments are classified into seven different types of fissile materials**
  - Plutonium Systems
  - Highly Enriched Uranium Systems (wt.% $^{235}U \geq 60$)
  - Intermediate and Mixed Enrichment Uranium Systems ($10 <$ wt.% $^{235}U < 60$)
  - Low Enriched Uranium Systems (wt.% $^{235}U \leq 10$)
  - Uranium-233 Systems
  - Mixed Plutonium - Uranium Systems
  - Special Isotope Systems

IFPE

The Public Domain Database on Nuclear Fuel Performance Experiments for the Purpose of Code Development and Validation

International Fuel Performance Experiments (IFPE) Database

(status 2 February 2004)

The Aim of the IFPE Database Project

The aim of the project is to provide in the public domain, a comprehensive and well-qualified database on Zr clad UO_2 fuel for model development and code validation. The data encompasses both normal and off-normal operation and include prototypic commercial irradiations as well as experiments performed in Material Testing Reactors. This work is carried out in close co-operation and co-ordination between OECD/NEA, the IAEA and the IFE/OECD/Halden Reactor Project.

Activities within the IFPE Database Project

- acquisition of data through discussion and negotiation with originators
- compilation of the data into a standard form and content as agreed by an Expert Group set up for supervising the work
- peer review of the data by independent experts
- integration and indexing of the data into the IFPE database, inclusion of all used reports in electronic form.
- distribution to interested parties and assistance where necessary in use of datasets.

IRPhE

INEEL
Home of Science and Engineering Solutions

Wednesday
March 13, 2002

IRPhEP

Background Information

Purpose/Benefits

Proposed Activity

Format for Submissions

Overview of ANL Fast Critical Experiments

International Criticality Safety Benchmark Evaluation Project (ICSBEP)

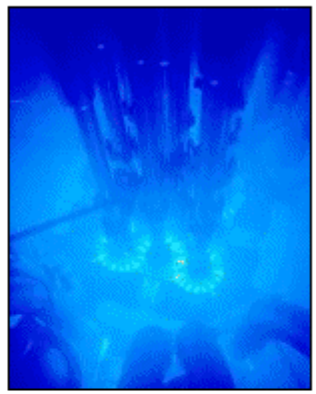
Advanced Nuclear Physics

International Reactor Physics Evaluation Project (IRPhEP)

Related links: International Criticality Safety Benchmark Evaluation Project

Background Information

Since the beginning of the Nuclear Power Industry, numerous experiments concerned with nuclear energy and technology have been performed at different research laboratories, worldwide. These measurements required a large investment in terms of infrastructure, expertise and cost; however, many were performed without a high degree of attention to permanent archival of results for future use. The results and techniques developed from these measurements remain of great value today and in the future. They provide the basis for recording, development, and validation of methods, and represent a significant collection of data for present and future research. This valuable national asset is in jeopardy of being lost. If the data are compromised, it is unlikely that any of these measurements will be repeated in the future.



INEEL's Advanced Test Reactor

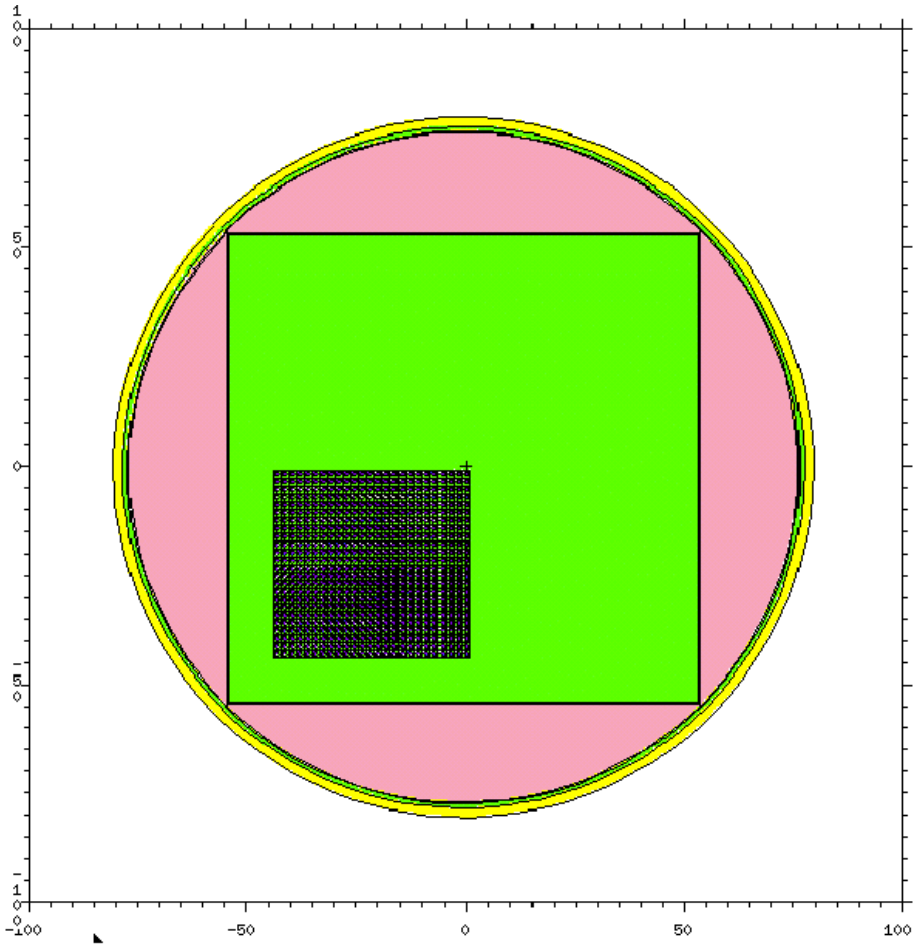
At present, there is an urgent need to preserve integral reactor physics experimental data including separate or special effects data for nuclear energy and technology applications and the knowledge and competence contained therein. The International Reactor Physics Evaluation Project (IRPhEP) was initiated by the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) in May of 2000.

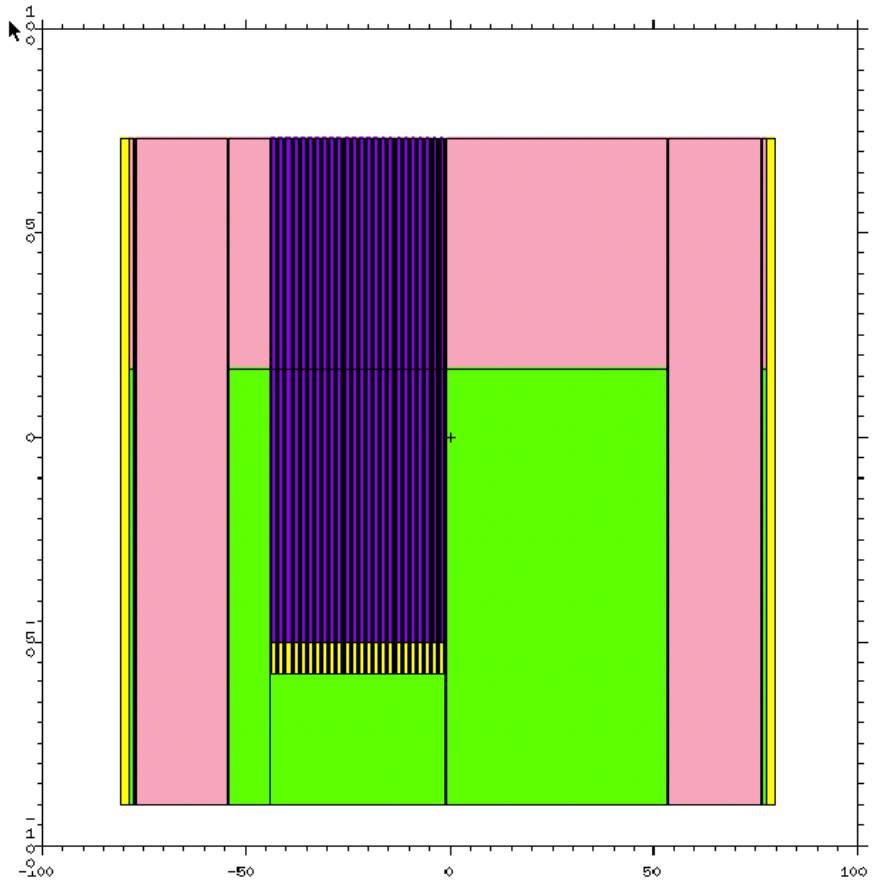
IRPhEP is a joint project between the Idaho National Engineering and Environmental Laboratory (INEEL) - Argonne National Laboratory (ANL) and is closely coordinated with the International Criticality Safety Benchmark Evaluation Project (ICSBEP).

SUMMARY OF IRPhE ACTIVITIES

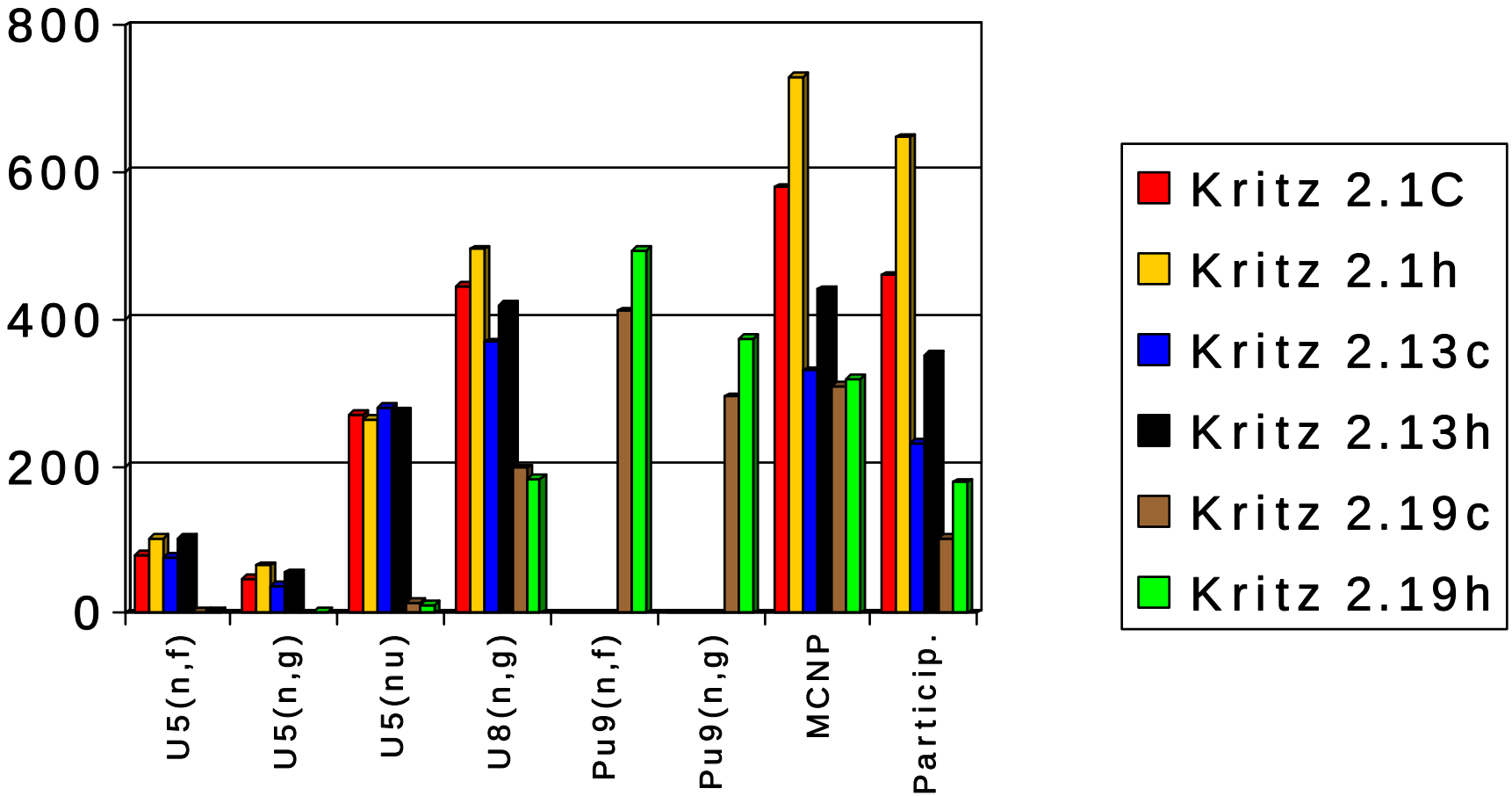
- **BFS-RESR-EXP-00**1: Critical Experiments with Pu, SiO2, Polyethylene (IPPE Obninsk)
- **BFS-RESR-EXP-00**2: Critical Experiments with Highly Enriched U, SiO2, Polyethylene (IPPE Obninsk)
- **DIMPLE-RESR-EXP-00**1: LW Low Enriched UO2 (3 wt.% 235 U) Rod Lattices Dimple S01 (Serco Assurance)
- **KRITZ-RESR-EXP-00**1: KRITZ-2:19 Experiment on Regular H2O/Fuel Pin Lattices With MOX Fuel (Studsvik)
- **PFACILITY-VVER-EXP-00**1: VVER Physics Experiments (KIAE)
- **VENUS-PWR-EXP-00**1: VENUS-2 PWR MOX Core Measurements (SCK-CEN)
- **ZR6-VVER-EXP-00**1: VVER Experiments (AEKI) (331 configurations)

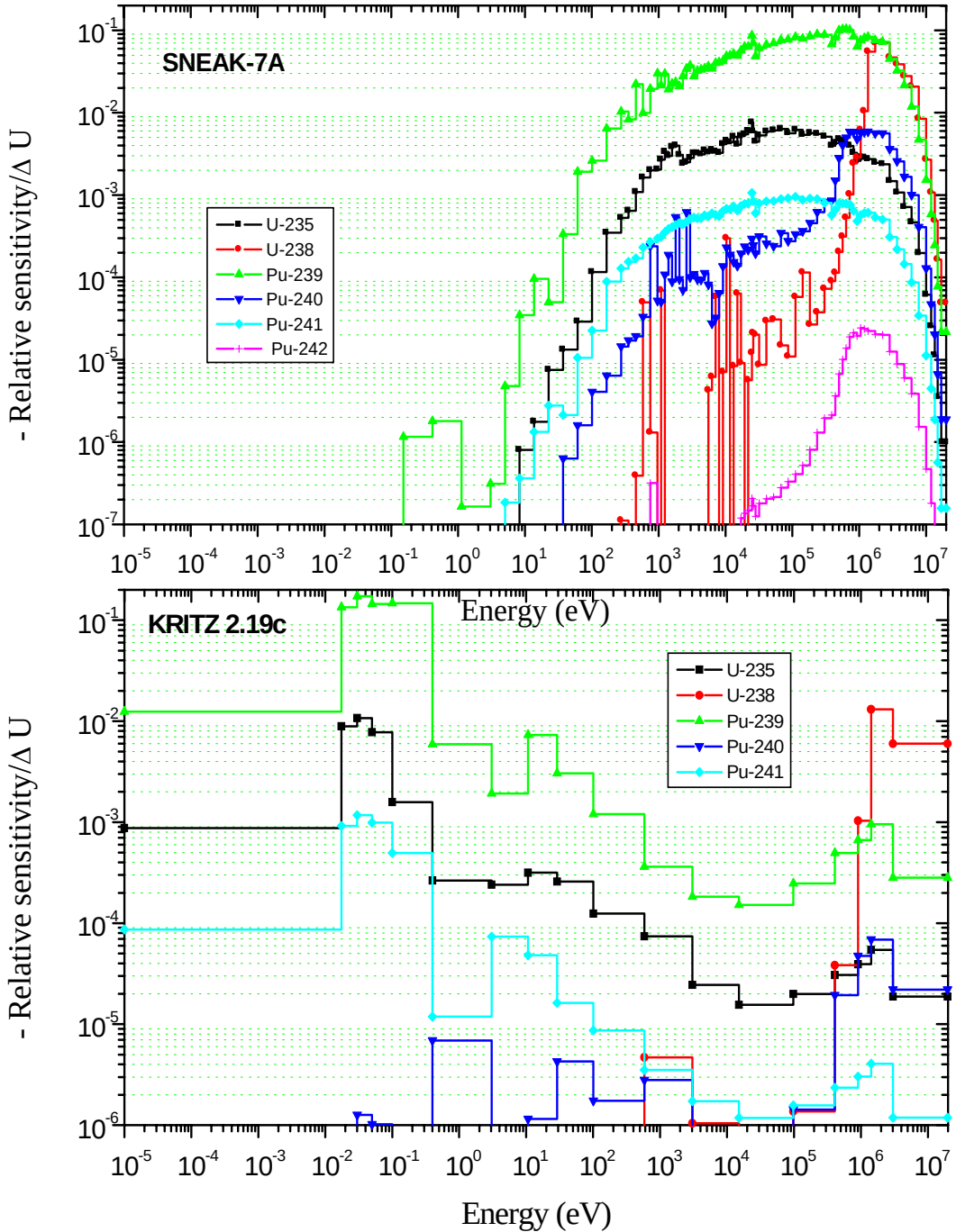
Kritz-2 Horizontal & Vertical Model (I. Remec, J. Gehin)





Uncertainties (in pcm) vs. C/E





KRITZ / SNEAK

Sensitivity of core eigenvalue to fission cross sections of U and Pu isotopes

Back

CCVM

CSNI Code Validation Matrix

INTEGRAL TEST DATA

This project is dedicated to the memory of Gianni Frescura, who managed it as Head of the NEA Safety Division from 1993 to 2003

Go to CCVM Separate Effects Test Data Page (updated June 2003)

Facility
GENERAL
BETHSY
DOEL2
FIST
FIX-II
LEIBSTADT
LOBI
LOFT
OTIS
PACTEL
PIPER
PKL
ROSA-III

CSNI Code Validation Matrix of Thermo-Hydraulic Codes for LWR LOCA and Transients

Data Collection at NEA Data Bank

Revised January 2004

Updates to latest version: Additional reports, photographs, and micrographs for LOFT/LP-FP2 and LOFT/L2-5 have been added.

Over the years the NEA Data Bank could collect a sizable subset of separate effects test reactor transient and LOCA integral test data (I.T.D.), as defined in the Code Validation Matrix of Document OCDE/GD(97)12. These data with accompanying documentation are now available on CD-ROMs. The writing format of the CD conforms to the standard ISO 9660. Each CD contains a copy of the INDEX file. It summarizes the complete contents of all CDs.

- The reports describing the experiments have been electronically scanned and transformed into PDF files. Each report is stored in a separate subdirectory.

EU-Concerted Action

QUADOS

Quality Assurance for Numerical Dosimetry

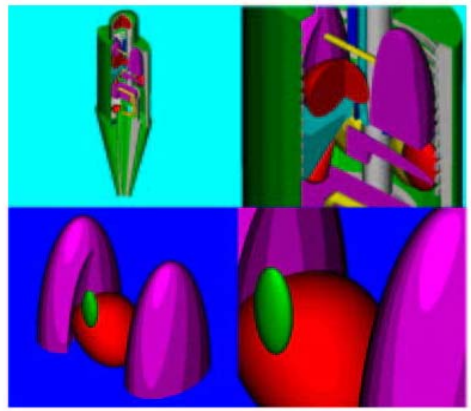
ENTE PER LE NUOVE TECNOLOGIE L'ENERGIA E L'AMBIENTE

ALMA MATER STUDIORUM UNIVERSITA' DI BOLOGNA

QUADOS

EURADOS
EUROPEAN DOSIMETRY GROUP

OECD
NUCLEAR ENERGY AGENCY



Monte Carlo techniques and computer codes are widely used to solve problems in nuclear science, technology and applications

Computer codes used as a "***black box***", user interaction performed via control cards, sometimes in detriment of the understanding of

"basic principles"
the physics insight

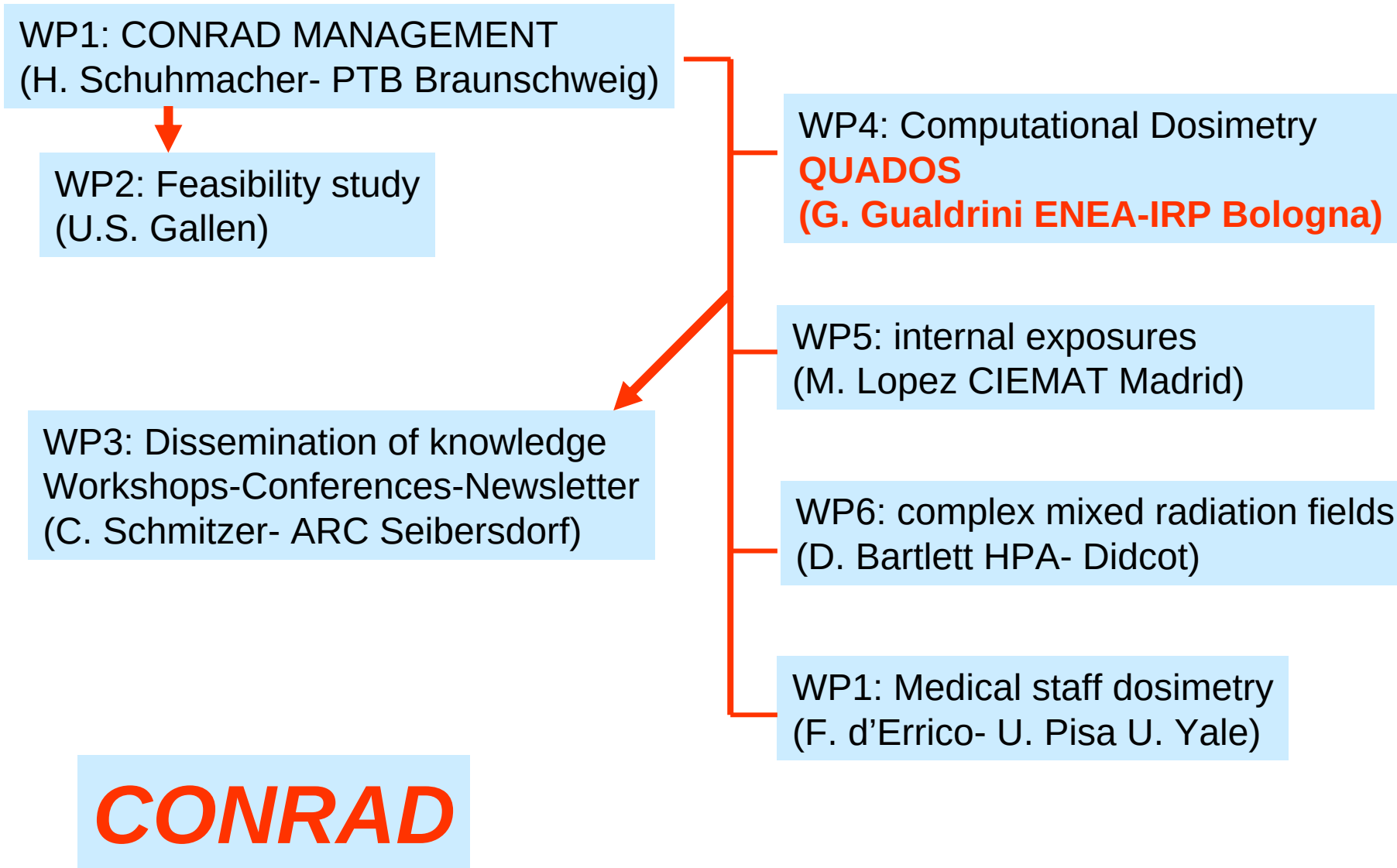
WORKSHOP

INTERCOMPARISON ON THE USAGE OF COMPUTATIONAL CODES IN RADIATION DOSIMETRY

BOLOGNA-Italy
July 14-16 2003

QUADOS Objectives

- The group designed a series of significant **reference problems**,
- **Verification of the correct usage of the computer codes,**
- Inform the community about the benefits to be obtained from **sensitivity and uncertainty analysis**,
- Inform the community about more sophisticated approaches that may be available to them.


WP1: CONRAD MANAGEMENT
(H. Schuhmacher- PTB Braunschweig)
WP2: Feasibility study
(U.S. Gallen)
WP3: Dissemination of knowledge
Workshops-Conferences-Newsletter
(C. Schmitzer- ARC Seibersdorf)
WP4: Computational Dosimetry
QUADOS
(G. Gualdrini ENEA-IRP Bologna)
WP5: internal exposures
(M. Lopez CIEMAT Madrid)
WP6: complex mixed radiation fields
(D. Bartlett HPA- Didcot)
WP1: Medical staff dosimetry
(F. d'Errico- U. Pisa U. Yale)
CONRAD

- WP4-WG6 - Computational Dosimetry (CONRAD WP4)
- Objectives
- A EURADOS specialists group for computational dosimetry has identified the quality assurance of the transport calculations that are widely used in dosimetry and the unfolding of spectral information as areas where coordination of research activities is urgently needed.
- The project combines various research coordination actions in the field of computational dosimetry applied to external and internal exposures at the workplace. The activities coordinated include intercomparisons and benchmark studies on the overall uncertainty assessment and on the application of advanced tools like voxel models and unfolding techniques for radiation spectra analysis. The results obtained in these projects will be presented and discussed in a workshop that will be optimally tailored to reach practitioners and to communicate the know-how on the correct use of complex computation tools as well as on the assessment of the uncertainties associated with numerical results. Special emphasis is given to a close collaboration with the partners undertaking other work packages (WPs) in the project (WP5, WP6 and WP7). The outcome will be a better understanding of the quality, i.e. reliability and uncertainty, for computational techniques in radiation protection.
- **Milestones and deliverables**
- Investigation of stakeholder needs for calculations with complex codes with particular emphasis on uncertainty assessment.
- The deliverables include reports on activities and achievements and a workshop.
- **Chairperson: Gianfranco Gualdrini**
  ENEA, Bologna, ITALY
  e-mail: guald@bologna.enea.it

CONCLUSIONS

- **Reactor design and safety parameters are biased due to uncertainties in nuclear data. Improved safety and better cost efficiency can be achieved by the reduction of uncertainties in design parameters.**
- **Uncertainties of calculated results can be estimated by sensitivity & uncertainty analysis and by comparison with benchmark experiments.**
- Sensitivity and uncertainty analysis can also **identify areas of weakness in data files and guide further evaluations.**
- **Some powerful calculational tools needed for such analysis can be obtained from the NEA-DB.**

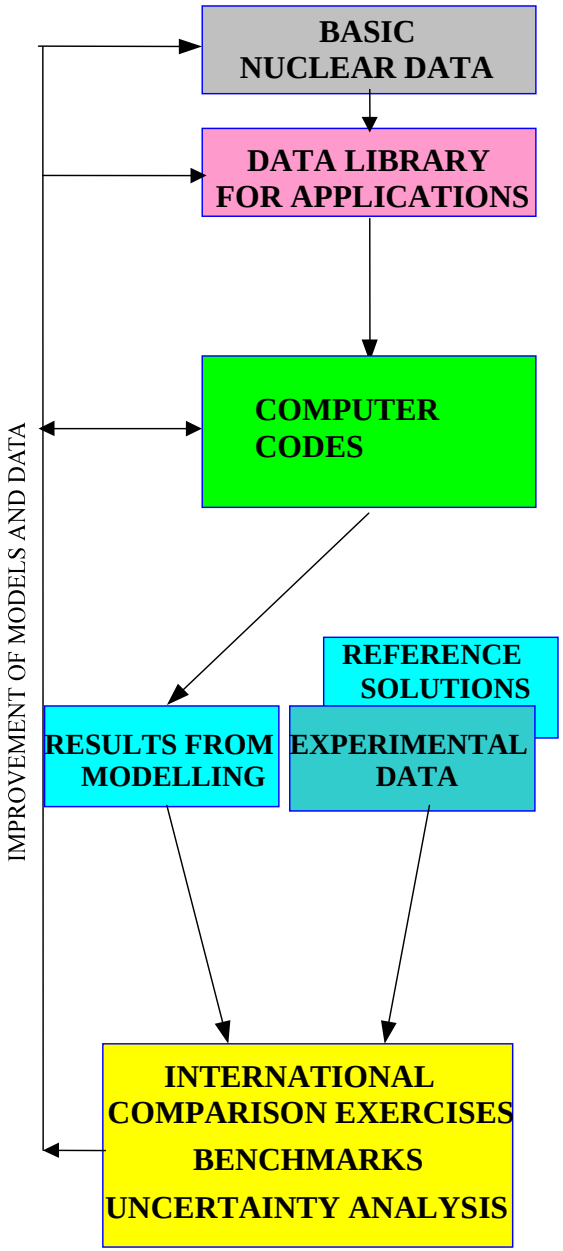
Web pages

OECD/NEA Data Bank:

- Computer program service: http://www.nea.fr/html/dbprog/
- Nuclear data : http://www.nea.fr/html/dbdata/
- Thermodynamic data : http://www.nea.fr/html/dbtdb/cgi-bin/tdbdocproc.cgi

INFORMATION TYPE

TOPICS



Basic data

- cross sections
- material properties
- basic data

Computer codes

- data processing
- core design
- core dynamics
- safety/accident analysis
- mechanics
- fluid dynamics
- heat transfer
- shielding/radiation protection
- impact on the environment

Integral data

- criticality
- fuel performance
- shielding experiments
- lattice and core physics experiments
- reactor operation
- thermal--hydraulic loops

Benchmarks - comparison exercises

- cells & lattices
- burn-up credit criticality
- transients/stability
- fuel cycle (plutonium recycling)
- shipping cask shielding
- pressure vessel dosimetry
- accelerator driven systems
- accelerator shielding