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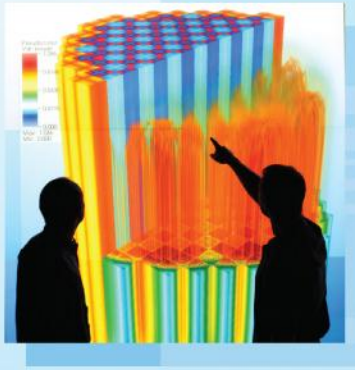

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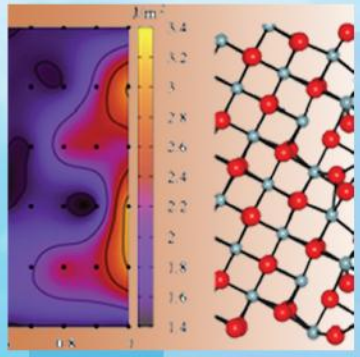

Fundamental science

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Consortium for Advanced Simulation of LWRs

Review of Experiments for CASL Neutronics Validation

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March 29, 2012

U.S. DEPARTMENT OF ENERGY | Nuclear Energy

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EXECUTIVE SUMMARY

The validation of a neutronics code or other nuclear design codes is a large undertakeing. The efforts required to validate a neutronics code often far exceed those to develop such code since large amount of experimental data are required to perform validation work. Fortunately, for neutronics codes validation, large number of experiments have been done dating back to the early days of the nuclear industry. There have been internationally coordinated efforts such as the International Criticality Safety Benchmark Evaluation Project (ICSBEP) and the International Reactor Physics Experiment Evaluation Project (IRPhEP) to preserve the data and knowledge learned from the past experiments. The outcome of these projects and other measurement data provide valuable data sources to perform CASL neutronics codes validation work. This document identifies all the available experimental data that are applicable to perform the validation work of CASL neutronics codes.

CONTENTS

EXECUTIVE SUMMARY iii

CONTENTS v

FIGURES vii

TABLES viii

ACRONYMS ix

1. Introduction 1

2. Critical Experiments 2
2.1. Strawbridge-Barry 101 Criticals 2
2.2. Babcock and Wilcox Spatial Critical Experiments with Urania-Gadolinia Fuel 2
2.3. International Criticality Safety Benchmark Evaluation Project (ICSBEP) 3

3. International Reactor Physics Experiment Evaluation Project (IRPhEP) 4
3.1. CREOLE PWR Reactivity Temperature Coefficient Experiment (CREOLE-PWR-EXP-001) 5
3.2. CREOLE PWR (CROCUS-LWR-RESR-001) 6
3.3. Reactor Physics Experiments in the IPEN/MB-01 Research Reactor Facility (IPEN(MB01)-LWR-RESR-001) 7
3.4. KRITZ LEU and MOX Experiments (three experiments available) 8
3.4.1. KRITZ-LWR-RESR-001: KRITZ-2:19 Experiment on Regular H2O/Fuel Pin Lattices with Mixed Oxide Fuel at Temperatures 21.1 and 235.9 °C 9
3.4.2. KRITZ-LWR-RESR-002: KRITZ-2:1 Experiment on Regular H2O/Fuel Pin Lattices with Low Enriched Uranium Fuel at Temperatures 248.5 °C 9
3.4.3. KRITZ-LWR-RESR-003: KRITZ-2:13 Experiment on Regular H2O/Fuel Pin Lattices with Low Enriched Uranium Fuel at Temperatures 243 °C 10
3.4.4. An International Benchmark Exercise on KRITZ Critical Experiments 10
3.5. Temperature Effects on Reactivity in Light Water Moderated UO2 Core with Soluble Poisons at TCA (TCA-LWR-EXP-001) 11
3.6. DIMPLE Low Power Reactor 12
3.6.1. DIMPLE-LWR-EXP-001: Light Water Moderated and Reflected Low Enriched Uranium (3 wt.% 235U) Dioxide Rod Lattices DIMPLE S01 13
3.6.2. DIMPLE-LWR-EXP-002: Light Water Moderated and Reflected Low Enriched Uranium (3 wt.% 235U) Dioxide Rod Lattices DIMPLE S06 13
3.7. VENUS Critical Facility 13
3.7.1. VENUS-PWR-EXP-001: VENUS-1 PWR UO2 Core 2-Dimensional Benchmark Experiment 14
3.7.2. VENUS-PWR-EXP-003: VENUS-3 PWR UO2 Core 3-Dimensional Benchmark Experiment 14
3.7.3. VENUS-PWR-EXP-005: Experimental Study of the VENUS-PRP Configurations No. 9 and 9/1 15
3.7.4. International Benchmark Exercise of VENUS-2 MOX Core Measurements 16

3.8. LR0-VVER-EXP-001: VVER Physics Experiments: Hexagonal Lattices (1.22-cm Pitch) of Low-Enriched U(2.0 - 3.3 WT.% U235)O2 Fuel Assemblies in Light Water with Central Control-Assembly Mockup 17

4. Post-Irradiation Examination (PIE) Isotopic Measurements 17

5. Operating Plants Data 18
5.1. Startup Physics Test Data 18
5.2. Hot Full Power Operating Measurement Data 18

6. Conclusions and Future Work 19

7. References 19

FIGURES

Figure 1. View of the CROCUS Reactor of the Swiss Federal Institute of Technology, Lausanne. 7

Figure 2. Some Details of the IPEN/MB-01 Core. 8

Figure 3. Vertical cross-section of the KRITZ reactor. 9

Figure 4. Horizontal cross-section of the benchmark model (Reference [6]). 11

Figure 5. General View of the DIMPLE Reactor. 12

Figure 6. VENUS-2 Core Geometry (Reference [8]). 16

TABLES

Table 1. Critical benchmark experiments applicable to CASL neutronics codes validation in VALID library 4

Table 2. Summary of Reactor Physics Benchmark Experiments from the IRPhEP Handbook That are Applicable to CASL Neutronics Codes Validation 5

Table 3. Summary of the KRITZ Critical Configuration Specifications.....11

Table 4. Available PWR Spent Fuel Samples for CASL Neutronics Codes Depletion Calculations Validation (Reference [9]) 18

ACRONYMS

2-D	Two-Dimensional
3-D	Three-Dimensional
AMA	Advanced Modeling Applications
ARO	All rods out
BOC	Beginning of Cycle
B&W	Babcock & Wilcox
CASL	Consortium for Advanced Simulation of Light Water Reactors
DeCART	Deterministic Core Analysis based on Ray Tracing
DRWM	Dynamic Rod Worth Measurement
HZP	Hot zero power
ICSBEP	International Criticality Safety Benchmark Evaluation Project
INL	Idaho National Laboratory
IRPhEP	International Reactor Physics Experiment Evaluation Project
JAERI	Japan Atomic Energy Research Institute
LIME	Lightweight Integrating Multi-physics Environment
MCNP	Monte Carlo N-Particle transport code
NRC	Nuclear Regulatory Commission
OECD/NEA	Organization of Economic Cooperation and Development/Nuclear Energy Agency
ORNL	Oak Ridge National Laboratory
PIE	Post-Irradiation Examination
PWR	Pressurized Water Reactor
RCS	reactor coolant system
RTC	Reactivity Temperature Coefficient
TCA	Tank-type Critical Assembly
TVA	Tennessee Valley Authority
VALID	ORNL's Verified, Archived Library of Inputs and Data
VENUS	Vulcain Experimental NUclear Study
VERA	Virtual Environment for Reactor Applications

1. INTRODUCTION

The verification and validation (V&V) of a neutronics analysis code or other nuclear design codes is a large undertaking since it must address correctness of the nuclear data library, the physical models and the numeric algorithms used in the code, the implementation of the physical models and the numeric algorithms, and its application within a nuclear design and analysis system. For this reason, a systematic V&V process is required. In the nuclear industry, the vendors historically employed a qualification process to verify and validate their nuclear design codes, which starts with the qualification of the basic methodology used in the code and proceeds through logical steps to the qualification of the codes as used with the entire design and analysis system.

Neutronics code qualification normally consists of three parts: 1). Comparisons with Monte Carlo and other neutronics codes calculations, 2). Comparisons to critical experiments and isotopic measurements, and 3). Comparisons against measured plant data. The first part, to certain extent, constitutes the verification of the methodologies and their implementations in a neutronic code. The second and third part constitute the validation of the neutronics codes for core design applications.

The validation of the CASL neutronics codes will also be carried out by comparing the calculated results with available experimental data. The measurement data include data from critical experiments, reactor physics experiments, Post-Irradiation Examination (PIE) of spent nuclear fuel, and operating plants.

The CASL VERA neutronics tools consist of Denovo, DeCart and Westinghouse's ANC. The ANC code has been validated and extensively used in Westinghouse's design and safety analysis processes. The DeCart code is considered as an intermediate tool before more advanced code Denovo is developed. Minimal efforts will be spent on validating the DeCart code. Hence the neutronics validation efforts for CASL will be focused on Denovo. Since the Denovo code is not ready to be validated yet, this report will discuss all the available data for CASL neutronics codes validation.

Section 2 disscusses the data from critical experiments. Section 3 discusses the data from reactor physics experiments. Section 4 disccusess the isotopic measurement data from PIE. Section 5 discusses the operating plant data from TVA.

2. CRITICAL EXPERIMENTS

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. It is unlikely that any of these measurements will be repeated in the future. The results and techniques developed from these measurements remain of great value today and in the future for CASL. They provide the basis for recording, development, and validation of methods, and represent a significant collection of data for present and future research. This section describes the critical benchmark experiments that are applicable to CASL neutronics codes validation. It should be pointed out that many of the critical experiments are documented for 2-D analysis (2-D information + experimentally measured axial buckling). Consequently, the CASL neutronics codes will need to be able to do 2-D calculations with an input axial buckling. Alternatively, some 3-D representation will need to be developed, which would present difficulty in some cases where the experimental data is not available.

2.1.Strawbridge-Barry 101 Criticals

The Strawbridge and Barry 101 criticals are uniform light water lattice critical experiments with several lattice parameters such as water to Uranium ratio, enrichment, experimental buckling, pellet diameter and boron concentration. These criticals contain 40 uranium oxide and 61 uranium metal cold clean experiments (Reference [1]). These critical experiments have historically been included in Westinghouse neutronics codes (e.g. PARAGON/ANC) qualifications since they cover a wide range of lattice parameters and therefore provide a severe test for the lattice physics codes to predict reactivities accurately over a broad range of conditions.

Since the Strawbridge-Barry criticals are uniform lattices for which experimental bucklings have been reported, these criticals can be treated as single pin cells in CASL neutronics codes validation. The ranges of lattice parameters covered by these criticals are:

Enrichment (a/o U235) : 1.04 to 4.069
Boron concentration (ppm): 0 to 3392
Water to uranium ratio- 1.0 to 11.96
Pellet diameter (cm): 0.44 to 2.35
Lattice pitch (cm): 0.95 to 4.95
Clad material: none, aluminum, stainless steel
Lattice type: square, hexagonal
Fuel density (glcm3): 7.5 to 18.9

2.2.Babcock and Wilcox Spatial Critical Experiments with Urania-Gadolinia Fuel

A large physics validation program sponsored by USAEC and Babcock & Wilcox (B&W) was conducted at B&W's Lynchburg Research Center during the 1970's. These experiments, which are documented in Reference [2], provided reactivity and power distribution measurements for typical PWR lattices at cold conditions for various configurations of fuel rods, guide thimbles, and several different burnable absorbers. The Babcock & Wilcox Spatial Criticals are critical experiments for a variety of lattice configurations such as PYREX rods, Gadolinia rods, control rods, and water holes

with UO2 fuel rods. The enrichment of UO2 fuel rods are 2.46 wt% and 4.02wt%, and the Gadolinia content is 4.00 wt%. Both reactivity and power distribution are measured and can be used to compare CASL neutronics codes calculations.

2.3.International Criticality Safety Benchmark Evaluation Project (ICSBEP)

The International Criticality Safety Benchmark Evaluation Project (ICSBEP) (Reference [3]) was initiated in 1992 by the Department of Energy Defense Programs Systems Engineering Division and is managed through Idaho National Laboratory (INL). It became an official activity of the Organization for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) in 1995. The purpose of the ICSBEP is to: 1). Identify a comprehensive set of critical benchmark data and, to the extent possible, verify the data by reviewing original and subsequently revised documentation, and by talking with the experimenters or individuals who are familiar with the experimenters or the experimental facility. 2). Evaluate the data and quantify overall uncertainties through various types of sensitivity analysis. 3). Compile the data into a standardized format. 4). Perform calculations of each experiment with standard criticality safety codes. 5). Formally document the work into a single source of verified benchmark critical data. The work of the ICSBEP is documented as an International Handbook of Evaluated Criticality Safety Benchmark Experiments. Currently, the handbook spans nearly 55,000 pages and contains 516 evaluations representing 4405 critical, near-critical, or subcritical configurations, 24 criticality alarm placement/shielding configurations with multiple dose points for each, and 200 configurations that have been categorized as fundamental physics measurements that are relevant to criticality safety applications. The handbook is intended for use by criticality safety analysts to perform necessary validations of their calculational techniques and is expected to be a valuable tool for decades to come. The ICSBEP Handbook is available both on DVD and the Internet. A DVD may be obtained by completing the DVD Request Form through http://icsbep.inel.gov. Access to the Handbook on the Internet requires a password. The password can be established by completing the Password Request Form.

The SCALE project at ORNL maintains a large number of the ICSBEP criticality benchmark experiments within ORNL's Verified, Archived Library of Inputs and Data (VALID). A total of more than 350 cases are available in the VALID library, which contains the input, output and other associated files (Reference [4]). The critical experiments available cover different broad categories of systems. These systems use a range of fissile materials including a range of uranium enrichments, various plutonium isotopic vectors, and mixed uranium-plutonium oxides. The physical form of the fissile material also varies and is represented as metal, solutions, or arrays of rods or plates in a water moderator. The neutron energy spectra of the systems also vary and cover both fast and thermal spectra.

There are a total of 118 cases available in VALID that are applicable to the CASL neutronics codes validation. Table 1 summarizes the listing of benchmark critical experiments applicable to CASL neutronics codes validation.

Table 1. Critical benchmark experiments applicable to CASL neutronics codes validation in VALID library

Evaluation	Cases	Fissile material	Moderator	Experiment description
LEU-COMP-THERM-001	8	UO2 2% wt	Light water	Water-Moderated U(2.35)O2 Fuel Rods in 2.032 cm Square-Pitched Arrays
LEU-COMP-THERM-002	5	UO2 4% wt	Light water	Water-Moderated U(4.31)O_2 Fuel Rods in 2.54 cm Square-Pitched Arrays
LEU-COMP-THERM-010	30	UO2 4% wt	Light water	Water-Moderated U(4.31)O_2 Fuel Rods Reflected by Two Lead, Uranium, or Steel Walls
LEU-COMP-THERM-017	29	UO2 2% wt	Light water	Water-Moderated U(2.35)O_2 Fuel Rods Reflected by Two Lead, Uranium, or Steel Walls
LEU-COMP-THERM-042	7	UO2 2% wt	Light water	Water-Moderated Rectangular Clusters of U(2.35)O_2 Fuel Rods (1.684 cm Pitch) Separated by Steel, Boral, Boraflex, Cadmium, or Copper Plates with Steel Reflecting Walls
LEU-COMP-THERM-050	18	UO2 5% wt	Light water	^{149}Sm Solution Tank in the Middle of Water-Moderated 4.738 wt % Enriched Uranium Dioxide Rod Arrays
MIX-COMP-THERM-001	4	Pu & natural UO2	Light water	Water-Reflected Mixed Plutonium-Uranium Oxide (20 wt % Pu) Pins
MIX-COMP-THERM-002	6	Pu & natural UO2	Light water	Rectangular Arrays of Water-Moderated UO_2–2 wt % PuO_2 (8% ^{240}Pu) Fuel Rods
MIX-COMP-THERM-004	11	Pu & natural UO2	Light water	Critical Arrays of Mixed Plutonium-Uranium Fuel Rods with Water-to-Fuel Volume Ratios Ranging from 2.4 to 5.6

3. INTERNATIONAL REACTOR PHYSICS EXPERIMENT EVALUATION PROJECT (IRPHEP)

The International Reactor Physics Experiment Evaluation Project (IRPhEP) (Reference [5]) was initiated as a pilot activity in 1999 by the Organization of Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Nuclear Science Committee (NSC). The project was endorsed as an official activity of the NSC in June of 2003 to preserve integral reactor physics experimental data including separate or special effects data for nuclear energy and technology applications and the knowledge and competence. The IRPhEP publishes the International Handbook of Evaluated Reactor Physics Benchmark Experiments annually. CASL participants can obtain a copy of the handbook by submitting a request through http://irphep.inl.gov.

There are a total of **thirteen** experiments from **eight** different reactors in the IRPhEP handbook that are applicable to the validation of CASL neutronics codes. Table 2 summarizes those experiments that are applicable to CASL neutronics codes validation. The following briefly describes the facilities and the experiments applicable to CASL.

Table 2. Summary of Reactor Physics Benchmark Experiments from the IRPhEP Handbook That are Applicable to CASL Neutronics Codes Validation

Experiment	K-eff	Power/ Reaction rate	Temp. Effects	Spectral Indices	Kinetics	Borated/ Unborated	UO2/ MOX
CREOLE PWR (CREOLE-PWR-EXP-001)	Yes	Yes	Yes			Yes/ Yes	Yes/ Yes
CROCUS (CROCUS-LWR-RESR-001)	Yes				Yes	Yes/ No	Yes/ No
IPEN MB1 (IPEN(MB01)-LWR-RESR-001)	Yes			Yes	Yes	No/ Yes	Yes/ No
KRITZ1 (KRITZ-LWR-RESR-001, -002, -003)*	Yes	Yes	Yes			Yes/ No	No/ Yes
TCA (TCA-LWR-EXP-001)	Yes		Yes			Yes/ Yes	Yes/ No
DIMPLE (DIMPLE-LWR-EXP-001, -002)*	Yes	Yes		Yes		No/ Yes	No /Yes
VENUS-1 (VENUS-PWR-EXP-001, -003, -005)*	Yes	Yes				No/ Yes	Yes/ No
VVER Reactor (LR0-VVER-EXP-001)**	Yes	Yes	Yes	Yes		Yes/ No	Yes/ No

*Multiple benchmarks available
**CASL neutronics codes may not set up to model VVER reactors. This experiment is included for completion of information collection.

3.1. CREOLE PWR Reactivity Temperature Coefficient Experiment (CREOLE-PWR-EXP-001)

The EOLE critical facility is a zero-power reactor located at CEA/Cadarache (France). This facility was built in 1965 in order to study LWR core designs in support of the French large-scale program of building PWR power plants, which was launched during the middle of the seventies. The CREOLE (Coefficients of Reactivity in EOLE) experimental program was conceived to supply accurate differential information on the Reactivity Temperature Coefficient (RTC) in the whole temperature range of interest in a large PWR (from room temperature up to 300 °C). The measurements were performed in the EOLE facility at CEA-Cadarache during the two last years of the seventies. The experimental facility consists of a pressurized central test loop in which it is possible to achieve operating conditions of a large PWR power reactor in terms of pressure and moderator temperature, a large vacuum-gap separation zone and a peripheral driver core of variable sizes surrounded by a water reflector.

The differential isothermal Reactivity Temperature Coefficient (RTC) of UO2 and UO2-PuO2 lattices were measured from 20 °C up to 300 °C in a central pressurized loop. Further, the integral temperature reactivity effect of $\Delta T \cong 280$ °C was obtained in terms of soluble-boron equivalence in the central loop and driver-zone critical-loading variations. The reactivity effects due to water temperature and density changes in the test loop were measured by the doubling-time technique during reactor divergence with all control rods extracted or through the adjustment of the driver-core critical sizes when necessary. Radial fission-rate distributions were measured by direct gamma scanning of the fuel rods in the central-loop and driver-core zones, and axial flux maps were achieved using fission chambers.

The measurements were carried out in 4 basic experimental configurations of the central test loop with 200 fuel-rod locations with a pitch of 1.26 cm, which is typical for the 17 x 17 PWR fuel assemblies:

- UO2 clean lattice (200 fuel rods of 3.1 % enrichment).
- UO2 poisoned lattice with 1166 ppm of boron in water (200 fuel rods of 3.1 % enrichment).
- UO2-PuO2 clean lattice (80 fuel rods with 3.2 % fissile Pu and 120 fuel rods with 2% fissile Pu).
- UO2-PuO2 clean lattice with water channels (72 fuel rods with 3.2% fissile Pu, 108 fuel rods with 2% fissile Pu and 20 water channels).

Five additional configurations were obtained from the previous ones by using aluminum overcladdings to simulate water-density changes. Therefore, the following measurements were performed in the CREOLE experiment:

- Measurements of the operating parameters: temperature, pressure and boron content
- Measurements of the basic technological parameters: geometry and material compositions
- Measurements of the critical sizes of different experimental configurations at room temperature
- Measurements of the temperature reactivity coefficients
- Measurements of soluble-boron reactivity worth and equivalence with the temperature reactivity effect
- Measurements with AG3 over-claddings to simulate water density variations
- Measurements of the reaction-rate distribution.

3.2.CREOLE PWR (CROCUS-LWR-RESR-001)

The CROCUS reactor, operated by the Swiss Federal Institute of Technology, Lausanne, is a simple two-zone uranium-fueled, H2O-moderated critical research facility. Figure 1 (Reference [5]) gives a view of the facility, a so-called zero power reactor, with a maximum allowed power of 100 W. The core is approximately cylindrical in shape with a diameter of about 60 cm and a height of 100 cm. In 1995, in the CROCUS reactor a configuration with a central zone of 1.806 wt.%-enriched UO2 rods and an outer zone of 0.947 wt.%-enriched uranium metal rods was made critical by raising the water level. Later on, the reactor was used for different experiments. Some configurations are evaluated in the IRPhEP handbook and are approved as benchmark configurations for the reactivity difference between the critical water level and some supercritical conditions.

CROCUS-LWR-RESR-001 is a benchmark on kinetics parameters in CROCUS. Two different kinds of measurements were carried out in the CROCUS reactor: a) Variation of the water moderator level. For this type of configuration, four different water levels were measured, one for the critical state (i.e. when the inverse reactor period is zero) and three higher water levels for supercritical states; b) Insertion of an absorber rod, adjustment of the water level to the new critical height, and then removal of the absorber rod. For this second set of experiments, a total of four configurations were measured, two for the critical states with the absorber rod inserted and two supercritical states obtained after withdrawing the absorber rod. The experiments at different water levels were performed in 1996 and the ones with absorber rods in 1997.

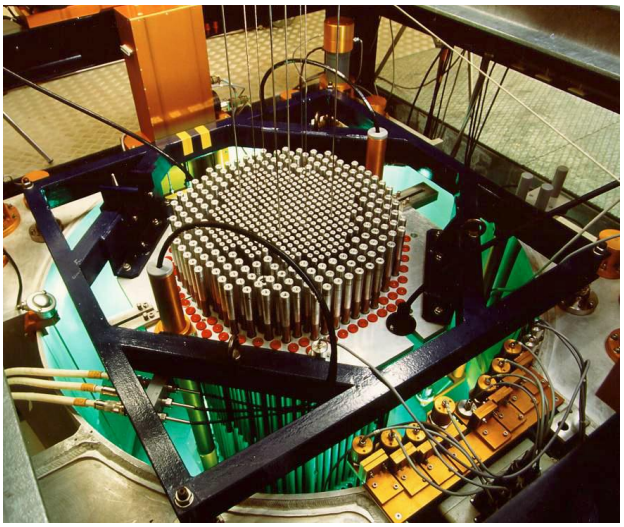

Figure 1. View of the CROCUS Reactor of the Swiss Federal Institute of Technology, Lausanne.

3.3.Reactor Physics Experiments in the IPEN/MB-01 Research Reactor Facility (IPEN(MB01)-LWR-RESR-001)

The IPEN/MB-01 research reactor is a zero power critical facility specially designed for measurement of a wide variety of reactor physics parameters to be used as benchmark experimental data for checking the calculation methodologies and related nuclear data libraries commonly used in the field of reactor physics. This facility is located in the city of São Paulo, Brazil and reached its first criticality on November 9, 1988. Since then it has been utilized for basic reactor physics research and as an instruction laboratory system. Figure 2 (Reference [5]) shows some details of the IPEN/MB-01 core. This facility consists of a 28x26 rectangular array of UO2 fuel rods 4.3486 wt.% enriched uranium as UO2 and clad by stainless steel (SS-304) inside a light water tank. The maximum allowed power is 100 W. The control of the IPEN/MB-01 reactor is via two control banks diagonally placed. The control banks are composed of 12 Ag-In-Cd rods and the safety banks by 12 B4C rods. The square pitch of the IPEN/MB-01 reactor was chosen to be close to the optimum fuel-to-moderator ratio (maximum $k\infty$). This feature favors the thermal neutron energy region and mainly the 235U events. This evaluation assesses the usefulness of the data acquired in the isothermal experiment as a benchmark for reactor physics codes and related nuclear data library. The experiments performed at the IPEN/MB-01 reactor serve as benchmarks to verify calculation methodologies and related nuclear data libraries. The experiments performed at the IPEN/MB-01 reactor were the following: critical configurations, buckling and extrapolation length, spectral characteristics, reactivity measurements, temperature reactivity coefficient, effective kinetic parameters, reaction-rate distributions, and power distribution. The criticality portion of the experiments has been documented under the ICSBEP.

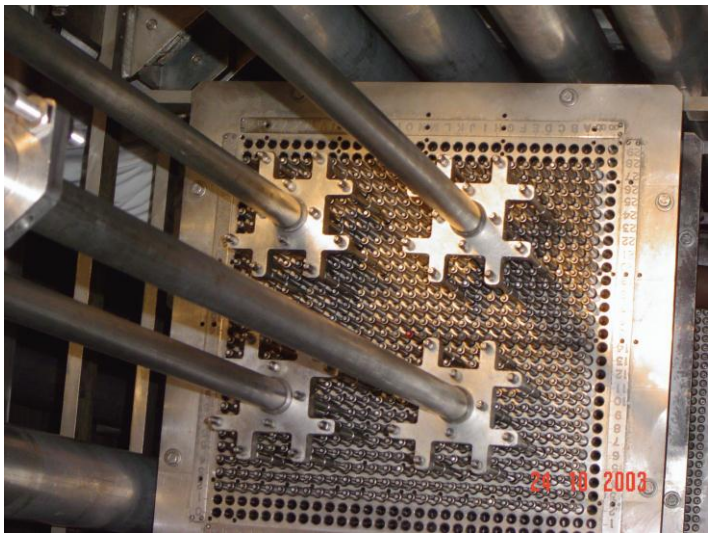

Figure 2. Some Details of the IPEN/MB-01 Core.

3.4.KRITZ LEU and MOX Experiments (three experiments available)

The KRITZ reactor operated at Studsvik, Sweden, during the first half of the 1970s. It comprised fuel rods in square-pitched lattices in a 5-m-high, 1.5-m-diameter cylindrical pressure tank. Figure 3 (Reference [3]) shows the vertical cross section of the KRITZ reactor. The "KRITZ experiments," performed in the period from September 1972 through February 1973, included several series of criticality experiments on light-water-moderated lattices with uranium dioxide rods, mixed-oxide rods, or both, at room temperature and at elevated temperatures up to 245 °C (473 IF) covering temperatures close to the range used in light water reactor cores. Criticality was attained by controlling the boron content in the water and by adjusting the water level. Critical levels were measured at low power, often as low as 10 W, to minimize the activation of the fuel. Measurements of the activated copper wires and gamma scans of the fuel rods were used to determine experimentally the axial buckling. For most of the cores measured relative powers for selected rods, obtained from gamma scans, are provided. The data released describe four experiments: three with uranium rods (KRITZ-1, KRITZ-2:1 and KRITZ-2:13) and one with mixed-oxide rods (KTITZ-2:19). Because of limited data available for KRITZ-1, the IRPhEP handbook focused on the other three configurations.

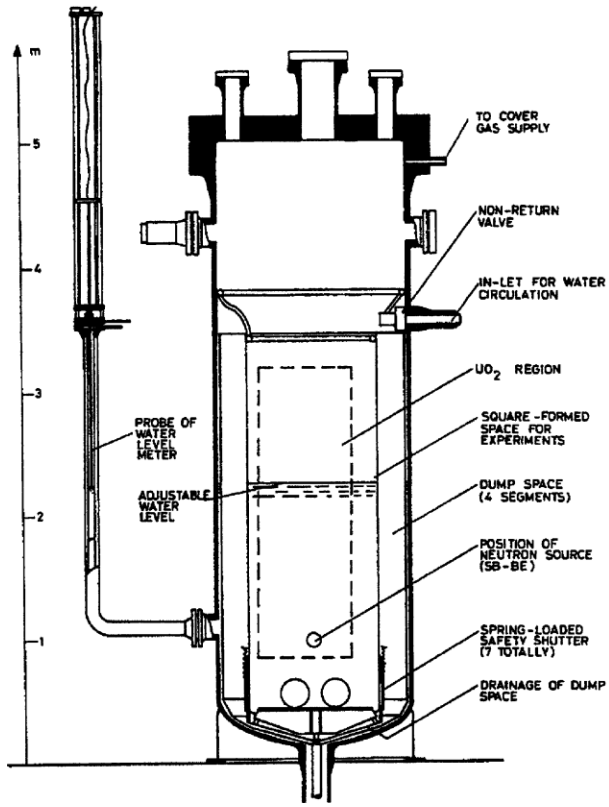

Figure 3. Vertical cross-section of the KRITZ reactor.

3.4.1. KRITZ-LWR-RESR-001: KRITZ-2:19 Experiment on Regular H2O/Fuel Pin Lattices with Mixed Oxide Fuel at Temperatures 21.1 and 235.9 °C

The objective of the KRITZ-2:19 experiment was to attain criticality of a rectangular array of mixed oxide Zircaloy-2 clad fuel rods in light water by regulating the concentration of boron in the water and by adjusting the water level. Criticality was achieved at isothermal conditions at room temperature (21.1 °C) and at elevated temperature (235.9 °C). Besides the critical boron concentrations and water level, the axial buckling was also determined, and the relative rod powers (fission-rate distributions) were measured for selected fuel rods. Measurements of the activated copper wires and gamma scans of the fuel rods were used to determine experimentally the axial buckling. The experiment with mixed-oxide rods at 18.00 mm pitch is considered. The experiments were performed in the KRITZ reactor in the period from September 1972 through February 1973. Criticality was attained by controlling the boron content in the water and by adjusting the water level. Critical levels were measured at low power, often as low as 10 W, to minimize the activation of the fuel. Both critical configurations described in the IRPhEP handbook – one at room temperature and one at elevated temperature – are considered acceptable for use as criticality benchmark experiments.

3.4.2. KRITZ-LWR-RESR-002: KRITZ-2:1 Experiment on Regular H2O/Fuel Pin Lattices with Low Enriched Uranium Fuel at Temperatures 248.5 °C

The objective of the KRITZ-2:1 experiment was to attain criticality of a rectangular array of low enriched uranium Zircaloy-2 clad fuel rods in light water by regulating the concentration of boron in

water and by adjusting the water level. Criticality was achieved at isothermal conditions at room temperature (19.7 °C) and at elevated temperature (248.5 °C). Besides the critical boron concentrations and water level, the axial buckling was also determined, and the relative rod fission rates were measured for selected fuel rods. Measurements of the activated copper wires and gamma scans of the fuel rods were used to determine experimentally the axial buckling.

3.4.3. KRITZ-LWR-RESR-003: KRITZ-2:13 Experiment on Regular H2O/Fuel Pin Lattices with Low Enriched Uranium Fuel at Temperatures 243 °C

The objective of the KRITZ 2:13 experiment was to attain criticality of a rectangular array of mixed oxide zircaloy-2 clad fuel rods in light water by regulating the concentration of boron in water and by adjusting the water level. Criticality was achieved at isothermal conditions at room temperature (22.1°C) and at elevated temperature (243.0 °C). Besides the critical boron concentrations and water level, the axial buckling was also determined, and the relative rod powers (fission rate distributions) were measured for selected fuel rods. Both critical configurations described in the IRPhEP handbook – one at the room temperature and one at the elevated temperature – are considered acceptable for use as criticality and reaction rate benchmark experiments.

3.4.4. An International Benchmark Exercise on KRITZ Critical Experiments

The KRITZ benchmark experiments have been extensively studied by an international benchmark exercise, which was launched in October 2000 in the framework of the joint activities of the OECD/NEA Working Party on the Physics of Plutonium Fuels and Innovative Fuel Cycles and the Task Force on Reactor-based Plutonium Disposition (Reference [6]). The aim of this exercise was to investigate the capabilities of the current production neutronics codes and nuclear data libraries to analyze MOX-fuelled systems, and to compare the accuracy of the predictions for the MOX- and UO2- fuelled configurations. Institutions from 7 countries participated in this exercise, providing 13 solutions. Table 3 summarizes the KRITZ critical configuration specifications. Figure 4 shows the horizontal cross-section of the benchmark model. Reference [6] provides comparative analyses of calculated and measured results, as well as intercomparisons of some of the results obtained by participants by calculation only. The computer codes used were deterministic codes such as THREEDANT, HELIOS and Monte Carlo codes. The results presented in reference [6] include the critical configuration core calculations and the fuel pin cell calculations. The critical configuration core calculations include: 1), the configuration multiplication factors at room and elevated temperatures, and 2), the relative rod powers for the rods for which the measurements were performed. The fuel pin cell calculations include: 1), the infinite multiplication factor (K_∞), and 2), absorption and fission rates for the selected nuclides (e.g. U-234, U-235, U-238, for UO2 fuel and Pu-239, Pu-230, Pu-241, Pu-242, Am-241, etc. for MOX fuel). The results documented in reference [6] provide valuable comparison results to validate CASL neutronics codes' predictive capability and nuclear cross section libraries used for light water reactor analyses.

Table 3. Summary of the KRITZ Critical Configuration Specifications

Experiment	Fuel	Fuel Diameter (mm)	Clad Outer Diameter (mm)	Rod Pitch (mm)	No. of Rods	Temp. (oC)	Boron Conc. (ppm)	Water Height (mm)
KRITZ-2:1	UO2, 1.86 wt.% 235U	10.58	12.25	14.85	44x44	19.7	217.9	652.8
						248.5	26.2	1055.2
KRITZ-2:13	UO2, 1.86 wt.% 235U	10.58	12.25	16.35	40x40	22.1	451.9	961.7
						243.0	280.1	1109.6
KRITZ-2:19	MOX 1.5 wt.% PuO2 91.41 at.% 239Pu	9.45	10.79	18.00	25x24	21.1	4.8	665.6
						235.9	5.2	1000.1

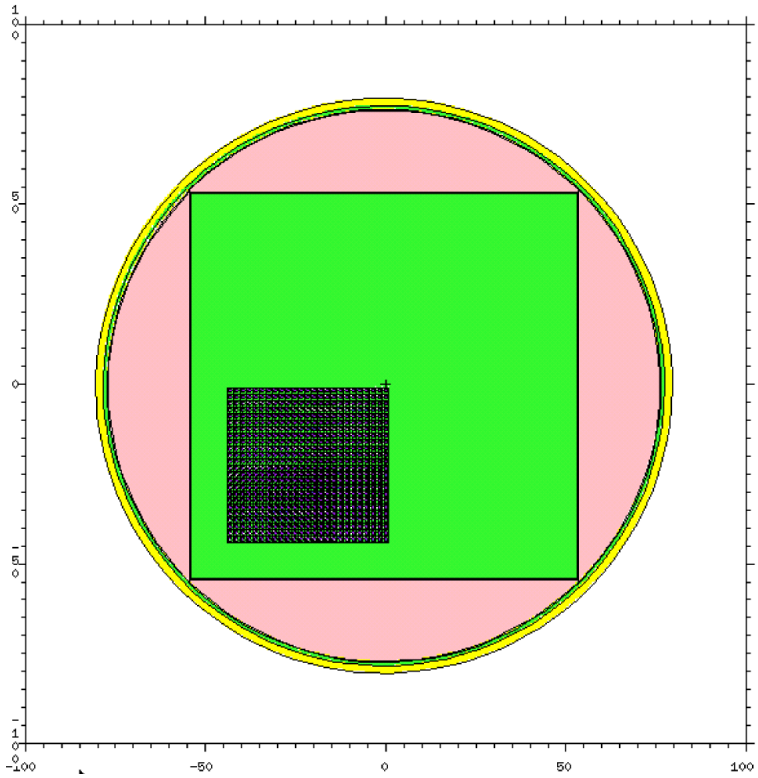

Figure 4. Horizontal cross-section of the benchmark model (Reference [6]).

3.5.Temperature Effects on Reactivity in Light Water Moderated UO2 Core with Soluble Poisons at TCA (TCA-LWR-EXP-001)

From the point of view of the nuclear criticality safety of the dissolvers in reprocessing plants, the temperature coefficient of reactivity is one of the most important neutronics quantities to evaluate the criticality safety margin. Soluble neutron poisons such as boric acid H3BO3 and gadolinium nitrate Gd(NO3)3 are useful to enlarge the capacity of the fuel dissolvers under the condition that the nuclear safety is assured. The operating conditions of the dissolvers widely vary with temperature, and with water-fuel volume ratio. Therefore, the accumulation of experimental data on the temperature coefficients of reactivity in these heterogeneous systems with soluble poisons is important for the advanced nuclear criticality safety design.

Experimental and computational studies on the temperature coefficient of reactivity have been performed using light-water moderated and reflected UO2 cores with soluble poisons such as boron and gadolinium. The experiments were carried out with the Tank-type Critical Assembly (TCA) in Japan Atomic Energy Research Institute (JAERI) from April 1988 to January 1989. The temperature coefficients of reactivity in the cores with soluble poisons were measured by changing temperature of the moderator and the reflector from the room temperature up to about 60°C. The geometries of the core configurations were simple. The experiments were well documented and carefully performed. There were no serious omissions of data. All experimental data regarding the temperature effect of reactivity are acceptable benchmarks. Light water moderated and reflected rectangular cores were constructed in the TCA to simulate the reactivity effect and temperature coefficient in a dissolver loaded with low enriched UO2 rods. The experiments were performed in order to obtain benchmark data of temperature effects on reactivity in simple cores with soluble poisons, and the dependence of temperature coefficients on the core configuration and the concentration of soluble poison in the moderator and reflector were mainly studied. The experiments were performed with respect to three kinds of cores, i.e., cores without poisons (named A-cores), ones with H3BO3 (B-cores), and ones with Gd(NO3)3 (C-cores).

3.6. DIMPLE Low Power Reactor

The DIMPLE is a low power reactor located at the U.K.A.E.A's Winfrith site. The DIMPLE facility consisted of a large aluminium primary vessel, 2.591m inner diameter, 4m high with a side-wall thickness of 0.65cm, surrounded by concrete shielding. Figure 5 (Reference [5]) shows the general view of the DIMPLE reactor. The reactor core was supported inside the tank by a steel chassis. A number of 'U' shaped beams were accurately arranged within the chassis to support two sets of aluminium lattice plates which it turn supported the fuel pins. The lower lattice plates were secured to the 'U' beams by a tubular stainless steel chassis. The upper lattice plates, approximately 60cm above the lower lattice plates, were attached to the 'U' beams by two support brackets. Two experiments have been accepted in the IRPhEP handbook as benchmark experiments – DIMPLE S01 and DIMPLE S06.

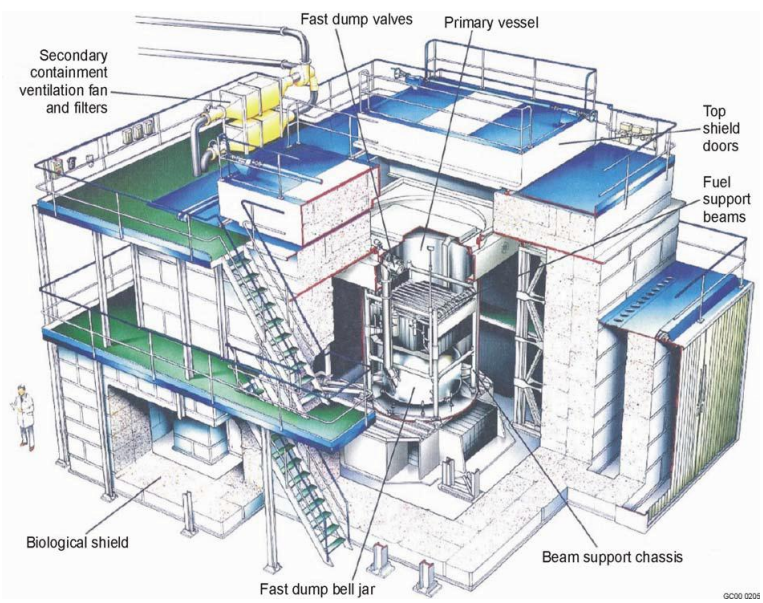

Figure 5. General View of the DIMPLE Reactor.

3.6.1. DIMPLE-LWR-EXP-001: Light Water Moderated and Reflected Low Enriched Uranium (3 wt.% 235U) Dioxide Rod Lattices DIMPLE S01

The DIMPLE S01 experimental program considered critical experiments with low enriched uranium dioxide fuel rods containing 3.0 wt.% 235U with light water moderation and reflection. These experiments were performed in the DIMPLE low power reactor at U.K.A.E.A's Winfrith site during 1983.

Comprehensive axial and radial reaction-rate distributions were measured with foils in the assembly S01A and these, along with fission chamber traverses (axial only), were used to derive experimental buckling values. The experimental configuration comprised a cylindrical array of fuel rods centrally located within a large aluminum vessel (2.6 m diameter and 4 m high) containing water. The array of rods was light water moderated and fully reflected to a critical height of around 50 cm above the base of the fuel stack in the rods. The fuel rods were located on a square pitch of 1.32 cm and were supported by an upper and lower lattice plate. The lower lattice plate was situated on an aluminum fuel support assembly.

3.6.2. DIMPLE-LWR-EXP-002: Light Water Moderated and Reflected Low Enriched Uranium (3 wt.% 235U) Dioxide Rod Lattices DIMPLE S06

The experimental program encompassed critical experiments with low enriched uranium dioxide fuel rods containing 3.0 wt.% 235U with light water moderation and reflection. The experiments were performed in the DIMPLE low power reactor at U.K.A.E.A's Winfrith site during the late 1980's and early 1990's. The experimental program extended previous studies in water-reflected cylindrical systems to power reactor geometries by assembling a cruciform array of 3% enriched uranium dioxide fuel pins. The array simulated the rectangular corner configuration of a Pressurized Water Reactor (PWR) and effectively represented twelve PWR fuel assemblies. Four primary versions of the cruciform assembly were constructed, the first being water reflected, as with the cylindrical systems. The assembly was then surrounded azimuthally by a stainless steel region simulating a PWR core baffle. A third version incorporated discrete burnable-poison pins and empty guide thimbles in a series of different arrays. The fourth version was an ex-core detector benchmark, which consisted of a realistic simulation of the core baffle, barrel, neutron shield pad and pressure vessel of a PWR. The experiments were performed in three separate phases, collectively known as the S06 series. The first phase, covering S06A and B, is the subject of this evaluation. The second phase, S06C, covered discrete burnable poison rods and empty guide thimble, consisted of 12 configurations, and is not the subject of this evaluation. The final phase, S06D, the ex-core detector benchmark study, consisted of 7 configurations. Neither S06C nor S06/D are described in the IRPhEP handbook.

3.7. VENUS Critical Facility

The VENUS critical facility is a zero power reactor located at SCK-CEN, Mol (Belgium). This facility was built in 1963-1964, as a nuclear mock-up of a projected marine reactor called VULCAIN; hence the name VENUS which means "Vulcain Experimental NUclear Study".
In 1967, this facility was adapted and improved in order to study LWR core designs and to provide experimental data for nuclear code validation. A great flexibility was looked for, as well as an easy handling of the fuel pins, handled one by one, while a great precision of the results had to be achieved.

In 1980, additional material was purchased with a view to studying typical 17×17 PWR fuel assemblies. Such an adaptation is easy: only new reactor grids and small devices adapted to the new fuel geometry are necessary. In 1982, special stainless steel pieces were manufactured in order to build a pressure vessel mock-up representative of a three-loop Westinghouse power plant. These first stainless steel pieces were delivered at the beginning of December 1982 and the reactor, loaded with this mock-up core, was made critical on 20 December 1982.

3.7.1. VENUS-PWR-EXP-001: VENUS-1 PWR UO2 Core 2-Dimensional Benchmark Experiment

The VENUS-1 experiment period was from 24 January 1983 to 23 June 1983. The VENUS-1 experiments were composed of (1) gamma scans to determine the core power distribution, (2) in-core and ex-core foil activations and (3) measurement of vertical bucklings in the core and core exterior. The VENUS-1 experiment was established for the validation of **2-dimensional** dosimetry analysis. In this format, the experiments were classified into three experiments, such as critical configuration, reaction-rate distribution measurement and power distribution measurements. The absolute power levels were described in the section of power distribution measurement.

The VENUS-1 experiment was established to build a pressure vessel mock-up representative of a 3 loop Westinghouse power plant. The experiments provide the measured reaction rates at dosimeters installed in various locations, pin-wise power distribution at the axial mid-plane, absolute power level and vertical buckling.

The evaluation of the RPV (Reactor Pressure Vessel) integrity for both PTS (Pressurized Thermal Shock) and end-of-life considerations requires the accurate determination of neutron fluence accumulated on the RPV. The evaluations of RPV integrity is validated through various measurements of the dosimeters extracted from the capsules.

The VENUS-1 experiments provide the reaction rate data measured at both in-core and ex-core locations. The measurement data are useful for the validation of the calculational methodology used for RPV irradiation analysis. In addition, as the VENUS-1 core configuration is similar to PWR core, the measured pin-wise power distribution is useful for the validation of fission source calculation.

3.7.2. VENUS-PWR-EXP-003: VENUS-3 PWR UO2 Core 3-Dimensional Benchmark Experiment

The VENUS-3 experiment was established to build a pressure vessel mock-up representative of a 3 loop Westinghouse power plant. The experimental configuration was made to be representative of typical irradiation conditions of a modern PWR vessel. The experiment was established for 3-dimensional dosimetry analysis. The activities were measured at various locations inside the vessel. The experiments provide the measured reaction rates at dosimeters installed in various locations and **three-dimensional** power distributions and absolute power level. The evaluation of the RPV (Reactor Pressure Vessel) integrity for both PTS (Pressurized thermal shock) and end-of-life considerations requires the accurate determination of neutron fluence accumulated by the RPV. The neutron influence can be validated through various measurements of the dosimeters extracted from the capsules. For some early built reactors, the PLSAs were loaded to reduce the neutron irradiation at the critical position of RPV. The VENUS-3 experiment was built for the mock-up of this modification. The VENUS-3 experiments provide the reaction-rate data measured along axial direction at both in-core and ex-core locations. Thus, the

experimental data are useful for the validation of three-dimensional neutron transport calculation. In addition, the measured pinwise power distribution is useful for the estimation of three-dimensional power distribution calculations. The criticality of VENUS-3 reactor was achieved to measure the reaction-rate and neutron source distributions in steady state condition. The VENUS-3 experiment was classified into three measured experiments: as a critical configuration, as a reaction-rate distribution measurement, and as power distribution measurements.

The VENUS-3 core evaluated in the IRPhEP handbook was made critical for the first time on March 16, 1988. The experimental program started on March 29, 1988 and was planned to run until the end of December 1988. At a later date, the facility was modified with the installation of pressure vessel internals, for the purpose of providing benchmarks for the PWR pressure surveillance programme, partly supported by Westinghouse Electric and the US Nuclear Regulatory Commission (USNRC). This was the VENUS-LWR-PVS (Light Water Reactor – Pressure Vessel Surveillance) programme that comprised three mock-up configurations of the core periphery of a PWR. This programme was carried out between 1983 and 1989. The LWR-PVS benchmark experiment in VENUS is aimed at validating the analytical methods needed to predict the azimuthal variation of the fluence in the pressure vessel.

3.7.3. VENUS-PWR-EXP-005: Experimental Study of the VENUS-PRP Configurations No. 9 and 9/1

The VENUS-PRP Configurations 9 and 9/1 were designed to study boundary effects between zones with different plutonium content and the influence of perturbations at the boundary. The following nuclear parameters were measured: criticality, spectral index $\sigma_f^{239}/\sigma_f^{235}$, power distributions within fuel rods and by regions, power sharing (MOX/UO2 power ratio).

The VENUS-PRP critical experiments' first aim of the programme was to provide experimental data for the validation of group cross-sections and design methods for Light Water Reactors (LWRs) with UO2-PuO2 mixed oxide (MOX) fuel.

The scale of the whole VENUS-PRP programme can be gauged through some relevant figures: The fuel stock for the VENUS facility consisted of 2800 fuel rods divided into several different types of chemical composition (UO2, UO2-PuO2, UO2-Gd2O3) and three different fabrication methods (pellets, homogeneous and heterogeneous vibro-pack oxides). It also included an additional 6600 pellets of eight different kinds used either separately, or in fuel rods capable of being disassembled. In order to make the daily operations easier, each fuel was named by a fraction with the 235U enrichment of UO2, as the numerator and the PuO2 content in the MOX as denominator. Both figures are expressed in wt.% and are nominal or rounded to the nearest integer. Criticality was achieved in approximately 100 different core configurations, by adjusting the water level. Experiments included, amongst others:
- 100 measurements of k_{eff};
- 200 irradiations;
- 4000 γ-scans of fuel rods;
- 120 measurements of reactivity;
- 500 measurements of the fine structure of the fission power distribution.

The VENUS-PRP Configuration 9 focused on the study of the power distribution across the boundary between a standard UO2 fuel region, enriched to 4% 235U (4/0 type) and a MOX fuel region made of UO2 enriched to 3% 235U with ~ 1% PuO2 (3/1 type), simulating a one cycle

burnt fuel (i.e. fuel with a burnup corresponding to that achieved after a single cycle of irradiation, or the order of 10 GWd/tHM).

3.7.4. International Benchmark Exercise of VENUS-2 MOX Core Measurements

To better understand the behavior of MOX fuel in challenging situations such as multiple recycle and high burn-up of plutonium in PWRs, the OECD/NEA launched a blind international benchmark exercise for the prediction of power distribution in the two-dimensional VENUS-2 MOX core experiments in 1999 (Reference [7]). This experiment-based benchmark was completed in 2000. The results showed that the calculations overestimated fission rates of MOX pins and slightly underestimated those of UO_2 pins. A three-dimensional VENUS-2 MOX core benchmark was then launched in 2001 for a more thorugh investigation of the calculation methods for MOX-fuelled systems(Reference [8]). Twelve participants contributed to the three-dimensional benchmark exercise, providing more than 20 solutions. Deterministic codes such as TORT, DANTSYS and PARCS as well as Monte Carlo codes were used by the participants. Figure 6 shows the VENUS-2 core geometry. In this benchmark exercise, both cell calculations and core calculations were performed. The K_∞ and reaction rates (absorption and fission) per isotope were calculated in the cell calculations for each fuel cell type. In the core calculations, the K_{eff} and normalized radial fission rate distribution at 325 fuel pin positions and normalized axial fission distribution of six fuel pins were calculated. The calculated resulted were compared with measured values.

Both the 2-D and 3-D VENUS-2 MOX core benchmark exercises provide excellent case studies for CASL to perform validation studies of its basic nuclear data and calculational methods and to identify possible improvements in nuclear data and physics modeling methods.

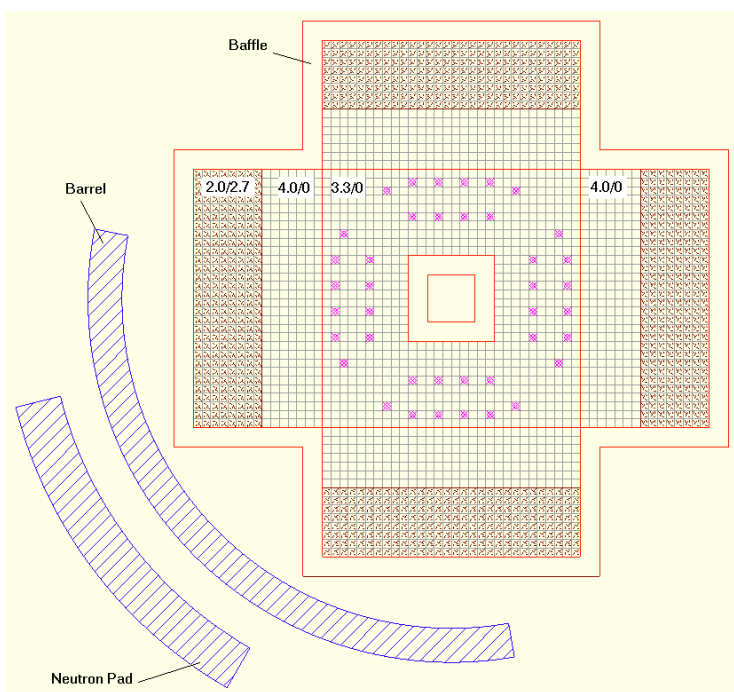

Figure 6. VENUS-2 Core Geometry (Reference [8]).

3.8. LR0-VVER-EXP-001: VVER Physics Experiments: Hexagonal Lattices (1.22-cm Pitch) of Low-Enriched U(2.0 - 3.3 WT.% U235)O2 Fuel Assemblies in Light Water with Central Control-Assembly Mockup

The LR-0 reactor is located in Řež near Prague in the Czech Republic. The first criticality was reached in December 1982. The reactor LR-0 is an experimental light-water-moderated zero-power reactor, originally designed for the research of the VVER-1000 and VVER-440 type reactor cores, spent-fuel storage lattices and benchmark experiments. The LR-0 fuel pins are shortened to about 1/3 the length of pins in VVER Nuclear Power Plants (NPPs). The fuel pins are available with enrichments 2 wt.%, 3 wt.%, 3.3 wt.%, 3.6 wt.%, and 4.4 wt.%. The LR-0 reactor fuel inventory consists of approximately 21000 VVER-type fuel pins.

Criticality measurements of different VVER cores were performed on the experimental zero-power reactor LR-0 (Nuclear Research Institute Řež plc) within the framework of projects for VVER-440 and VVER-1000 type core loadings. The main goal of these projects was to complete the experimental tasks, which could be utilized for verification of codes and libraries used for neutronic calculations in safety analyses of criticality of cores, storage, and transport-cask lattices for VVER type reactors. The proposed benchmark describes the experimental investigation of VVER-440 control-assembly induced power peaking in neighboring fuel assemblies. The experimental series, performed in 2004 to 2006, looked at three different modifications of a mockup of a control assembly placed at the center of the reactor: without Hf (old model used in NPPs), with 0.4-mm Hf plates (only for experimental purpose) and with 0.6-mm Hf plates (new model introduced into NPPs).

4. POST-IRRADIATION EXAMINATION (PIE) ISOTOPIC MEASUREMENTS

To accurately predict the isotopic composition of fuel being irradiated (depleted) in a nuclear reactor is a key factor in the reload design, licensing and operation of a nuclear power plant. It is equally important in the design, licensing, and operation of radioactive waste transportation systems, interim storage facilities, and a final repository site. To validate the CASL neutronics codes and data used in depletion calculations, experimental measurements of isotopic data are compared with numerical predictions for relevant post-irradiation fuel samples. The measured isotopic data are difficult to obtain. Fortunately, ORNL's SCALE project maintains a database to benchmark and validate the SCALE code package for its predictions of PWR spent nuclear fuel isotopic compositions. The spent fuel isotopic characterization validation reports (Reference [9]) can be found at http://scale.ornl.gov/validation_spent.shtml. As discussed in Reference [9], there are 118 PWR spent fuel samples available from low-, moderate-, and high-burnup spent fuel assemblies from nine PWRs. Table 5 summarizes the available spent fuel samples (Reference [9]). The initial fuel enrichments and burnup values for the samples considered vary from 2.453 to 4.657 wt % ^{235}U and from 7.2 to 70.4 GWd/MTU, respectively. The data cover a wide range of burnup conditions. The data is readily available at ORNL for CASL to validate its neutronics codes depletion calculations. In reference [9] the SCALE 2-D depletion mdoles and the TRITON/NEWT models were developed for the spent fuel samples listed in Table 4. The depletion calculation results were reported as the ratio of the experimental-to-calculated isotopic concentrations to provide the isotopic cpmposition validation methodology for SCALE. Analagous calculations can be carried for these spent fuel samples with CASL neutronics codes to validate CASL's depletion calculation methodologies.

Table 4. Available PWR Spent Fuel Samples for CASL Neutronics Codes Depletion Calculations Validation (Reference [9])

Reactor	Assembly Design	Enrichment (wt % U-235)	No. of Samples	Burnup (GWd/MTU)
Trino Vercellese	15 x 15	2.719, 3.13, 3.897	31	7.2 – 25.3
Kernkraftwerk Obrigheim (KWO)	14 x 14	2.83, 3.00, 3.13	27	15.6 – 37.5
Turkey Point-3	15 x 15	2.556	5	30.5 – 31.6
H. B. Robinson-2	15 x 15	2.561	4	16.0 – 31.7
Calvert Cliffs-1	14 x 14	2.453, 2.72, 3.038	9	18.7 – 46.5
Takahama-3	17 x 17	2.63, 4.11	16	7.8 – 47.3
TMI-1	15 x 15	4.013, 4.657	19	22.8 – 55.7
Gösgen	15 x 15	3.5, 4.1, 4.3	6	29.1 – 70.4
GKN II	18 x 18	3.8	1	54.1

5. OPERATING PLANTS DATA

Since TVA is a core partner of CASL and Watts Bar unit 1 and 2 data can be made available to CASL, these operating plant data provide valuable data source for CASL neutronics codes validation. The startup physics test data and operating plants measurement data can be used to perform validation calculations.

5.1. Startup Physics Test Data

The beginning of cycle (BOC) hot zero power (HZP) startup physics test measurement data such as: 1). critical boron concentration, 2). isothermal temperature coefficient, and 3). control rod worth, etc. The startup physics tests measurement data were obtained following startup physics tests procedure. The critical boron concentrations are measured at bite position. Next, the control rods are withdrawn to all-rods-out (ARO) position and then the reactivity change is measured by a reactivity meter. The reactivity change is converted to an equivalent boron concentration by dividing it by the calculated boron worth and is added to the above measured critical boron concentration. The value of the isothermal temperature coefficient is obtained by changing the inlet temperature and measuring the reactivity change. For the measurement of the control rod worth, Dynamic Rod Worth Measurement (DRWM) or the dilution/boration method is used. DRWM method inserts the control rod continuously without boron concentration change and measures the reactivity change. The dilution/boration method changes the boron concentration continuously and compensates the reactivity change by the control rod. The reactivity change is measured by a reactivity meter. (Reference [10])

5.2. Hot Full Power Operating Measurement Data

The hot full power operating measurement data include: 1). Critical boron concentration versus burnup, 2). Assembly average power distribution, and 3). axial power distribution. TVA can provide such measurement data for Watts Bar unit 1 and 2 to provide comparisons for code predictions.

6. CONCLUSIONS AND FUTURE WORK

Large number of experiments have been carried out worldwide in various facilities in the past several decades, which have provided invaluable data to validate neutronics codes for nuclear reactor design work. There have also been internationally coordinated efforts such as ICSBEP and IRPhEP to preserve the data and knowledge learned from those experiments. These projects are very beneficial for CASL's neutronics validation work.

This document has identified the available data for CASL neutronics validation. When the CASL neutronics code such as DENOVO is further developed to a mature stage, the actual validation calculations can be performed using the measurement data from the experiments identified in this document.

All the experiments identifided in Section 2 for Critical Experiments can be used to validate CASL neutronics codes prediction capability of the critical eigenvalue.

Among the experiments identified in Section 3, which are available from International Reactor Physics Experiment Evaluation Project (IRPhEP) handbook, the VENUS benchmark experiments and the KRITZ benchmark experiments have been well studied previously with various other neutronics codes. For example, the VENUS-2 experiments have been studied to validate the DeCart code and the NEWT code previously (references [11] & [12]). These present the best cases to validate CASL neutronics codes' capability to calculate critical eigenvalue, power distribution, kinetics parameters, isotopic compositions, etc. Therefore these two benchmark experiments should be first used to perform actual CASL neutronics codes validation calculations.

PIE data identified in Section 4 can be used to validate CASL neutronics codes depletion calculations. TVA's Watts Bar unit 1 & 2 plant data can be used to validate CASL neutronics codes capability to predict real plant operations.

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