Proceedings of the Experts Meeting on

EXPERIMENTAL NEEDS IN CRITICALITY SAFETY

Albuquerque, New Mexico, USA
25-26 September 1995

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-PERATION AND DEVELOPMENT

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter; Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all OECD Member countries, except New Zealand and Poland. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of the NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

- *encouraging harmonization of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
- *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
- *developing exchanges of scientific and technical information particularly through participation in common services;*
- *setting up international research and development programmes and joint undertakings.*

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

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CRITICALITY SAFETY FACILITIES DESCRIPTION

United States and Canada

CRITICALITY FACILITIES AT THE LOS ALAMOS NATIONAL LABORATORIES

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REVIEW OF CRITICALITY EXPERIMENT FACILITIES IN JAPAN

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Abstract

Construction of two types of criticality experiment facilities, Static Experiment Criticality Facility (STACY), and Transient Experiment Critical Facility (TRACY), has been completed at the Nuclear Fuel Cycle Engineering Research Facility (NUCEF) of JAERI Tokai Establishment. STACY and TRACY were created in order to facilitate nuclear criticality safety research related to the reprocessing of light water reactor spent fuels. They handle low-enriched uranyl-nitrate solution fuel or plutonium-nitrate solution fuel, and thus are considered to be solution-type criticality facilities.

The JAERI Tank-type Critical Assembly (TCA) performs various reactor physics experiments on LWR fuel rod systems. The TCA also conducts criticality safety experiments related to fuel storage and transportation systems, and further involves itself with regard to the development of subcriticality measurement techniques.

In the Kyoto University Critical Assembly (KUCA), many fundamental criticality safety experiments have been performed using A-core and B-core with solid moderators such as polyethylene and graphite. Other experiments have also been performed using C-core and light water moderators.

The Power Reactor and Nuclear Fuel Development Corporation's (PNC) Deuterium Critical Assembly (DCA) has been conducting reactor physics experiments on ATR until just recently. The DCA facility was modified to perform subcriticality measurements; the goal of this modification is to contribute to the criticality safety management of nuclear fuel cycle facilities.

A more detailed description of the function and activities of each of the above-mentioned facilities follows in this paper

Introduction

Japan Atomic Energy Research Institute (JAERI) began conducting criticality safety studies around 1980, doing so with specific objectives in mind. Included among these objectives were developing calculation methods, acquiring experimental data, and preparing standards needed for criticality controls of out-of-reactor facilities, specifically in the down-stream of the nuclear fuel cycle.

Construction of two types of criticality experiment facilities, Static Experiment Criticality Facility (STACY), and Transient Experiment Critical Facility (TRACY), has been completed at the Nuclear Fuel Cycle Engineering Research Facility (NUCEF) of JAERI Tokai Establishment. STACY and TRACY were created in order to facilitate nuclear criticality safety research related to the reprocessing of light water reactor spent fuels. They handle low-enriched uranyl-nitrate solution fuel or plutonium-nitrate solution fuel, and thus are considered to be solution-type criticality facilities.

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The Power Reactor and Nuclear Fuel Development Corporation’s (PNC) Deuterium Critical Assembly (DCA) has been conducting reactor physics experiments on ATR until just recently. The DCA facility was modified to perform subcriticality measurements; the goal of this modification is to contribute to the criticality safety management of nuclear fuel cycle facilities.

A more detailed description of the function and activities of each of the above-mentioned facilities follows in this paper. For a brief summary, refer to Table 1.

STACY

STACY is designed to conduct critical experiments performed with a low-enriched nitrate solution, a plutonium solution and a mixture of the two [A1,A2,A3]. The main purpose of this facility is to obtain fundamental critical data for verification of the criticality safety calculation code and the nuclear data library, and also to evaluate the safety margin of the criticality safety design. Another function STACY fulfils is the accumulation of systematic data of kinetic parameters, including prompt neutron life time, delayed neutron fraction and reactivity coefficient of temperature which dominates the transient phenomena in the abnormal condition.

Experimental techniques for subcriticality measurement are also being investigated at STACY. This is being accomplished through the use of either a single- or multiple-unit core system containing fuel solution.

Uranyl-nitrate solutions of 10 wt.% and 6 wt.% enrichment are used and maximum uranium concentration amounts to 500 gU/l. Plutonium concentration is limited to 300 gPu/l. Many core configurations are being tested, including isolated cylindrical and slab type tanks, two-unit interacting

Table 1. Summary of criticality safety experiment facilities in Japan

FACILITY	FUEL	POTENTIAL OF EXPERIMENTAL POSSIBILITIES	RESEARCH PLAN	NOTE
STACY	Uranyl-nitrate solution (10.6 wt%)	Evaluation of criticality safety margin for nuclear fuel cycle facilities ^{235}U enrichment, fuel concentration, acidity	Basic studies on criticality of homogeneous and heterogeneous core	(Experimental technique)
	Plutonium-nitrate solution U+Pu mixture	Plutonium enrichment Soluble poison	Evaluation of neutron interaction effect	Solution level worth method Pulsed neutron source method
	PWR-type fuel rod	Kinetic parameters Solution temperature	Reactivity effect of structural material and soluble poison	Reactor noise method
TRACY	Uranyl-nitrate solution (10 wt%)	Measurement of total fission (Inserted reactivity, Reactivity addition rate) Measurement of radiation dose and radionuclide transport	Fundamental property of reactivity feedback mechanism Exposure evaluation Confinement of fission product	Pulse neutron source method
TCA	UO_2 fuel rod MOX fuel rod	Criticality condition of heterogeneous system Water to fuel volume ratio Fixed absorber Soluble poison	Reactivity of fission product Development of subcriticality measurement technique	Water level worth method Exponential method Source multiplication method
KUCA	Highly-enriched uranium	Measurement of large subcriticality Experiments on tight-pitch lattice core and coupled core Measurement of effective delayed neutron fraction	Basic studies on coupled core system Reactivity of non-uniformly distributed core system Development of subcriticality monitor	Feynman-α Pulse neutron source method
DCA	UO_2 fuel MOX fuel	Rational management for the nuclear fuel cycle facilities Subcriticality measurement Enriched uranium or plutonium fuel pin Volume ratio of fuel to moderator Share due to fuel arrangement	Development of subcriticality monitoring system	Feynman-α Mihalczo method

tanks (which will be accomplished by changing the tank dimension) and fuel concentration. Tests are not conducted exclusively with a homogeneous core of fuel solution; a series of experiments is also being performed on a heterogeneous core containing an array of low enriched oxide fuel rods immersed in the fuel solution. In addition to altering fuel conditions such as concentration, acidity and temperature, many types of reflectors and soluble poisons are used to measure the reactivity effects of these materials. STACY achieved initial criticality on February 23, 1995, and the first series of experiments was initiated with a 10 wt.% enrichment uranyl nitrate solution and a cylindrical tank of 60 cm in diameter [A4].

TRACY

TRACY is designed to perform transient experiments above super criticality in order to investigate the physical and chemical characteristics of criticality accidents. Experiments simulate criticality accidents with regard to the low-enriched uranyl-nitrate solution handled in reprocessing plants. The main purpose is verification of the ability to contain radioactive materials and to confirm the total energy release in the reprocessing plant. In the transient experiments, the feedback mechanism of reactivity and total amount of energy release are investigated by changing parameters such as solution concentration, reactivity addition rate, inserted reactivity and initial neutron density. A cylindrical tank of 50 cm in diameter is used with and without reflector. A uranyl-nitrate solution of 10 wt.% enrichment is initially used and maximum concentration is 500 gU/l. TRACY’s operation consists of static operation and transient operation. The static operation is similar to that of STACY, while the maximum power is 10 kW. In the transient operation, one of three operation modes (i.e. pulse operation by rapid withdrawal of the transient rod, ramp operation by slow withdrawal of the transient rod, and ramp operation with a slow feed of fuel solution to the core tank) is chosen. The maximum reactivity and total fission per operation is three dollars and 1×10^{18} fissions, respectively [A5]. TRACY is scheduled to attain the initial criticality by the end of 1995 and to commence the transient operation by early 1996 in order to obtain licensing from the Science and Technology Agency (STA) of Japan.

TCA

Experimental studies using the Tank-type Critical Assembly (TCA) have played an important role not only in providing measured data but also for reactor physical understandings of criticality parameters. This machine can be used as a very effective means to mock up several materials and configurations encountered in nuclear facilities with various neutronic conditions. The following are the principal areas of criticality safety studies currently utilising the TCA.

Development of subcriticality determination method

Knowledge of a degree of subcriticality is a fundamental aspect of criticality safety control. This kind of knowledge – if obtained in a timely manner during the operation of nuclear facilities – could lead to the application of advanced control methods such as a more positive usage of neutron absorbers.

For the purpose of comparing the capability of subcriticality determination, several subcritical experiments have been examined, such as exponential [Bl,B2], pulsed neutron source [Bl,B2], neutron noise [B3,B4] and neutron source multiplication [B5] experiments. From the conclusions of these experiments, it should be noted that in order to obtain the widely used k_{eff} value of relatively higher subcritical states, theoretical corrections through some neutron balance calculations are always required. This means that detailed information regarding the physical properties of the nuclear system must be given in advance, restricting the effective area of subcritical measurement to a narrow range. In an attempt to surmount this problem, a new method based on neutron multiplication was recently proposed, in which intensities of higher energy prompt gamma-rays associated with neutron emissions from a neutron source and its induced fission are correlated to observe a degree of subcriticality defined by using the real neutron multiplication [B6].

Basic studies on several criticality parameters

As a means of contributing to the advancement of criticality control methods utilising neutron absorbers, reactivity effects of fixed absorbers and soluble poisons were measured in systematic experiments. It is of some interest that the effect of uniformly distributed small absorber rods [B7] is treated as a decrease in the infinite neutron multiplication factor or the material buckling, while that of localised large absorber plates [B8] is regarded as a decrease in the non-leakage probability of neutrons or an increase in the geometric buckling in the direction crossing the absorber plate. The latter observation was made not only with regard to the measurements of the neutron isolation effects of water [B9], concrete [B10] or borated stainless steel [Bl1] layers between coupled cores and of several materials inside an annular core [B12], but also in the experiments of steel [B13] and concrete [B10] reflectors. As for the effects of soluble poisons, experiments concerning the temperature coefficient of reactivity were carried out with Gd and B solved in water around fuel lattices [B14], following consideration of the case of homogeneous cores [B15]. It was discovered that Gd has an effect of shifting the coefficient towards positive, whereas B (l/v-absorber) has no significant effect on either core.

Another important domain of the TCA experiments concerns abnormal conditions in criticality safety control. The reactivity effects of non-uniform fuel distribution [B16] and deformation of core configuration (sloshing effect in earthquakes) [B17,B18] are typical examples. The latter experiment gave rise from the need to determine the geometric buckling Bg^2 of general core configurations. For this purpose, Bg^2 values of some regular polygons were measured [Bl9], and theoretical studies concerning the core deformation were developed [B20,B21,B22].

In these TCA experiments, reactivity (in most cases) is measured from the change in buckling [B8]. The necessary condition for this measurement – that the buckling coefficient of reactivity be almost unchanged between the two states, before and after the perturbation – has been confirmed experimentally [B23] and theoretically [B24]. These experiments have also resulted in the production of benchmark data for the verification of calculation codes and the nuclear data library.

One example is the measurement of conversion ratios in various neutron spectra [B25,B26,B27,B28], which pointed to the possibility of over-estimation of the ^{238}U capture cross-section, especially within the resonance range. Furthermore, in an effort to develop calculation methods usable for rapid prediction of criticality safety conditions, the boundary element method (BEM) has been applied to solve the neutron diffusion equation [B29,B30]. This method will likely become a useful tool not only in facilitating prediction of experimental conditions but also in the monitoring of on-site subcriticality in nuclear facilities.

KUCA

The Kyoto University Critical Assembly (KUCA) is a multi-core-type critical assembly established in 1974 as a facility conceived for the joint use and study by researchers from all Japanese universities. Among the three cores of KUCA, solid moderators such as polyethylene or graphite are used in the A- and B-cores, whereas light-water is used as the moderator in the C-core. A Cockcroft-Walton-type accelerator is also installed on the premises.

As a rule, the KUCA is operated from 9:30 to 17:00, from Tuesday to Friday every week. In the fiscal year 1994, total operating time was 114 days (871 hours), and 149 researchers (including students) participated in various forms of experimentation.

With regard to reactor physics experiments, several studies on nuclear criticality safety have been carried out with KUCA. Using the A-core in connection with the Cockcroft-Walton-type accelerator, the large subcriticality was measured by the pulsed neutron method [C1,C2,C3,C5]. Using the C-core, the criticality of the coupled-core was systematically studied as a function of the core separation distance [C3,C4,C6,C7]. In the C-core, the Feynman-α method was successfully applied to the measurement of large subcriticality [C8], the area ratio method was developed for the subcriticality measurement in a coupled-core system [C10], and the source multiplication method was examined for application to the subcriticality measurement [C12]. Using the B-core, the reactivity effect of a non-uniformly distributed fuel system was systematically examined [C9,C13,C18], the fuel bunching effect was measured in a tight-pitch lattice core [C11], and the effective delayed neutron fraction was measured by the covariance method [C19]. The development of a subcriticality monitor is being pursued through the use of the Feynman-α method in the B-core [C14-17,C20].

DCA

DCA (Deutrium Criticality Assembly) was constructed in 1969 as part of experimental facilities for the research and development programme of the Advanced Thermal Reactor (ATR).

Diameter of core tank (10 mm thick AL) is about 3 m and height is 3.5 m. Fuel and other core set-ups are arranged between lower and upper Al grid plates, and are suspended as a whole by the upper grid plate and tightened by the lower grid plate in the tank. Operation or control of DCA is mainly accomplished by changing the heavy water level.

Recently, the subcriticality measurement has been performed by employing DCA to develop a subcriticality monitoring system which contributes to the criticality safety design for nuclear fuel cycle facilities. In order to carry out this experiment, the DCA facility required some modification. To this end, the subcriticality test region was equipped with fuel pins and moderator at the central part of the DCA core. Since the area outside of the test region is used as the driver region of a reactivity through the utilisation of ATR type fuels, only the independent test region cannot achieve the critical.

The subcriticality measurements due to the reactor noise method such as the Feynman-α method, the Mihalczo method and so on, will be performed by using this modified DCA facility.

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Europe

THE CRITICALITY SAFETY LABORATORY OF VALDUC (FRANCE): POTENTIALITY, FACILITIES AND EXPERIMENTAL POSSIBILITIES

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Abstract

In order to control the criticality risk in laboratories or plants where nuclear material in sufficient quantities are present, qualified calculation codes are needed, both for the design of these facilities and for safety assessment purposes. Therefore, reference experiments (also called "benchmarks") are required, particularly since the nuclear industry is seeking to improve the economy of these facilities, while reducing the constraints linked to the criticality hazard. The consideration of new elements in criticality calculations, such as, for example, the presence of fission products, requires experimental qualification.

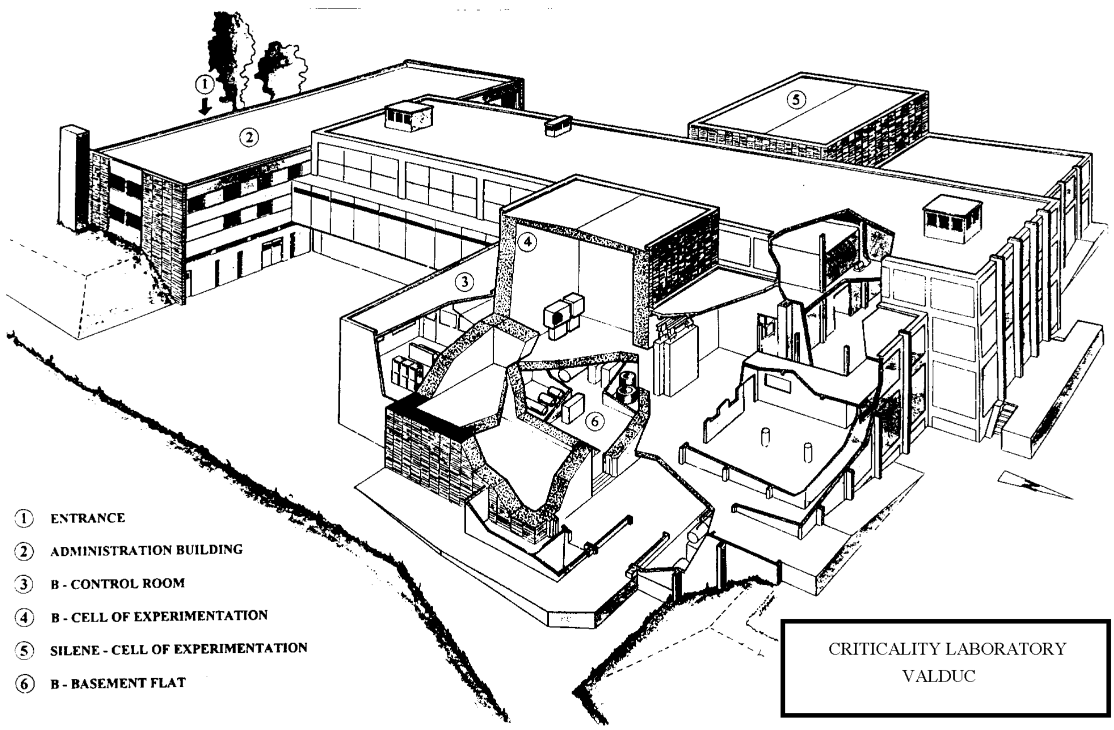
The Criticality Safety Laboratory of Valduc is the answer to the IPSN's mission of safety assessment in the field of criticality risk control. Its considerable potentiality with respect not only to its skills and expertise, but also to its facilities and related equipment, enables it to perform representative criticality experiments of high quality.

Technical potential: Criticality facilities, available nuclear materials and related equipment

The Criticality Laboratory of Valduc (Fig. 1) was built in the sixties.

Today, it chiefly accommodates three criticality facilities, called “B Apparatus,” Silene, and Maracas, which are supported by related equipment that is essential for the preparation, characterisation, reprocessing, and management of the nuclear materials required for the execution of experiments. A criticality laboratory must, therefore, have a fairly “heavy” minimum technical infrastructure at its disposal, whatever the desired number of experiments to be performed. We shall see that the same holds true for requirements in human potential. The respect of regulations also results in constraints and demands both technical and human.

Figure 1. General view of the Valduc Criticality Laboratory



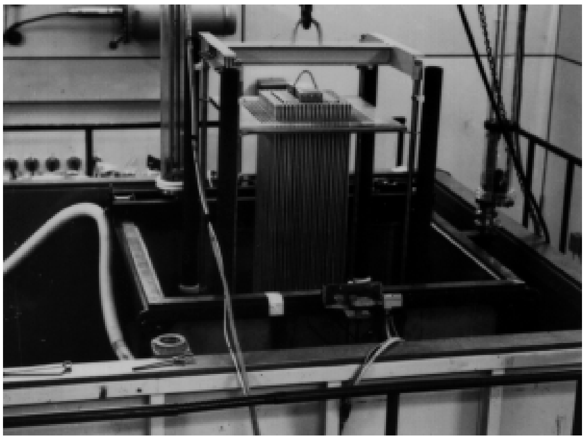
Criticality facilities

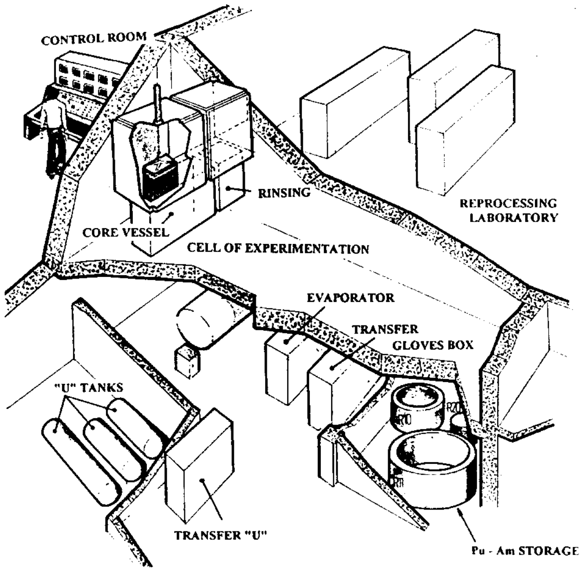
The “B Apparatus” (Fig. 2)

This is a sub-critical facility that enables the reproduction of a great diversity of criticality configurations representative in particular of the storage, transport, and reprocessing of fissile materials in the form of rods or fissile solutions, with or without the presence of a moderator or a neutron poison (boron, gadolinium, fission products, etc.).

The fuel elements of reactor “type” (clad sintered uranium oxide rods, and clad U-Pu oxide rods) can be separated by shielding, immersed in a solution, surrounded by reflectors, or placed in a moderating medium. The subcritical approach is performed by progressively introducing water or a solution via the bottom of the experimentation tanks.

Figure 2. General view of the "B Apparatus" critical facility





A Pu processing laboratory enables purifying the solution in resin columns and concentrating the solutions with evaporators.

The facility has performed more than 3000 experiments since its construction. It is now undergoing extensive renovation work that will be completed at the beginning of 1997, in order to meet the needs of future experimental criticality programmes as defined by French specialists.

The "Maracas" Machine (Fig. 3)

Maracas is a subcritical facility of "split-table" type (a system for bringing two elements progressively closer to each other), able to contain up to six tonnes of nuclear material on two tables, one of which is stationary, while the other is mobile. This facility was used from 1982 to 1986 to meet the requirements of a criticality programme implementing over five tonnes of UO_2 powder with variable moderation ratios.

This type of facility is perfectly suited to the performance of criticality experiments focused on mediums of "powder" or "metal" types.

Maracas is shut down at present.

The Silene pulsed reactor (Fig. 4)

The Silene experimental reactor was constructed to study criticality accidents and their radiological consequences.

Silene operates with a liquid fuel (93% ^{235}U enriched uranium nitrate) that is introduced into an annular tank located in a large concrete test room. The solution is introduced up to supercritical level in the presence of an absorber rod. Divergence is achieved by ejection of the rod. The reactor is shut down by draining the solution into a storage tank.

Figure 3. Maracas "split-table"

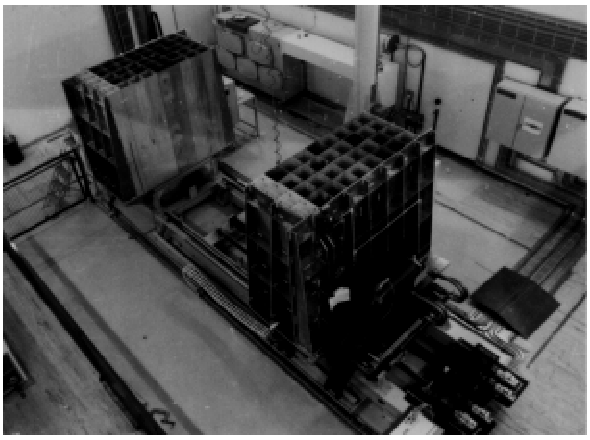
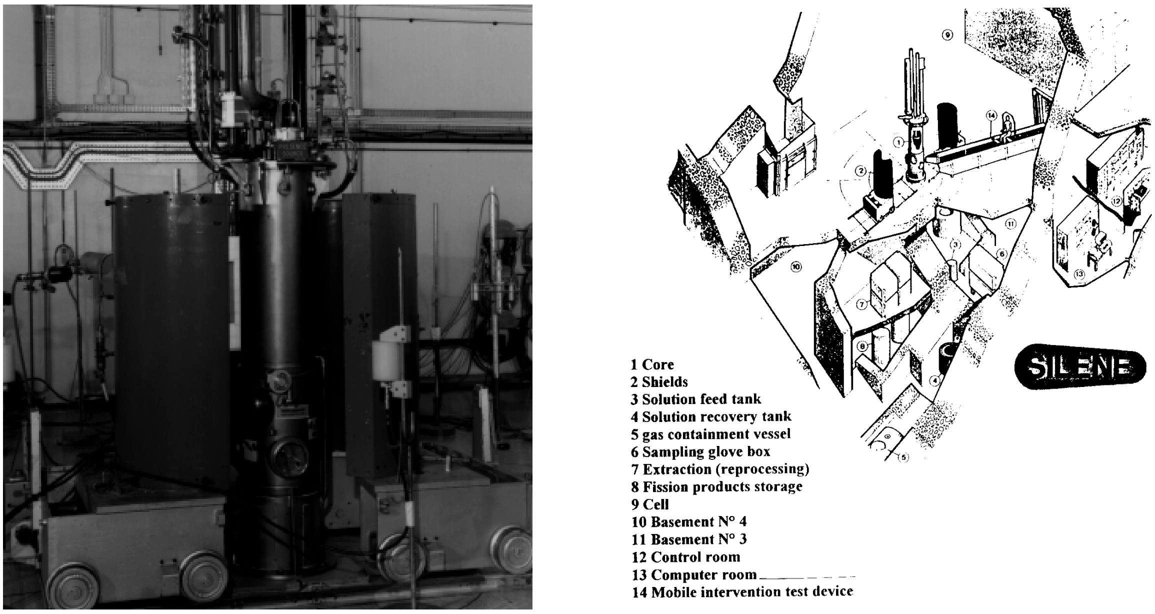


Figure 4. General view of the Silene reactor



An uranium processing laboratory enables extracting fission products from the uranium solutions on banks of mixer-settlers and concentrating the solutions in an evaporator.

The kinetics of the power excursions that can be reproduced vary from a few minutes to a millisecond. In a "pulse" type operation, the authorised reactivity step is \$3 whereas in a "reactivity ramp" operation, up to \$7 can be inserted.

Available nuclear materials

At present, the potential of the Valduc Criticality Laboratory in fissile materials is as follows:

- Approximately 50 kg of plutonium in the form of plutonium nitrate solution (76% ^{239}Pu and 20.6% ^{240}Pu);

- 27 kg of uranium in the form of enriched uranyl nitrate (93% ^{235}U);
- 1300 LWR-type rods containing enriched UO_2 (4.75% ^{235}U);
- 2500 “High Burn-up” fuel rods, containing UO_2-PuO_2 mixed oxide (1.1% Pu and 1.57% ^{235}U).

Other materials can of course be brought to the laboratory according to need for specific experiments, but the range of available fuels is proof of the laboratory’s existing possibilities.

Related equipment

In order to perform high quality criticality experiments that can serve as benchmarks, and to be able to meet requirements with respect to regulations and the management of nuclear materials, it must be possible to:

- Prepare nuclear materials under requisite experimental conditions;
- Characterise these materials (concentration, acidity, isotopic analyses, etc.) using appropriate physico-chemical methods;
- Reprocess them in chemical engineering facilities to purify them of either fission products, neutron poisons, or the americium that is formed; and
- Manage them from the physical monitoring point of view, as well as from that of the effluents and wastes generated during their various treatments.

The building that accommodates the IPSN criticality facilities therefore comprises two types of equipment:

- The conventional equipment of any basic nuclear facility, that is to say:
  - A ventilation system with inlet and outlet filters;
  - A remote monitoring network that manages the information from sensors designed for non-nuclear conventional safety purposes (fire detectors, flooding, etc.);
  - A radiological protection surveillance system connected to area monitoring devices (irradiation, contamination, etc.);
  - Nuclear material storage;
  - A scientific computer calculation network.
- Specific equipment related to both the experimentation and materials:
  - Physico-chemical analysis laboratories for the uranium and plutonium;
  - Nuclear radiation measurement laboratories (α, β, γ);
  - Reprocessing laboratories for the uranium and plutonium solutions;
  - Facilities for waste and effluent management;
  - An electronics and nuclear instrumentation laboratory.

Human potential: Expertise and specialisation

Expertise and specialisation must be suited to the specificities and demands of the activities described above. Therefore, the Criticality Laboratory has at its disposal personnel skilled in a wide range of fields:

- Physics/Neutronics:

  - Interpretation of experiments;
  - Modelling;
  - Design of technical devices;
  - Use and development of specific techniques related to criticality;
  - Measurement of nuclear radiation (α, β, γ).

- Chemistry:

  - Operation of chemical engineering facilities;
  - Physico-chemical analyses for nuclear material characterisation;
  - Operation and management of waste and effluent processing facilities.

- Electronics:

  - Design of instrumentation and control systems for criticality facilities;
  - Implementation of nuclear instrumentation.

- Nuclear facility operation:

  - The control and operation of nuclear facilities necessitates specific personnel training (reactor operators, etc.).

In more general fields, other skills are needed in order to meet regulations or safety requirements, as for example:

- The protection of nuclear materials;
- Quality assurance;
- Protection;
- Nuclear Safety;
- Administration and management.

Assets and potentiality of the Valduc Criticality Laboratory

The above description of the technical and human potential of the IPSN/Valduc Criticality Laboratory demonstrates that the facility can rely not only on its technical assets, but also on the expertise of its personnel to meet both present and future experimentation needs in the criticality domain. The “B Apparatus” enables the reproduction of highly varied criticality configurations involving fuel rods, solutions, reflectors, neutron poisons, shielding, etc., and, on site, there is a practically complete range of nuclear materials available for performing experiments at short notice. The Maracas split table could be reused if a need for experiments implementing fuels in the form of powders were to be expressed.

The Silene reactor provides the means of acquiring basic knowledge of the consequences of an accidental criticality excursion. Thanks to this facility, important studies are being carried out in the fields of radio-biology, dosimetry, intervention and the testing of criticality accident detectors. Even if all precautions are taken in nuclear facilities for protection against the risk of criticality, the possibility of such an accident can never be completely ruled out, and it is therefore important to have access to data bases that enable evaluating the consequences of an accidental situation and developing a strategy for procedures in the event of an accident.

Finally, it should be noted that the Valduc laboratory benefits from over thirty years of know-how and expertise acquired in the domain of criticality expansion.

THE FRENCH EXPERIMENTAL FACILITIES IN CRITICALITY PHYSICS: ONGOING AND FUTURE PROGRAMMES

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Abstract

The Nuclear Reactor Directorate (DRN) of the French Atomic Energy Commission (CEA) operates two zero-power reactors. The first of these is the EOLE reactor, and the other is the MINERVE reactor located at Cadarache. These experimental facilities are mainly devoted to reactor physics [1,2] of Light-water Reactors (LWR) [3], Advanced Pressurised-water Reactors (APWR) [4] and High-conversion Reactors (HCR) [5]. However, for fifteen years, CEA has performed specific experiments dedicated to criticality safety in the EOLE and MINERVE reactors. The goal of these critical experiments has been to supply neutronics basic information on fuel cycle plants in order to enlarge mock-up experiments carried out in the Valduc facilities [6] of the Nuclear Safety Institute IPSN.

The experimental facilities

MINERVE is an experimental pool reactor. The core comprises a peripheral enriched UAl driver zone and a square central cavity the dimensions of which may be varied in order to accommodate either a LWR or a FBR test lattice in a watertight cylinder. The MELODIE PWR block is made up of 800 UO_2 rods in a 1.26-cm square pitch. A vertical oscillator is used to measure with an automatic rotative control rod, the sample in-lattice worth; irradiated and fresh samples of fissile and fertile material, or absorber material are investigated. The samples are oscillated in the central cell between the midplane and the upper reflector. In order to measure resonance Doppler broadening, a high frequency furnace may heat, if necessary, the sample during a half-period [7,8].

EOLE is a very versatile facility for lattice studies; either a coupled core with an MTR or LWR driver zone or a large homogeneous core may be built and contained in the aluminium alloy tank of the reactor. Criticality is obtained by poisoning the moderator. A truly unique aspect of EOLE is that it is equipped with two separate water circuits: one for the test zone, the other for the driver core. It is possible to introduce soluble poison (boron, gadolinium). A typical device featured in the central cavity is the CREOLE test rig [9]. This is used to measure the temperature coefficient of UO_2 or UO_2/PuO_2 lattices under PWR operating conditions up to 300°C and 120 bars [10].

These two facilities enable us to perform various measurement types: 3-D flux distributions, power maps, buckling by spectrometry on fuel pins and foils (Mn, Dy, ^{235}U, ^{238}U). Reactivity worths (burnable poisons, control rod materials, moderator temperature effects, etc.) are generally deduced from equivalent moderator poisoning and checked through the doubling time technique or determination of the new critical size.

Unfortunately, critical experiments do not allow to meet the capture cross-section accuracy required for major nuclides, and furthermore, to qualify the fuel depletion calculation. Therefore, we used characterisation of spent fuels from Framatome PWRs [11].

After cooling, assemblies are transported in a leak cask to CEA-Saclay centre. In the LECI laboratory, samples are cut in radiation shielding cells and put in small individual containers. These containers are shipped to “clean” cells at COMIR-Cadarache where samples are decontaminated and dissolved in a hot acid solution. Solution containers are then sent to specific CEA facilities for chemical analyses.

We used chemical analysis and micro-probe measurements of PWR spent fuel to check burnup calculations and to improve cross-sections through sensitivity analysis. Investigated assemblies irradiated during 1, 2, 3, 4 or 5 reactor cycles enabled us to scan a wide range of burnup (up to 65 000 MWd/t); from highly depleted pellets, the accuracy of minor actinide cross-sections (^{237}Np, ^{236}Pu, ^{238}Pu, ^{241}Am, ^{242}Am, ^{243}Am, ^{242}Cm, ^{246}Cm) achieved through the interpretation [12] is better than that reached through differential measurements. The accuracy requirements were met for fuel inventory and actinide safety-criticality calculations.

Achieved experimental programmes

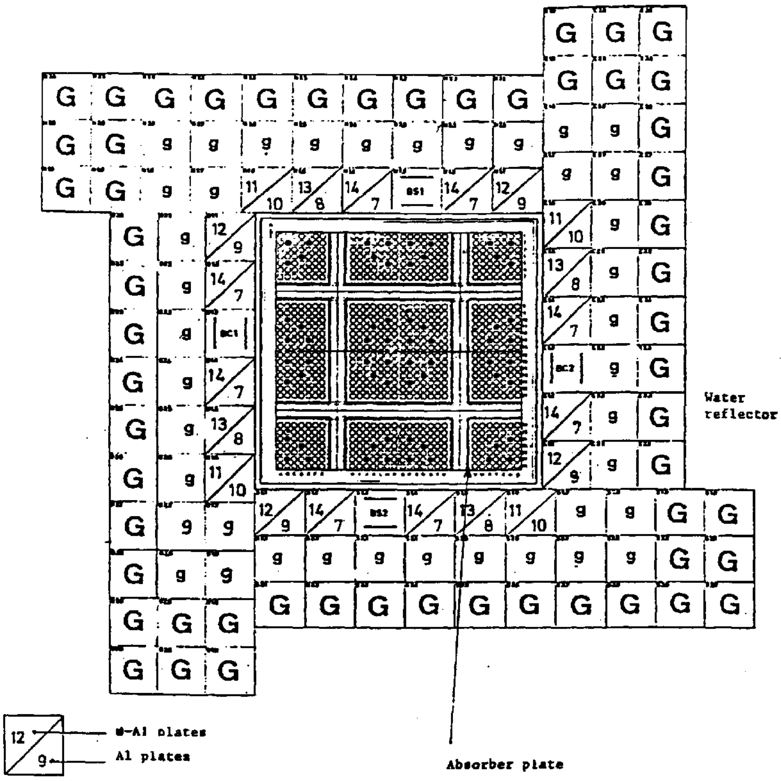
At various fuel cycle steps (assembly manufacturing, nuclear plant, reprocessing plant), assembly storage is needed for fresh and spent fuels. The CRISTO experiments in the EOLE reactor were conceived in order to qualify the calculation schemes of these storage lattices and to validate criticality calculations in accidental configurations. This experiment provides the multiplication factor of subcritical lattices for which a traditional critical experiment is not feasible.

The first experiment, *CRISTO I*, carried out in 1977-1978, used a wide pitch (water gap = 8 and 10 cm) and a grey absorber (2 and 4 mm stainless steel) [13].

The aim of the *CRISTO II* experiment performed in 1980-1981 [14] was the study of high density storages: water gap e_{H_2O} = 4.5 cm and 6.5 cm. The experiment allows the comparison of the anti-reactivity introduced by various kinds of neutron absorber plates (stainless steel, borated steel, cadmium...). Moreover the experiment provides the k_∞ curve, and consequently the saturation point, as a function of the increase in the absorber efficiency (from 0 to 20 mg/cm^2 of ^{10}B).

The storage area is simulated by a section containing four bundles of 14×14 UO_2 pins, loaded in a square cavity as graphed in Figure 1. The fuel characteristics are similar to those of a 17×17 PWR with 3 % ^{235}U-enriched UO_2 rods. The cavity is surrounded by a MTR driver core which enables the reactor criticality; progressively larger loading of enriched U-Al plates allows us to investigate absorbers which are increasingly strong in the test zone: stainless steel, borated steel plates doped at 2.5, 5 and 10 mg/cm^2 in ^{10}B, B_4C and cadmium.

Figure 1. CRISTO II experiment



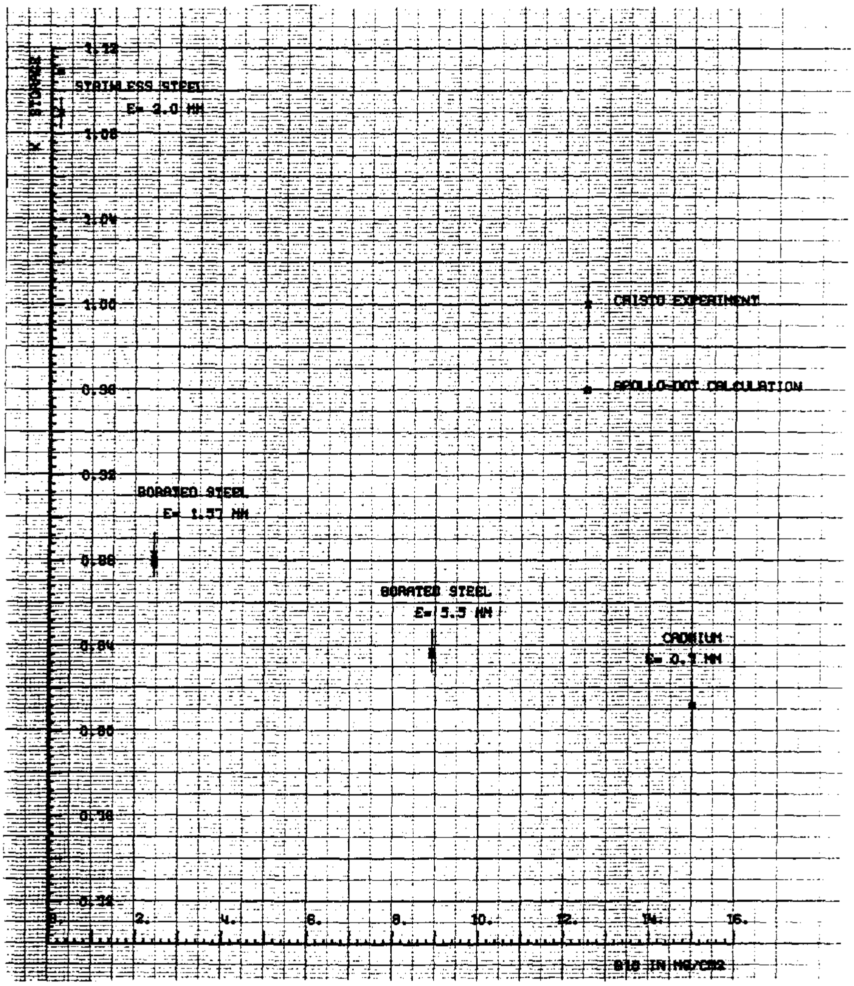
The core calculations were checked against the following measurements:

- Radial and axial fission distributions in both test zone and driver core (γ-scanning and fission chamber measurements),
- Thermal flux measurements inside the water gap (Mn detectors),
- Spectral indices (depleted and enriched U detectors).

The measured multiplication factor k_∞^{sto} is obtained from an equivalence with a regular lattice (same UO_2 rods, but the pitch lattice is 1.58 cm), which is loaded in the whole cavity and poisoned with boric acid until the criticality of the reactor is reached. k_∞^{reg} of this "equivalent" regular lattice is inferred from radial and axial buckling measurements.

The k_∞ calculation experiment comparison plotted on Figure 2 is particularly satisfactory. The grey absorber (2 cm SS plate) and the reference storage configuration without absorber plates are not accurately calculated. Thanks to the thermal flux measurements (Mn foils) throughout the water gap, it was pointed out that the discrepancy originates from a 10% underestimation of the strong thermal flux level between the assemblies.

Figure 2. Calculation-experiment comparison of k_∞ storage (4.5-cm water gap)



CRISTO III experiments [15] are related to a criticality dissolver risk in a reprocessing plant. The soluble gadolinium reactivity worth was investigated with respect to the boron worth. *CRISTO III* experiments permit the verification of the calculation schemes of UO_2 rods fuelled in close-packed lattices with a poisoned (gadolinium, boron) moderator. The first *CRISTO III* experiment was performed in 1980 and corresponds to a tight fuel array (∅ fuel = 9.4 mm, square pin pitch ∅ = 9.6 mm). The second campaign, achieved in 1982, is a ∅ 11.5-mm pin pitch experiment.

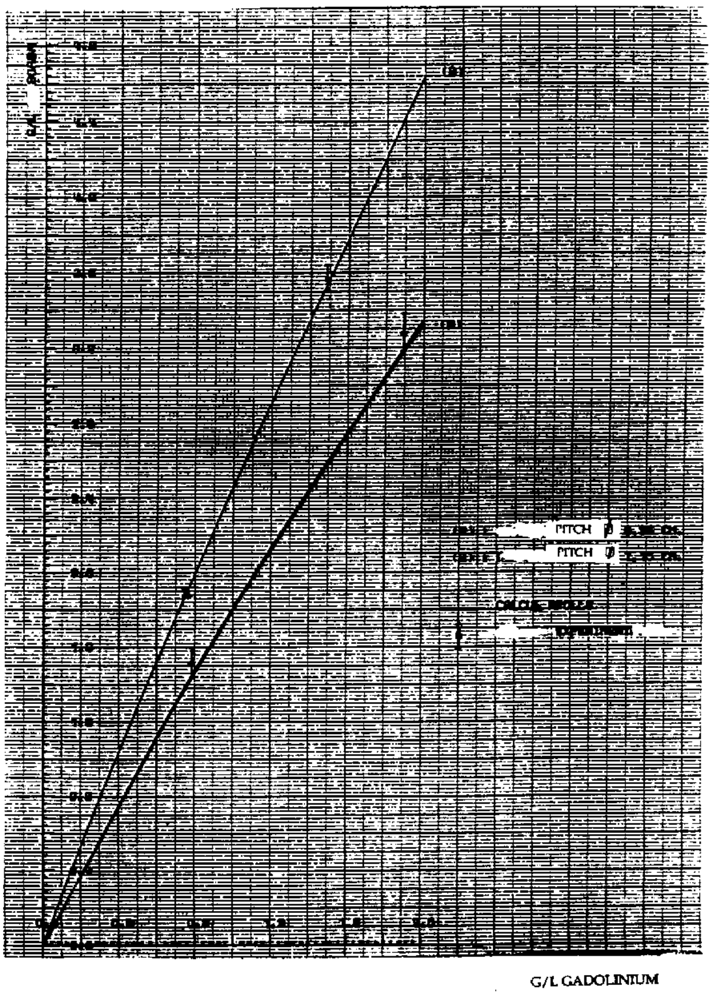
Three kinds of measurements were performed:

- Fission rate distributions, in both the test zone and the surrounding LWR driver core, by gamma spectrometry and fission chamber measurements. The aim of these axial and radial measurements is to check the 3-D core calculations;

- Buckling measurements and spectral indices (^{235}U, ^{238}U and ^{239}Pu miniature fission chamber) in the test zone. These experiments allow us to deduce the nuclear properties of these unmoderated lattices: material buckling, infinite multiplication factor k_∞;

- Research of the equivalent poisoning in boron and gadolinium up to 2g/l Gd for the same driver core loading. These critical measurements yield the equivalence boron/gadolinium in infinite lattice. The experimental results are compared with APOLLO calculations in Figure 3. The relative efficiency of the soluble gadolinium (roughly twice the soluble boron) increases with the moderation ratio because gadolinium is a more thermal absorber than boron.

Figure 3. Reactivity worth equivalence between soluble gadolinium and boron

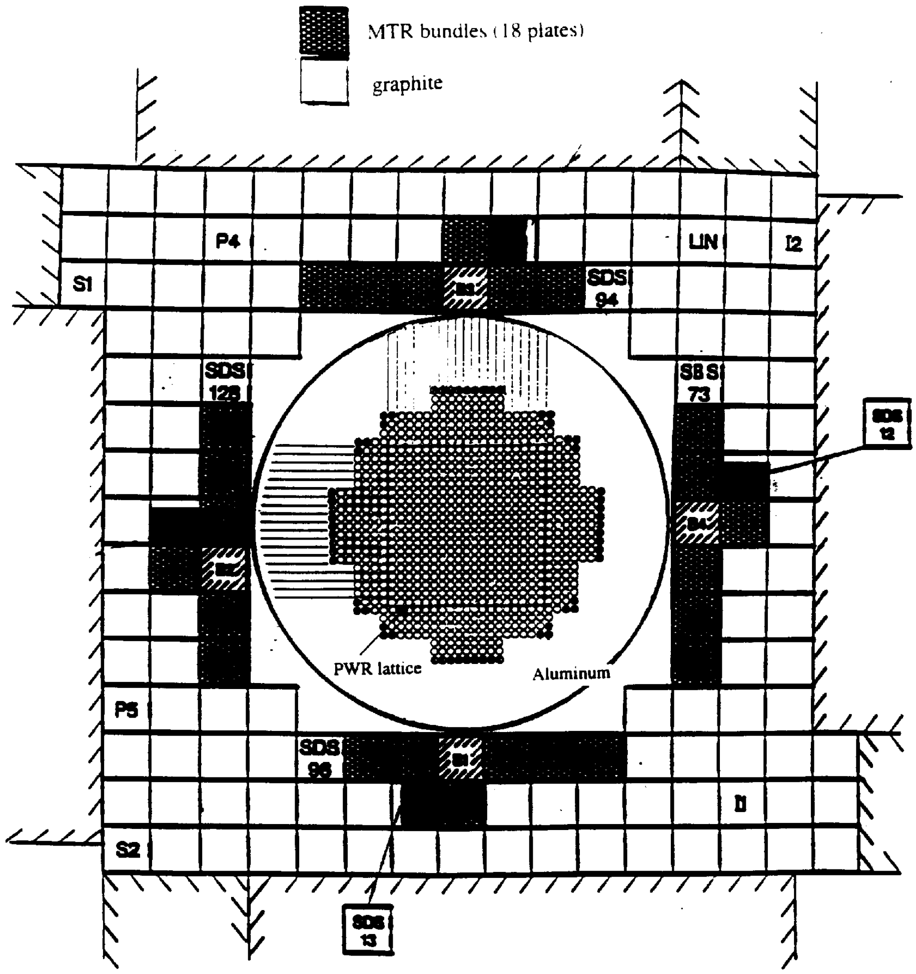


The on-going experimental programme: Burnup credit qualification

The programme of CEA-Cadarache involves two kinds of experiment:

- Measurements of heavy nuclide and fission product concentrations versus burnup in PWR spent fuel rods,
- Measurements of burnup credit nuclide and spent fuel reactivity worth in the PWR test zone of the MINERVE reactor (see Figure 4).

Figure 4. The Minerve reactor with the PWR test zone configuration



The qualification of spent fuel inventory calculations

A large experimental programme based on spent fuel chemical analysis has been carried out since 1993. Uranium, plutonium, americium and curium have been analysed in PWR samples. Furthermore, fission product chemical elements (Sm, Rh, Nd, Cs, Mo, Eu, Gd, Ag, Ru) have been measured relative to uranium, as well as their isotopic composition. The chemical analysis programme for UO_2 assemblies is summarised below:

FRENCH PWR	FUEL ASSEMBLY	IRRADIATION
BUGEY 3	17×17 3.1%	1.5 cycle and 3 cycles
GRAVELINES 3	17×17 4.5%	2 cycles and 5 cycles

The reactivity worth measurements for burnup credit in PWR fuel cycle

The goal of the current reactivity worth measurements by the oscillation technique in the MINERVE reactor is the qualification of the calculational tools and data used to predict the poisoning worth of the selected 'BUC' nuclides from PWR spent UO_2 fuels.

We manufactured three kinds of PWR-type oscillation samples:

1. *Separated FP isotopes in UO_2 pellets* (natural uranium). The characteristic of these fission products samples are summarised in Table 1. The mass of each fission product isotope by sample was optimised in order to obtain a similar reactivity worth corresponding to the maximum accuracy in MINERVE worth measurements.

2. *Simulant samples containing a natural element* in a Al_2O_3 matrix: Ag, Ru, Rh, Mo and Cs samples. The measured reactivity worth of the Ag, Ru and Mo elements is mainly due to the burnup credit nuclides: ^{109}Ag, ^{101}Ru and ^{95}Mo respectively. The sample content in both types of fission product simulant samples was obtained from chemical or mass spectrometer analysis on pellets from the same batch.

3. *Irradiated samples* built with UO_2 fuel rod cuts from various French PWRs. These fuel cuts are performed in the rods located at the centre of the 17×17 assemblies (asymptotic spectrum) (see Table 2).

Table 1

SAMPLE UO_2+X_2O_3	ADDITIVE X	MASS X (ppm)
UF		0
Sm	Sm-nat	460
^{9}Sm	^{149}Sm	69.3
^{2}Sm	^{152}Sm	10750
^{7}Sm	^{147}Sm	18500
Nd	Nd-nat	69900
^{3}Nd	^{143}Nd	10700
^{5}Nd	^{145}Nd	59500
^{5}Gd	^{155}Gd	121
^{3}Eu	^{153}Eu	9000
Rh	^{103}Rh	5350
SAMPLE NON-SINTERED UO_2	**ADDITIVE X**	**MASS ADDITIVE (g)**
UC		0
Cs%1.6	Cs_2CO_3	0.62
Cs%10	Cs_2CO_3	3.77
^{5}Mo	$Mo_{95}O_3$	5.66
^{9}Tc	$KTcO_4$	4.32

Table 2

PWR IRRADIATION CYCLES		ASSEMBLY-ROD-HEIGHT (mm)	BURNUP (GWd/t)
BUGEY 3	1.5 CYCLE	FGC53-G11-1960	20
	2 CYCLES	FGC53-K7-1960	25.5
	3 CYCLES	FGC53-K11-1960	38
FESSENHEIM 2	2 CYCLES	FEB34-K11-2020	30
	2 CYCLES	FEB34-K11-1500	31
	4 CYCLES	FEC57-H11-3000	49
	4 CYCLES	FEC57-H11-1960	49.5
	5 CYCLES	FEC57-G10-3000	59
	5 CYCLES	FEC57-G10-1969	60
GRAVELINES 3	2 CYCLES	FF06E2BV-K08-1960	27
	5 CYCLES	FF06E3BV-J07-1860	62

The contents of these irradiated samples are obtained from dissolution and chemical analyses on a 2-cm rod cut near the rod piece used for the oscillation sample.

These spent fuel samples aim to calibrate the overall Actinide + FP burnup credit versus irradiation. The variety of the irradiated samples used supply the total BUC worth in the 20-60 GWd/t burnup range.

Furthermore, Gravelines fuels are representative of the current high ^{235}U enrichment e = 4.5%, and high burnup BU = 60 GWd/t corresponding to PWR fuel management with 1/4 core loading.

These three types of samples – separated isotope, simulant samples and irradiated samples – have an identical geometry and can be viewed as 100 mm long sections of PWR pins, contained in thin watertight zircaloy sleeves, L_{ext} = 103.5 mm.

The samples were oscillated in the centre of the PWR block implemented at the MINERVE reactor. This configuration, called the R1-UO_2, is presented in Figure 4. This R1-UO_2 experiment was carried out in three successive experimental campaigns: the separated FP samples in 1993, the spent fuel samples in 1994 and the natural FP in Al_2O_3 matrix in 1995. The slowing-down density at thermal cut-off, which is representative of the spectrum hardness, amounts to q = 0.65 in the R1-UO_2 lattice.

In order to study the full range of spectra spanning the potential reactivity peak in dissolver media, three complementary experiments were planned:

- R2-UO_2 configuration in MINERVE, with a softer spectrum corresponding to the optimum moderation ratio (q = 0.8), to be performed at the end of 1995;
- DIMPLE configuration with a pure thermal spectrum (D_2O or H_2O central zone);
- DIMPLE core with highly enriched 7% UO_2 rods. Worth measurement at the centre of this lattice enhances the FP resonance capture contribution. This DIMPLE spectrum corresponds to dissolver media poisoned with soluble gadolinium. DIMPLE experiments were performed during 1993 and 1994 in the framework of the CERES collaborative programme between AEA and CEA.

The MINERVE reactivity worth measurements were computed with the criticality calculation package CRIBLE [16] based on the APOLLO-1 code and its CEA-86 library (JEF-1 file for fission product data).

ISOTOPE	CRIBLE (CEA86)	EXP. UNCERT. (1σ)
Sm ^{9}Sm ^{2}Sm ^{7}Sm	-7.4 % -8.2 % 5.1 % 9.4 %	+ 1.8 % + 2.0 % + 3.0 % + 5.0 %
Nd ^{3}Nd Nd	-1.0 % -10.2 % 7.1 %	+ 3.0 % + 3.0 % + 10.0 %
^{5}Gd	-0.3 %	+ 2.0 %
^{3}Eu	0.0	+ 2.0 %

The future programme: Burnup credit in MOX fuel and BWRs

A large programme has been carried out by the CEA since 1991 in view of supporting the application of reactivity burnup credit in the PWR spent fuel cycle.

A similar programme will be initiated in 1996 to investigate burnup credit in MOX assemblies burnt in French PWRs.

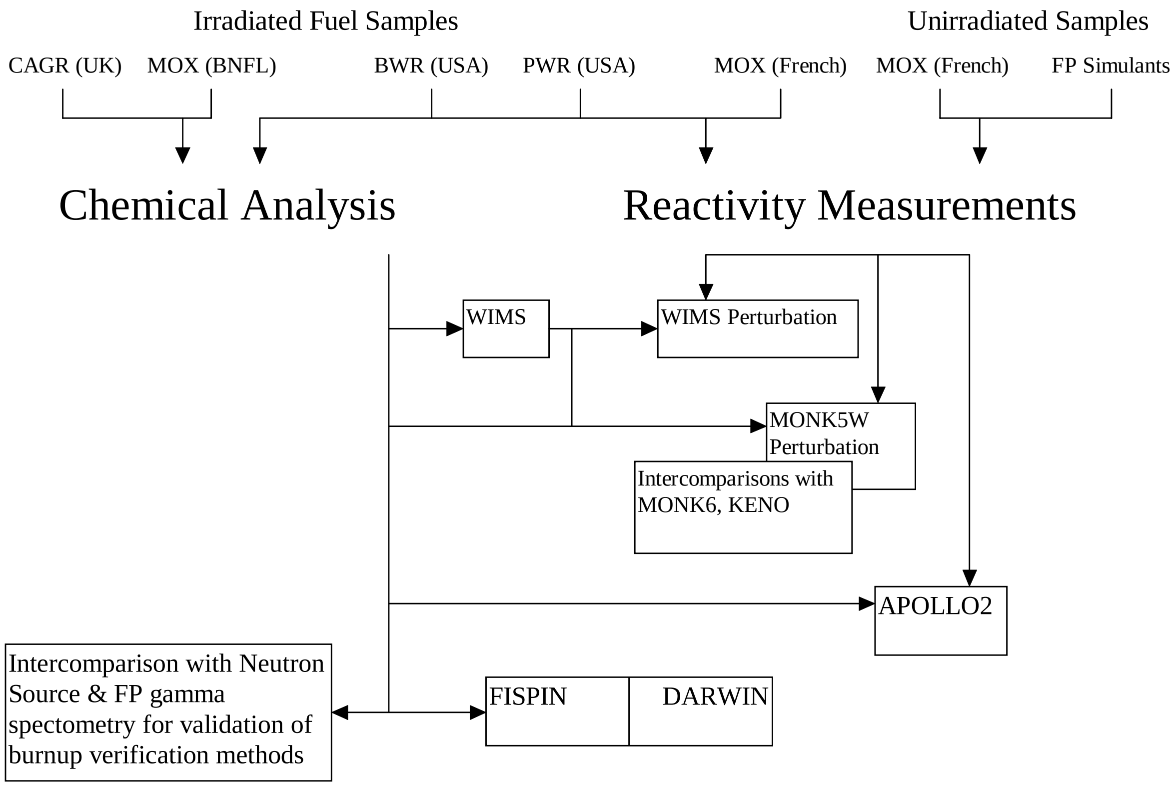
Chemical analyses, involving FPs, will be carried out in various pins from MOX assemblies irradiated in the Saint-Laurent B1 plant. In the same 17×17 assembly, fuel cuts from the three enrichment zones will be analysed.

An oscillation experiment will be performed. A fresh MOX lattice will be implemented at the centre of the MINERVE reactor. We manufactured irradiated MOX samples at increasing burnups from SLB1 assemblies. The oscillation of this MOX spent fuel in the MINERVE MOX lattice will provide the reactivity loss with respect to fuel burnup. Furthermore, fresh MOX samples and calibration samples (UO_2, borated UO_2) will be oscillated. In order to check the reactivity worth of each burnup credit FP nuclide, we will measure the specific samples doped with one FP.

An experimental programme devoted to BWR fuel will be started up in 1996. In order to check BWR fuel inventory, 15 cuts from three various pins of a BWR 9×9 assembly will be characterised in 1996-1997. Chemical analyses will be performed, including minor actinides and the burnup credit fission products. Oscillation samples from rod cuts at various heights will be manufactured. These BWR fuel samples irradiated at BU = 42 GWd/t will be oscillated in the MINERVE reactor in 1997.

This extensive experimental programme on burnup credit in BWR and MOX fuels is strengthened by the CERES III collaborative programme [17] between AEA, CEA and Sandia National Laboratory (Figure 5).

Figure 5. The CERES III programme



Qualification needs and experiment proposal

Calculations of the future French fuel cycle are facing three challenges:

1. *Highly enriched LWR assemblies with high burnup*;

2. *Extensive use of plutonium*. The plutonium from the La Hague reprocessing plant is currently recycled in 16 PWRs. The Pu recycling is also considered in specific reactors: the RMA reactor (over-moderated PWR with 100% MOX fuel) and the CAPRA reactor (Fast Breeder Reactor, with high enrichment e_{Pu} = 45% in order to burn the plutonium). Furthermore, MOX fuel reprocessing is envisaged; the corresponding degraded plutonium should be used in RMA or CAPRA reactors.

3. *Np, Am, Cm separation in the reprocessing plants, and MinAc transmutation in specific assemblies*. Am and Cm use raises severe criticality constraints because the critical mass of ^{242}Am and ^{245}Cm is hydrogenated media corresponds to some grams.

In order to improve the reliability of criticality calculations of Pu and MinAc recycling, we propose to perform actinide reactivity worth measurements in the Minerve reactor. Specific oscillation samples could be made with separated isotopes such as ^{236}U, ^{242}Pu, ^{237}Np, ^{241}Am, ^{244}Cm, ^{245}Cm, ...

Conclusion

The EOLE and MINERVE reactors are powerful tools for the investigation of the physics of criticality-safety. Fundamental experiments have been performed which have supplied basic data, particularly the capture cross-sections of gadolinium and fission products. The on-going programme is dedicated to burnup credit in PWR fuel, and will be carried on with burnup credit in MOX and BWR fuel until 1997. In order to check actinide nuclear data, a new experiment is proposed in the MINERVE reactor. The reactivity worth of each heavy isotope will be accurately measured by an oscillation technique. This experimental information would be particularly relevant for high burnups, Pu recycling and the separation, storage and transmutation of actinides.

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Russia

OVERVIEW OF CRITICALITY SAFETY FACILITIES IN RUSSIA: PROGRAMME OF WORK AND FUTURE NEEDS

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Abstract

This report briefly presents the results of experiments relevant to Nuclear Criticality Safety (NCS) that were carried out in three Russian institutes under the auspices of MINATOM. Activities of other organisations within the CIS are also discussed. Proposals for increased international co-operation are made which designate the OECD/NEA Working Group as an active participant. The suggested initiatives mainly concern NCS investigations with regard to current treatment conditions of fissile materials (FM).

Introduction

NCS research began in the scientific organisations of the former USSR during the 1940s (RBC-I, VNIIEF) and the 1950s (JINN, Dubna, IPPE, VNIITF). These investigations were concentrated in two areas: nuclear weapons development and nuclear power engineering. The sphere of NCS interest gradually developed, and at present relates mostly to problems involving the treatment of low- and intermediate-enriched wastes, and also with power and weapons grade plutonium.

The amount of circulating fissile materials (FM) has markedly increased in recent years, partly as a result of nuclear disarmament in the USA and Russia. Containers composed of new materials – which had not been previously investigated – began to be commonly used for the transport and storage of irradiated and non-irradiated FM. As a result, it has become necessary to study the behaviour of large systems of neutronically connected FM assemblies, and the influence of reflector materials on large systems of heterogeneous FM assemblies, such as packed wastes or reactor fuel.

FM systems are assembled in which fission chains on intermediate energy neutrons take place, such as dioxides of low- and intermediate-enriched uranium, dioxides of ^{239}Pu or ^{235}U, and systems with large amounts of neutron absorbers in thermal, 1/v or resonance energy ranges.

All of the above-mentioned circumstances lend to the conclusion that the intensification of NCS experiments and calculations would be extremely beneficial.

General information on NCS experiments undertaken

General data concerning NCS experiments, presently performed in three Russian institutes (IPPE, VNIIEF, VNIITF), are presented in Table 1.

Table 1. Performed NCS experiments (general information)

	VNIIEF [1]	**VNIITF [2]**
Existing facilities	FKBN-2M	FKBN-M (including ROMB-model)
Configuration of critical assemblies	Spherical assemblies	Spherical assemblies, cylindrical assemblies built of disks and rings
Fissile materials	^{235}U, ^{238}U, ^{239}Pu (metals and imitator of water solution of ^{235}U salt)	^{235}U, ^{239}Pu (metals, highly enriched)
Reflector and moderator materials	^{238}U, Be, BeO, Fe, Al, W, Pb, Ni, Cu, Mo, Ti, C, Zr, ^{10}B, B_4C, Cd, $(CH_2)_n$, H_2O, concrete	^{238}U, Be, BeO, Fe, Al, W, Pb, Ni, Nb, Mo, Ti, Cd, $(CH_2)_n$, H_2O, concrete
Number of experiments	About 1000	About 350
Number of benchmark-type experiments	About 200	About 250

IPPE has the greatest potential to perform experiments requiring large amounts of FM solutions. IPPE has performed – and continues to perform – numerous experiments arranging FM solutions in various configurations: individual vessels of cylindrical, spherical, conical and rectangular forms, the sets of the vessels in regular lattices and in other arbitrary combinations (including those often found in power reactors and other facilities).

VNIIEF has a wide range of sets of hemispheric details that permit investigation of metallic FM in the solid phase; an assembly of imitator of uranium-salt-water solution. The range of details of spherical assemblies is not as great, but benchmark-type experiments can be performed in various cylindrical configurations.

While speaking of Russian organisations, the Russian scientific centre known as the Kurchatov Institute (RBC-I) should be mentioned. This facility could greatly advance the investigations concerning NCS problems, especially with regard to the use of uranium assemblies (metal). The activities of RBC-I, however, are not covered in this report.

The study of NCS with regard to fast nuclear reactors is undertaken in the scientific laboratories of Beloyarskaya NPP (Eketherinburgh).

Other scientific organisations that develop and exploit NPP have too few prospects of performing NCS experiments, and thus are not mentioned here.

As for the activities of other states belonging to the CIS, only Kazakhstan's IGR reactor, which still operates, is referred to. In recent years, VNIITF and RBC-I have conducted joint studies at this facility concerning severe accidents in NPP linked to fuel element melting.

List of NCS experiments that may be performed at IPPE, VNIIEF and VNIITF

Scientific organisations in Russia possess the facilities, experience and skilled specialists which permit experimental and calculational investigations to be conducted in the following NCS fields.

IPPE

- The performing of new critical experiments with water solutions of uranyl nitrate at uranium enrichments equal to 10, 15 and 20 per cent (four-six concentrations for each enrichment). The diameter of the core varies depending on the enrichment and concentration of uranium, from 300 mm to 700 mm, the solution volume from 21 l to 230 l. The interval of uranium concentration value will be from 150 g/l to 420 g/l. This process is supposed to measure the effectiveness of different bottom reflectors such as steel, polyethylene, layers of moderating materials lined with cadmium and so on, of different thicknesses;

- Measurement of critical parameters of heterogeneous systems with a water moderator and reflector for fuel rods and subassemblies of BN and RBMK reactor.

These experiments are required not only for the verification of codes and the experimental substantiation of nuclear criticality safety, but also for the safe storage and transportation of fresh and spent fuel.

VNIIEF

- Obtain experimental and calculated data for development of blankets for electro-nuclear transmutation of radioactive wastes and plutonium (using composite assemblies of ^{239}Pu and ^{235}U, together with isotope ^{237}Np);

- Test calculations connected with the behaviour of low-active wastes (using assemblies with large inner cavities);

- Specify nuclear data for "resonance" reflectors (assemblies with Be, Ch_2-moderators and Fe, ^{238}U-reflectors);

- Measure reactivity perturbations by means of low-mass samples (of ^{234}U, ^{236}U, ^{237}Np, ^{241}Am, ^{243}Am, and so on, as well as by fragments of fresh and burnt fuel elements).

VNIITF

- Continue benchmark-type experiments with sets of discs in cylindrical configurations to simulate neutron data of uranium-hydrogen systems;

- Continue benchmark experiments with different reflectors in spherical and cylindrical configurations for correction and validation of codes for NCS calculations, particularly in the "unresolved resonance" ranges;

- Carry out experiments in order to investigate accident situations in nuclear reactors; study issues such as storage transport, fuel reprocessing, and the development of active targets for electro-nuclide transmutation facilities;

- Investigate criticality anomalies of ^{235}U and Pu possessing composite reflectors;

- Perform experiments aiming toward the specification of neutron cross sections of different materials;

- Perform experiments in order to develop an active target for electro-nuclear transmutation blankets.

Need and realisation of plans and proposals on international co-operation

Many organisations are presently experiencing difficulties such that the continued pursuit of NCS experiments remains uncertain. These include:

- Decreases in scientific and technical connections between institutes, both inside Russian and particularly among countries in CIS;

- Strong decreases in the funding of scientific research;

- Decreases in the number of highly-skilled specialists and a significant lack of training for new specialists;

- Wear and tear of equipment.

There exists a need to:

- Support current exploitation of existing equipment;
- Modernise the control and data processing systems;
- Increase the amounts of our test assemblies (new materials and configurations);
- Increase scientific and technical information exchange amongst specialists from different countries.

Possible means of international support and mutual assistance include:

- Favourable references from foreign laboratories in relation to projects in being pursued in Russia, Kazakhstan, and Georgia, relevant to problems of NCS and NPP, supposed to funded by ISTC;
- Direct contacts of foreign partners with Russian institutes concerning NCS issues;
- Establishment of working groups composed of Russian and CIS specialists, international organisations and foreign laboratories. These groups would conduct joint NCS research;
- Sponsorship of CIS specialists with regard to participation in international symposia, conferences and meetings relevant to NCS.

Conclusion

Several leading scientific and research institutes of the Ministry of Atomic Energy perform a great number of NCS experiments, including roughly one hundred benchmark-type experiments.

A combination of existing equipment and highly-skilled specialists permits, for the time being, continued NCS research with the intended objective of resolving urgent issues relevant to FM treatment.

The current state of affairs in Russia and other countries is such that intensification of international co-operation in the NCS field is required in order that research may continue in this domain. Support may be offered in the following forms: direct contracts, ISTC projects, joint research groups, etc.

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TECHNICAL CAPABILITIES AND NEAREST FUTURE PROGRAMME OF CRITICAL FACILITY OF IPPE NUCLEAR SAFETY DIVISION

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Abstract

There is only one facility in IPPE and Russia which can be used for experiments with uranium solutions. This is the SPH – Solution Physical Heterogeneous – facility located in the experimental building of IPPE Nuclear Safety Division.

It is a universal facility which possesses a license for performing critical experiments with "zero-power assemblies" (no more than 100 wt) which can have different kinds of cores with either a uranium solution or heterogeneous uranium-water ones with different values of uranium enrichment. The device responsible for the fixing of control and scram rods can be placed at different points according to the design that must be developed for each assembly.

SPH experimental facility of Nuclear Safety Division

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The following characteristics of the SPH facility should be noted:

- Dimensions of box (facility room) – 5.5×7.5×8.5 m,
- Loading crane cans operate with details, components having a mass of less than one ton,
- Gate dimensions – 2×2 m,
- Solution storage has 17 tubular vessels. Total volume is 2 m^3, unit volume is 80-120 l. There are now solutions with different concentrations of 91%-uranium with total mass of uranium of about 75 kg,
- System solution dosage includes two lines, each of them having the volume measurement vessel (tube) with total volume equal to 10 l and minimal volume equal to 50 ml,
- System of water dosage has a special device similar to that of petrol station device. Minimal dosage volume is equal to 0.2 l,
- Split-table device has three platforms, of which at least two are moving. Equipment with a mass no greater than 500 kg can be placed on each platform.

There are different kinds of nuclear materials in the central institute storage which can be used for experiments in the SPH facility. For example:

- Uranium dioxide with different enrichment, rods with different kinds of uranium fuel enriched from 2 to 96% and different sizes,
- Special pellets with dioxide and metal HEU (36 and 90%), LEU, depleted uranium and plutonium having different isotopic compositions with stainless steel claddings (diameter about 35 mm, thickness about 5 mm).

The capability of the facility is clear from the summary listing of the criticality experiments fulfilled in the past.

Summary listing of criticality experiments carried out to date at SCC RF IPPE

The first criticality experiment was carried out at IPPE in 1959. Experiments were mainly performed with highly-enriched uranium (90% enrichment) and a uranyl nitrate solution.

A large number of experiments were carried out in order to investigate critical parameters of uranium solutions of bare, and reflected tanks had a cylinder, sphere or rectangular geometry. No experiments were conducted with low-enriched uranium.

A few critical experiments were performed to determine the relative effect of some reflector materials on critical parameters.

An experimental programme concerning the interaction between subcritical units was also pursued. Cylindrical vessels and rectangular parallelepipeds with square bottoms were used as subcritical units. The interaction was studied not only in air, but also in water. Subsequently, the interacting systems of two annulars was studied. Each model consisted of an annular with an outer diameter equal to 800 mm and a width of 100 mm.

Measurements regarding the effectiveness of rod arrays were made in cylindrical and rectangular geometries with the same tank sizes, rod number and material, pitch dimensions and solution concentrations as variables.

Another area of investigation considered the critical parameters of aqueous solutions composed of 90% U-enriched uranyl nitrate and contained in vessels of annular cylindrical geometry. The annuli were formed by combinations of four stainless steel cylinders with diameters equal to 80, 124, 148, and 197 cm. The experiments were performed with inserts in the form of cylinders, layers of moderating materials lined with cadmium or made of boron carbide.

Critical and subcritical experiments were made with models of tubular equipment having the form of cross and the letter “Y”. The first model was composed of three tubes with a 14-cm diameter, which were connected with tube lintels of the same diameter. The distance between tubes and lintels was 452 mm. The second model possessed a similar geometry to that of the previous one, however the pipe diameter was 195 mm, the distance between them was 519 mm and the walls were 12 mm thick. The third model was composed of six tubes located in the angles of a regular hexagon and a central tube with a 140-mm diameter and a 120-mm height. The distance between the tubes was equal to 260 mm.

At the beginning of the 1970s, a series of experiments was conducted with conic-angular systems. In these experiments the core was composed of two cones located coaxially; each of the cones possessed a 90-degree angle.

Critical experiments with models of real vessel safety models having a complicated geometry were performed. Some attempts were made to investigate the criticality of real plant vessels and real chemical substances. Experiments with real vessels were not critical and therefore cannot be used for code evaluation.

Experiments with real chemical substances were carried out using the so-called mockup critical experiments. In these experiments, the insert was surrounded by an annular uranyl nitrate aqueous solution core (reference zone), the composition and size of which can be changed. The special analysis of critical mockup experiments was performed in order to obtain critical parameters for insert fissile material. Investigations were performed with substances which cannot be used directly in critical experiments: high-enriched uranium dioxide powder with a real (6 g/cc) density; moist powder with enrichment of uranium of 2.4, 3.6, 4.4% (ratio H/U is equal to 7.8); uranium tetrafluoride, etc.

Critical heterogeneous experiments were made with 1.12-cm-diameter rods filled with uranium (90%) dioxide pellets in water lattices. Similar experiments were performed with oil as moderator and rods of a 1.0-cm diameter.

Experiments with lattices of cylinders filled with HEU-dioxide powder were performed for the validation of calculation methods used for the analysis of emergency situations which occur in storage and during transportation. The diameter of the cylinders varied from 8.3 to 15.3 cm. Cylinders were installed into a tank in regular lattices of a square or triangular configuration. Criticality was achieved by changing the moderator level. Nineteen cylinders were available for a series of experiments carried out with 8.3-cm-OD cylinders, nine for a series with 12.5-cm-OD cylinders, and nine for a series with 15.3-cm-OD cylinders.

The experimental programme concerning solutions of uranium and gadolinium was carried out in a simple cylindrical geometry with bottom and axial reflectors. The core diameter varied from 250 to 600 mm. Uranium concentration fluctuated within the range of 50-450 g U/l with a uranium enrichment of 90%. The concentration of a natural mixture of gadolinium isotopes varied between 0-10 g/l.

Experiments conducted with slabs (and various partitions between the slabs) were performed in order to specify partition absorption and to verify methods of interaction calculation. The experimental facility consisted of a platform which provided remote tank movement. The tanks were made of steel, and possessed a solution layer thickness of 90-110 mm. The measurements were made using partitions of variable thickness. The following materials were used for partitions: bricks, concrete, polyethylene including the one containing boron.

The preceding list of experiments demonstrates the capability of the SPH facility of the Nuclear Safety Division. Some of these experiments can be classified as criticality experiments for nuclear safety, whereas others may be considered as benchmark experiments.

Nearest future programme of critical experiments

1. To perform new critical experiments with water solutions of uranyl nitrate at uranium enrichments equal to 10, 15, and 20% (four-six concentrations for each enrichment).

   Conditions of experiments are similar to those presented in LA-12683 “Forecast of Criticality Experiments and Experimental Programmes Needed to Support Nuclear Operations in the United States of America”.

   As is usual, this work includes the development of assembly design, the manufacturing of assembly details, the assembling, the fulfilling of experiments, the preparing and reprocessing of solutions, and the preliminary estimation of experimental results.

   The diameter of the core will vary from 300 to 700 mm in dependence of enrichment and concentration of uranium, and the solution volume from 21 to 230 l. The interval of the uranium concentration value will vary from 150g/l to 420 g/l. It is supposed to measure the effectiveness of different bottom reflectors such as steel, polyethylene, layers of moderating materials of different thickness lined with cadmium, etc.

2. The measurement of critical parameters of heterogeneous water-moderated systems and reflector for fuel rods and subassemblies of BN and RBMK reactors. These experiments are

needed both for the verification of codes and the experimental substantiation of nuclear criticality safety in storage and transportation of fresh and spent fuel.

The following parameters must be evaluated:

- Critical pitch for infinite lattices of rods and subassemblies,
- Minimal critical number of rods and subassemblies in water (for optimum conditions).

SOME NEEDS IN VALIDATION OF CRITICAL SAFETY BEING SATISFIED USING PRESERVED EXPERIMENTAL INSTALLATIONS

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Abstract

The aim of this report is to raise the possiblity of satisfying the need for additional experimental validation of critical safety calculations by expanding upon experimental programmes being conducted at the critical facilities maintained in our Institute.

Criticality of the tight uranium-water lattices

Criticality of the tight uranium-water lattices in which the majority of fissions initiated by intermediate neutrons is presently calculated with insufficient realiability because an adequately representative set of critical experiments does not exist. This shortcoming may be eliminated to a great extent based on the results of expriments conducted at the MATR critical facility located in IPPE. This facility was specially designed for the study of uranium-water lattices. Experiments on this assembly may be performed at a water temperature of up to 550°C.

Uranium oxide fuel rods with 5 per cent enrichment are canned by stainless steel. The diameter of the fuel pin is 7.6 mm. Lattice pitch is determined by the matrix constructed for each experiment. The possiblity of water displacement from the core is anticipated by means of intoducing a thin wall of empty tubes in the lattice. Boron or another equally efficient absorber (gadolinium, for example) can be introduced into the water. The question regarding the use of more or less enriched fuel may be decided in the future if necessary.

Proposed experiments could be a useful addition to the four experimental series performed on the MATR facility in the past but not yet been carefully evaluated. In the first three series of experiments, 5 per cent enrichment uranium was used. Lattice pitch provided the hydrogen to uranium ratios 110, 60 and 30. In the fourth series the ring fuel elements were used with enrichement of 17 per cent and lattice provide the hydrogen to uranium ratio near 250. Lattice with pure water and water poisoned by gadolinium or cadmium was investigated throughout the entire temperature diapason. In our view, a more complete evaluation of these experiments could be of great interest.

Experiments involving the use of plutonium fuel cannot be conducted at MATR. Such experiments have, however, been planned for the IPPE-BFS facility. The objective would be to insert a subcritical plutonium-water lattice into the driver zone. The execution of this project depends upon the delivery of plutonium fuel elements to BFS. Efforts made to contract terms for the delivery of fuel rods have repeatedly failed. Furthermore, experiments at BFS can only be performed at room temperature.

Criticality of the minor actinides multiplying system

The criticality of the minor actinides multiplying system is calculated predominantly on the basis of evaluated neutron data measured in differential experiments, or sometimes on the basis of data obtained by theoretical calcualtions. The accuracy of this data should be the better. There is no doubt that the reliability of criticality estimations based on multiplying systems with neptunium, americium and curium is no longer sufficient. However there currently exists no possibility of direct experimental ivestigation of the critical characteristics of such systems. While it is true that 10 kg of neptunium oxide in the form of 1-cm thick pellets is currently available at BFS, the critical systems which could be assembled using this material would scarcely be of interest from the critical safety point of view.

Despite this, experience has shown that the accuracy of criticality predictions can be considerably increased if the constants used in the calculations are tested and, if necessary, adjusted using the results of measurements of minor actinides cross section and reactivity worth ratios performed in the cores of the critical assemblies with different neutron spectra. A number of such experiments are performed on the critical assemblies and power reactors in Russia and other countries. However the data compiled are not verified as thouroughly as that accumulated within the framework of the International Criticality Safety Benchmark Experiment Project. We are certain that if such a project were initiated, quite a number of contradictions would be found in the existing data. Elimination of these contradictions and enlarging upon the data currently available through specially planned experiments on the critical assemblies would allow the creation of a reliable data base available for testing and adjusting the constants used in the calculations of the minor actinides criticality systems.

If the present Experts Meeting recommends the advisability of this subject and eventually moves to form an organised project, the qualified specialists from our Institute would be availabe to participate in the capacity of evaluation, as well as through the means of providing additional measurements.

A SHORT REVIEW OFCRITICAL EXPERIMENTS PERFORMED AT THE KURCHATOV INSTITUTE

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Abstract

Starting with experiments that led to the creation of the first national reactor, reactor neutronics experiments have been a necessary stage of development and improvement for nuclear power systems. As early as the first critical experiments carried out by Igor V. Kurchatov [1], support was provided for the unique merits of zero-power reactors/critical assemblies: a range of parameters of kinetic response on a variety of kinetic conditions, eminently convenient for practice, as well as invariability over a wide scope of the most important functionals of neutron flux to reactor power. A short review of critical experiments performed at the Kurchatov Institute is presented here.

Starting with experiments that led to the creation of the first national reactor, reactor neutronics experiments have been a necessary stage of development and improvement for nuclear power systems. As early as the first critical experiments carried out by Igor V. Kurchatov [1], support was provided for the unique merits of zero-power reactors/critical assemblies: a range of parameters of kinetic response on a variety of kinetic conditions, eminently convenient for practice, as well as invariability over a wide scope of the most important functionals of neutron flux to reactor power.

The fundamental physical peculiarities of reactors, combined with the characteristics of delayed neutrons and with the relation between the number of fission reaction carriers and environmental nuclei, determined the possibilities for modelling in-reactor neutron physics processes with the help of critical assemblies, and the universal introduction of the latter into experimental work.

Over the course of neutron physics experimental development throughout the world, it was revealed that a broad spectrum of problems under study possessed a common objective: that of attempting to upgrade the accuracy of predicting parameters in reactors being improved or under development. At one end of this spectrum of problems lie the mock-up experiments, in which an exact as possible simulation of geometry and material characteristics of designed reactor provides for the minimum contribution of calculational error into the prediction of reactor parameters under study. At the other end of the spectrum lies problems related to integral experiments – identification, based on the simplest critical assemblies and most powerful calculational methods, free form physical approximations, of probable errors and the line to obtain more precise differential nuclear data.

Reactor experiment strategy, “specific weight” of experiments of that or another type in physical validation of specific designs are closely related to state-of-the-art calculation theory and methods and strongly depend upon the peculiarities of the reactor area under consideration. Since the 1950s the Institute of Atomic Energy (now the Russian Research Centre “Kurchatov Institute”) has been involved in investigations of nuclear reactors intended for various purposes. At that time, the construction of critical assemblies was an immediate necessity. Of course, their number, design and purpose have evolved over time. A description of the critical assemblies’ present state can be found in Attachment 1.

In the area concerning reactors intended for use in atomic power plants the work moved in several different directions. Critical experiments in facilities with a water moderator were started in the 1950s and have been conducted up to the present. During this period a certain amount of study was devoted to reactor system sin the enrichment range from natural uranium to about 6% enrichment in ^{235}U for a wide range of water-uranium ratio. The generalised data from the first stage of experimental investigations in the area of commercial water-water type power reactor physics (including multiplication parameters for lattices of various enrichment) are provided in [2].

The research programme concerning WWER lattices was radically extended through the use of critical assemblies in Hungary (ZR-6) [3] and Czechoslovakia (LR-0) [4]. The former was used primarily for the precision measurement of a set of reactor functionals (material parameter, spectral indices, reactivity coefficients, etc.) in the enrichment range 1.6-4.4% and temperature range 20-1300°C, and the latter for the investigation of systems with burnable poison (boron, gadolinium). It should be noted that the ZR-6 assembly was used to thoroughly investigate the effect of technological uncertainties (material composition, core geometry, etc.) on various neutronics parameters.

An extensive set of critical experiments was performed on the uranium-graphite RBMK facility, representing a physical model of this power reactor. The inability to vary the lattice spacing was compensated for by the enrichment of ^{235}U up to 2.4%, as well as by the capacity to vary graphite content in the reactor cell and the amount of water in the channels. A portion of the results of this study was published [5].

The broadest range of systems studied consists of the small nuclear power systems intended for various purposes (i.e. power supply of decentralised customers, transport, space, etc.). Critical assemblies utilising a water and zirconium hydride moderator were made of fuel elements of a great variety of designs in the range of enrichment in ^{235}U from 5 to 95%, ratio between hydrogen and ^{235}U nuclear concentrations from 25 to 1000, temperature from 20 to 3000°C.

A brief description of the above-mentioned critical experiments can be found in Attachment 2. It should be pointed out that some of these experiments were possessed of very simple geometry and well-described composition, allowing them to be classified as benchmark experiments [6, 7]. In addition to experimental work, a large amount of effort at Kurchatov Institute has been devoted to theoretical and calculational methods and codes designed for the study of nuclear criticality safety issues for multiplication systems.

Along with the facilities described above, the Kurchatov Institute operates two solution reactors at the RRC KI, ARGUS and IIN, at which numerous critical experiments with uranil sulphide (UO_2SO_4) water solutions have been performed. Both of these reactors continue to operate, and critical experiments can be performed there now and into the future.

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Attachment 1

RRC “Kurchatov Institute” Critical Assemblies

N	NAME	TYPE	PURPOSE	POWER, W
1	**SF-1**	Uranium-water reactor prototype	Reactor core investigation at moderator temperature of up to 300°C and pressure up to 200 atm	100
2	**SF-3**	Uranium-water rreactor prototype	Reactor core investigation at moderator temperature of up to 900°C	100
3	**SF-5**	Uranium-zirconium hydride critical assembly	Recator core investigation at temperature of up to 3000°C	100
4	**SF-7**	Uranium-water reactor prototype	Reactor core investigation at moderator temperature of up to 900°C	100
5	**Kvant**	Uranium-water reactor prototype	Irradiation of samples, moderator temperature of up to 3000°C	1000
6	**Delta**	Uranium-water reactor prototype	Reactor core investigation at moderator temperature of up to 900°C	100
7	**Narciss**	Heterogeneous critical assembly with highly enriched uranium, hydride moderator, beryllium	Reactor core investigation under ambient conditions	10
8	**Astra**	Uranium-graphite core with enriched uranium, graphite reflectors	Reactor core investigation under ambient conditions	100
9	**Grog**	Uranium-graphite core with graphite reflector	Reactor core investigation under ambient conditions	100
10	**Ug**	Physical models uranium-graphite reactors, coolant simulator in the core channels – water, graphite reflector	Investigation of uranium-graphite channel reactor physics	100
11	**RBMK**	Uranium-graphite channel assembly. Experiments with and without water in reactor channels	Investigation of RBMK-type reactor core under ambient conditons	25
12	**Ephir-2M**	Uranium-water reactor prototype	Reactor core investigation under ambient conditions	100

Attachment 1 (cont.)

13	**Mayak**	Uranium-water reactor prototype	Investigation of the pulsed mode of reactor operation	10
14	**B-1000**	Prototype of water-water reactor with fuel elements enriched in ^{235}U up to 4.4%	WWER reactor core investigation under ambient conditions	200
15	**P**	Prototype of water-water reactor with fuel elements enriched in ^{235}U up to 4.4%, 6.5%, 10%	WWER reactor core investigation under ambient conditions	200
16	**MR**	Physical model of reactor physical model	MR reactor core investigation research reactor MR	100

Attachment 2

LEU-COMP-THERM-001

Regular hexagonal lattice with one-region cylindrical core, lattice spacing 11.0 and 12.7 mm. The determination of criticality parameters *Hcr* and *dp/dH* is based on measurement of reactivity *p* as a function of moderator height *H* near the critical level. Fuel rods – WWER-type fuel – uranium dioxide (UO_2), enrichment – 4.4% ^{235}U, cladding – zirconium-niobium alloy (1% Nb), moderator – distilled water. Room temperature.

LEU-COMP-THERM-002

Regular hexagonal lattice with one-region cylindrical core, lattice spacing 11.0 and 12.7 mm. The determination of criticality parameters *Hcr* and *dp/dH* is based on measurement of reactivity *p* as a function of moderator height *H* near the critical level. Fuel rods – WWER-type, fuel – uranium dioxide (UO_2), enrichment – 6.5% ^{235}U, cladding – zirconium-niobium alloy (1% Nb), moderator – distilled water. Room temperature.

LEU-COMP-THERM-003

Regular hexagonal lattice with two-region cylindrical core (inner zone with reprocessed uranium), lattice spacing 12.7 mm. The determination of criticality parameters *Hcr* and *dp/dH* is based on measurement of reactivity *p* as a function of moderator height *H* near the critical level. Fuel rods – WWER-type, fuel – uranium dioxide (UO_2), enrichment – 4.4% ^{235}U, content of ^{236}U – 0.31%, cladding – zirconium-niobium alloy (1% Nb), moderator – distilled water. Room temperature.

LEU-COMP-THERM-004 (depository)

Experiments carried out to investigate depository safety for accidents with low density of coolant. Regular hexagonal lattice with two-region cylindrical core, separated by a water gap with/without aluminium tubes, lattice spacing 12.7 mm. The number of experiments – over 30. The determination of criticality parameters *Hcr* and *dp/dH* is based on measurement of reactivity *p* as a function of moderator height *H* near the critical level. Power distribution is determined by the gamma-activity method. Fuel rods – WWER-type, fuel – uranium dioxide (UO_2), enrichment – 4.4% ^{235}U, cladding – zirconium-niobium alloy (1% Nb), moderator – distilled water and boron acid.

LEU-COMP-THERM-(005-007)

Critical experiments on the assemblies of WWER – 1000 type hexagonal bundles (7, 19, 31) with various UO_2 fuel enrichment (2.0, 3.0, 3.3, 3.6, 4.4%) in water moderator. Critical experiments and fission rate distribution measurements on the assemblies of WWER-type hexagonal cassettes with

UO_2-Gd fuel of various enrichment (3.6 and 4.4%) and Gd concentration (2.0 and 6.0%). Critical experiments on the assemblies of WWER-type hexagonal bundles and boron containing shroud tubes-mock-ups of the hermetically sealed spent fuel storage. These experiments were carried out on the LR-0 zero-power facility in Czechoslovakia.

LEU-COMP-THERM-(008-010)

The uranium-graphite facility is a physical model of RBMK power reactor. Graphite bricks of the facility are mode in the form of right-angle prisms with outer dimensions of 4.50 by 4.50 by 4.10 m^3. The square lattice of 324 fuel channels has a spacing of 250 mm. Experiments were performed at room temperature and atmospheric pressure. The experiments were performed on six critical assemblies with 1.8, 2.0 and 2.4% – enriched uranium dioxide fuel. The next series of results includes three types of assemblies of 2.0% – enriched uranium dioxide fuel. The assemblies contained typical inhomogeneities, such as rows of unloaded channels, water-filled channels, and additional neutron absorbers. Four types of assemblies were investigated with “modernised” graphite bricks, where the content of graphite in the core cell was reduced by 20%. These are assemblies of 2.0% – enriched uranium dioxide fuel with a uniform loading, a number of rows of unloaded channels, water-filled channels, and additional neutron absorbers. Each of the above assembly types was investigated in two states: with and without water in the channels with fuel. The criticality (k_{eff} = 1) was registered for all of the assemblies; the values of k_{eff}, or a reactivity margin, was measured with all absorber rods withdrawn. Distributions of the neutron flux density in critical state were measured by activation method and small fission chambers. The sources and values of errors in the experiment results were assessed.

LEO-COMP-THERM-011

UO_2 rod water-moderated lattice is presented. cylindrical fuel rods consist of stainless steel clad 85.6 cm long, 0.51 cm diam UO_2, enriched to 10%, and 0.5 cm diam UO_2 enriched to 7.5%. Variety of critical assemblies with hexagonal lattices, having spacing 0.7, 0.8, 1.22, 1.4, 1.83, 1.852, was studied at room temperature. Critical experiments for the lattices with fuel rods enriched to 10% and the spacing of 0.7, 1.4 and 1.852 were carried out in the temperature range of 20-3000°C. All these lattices in the absence of any control rods were fully reflected and their perimeters were made as close to circular as possible. Critical assemblies consisting of hexagonal clusters of such fuel rods were also studied. Besides, a set of critical experiments for square lattices with these fuel rods was performed. The lattices with a variable spacing, as well as assemblies, consisting of two separate parts with a water gap between them, were studied in these experiments.

LEU-COMP-THERM-001

The critical assemblies consisting of 19 air-cooled hexagonal fuel blocks with carbon and water reflector were studied. In the central block control rods were placed. The lattice spacing was 6.55 cm. Fuel blocks consisted of stainless steel clad 36 cm long UO_2 enriched to 20.8%, mixed with ZrHx having hydrogen-zirconium ratio of 1.88. These experiments were performed in the temperature range of 20-2400°C. Thermal neutron flux distributions and (eff were measured.

HEU-COMP-THERM-001

In these experiments there were used four fuel rod types, differing, only in uranium in concentration. The fuel block consists of UO_2 (80% enrichment), mixed and pressed with aluminium powder so that uranium concentration in each of the next block type was 30% higher than in the previous one. The uranium density range is 0.66 g/cm^3 – 1.5 g/cm^3. All fuel blocks had the same cylindrical form and the same dimensions – diameter 7 mm, height 60 mm. Each of the fuel blocks was enclosed in the leak-tight Zr cover. Fuel rod consisted of 16 fuel blocks and two 20 cm high stainless steel rods at each end of the fuel rod. Each of the fuel rods had the stainless steel cladding (diameter 13 mm) and could be dismantled. Two series of experiments were carried out with the above fuel rods:

1. Critical assemblies made of the fuel rods arranged in circular water lattices with the range of average hydrogen – ^{235}U concentration ratio of 50 to 600. For one of the fuel block types experiments were performed in the vessel with pressure 120 atm in the temperature range 20-2500°C. For the other three types experiments were performed in the temperature range 20-900°C. No control rods were in the cores in critical states. Critical numbers of the fuel rods and fission density distributions were measured.

2. Critical assemblies made of fuel rods arranged in hexagonal lattices and collected in hexagonal clusters. Fuel blocks were regularly replaced by Al or B_4C rods in some of the fuel rods. All these lattices were fully reflected, with the radial reflectors consisting of the aluminium alloy shroud covered with Cd and surrounded by water, water was also used as the top and bottom reflectors. Critical state for each assembly was attained through adjustment of B_4C rods' heights in the fuel rods. All these experiments were performed at room temperature only. Besides the critical number of fuel rods, measured were detailed fission density distributions.

HEU-COMP-THERM-002

These were metal rod hexagonal water moderated lattices. They consisted of Zr- clad 90 cm long square 0.47 by 0.47 cm U-Zr alloy (20 wt% U, 80 wt% Zr) fuel rods enriched to 90%. Four critical assemblies with these fuel rods, differing only in the lattice spacing (0.7, 0.9, 1.2, 2.4 cm) were studied at room temperature. The lattices with no control rods were fully reflected and their perimeters were made as close to circular as possible.

HEU-COMP-INTER-001

The assembly is a small heterogeneous, intermediate reactor, fuelled with highly enriched uranium dioxide (96% ^{235}U), moderated with zirconium hydride. The reactor core contains 37 fuel elements and is surrounded by a radial beryllium reflector that contains 12 rotatable control drums wit poison segments. Several benchmark critical experiments have been performed on intermediate critical assembly for the Nuclear Criticality Safety Program.

SESSION II

EXPERIMENTAL NEEDS COVERED BY CURRENT PROGRAMMES

**CRITICALITY SAFETY PROGRAMME OF CEA:
1996-2000, AND FUTURE NEEDS**

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Abstract

Firstly, this document summarises the experimental programmes in the field of criticality that are projected for the IPSN laboratory in Valduc for the period 1996-2000.

These programmes are detailed in the paper written by F. Barbry entitled "French Criticality Experimental Programmes, Potentiality of the IPSN/Valduc Criticality Laboratory: ICNC'95. These programmes are supported by IPSN and COGEMA.

Secondly, the main items needed for future experimental programmes are exposed.

The purpose of all French critical experiments is to obtain experimental reference data in order to establish the validity of computational methods so as to accurately compute k_{eff} for systems containing fissile material in all fuel cycle operations (fabrication, transport, storage, reprocessing, waste, etc.).

Experimental programme 1996-2000

Compact LWR rod arrays in water

After the renovation of the "B facility" in Valduc Centre in 1996, critical experiments will be re-initiated with the study of **undermoderated arrays** of LWR rods in water. This programme will take place at the end of two important test programmes using UO_2 and mixed oxide rods, including mixed arrays and non-symmetrical configurations.

The objective of this programme is to simulate high density storage of spent fuels and special conditions of transport.

Fission products (1st set)

Many fission products contained in the burned-up fuels get an important neutron poisoning effect, which results as part of the "burn-up credit".

The purpose of the experimental programme in Valduc is to qualify the neutron properties of the main fission product contained in the LWR irradiated fuels. A first set of fission products will be considered (^{149}Sm, ^{103}Rh, ^{133}Cs, ^{143}Nd, ^{152}Sm, ^{155}Gd) for experiment during the period 1996-2000.

Some of these will be studied separately, while others will be studied simultaneously with varying proportions in the burned-up fuels. A survey of the projected experiments can be found in different papers from IPSN in the ICNC'95 report.

Dissolution of LWR fuels

In 1990, the OECD International Calculation Working Group chaired by G.E. Whitesides demonstrated the difficulties involved with calculating "dissolver" situations, in which fissile material in solid form is surrounded by fissile material in solution. This work revealed the lack of experimental data available for such configurations.

The projected programme at Valduc will supply experimental data for this field. Critical experiments will be conducted with the core of LWR rods moderated by nitrate solutions of natural uranium.

Non-fissile structural materials (1st set)

The criticality calculations for fuel cycle facilities or for transport involve a large variety of non-fissile materials called "structural" unknown for reactor core calculations (copper, titanium, molybdenum, tantal, carbon, lead and organic materials such as polyethylene $(CH_2)_n$, Plexiglas $(C_5H_8O_2)$n, PVC $(C_2H_3Cl)_n$...).

The projected experiments will consist of the plant tanks' core containing a high enriched uranium nitrate solution directly reflected by the studied materials, in order to establish the absorption and reflection properties of these substances.

Future needs of critical experiment

The present list of future needs for critical experiments should be considered as indicative, as it results from a preliminary investigation by a group of CEA criticality specialists. Further investigations – based on neutronic studies – need to be carried out in order to examine the accuracy of these proposals, which relate to safety assessment needs and industrial programmes.

High irradiated Pu solutions (high ^{240}Pu content)

The reprocessing of MOX irradiated fuels produces large quantities of plutonium with a high content of ^{240}Pu (# 30%). Up to now, reprocessing plants have been designed taking into account a minimum of 17% ^{240}Pu. In the future, a less stringent design which takes into account the increasing value of the ^{240}Pu content could be applied. Such criticality calculations require a qualification of the codes and libraries for high irradiated plutonium.

Critical experiments could be carried out with plutonium nitrate solution (in the range 30-100 g/l) contained in a cylindrical vessel.

Fission products (2nd set)

After the study of the 6 major fission products, a second set of 8 fission products could be considered (^{95}Mo, ^{99}Tc, ^{101}Ru, ^{109}Ag, ^{145}Nd, ^{147}Sm, ^{150}Sm, ^{153}Eu). This experiment could be performed with lattices of LWR rods moderated by a nitrate solution composed of mixtures of these fission products in proportions representative of irradiated fuels.

PuO_2 or PuO_2-UO_2 low moderated powders

MOX fuel fabrication requires the handling of large homogeneous quantities of PuO_2 and PuO_2-UO_2 powders. The subcriticality of these large amounts of Pu is based on the low density of the powders and the low content of hydrogenous materials. At present, these quantities have to be limited considering the lack of experimental data for the validation of the calculations. An increase of the capacity of future MOX fuel fabrication plants will probably necessitate critical experiments to validate the codes and libraries.

Actinides

Some research for the “future reprocessing” of irradiated fuels suggests the separation of Am and Cm from the fission products. Some isotopes of Am and Cm have very low critical masses (^{242}Am: 19g, ^{243}Cm: 116 g, ^{245}Cm: 43 g). The knowledge of these values could be insufficient for the design of industrial facilities. It will be necessary to perform experiments to validate the critical values of these isotopes.

Other needs

It would be useful to qualify some “safety margins” in fuel cycle facilities:

- the subcriticality of **very low enriched uranium** (1.2%) recovered by LWR reprocessing,

- the “supplementary” margin with **random stacks of fuel pieces** involved in the dissolution process of irradiated fuels...

APPLICABILITY OF ZPR CRITICAL EXPERIMENT DATA TO CRITICALITY SAFETY

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Abstract

Zero Power Reactor (ZPR) critical experiments carried out over the past 30 years comprise a potential source of well-documented, high-quality measurements that could add significantly to the body of criticality benchmarks, particularly for metal systems. Traditionally these measurements have been largely ignored by the criticality community because of their seemingly insurmountable complexity. However, a new analytical tool that can build exact Monte Carlo input from historical reactor loading records makes it possible to generate tractable benchmark problems from past experiments at a small fraction of the cost of performing new measurements. Further, benchmark experiments can be used collectively in an evaluation code to produce unbiased estimates and uncertainties for new systems of interest as well as for evaluating the utility of proposed new benchmarks and experiments.

Experiments past for problems present

More than a hundred zero power reactor (ZPR) critical assemblies were constructed, over a period of about three decades, at the Argonne National Laboratory ZPR-3, ZPR-6, ZPR-9 and ZPPR fast critical assembly facilities. To be sure, the original reason for performing these critical experiments was to support fast reactor development. Nevertheless, data from some of the assemblies are well suited to form the basis for valuable, new criticality safety benchmarks. The purpose of this paper is to describe the ZPR data that would be of benefit to the criticality safety community and to explain how these data could be developed into practical criticality safety benchmarks.

Of the three classes of ZPR assemblies – engineering mock-ups, engineering benchmarks and physics benchmarks – the physics benchmarks tend to be most useful for criticality safety. Because physics benchmarks were designed to test fast reactor physics data and methods, they were as simple as possible in geometry and composition. The principal fissile species was ^{235}U and/or ^{239}Pu. Fuel enrichments ranged from 9% to 95%. Often there were only one or two main core diluent materials, such as aluminium, graphite, iron, sodium or stainless steel. The cores were reflected (and totally insulated from room return effects) by one or two layers of materials such as depleted uranium, lead or stainless steel. Despite their more complex nature, a small number of assemblies from the other two classes would make useful criticality safety benchmarks because they have features related to criticality safety issues, such as reflection by soil-like material.

How it was done

Criticality, the simplest and most direct integral measurement, was always measured with high precision [1]. Essentially, it consisted of specifying the contents and conditions of the "as-built" configuration. The inverse count rate was monitored in the approach to critical. The excess reactivity of slightly supercritical configurations was determined by the positive period technique or by using calibrated control rods. The reactivity of slightly subcritical configurations was determined by rod drop or source jerk inverse kinetics methods. Near-critical configurations were generally within 0.2% Δk of unity. Measurements were made for such things as temperature and matrix interface gap adjustments to criticality. All the uncertainty contributions are small. The largest source of uncertainty generally was the mass and compositions of the plate materials making up the assembly. The total criticality measurement uncertainty typically is less than 0.1% Δk.

In a few instances, a configuration of particular interest may only be available in a loading that was subcritical by as much as a few dollars. High precision subcritical source multiplication measurements were made on these configurations [2]. Typically, between 16 and 64 in-core fission chambers were used, and corrections were made for changes in detector efficiency and source importance. The measurements were calibrated using a slightly subcritical reference configuration. The total uncertainty is at most 2% for any level of subcriticality.

In addition to criticality, an experimental campaign also typically included measurements of numerous other integral quantities. The quantities measured often included reaction rate ratios, reaction rate distributions, kinetics parameters, neutron spectrum, small-sample worth distributions, control rod worths, coolant voiding worth, and ^{238}U Doppler worth. At least some of these data would have diagnostic value in a criticality assessment; they could help identify root causes of criticality mispredictions or they could help in an assessment of whether good accuracy of a criticality prediction was fortuitous. These supplemental data can be applied using a data adjustment formalism, as described below.

The making of a benchmark

The as-built ZPR critical assemblies are much too complicated to serve directly as criticality safety benchmarks. Even the simplest ZPR assembly had far more geometric detail than well-known Los Alamos benchmark fast critical assemblies such as Godiva [3]. An XY (radial) slice through a simple assembly is depicted in Figure 1. Each box represents a 5.5-cm square matrix tube containing a plate-loaded drawer. A drawer is loaded in each half of the matrix, which is split at the axial midplane. An axial segment of a matrix tube with its loaded drawer constitutes a unit cell. The XY cross section of such a cell is shown in Figure 2. The columns in Figure 2 are 5.1-cm-tall plates of the various materials that comprise a reactor region. There were always complications introduced by in-core instrumentation and operational control rods, no matter how basic the assembly loading.

Figure 1. ZPPR/12 assembly interface design

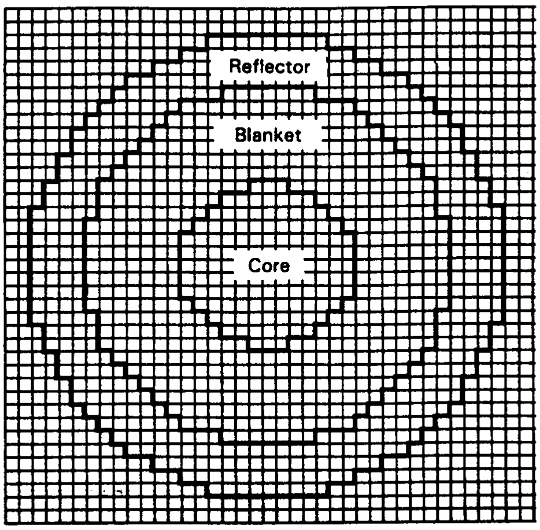
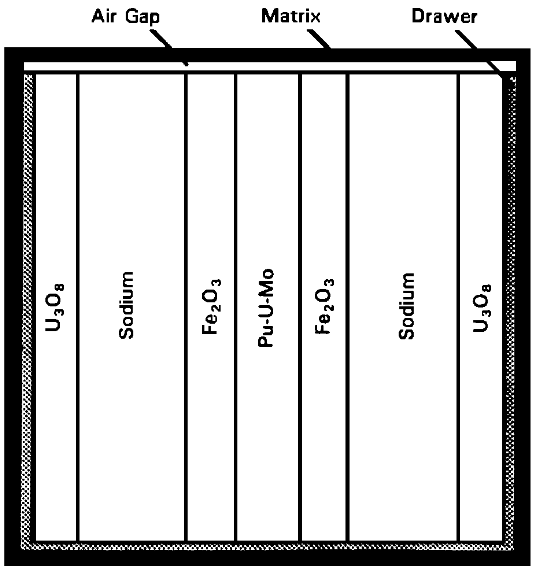


Figure 2. Typical core unit cell



Standard analysis of critical experiments employed a sophisticated deterministic calculational scheme developed over many years to deal with plate cell heterogeneity, neutron streaming and three-dimensional issues [4]. A version of this analysis scheme was used to derive simple benchmarks for the Cross Section Evaluation Working Group (CSEWG) [5]. These benchmarks were composed of a few homogeneous regions with smooth boundaries, eliminating the difficult analysis issues associated with plate critical assemblies. However, if a realistic uncertainty is assigned to the calculated conversion between the as-built assembly and the simple benchmark, the total uncertainty in the criticality of the benchmark is much larger than the uncertainty in the original criticality measurement.

A more accurate approach is now available to define the criticality safety benchmarks from ZPR experiments. Benchmarks can be derived from as-built critical configurations using continuous energy Monte Carlo. Monte Carlo models of the as-built configuration and the simplified benchmark configuration would be generated. The measured eigenvalue would be adjusted by the difference between the eigenvalues of the two models. The fully detailed models would be created using the BLDVIM code, which was developed in the last years of ZPPR operation. BLDVIM reads the assembly description on the ZPPR computer database and prepares the model input to the VIM continuous energy Monte Carlo code. BLDVIM yields a high fidelity, quality assured model, and makes quite tractable a task that would take months of effort to do by hand for even a single loading. The VIM code has a geometry option designed to be computationally efficient for plate critical assemblies. VIM accuracy has been proved through many years of use in conjunction with the analysis of ZPR critical experiments and with British and Los Alamos benchmarks. The features that are difficult to deal with deterministically (e.g., plate heterogeneity) are no problem for continuous energy Monte Carlo. Thus, the adjusted eigenvalue of the benchmark model would be very accurate; a total uncertainty on the order of 0.2% Δk is expected with this approach, which is much better than the ≈0.5% Δk accuracy of the ZPR CSEWG benchmarks. To minimise the Monte Carlo extrapolation of the experimental data, the benchmark model should deviate as little as practical from the exact model and, at the same time, be easily calculable by a variety of standard criticality safety codes. The viability of this approach has already been demonstrated by its application to ZPPR Assembly 21 [6], the last of the plate critical assemblies and the only one dedicated specifically to criticality safety.

Meet the candidates

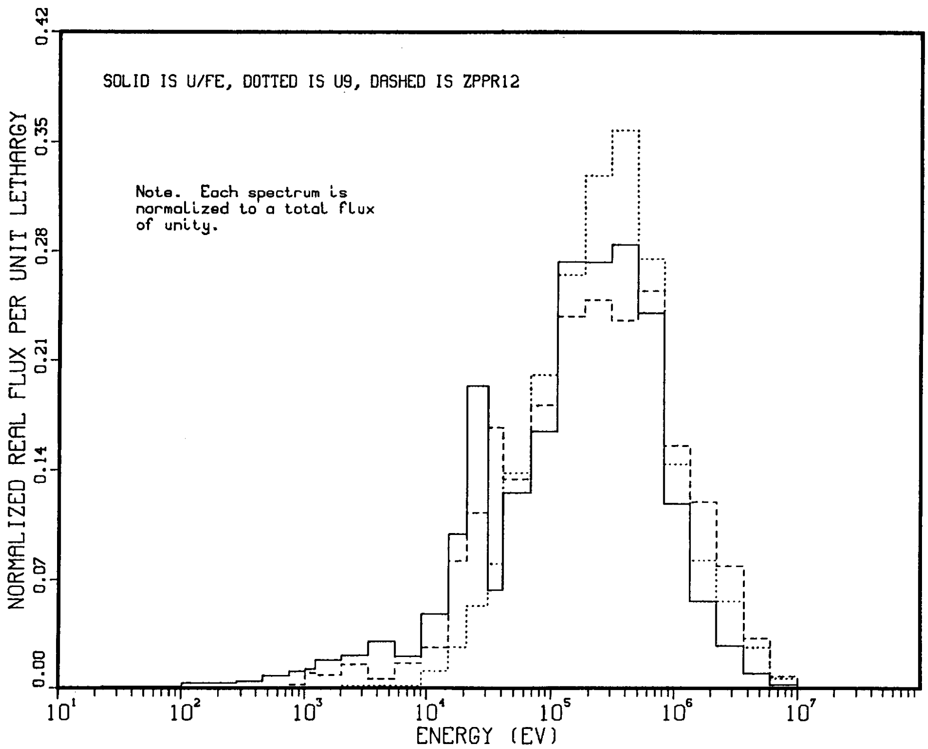
Many of the assemblies that would add significantly to the criticality safety database are listed in Table I. The assemblies are identified by facility and assembly number, e.g., ZPR-9/34 is Assembly 34 at the ZPR-9 facility. Brief sketches of some features of each assembly are included in Table I. In general, complicated assemblies were excluded unless they had some important and unusual characteristic. Three characteristics were used in the selection of assemblies: 1) a simple core composition that could help identify problems with neutron cross section data, 2) a variation of reflector material, again with concern about cross section data problems, and 3) a geometric feature related to difficulties in criticality safety calculations. Many of the candidates come from two series of diagnostic benchmark cores, the ZPR-3 plutonium-fuelled benchmark series [7], and the ZPR-6 and 9 Diagnostic Cores Programme [8].

Table 1. ZPR assemblies useful for criticality safety

Assembly	Attributes
ZPR-9/35	All-U core and reflectors; 9% enriched core; hard, narrow spectrum.
ZPR-9/34	Core: 93% enriched U, Fe diluent; SS reflectors; broad spectrum.
ZPR-3/23	Core: 93% enriched U, Al diluent; DU reflectors.
ZPR-3/14	Core: 93% enriched U, C diluent; DU reflectors.
ZPR-6/10	Core: ^{239}Pu, C and SS diluents; SS reflectors. Soft, broad spectrum.
ZPR-3/53	Core: ^{239}Pu, C and DU diluents; DU reflectors.
ZPR-3/54	Core: ^{239}Pu, C and DU diluents; SS reflectors.
ZPR-3/58	Core: ^{239}Pu, C diluent; DU reflectors.
ZPR-3/59	Core: ^{239}Pu, C diluent; Pb reflectors.
ZPPR/8	Radially heterogeneous Pu oxide LMFBR; U vs. Th in blankets.
ZPPR/12	Pu oxide LMFBR comp. but simple geometry core; DU vs. U_3O_8 blankets.
ZPPR/13A	Core comp. similar to ZPPR/12 but alternating annular rings of core and blanket.
ZPPR/15	Metal-fuel LMFBR comp.; ^{239}Pu vs. ^{235}U fissile; Zr vs. SS in a zone.
ZPPR/20	Core: ^{235}U with Li, Nb, Re; compare reflection by BeO, CH_2, SiO_2.
ZPPR/21	Compare ^{235}U vs. ^{239}Pu fissile; Zr, SS & DU diluents; C reflectors.

Figure 3 shows plots of core neutron spectrum from three of the assemblies. It can be seen that there is a wide variation in energy range and emphasis.

Figure 3. Comparison of core neutron spectra from three ZPR assemblies



Features of interest include the following:

- ZPR-9/35, also known as the U9 benchmark, had a simple geometry and the simplest composition of any ZPR assembly. Except for the stainless steel (SS) matrix tubes and drawers, it consisted entirely of uranium – 9% homogenised enrichment in the core and depleted uranium metal (DU) reflectors.

- The alternating rings of blanket and core in ZPPR/13A bear some resemblance to a storage configuration with rows of fissile material separated by non-fissile material. The fuel composition was Pu-^{238}U oxide and the spectrum was moderately hard. ZPPR/12 had a core with a similar composition to that of ZPPR/13A but the core geometry was a simple cylinder, which should make a comparison between these two assemblies useful.

- ZPPR/8 is the only ZPR assembly that contained thorium. Some configurations of this engineering benchmark differed only by depleted uranium replacing thorium in some blanket regions.

- The ZPPR/15 configurations provide a comparison between ^{235}U and ^{239}Pu metal fuels in very similar critical configurations. The enrichment was 15-20% and the spectrum was moderately hard. The effect of replacing steel with zirconium can be deduced by comparing two of the configurations.

- Highly enriched U and ^{239}Pu can be compared using ZPR-3/14 and ZPR-3/58. Both of the cores had graphite as the principal diluent and had DU reflectors.

- Different diluents used with ^{239}Pu can be compared using ZPR-3/58, ZPR-3/54 and ZPR-6/10. Similarly, different simple diluent mixes used with highly enriched U can be compared using ZPR-9/34, ZPR-9/35, ZPR-3/14 and ZPR-3/23.

- ZPPR/21 was constructed specifically to address criticality safety for Argonne's Fuel Conditioning Facility. Both ^{235}U and ^{239}Pu metal fuels were mocked up. The enrichment was 50-60% and the spectrum was hard.

- The effects of different reflector material can be evaluated using several of these cases. ZPR-3/53 and ZPR-3/54 were the same except for DU and SS reflection, respectively. Similarly, ZPR-3/58 and ZPR-3/59 differed only by DU vs. Pb reflection. Variants of ZPPR/12 differed by use of DU vs. depleted uranium oxide for the reflectors. ZPPR/20 had variants with different reflecting materials including sand (SiO_2), which could be of particular interest for waste burial.

A good example of the relevance of these assemblies to the criticality safety community is ZPR-9/34, the Uranium/Iron Benchmark. An article in a recent issue of the Criticality Safety Quarterly[1] describes an investigation of large discrepancies among criticality predictions for simple metal/^{235}U systems. The article states, incorrectly, that "there are no experiments that adequately represent the characteristics of these metal/^{235}U systems, and so the ability to predict the 'correct' critical configuration is not known." In fact, the Uranium/Iron Benchmark, which was built in 1980, is closely related to the iron/^{235}U system described in the article, since it was a critical assembly composed predominately of iron and 93% enriched uranium. Furthermore, the problems with treatment of resonance cross section behaviour in the neutron cross sections of iron and other structural materials uncovered in the investigation were discovered previously, in connection with analysis of the Uranium/Iron Benchmark[2]. There is other information from ZPR experiments that is relevant to this investigation, including the aluminium/uranium critical assembly ZPR-3/23, and code comparisons for model problems inspired by ZPR critical assemblies.

Putting it all together – data adjustment

In addition to physics measurements directly relevant to criticality safety, there exists an extensive database of supporting measurements and calculations resulting from decades of fast reactor research conducted at the ZPRs, as well as independent data from Los Alamos National Laboratory and the United Kingdom. This database currently contains over 250 integral measurements of parameters such as k-eff, control rod worths, boron worths, and reaction rate ratios. The supplementary database contains a wealth of information that can be used to validate and extrapolate measurements of primary concern.

The full potential of the database is realised through the use of formal data fitting procedures incorporated in the GMADJ (**G**auss-**M**arkov **Ad**justment) code [9]. GMADJ employs a generalised least-squares fitting procedure that combines information contained in the integral measurements with the pre-evaluated data library via sensitivity coefficients. The system is over-determined and a best (minimum variance, unbiased) estimate of any integral parameter is obtained. The procedure also makes use of covariances associated with both integral and differential data.

Use of the supplementary database by way of the GMADJ procedure provides several powerful capabilities. This procedure has proven effective [10] as a consistency check for experimental data as well as a means of validating experimental techniques. For example, performing the least-squares fit using independent data immediately points to measurements or calculations that are inconsistent with the rest of the database, therefore providing an effective layer of quality control. GMADJ also supplies biases and uncertainties of parameters not directly measurable (for both supplementary data and data of primary concern), thus effectively extending the knowledge from a given experiment. One aspect of the data fitting procedure which may be particularly useful for criticality analyses is the ability to predict biases and uncertainties relating to systems for which no measurements exist. This is accomplished by extrapolating the existing database via sensitivity coefficients. GMADJ also produces a set of correlation coefficients after fitting which provides information on the relevance of a measurement in one system to a measurement in another system, making GMADJ a useful tool for correlation analyses. The ability to use all available physics information via the GMADJ least-squares fitting procedure compliments the directly relevant measurements and enhances any criticality data analysis.

Making it happen

Some effort would be needed to confirm that all the selected experiments are free from significant flaws. In the last 20 years of operation, care was taken to make sure that room return effects were totally negligible. However, this could be an issue for two of the older assemblies, ZPR-3/54 and ZPR-3/59. Since all the ZPR-3 experiments are quite old, the data from them would have to be scrutinised carefully.

Many of the selected assemblies are not on the ZPPR computer database but their descriptions could be entered from hard-copy archival records. This would involve transcribing drawer master and matrix loading documents. The plate description library currently covers most of the materials but a few plate types would have to be added. Once the computer database is expanded, all the cases could be processed using the proposed modelling scheme.

The capability to process the data cannot be assured far into the future. The last Argonne critical facility, ZPPR, ceased operation several years ago. Its personnel have dispersed throughout the laboratory and beyond. The only computer on which the ZPPR relational database can be used is increasingly subject to breakdowns. Moreover, the programmatic funding source (IFR Programme) that provided the funds to maintain ZPPR in a non-operational standby state has been terminated.

Criticality safety benchmarks from ZPR fast neutron critical assembly experiments would significantly broaden the scope of the benchmark database and thus provide a valuable contribution in the increasingly important study of criticality safety. The wealth of criticality data that heretofore has been largely inaccessible to the criticality safety community could be used to test criticality safety tools and cross section data. The data adjustment formalism offers a way to apply the data to unmeasured configurations and estimate the uncertainty in the criticality predictions.

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CRITICALITY FACILITIES AND PROGRAMMES AT SANDIA NATIONAL LABORATORIES

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Abstract

The reactor facilities at Sandia National Laboratories have hosted a number of reactors and critical experiments. A critical experiment is currently being done to support an ongoing investigation by the US Department of Energy concerning the consequences of taking fuel burnup into account in the design of spent fuel transportation packages. A series of experiments, collectively called the Spent Fuel Safety Experiment (SFSX), has been devised to provide integral benchmarks for testing computer-generated predictions of spent fuel behaviour. A set of experiments is planned in which sections of unirradiated fuel rods are interchanged with similar sections of spent pressurised-water reactor (PWR) fuel rods in a critical assembly. By determining the critical size of the arrays, one can obtain benchmark data for comparison with criticality safety calculations.

The SFSX provides a direct measurement of the reactivity effects of spent PWR fuel using a well-characterised, spent fuel sample. The SFSX also provides an experimental measurement of the end-effect, i.e. the reactivity effect of the variation of the burnup profile at the ends of PWR fuel rods. The design of the SFSX is optimised to yield accurate benchmark measurements of the effects of interest, well above experimental uncertainties.

Reactor facilities

A number of reactors have been operated at the reactor facilities in Sandia National Laboratories Technical Area V. Currently, three reactors are available for operation in the area: the Annular Core Research Reactor (ACRR), the Sandia Pulsed Reactor III (SPR-III), and the Sandia Pulsed Reactor II (SPR-II) [1].

The ACRR is a water-moderated reactor with fuel rods containing a sintered mixture of uranium dioxide and beryllium oxide. The reactor core is at the bottom of a 10-meter-deep pool. The reactor has a 23-cm-diameter dry irradiation cavity in the centre of the core. Larger irradiation cavities can be attached to the outside of the core as experimental needs dictate. The ACRR is used for both pulsed and steady-state irradiations for fissile experiment irradiations and radiation effects studies. This reactor is currently being modified to produce medical isotopes.

The SPR-II and SPR-III are metal-fuelled reactors used primarily as short-pulsed neutron sources. These reactors are fuelled with annular assemblies of fully enriched uranium metal alloyed with 10 weight percent molybdenum. The neutron spectrum of these reactors is nearly identical to the fission neutron spectrum. These reactors are used primarily for radiation effects studies.

A critical assembly was operated in the area in the late 1980s to support the Space Nuclear Thermal Propulsion programme. A water-moderated UC-ZrC-fuelled assembly was constructed to mock up the behaviour of a proposed nuclear rocket engine test facility.

Currently, a critical experiment known as the Spent Fuel Safety Experiment, is being designed and constructed. This experiment is described in detail below.

The Sandia reactor facilities have been used for radiation effects studies, nuclear reactor accident experiments, critical experiments, and a number of other programmes. There has not been a continuous critical experiments programme at the laboratory. Criticality experiments have been performed for sponsors on an as-needed basis. This fact has both disadvantages and advantages for a critical experiment programme. The disadvantage is that an experiment team must be assembled for each experiment and use of the facilities must be scheduled in conjunction with the other experimental programmes. The advantage is that the infrastructure necessary to do the experiments already exists at the laboratory. Thus the sponsors of critical experiments need only pay for that portion of the infrastructure that is used during the experiment. Maintenance of the infrastructure is largely funded by the other reactor programmes.

The Spent Fuel Safety Experiment

A key issue related to acceptance of spent fuel burnup credit is the verification of the impact of the composition of the spent fuel on the system self-multiplication factor. The Spent Fuel Safety Experiment (SFSX) directly addresses this issue. A second issue that SFSX addresses is the effect of axial burnup distribution in spent PWR fuel. The “end-effects” issue has been a pivotal concern since the inclusion of burnup credit was proposed for transportation and storage of spent PWR fuel. The objective of SFSX is to provide a benchmark for criticality calculations of spent PWR fuel. Desirable features of the SFSX benchmarks are (1) they provide critical experiments using actual spent PWR fuel, (2) they provide critical data at ambient temperature, eliminating moderator temperature concerns, (3) they provide a fresh fuel critical experiment as a reference point enabling the determination of the worth of the spent fuel and (4) they provide a benchmark for criticality calculations with axially varying fuel burnup and a basis for the determination of the reactivity worth of the under-burned ends relative to the spent fuel centre sections.

The ability to accurately predict the self-multiplication factor for spent PWR fuel configurations depends on two critical factors: (1) accurate determination of the constituents of the spent fuel (isotopics) and (2) accurate calculation of the multiplication factor for the known isotopics and geometry. The ATM-104 material [2], briefly described below, provides chemically analysed (known isotopics) spent PWR fuel. The isotopes for which the ATM-104 fuel was analysed can be used precisely in comparing calculated and measured values of the self-multiplication factor for this benchmark critical experiment.

Experiment fuels

Two types of fuel are available for the SFSX critical experiments, a large amount of unirradiated or fresh fuel, and a smaller amount of irradiated or spent fuel. The critical assemblies consist primarily of the fresh fuel which provides a neutron environment in which the effects of the spent fuel are measured.

The spent fuel sections are cut from fuel rods in assembly D047 of the Calvert Cliffs Nuclear Power Plant (Unit 1), which achieved an assembly-averaged burnup of about 42 MWd/kgM. Before irradiation in the reactor, the fuel rods consisted of 3.04 % enriched UO_2 pellets with a pellet diameter of 0.9563 cm. The rods are clad in zircaloy-4. Selected rods from this assembly were characterised at the Pacific Northwest Laboratory [2]. The fuel sections used in the SFSX are taken from the fuel rods immediately surrounding a rod that was destructively characterised. After cutting, the spent fuel sections are placed in a second unirradiated, sealed zircaloy-4 rod which prevents water infiltration to the fuel and fission-product migration to the rest of the assembly.

Axial gamma scans of several rods from the ATM-104 bundle were made for ^{137}Cs activity (an indicator of burnup) and reported in Reference 2. The implied burnup for these rods is relatively flat over most of the length of the rod with the exception of about 60 cm on the ends where the burnup drops sharply. With a lower average burnup, the end sections are more reactive than the centre sections. The 50-cm height of the unirradiated fuel in the SFSX assembly was chosen to give the experiment sensitivity to this relatively short region of rapidly varying burnup.

The fresh fuel [3] is 4.31 % enriched UO_2 pellets with an outside diameter of 1.27 cm. The fuel is clad in zircaloy-4. The height of the fuel pellet stack in the fresh fuel is nominally 50 cm.

Experiment design

The SFSX is a fuel-replacement experiment. As such, it consists of a series of three approach-to-critical experiments. The first experiment uses only unirradiated or fresh fuel elements, described below. This provides the first benchmark and a demonstration of the approach to critical procedures. The primary benefit of this fresh fuel data point is to provide a basis for the determination of the spent fuel worth.

For the second experiment, the seven centrally located unirradiated fuel elements are replaced by spent fuel taken from the midsections of several of the fuel rods from the ATM-104 assembly. The centre sections of the spent fuel rods are selected specifically to eliminate considerations of axial burnup variations. The burnup is relatively uniform along 50 cm near the centre of the fuel rods.

Replacing the fresh fuel with the spent fuel results in a decrease in multiplication, directly providing a reactivity worth of the spent fuel. Fresh fuel is added to the remaining outer locations in the assembly until a delayed critical configuration is reached. This configuration is the benchmark data point for calculation of a spent fuel critical.

The third experiment is similar to the second except that the spent fuel originates from the end sections of the ATM-104 fuel. The distribution of burnup effects, particularly the under-burned ends of the fuel rods, is addressed by this third measurement. Because the burnup of the end fuel section has a large gradient, this measurement, in addition to providing a comparison of the worths of end and centre sections, provides a challenging benchmark for axial nodalisation in criticality calculations.

The SFSX assembly is designed for optimum sensitivity to the effects of interest rather than to mock up a specific fuel configuration. The 2.8-cm-fuel rod pitch was chosen to minimise the total fuel loading at delayed critical (DC). The fuel height was set at 50 cm to give adequate sensitivity to the end effects in the spent fuel.

Each experiment is run as a standard approach-to-critical experiment. During each approach to critical, the count rate data are collected for successively larger fuel arrays and used to generate plots of inverse multiplication compared to the corresponding number of fuel rods in the assembly. From the plots, estimates are made of critical fuel mass for each experiment. The fuel additions are continued until the plots indicate that the next fuel rod addition will result in a delayed critical configuration. The fuel loading at which the plots indicate that the assembly is at exactly delayed critical is the benchmark datum for the experiment. The next fuel rod is then added to obtain a fuel configuration slightly beyond delayed critical and the resultant positive period is measured. The positive period is also reported and can be used with the calculated delayed neutron fraction for the assembly to obtain the self-multiplication factor.

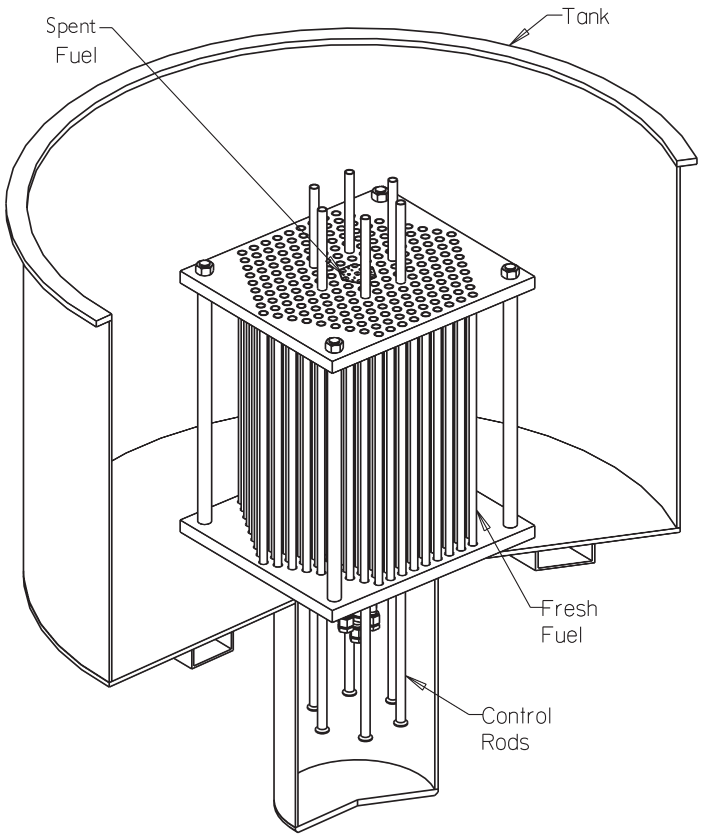
The relative reactivity worth of the spent fuel can be assessed by comparing the measured delayed critical fuel loading with and without spent fuel. An analytical effort to model the experiments and anticipate the core loadings required to yield the delayed critical conditions is carried on in parallel with the experiments. The primary analytical tools are the Monte Carlo neutronics code KENO-Va and a 27-group cross-section library that includes cross-sections for many fission products, both from the SCALE package of codes and data [4].

Assembly design

The SFSX is a water-moderated and -reflected assembly with the fuel rods in a 2.8-cm triangular pitch. Figure 1 shows an overall view of the SFSX hardware. The assembly tank is a right circular cylinder approximately 120 cm in diameter and 90 cm tall with a 30-cm diameter cylindrical projection out the bottom. The assembly fuel is supported by two 2.54-cm-thick aluminium grid plates. The top grid plate is drilled through with the grid pattern for insertion and removal of the fuel. The bottom grid plate consists of a sandwich of two 1.27-cm-thick plates, the upper one being drilled through with the grid pattern and the lower being solid to support the fuel. The spacing between the grid plates is 50 cm. The grid plates are mounted in the tank so that a water reflector of at least 10-cm thickness surrounds the assembly.

The unirradiated fuel is fabricated with a nominal 50-cm fuel height. The cladding is zircaloy-4 with a 1.27-cm-thick lower end plug and a 2.54-cm-thick upper end plug. When the fuel rods are inserted in the assembly, the top of the lower end plug is at the elevation of the top of the lower grid plate.

Figure 1. Overall view of the SFSX critical assembly



The upper and lower edges of the upper end plug are at the same elevations as the upper and lower surfaces of the upper grid plate. As a result, the end plugs in the fresh fuel exactly fill the holes drilled in the grid plates. The assembly can be modelled as a fuel/clad/water region with solid metal plates above and below surrounded by a water reflector.

The assembly grid plates have hexagonal holes machined in them that remove the central seven fuel rod positions. During the critical experiment using all fresh fuel, these holes are filled by 2.54-cm-thick auxiliary grid plates that complete the grid pattern. The auxiliary grid plates are removed during experiments that include spent fuel. The seven spent fuel rod sections are mounted in a bundle with upper and lower grid plates that fit in the hexagonal holes in the assembly grid plates. The spent fuel bundle is attached to an actuating mechanism that moves the bundle from the centre of the assembly down into the projection in the bottom of the assembly tank. The spent fuel bundle is driven to this lower position at times when personnel are in the assembly area to reduce radiation levels in the vicinity.

During operation, the assembly is controlled by six identical fuel-followed control rods. The control rods are attached through independent electromagnets in pairs to three control element drives. Each control rod drops independently during an assembly scram. The control rods have a lower fuelled section that matches the grid plate-fuel-grid plate configuration of the rest of the assembly. Above the fuelled section is a 10-cm-thick polyethylene section below a 50-cm boron-carbide absorber section. The polyethylene is included to decouple the absorber section from the assembly with the control rods fully raised. With a control rod up, the grid position occupied by the control rod is nearly identical to one occupied by a fresh fuel rod.

Results

Several parameters of the SFSX critical assemblies were calculated using the SCALE code and cross-section package [4]. These parameters, for both critical assemblies that include spent fuel, are shown in Table 1. The first line of the table gives the smallest total number of grid locations occupied by both fresh and spent fuel rods when the assembly exhibits a positive period. The second line gives the reactivity worth of an incremental fresh-fuel rod which can be used to translate the reactivities given in the following table lines into array size increments. The third and succeeding lines give these reactivities ($\Delta k/k$) for several modifications to the assembly fuel. These data can be used as figures-of-merit for the assemblies.

The third line of Table 1 gives the reactivity difference produced by replacing the spent fuel bundle with assembly fresh fuel. The fourth line shows the sensitivity of the assembly to the difference between the spent fuel centre sections and the end sections. The fifth line in the table gives the reactivity effect of removing the fission products from the spent fuel, thus showing the capability of the assemblies to detect the difference between taking full burnup credit and taking credit only for fissile material depletion including actinide build-up. The sixth line shows the reactivity effect of removing the fission products and the absorbing plutonium isotopes, the difference between full burnup credit and assuming only fissile depletion and fissile plutonium build-up. The final line in the table gives the reactivity effect of replacing the spent fuel with unburned PWR fuel (the composition of the test fuel before it was irradiated).

The calculated results in Table 1 show that the SFSX critical experiments have sufficient sensitivity to detect the fission products in the spent fuel (a 0.60 % reactivity difference is more than four fresh fuel rods difference in the array size). This is the difference between the full burnup credit model and the model that assumes only fissile depletion and actinide build-up.

Table 1. SFSX calculated parameters

<table>
<tr><th colspan="2"></th><th>Seven Spent Fuel Centres</th><th>Seven Spent Fuel Ends</th></tr>
<tr><td colspan="2">Total Grid Locations Occupied at DC+*</td><td>156</td><td>154</td></tr>
<tr><td colspan="2">Incremental Fresh Fuel Rod Worth at DC+ (%)</td><td>0.134</td><td>0.136</td></tr>
<tr><td></td><td>Replace Spent Fuel with Assembly Fresh Fuel</td><td>3.41 ± 0.06</td><td>3.08 ± 0.06</td></tr>
<tr><td></td><td>Replace Spent Fuel Centre Sections with End Sections</td><td colspan="2">0.30 ± 0.04</td></tr>
<tr><td>Δk/k (%)</td><td>Remove Fission Products</td><td>0.62 ± 0.04</td><td>0.56 ± 0.04</td></tr>
<tr><td></td><td>Remove Fission Products and Absorbing Plutonium Isotopes</td><td>0.99 ± 0.04</td><td>0.72 ± 0.04</td></tr>
<tr><td></td><td>Replace Spent Fuel with Fresh PWR Fuel</td><td>1.97 ± 0.05</td><td>1.65 ± 0.05</td></tr>
<tr><td colspan="4">Array size at DC+ with all assembly fresh fuel: 131
Incremental fresh fuel rod reactivity worth at DC+: 0.167%
* DC+ is the smallest array expected to be supercritical</td></tr>
</table>

The SFSX assemblies are flexible and can be used to perform many other critical experiments. Experimental parameters that could be varied are the fuel rod pitch (to vary the neutron spectrum), the number of spent rods in the bundle, and the type of spent fuel used (e.g., different burnup). The assemblies can also be used to address the effects of individual fission products by inserting fuel rods doped with the fission product of interest.

Conclusions

The SFSX provides a direct measurement of the reactivity effects of spent fuel using a well-characterised, spent fuel sample (ATM-104). The SFSX provides an experimental measurement of the end-effect, i.e. the reactivity effect of the variation of the burnup profile at the ends of the fuel rods. The design of SFSX is optimised to yield accurate benchmark data for the effects of interest. The reactivity effects of the spent fuel fission products, as well as the end effects, can be measured well above experimental uncertainties.

The SFSX is a fuel-replacement experiment designed to measure the critical array size for three fuel configurations. The first configuration includes only unirradiated fuel. The second includes a seven-rod bundle of fuel sections cut from the centre of a PWR fuel assembly where the burnup is relatively constant. The third configuration includes a seven-rod bundle of fuel sections from the ends of a PWR fuel assembly with steeply varying burnup. The composition of both fuel types is well known, having been measured previously.

The measured array size data constitute integral benchmarks against which the codes and cross-sections used to perform criticality calculations on spent nuclear fuel containers can be tested.

Acknowledgements

This work has been supported by the United States Department of Energy under Contract No. DE-AC04-94ALB5000.

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CRITICALITY SAFETY RESEARCH PROGRAMMES IN JAPAN

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Abstract

In order to ensure and improve the safety of nuclear installations, the Japanese Nuclear Safety Commission has established a new five-year (from fiscal 1996 to 2000) national programme for safety research. This programme comprises the safety research programmes of nuclear facilities, and radioactive materials transport; under this last category is included the programme for criticality safety research.

The criticality safety research of the five-year programme consists of the seven following areas:

1. Research on criticality safety evaluation techniques,
2. Measurement of criticality data,
3. Research on criticality safety of chemical processes,
4. Research and development of subcriticality measuring systems,
5. Development of criticality safety control systems for spent fuel storage,
6. Research on criticality safety control and analysis of the MOX fuel fabrication plant,
7. Criticality safety research on LWR fresh (including MOX) and spent fuel transportation casks.

The contents of the above research items are presented in this paper.

Research on criticality safety evaluation techniques

1) Data collection on criticality conditions of low-enriched uranium, plutonium and their mixed solution and powder, and minimum critical mass of minor actinide nuclides for revising the criticality safety handbook of Japan,

2) Development of computer codes for analysing super critical phenomena of low-enriched uranium and plutonium solution,

3) Analyses of miscellaneous phenomena such as reactivity changes due to fuel particle size, non-homogeneous fuel density distribution, etc., and neutron isolation length of structural materials such as concrete, SUS, and so on,

4) Probabilistic study on criticality accidents for obtaining accident analysis scenarios.

Measurement of criticality data

1) Measurement of criticality conditions of low-enriched uranium and MOX fuel solution,

2) Measurement of super critical phenomena of low-enriched uranium solution including neutron kinetics, reactivity feedback and radiation dose evaluation,

3) Development of an alarm system for criticality accident,

4) Measurement of reactivity worth of FPs, actinide nuclides and scuttle material.

Research on criticality safety of chemical processes

1) Atomic number density of each nuclide in MOX fuel solution,

2) Chemical properties of a reprocessing process under transient condition,

3) Criticality conditions of fuel solved in solvent such as TBP,

4) Separation and deposition of fissile materials from solution.

Research and development of subcriticality measuring systems

1) Subcriticality measurement with uranium and plutonium fuel by different methods such as neutron multiplication, pulse or noise methods for development of subcriticality monitor,

2) Development of subcriticality measurement technology.

Development of criticality safety control systems for spent fuel storage

1) Accumulation of isotopic composition data in spent fuel of PWR and BWR,

2) Development of criticality safety control systems taken into account of burnup credit,

3) Development and evaluation of burnup codes and burnup measurement instruments.

Research on criticality safety control and analysis of the MOX fuel fabrication plant

1) Preparation of computer code systems for criticality control of MOX fabrication facilities,

2) Accumulation of criticality safety data of a real plant,

3) Development of a criticality safety monitoring system.

Criticality safety research on LWR fresh (including MOX) and spent fuel transportation casks

Research items are generally the same as found in the section entitled *Development of criticality safety control systems for spent fuel storage*.

EXPERIMENTAL NEEDS IN CRITICALITY SAFETY SURVEYED IN JAPAN

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Abstract

Experimental needs in criticality safety have been surveyed in Japan through a questionnaire. Twenty-eight answers from 19 experts have been summarised and arranged in four tables, i.e. needs in criticality experiments, reactivity worth measurements, subcritical and supercritical experiments.

Introduction

In order to clarify the experimental needs in criticality safety of the Japanese specialists, a survey was carried out in August 1995, prior to this specialists' meeting. A questionnaire was sent to 42 people, many of whom are members of the specialists' committee on nuclear criticality safety, chaired by Prof. R. Kiyose. The questionnaire is provided in the annex.

We received 28 answers from 19 people. The results are summarised and arranged in four tables – Table 1 to Table 4. Each table contains four items, i.e., objective, fuel, experimental facility, and note (the latter giving the name of the person suggesting the needs).

Table 1 shows needs in critical experiments. There are many requests for experiments related to burnup credit (No. A1-A5), MOX fuel (No. A6-A10), and advanced fuel recycling(No. A11-A13). Other requests for critical experiments concern compact storage (No. A14), the poisons inside the fuel tank (No. A15), the dissolver problem (No. A16-A17), the partial moderation (No. A18) and the interactions among fuel units (No. A19).

The various needs in reactivity worth measurements are collected in Table 2.

The needs in subcritical experiments are summarised in Table 3. The data are expected not only to be applicable to the development of subcritical monitors, but also to lead to other evaluation methods than the present method based on criticality. Supercritical experiments requested, as shown in Table 4, have the possibility of greatly improving our understanding of criticality safety.

Table 1. Critical experiments proposed and requested in Japan

No.	Objective	Fuel	Exp. Facilities	Note
A1	BUC	Spent fuel		Involved in the subcriticality and isotopic measurements of LWR SF at JAERI
A2	Benchmark data for burnup calculation and BUC	Spent fuel assembly	TCA or DCA	Proposed by Y. Yamane (Nagoya Univ.)
A3	Treatment of high burnup fuel or a fuel containing MAs for transmutation	U or Pu containing FP and MA		Proposed by T. Misawa (Nagoya Univ.)
A4	Effective use of BUC in SF management from high burnup LWRs involving MOX fuel, and FBRs in the near future	FPs and MAs	TCA	Proposed by T. Suzaki (JAERI)
A5	Direct validation of whole process of the criticality safety design taking BUC	Actual fuel assembly	Transfer facility and shielding capability should be enlarged	Requested by K. Itahara *et al.* (MHI)
A6	Criticality safety analysis of LWR-MOX fuel fabrication facilities	MOX fuel (powder and rods)		Requested by S. Mitake (NUPEC)

Abbreviations (SF: Spent Fuel, FP: Fission Product, MA: Minor Actinide, BUC: Burnup Credit)

Table 1. Critical experiments proposed and requested in Japan (*cont.*)

No.	Objective	Fuel	Exp. Facilities	Note
A7	MOX fabrication, storage	MOX fuel (powder and pellet)	A new facility with humidity control for powder	Requested by Y. Nomura (JAERI)
A8	Construction of MOX fabrication facility	MOX powder	NUCEF (should be modified)	Requested by H. Aoyagi (JNF)
A9	Benchmark data for calculational codes	MOX fuel	EOLE (France), VENUS (Belgium), TCA	Requested by I. Mitsuhashi (Toshiba)
A10	Heterogeneity and/or non-uniformity effect	Granulated UO_2 and MOX	A new facility with various grain sizes	Requested by H. Okuno (JAERI)
A11	Development of the advanced fuel recycle	MOX of Pu, U and MA	NUCEF (should be modified)	Requested by I. Nojiri (PNC)
A12	Validation of the calculation system to be used fuel cycle facility design for improved nuclear fuel cycle applications	U-Pu-MA mixtures	Some existing critical facilities for Pu solution or powder	Requested by K. Itahara *et al*. (MHI)
A13	Advanced reprocessing method (actinide recycle process)		NUCEF	Requested by T. Fukasawa (Hitachi)
A14	Compact storage	High burnup fuel and/or MOX fuel		Requested by H. Maruyama (Hitachi)
A15	Advanced criticality safety design of reprocessing plant	Organic fuel solution and fuel powder	NUCEF/STACY (should be modified)	Proposed by K. Nakajima (JAERI)
A16	Criticality safety analysis of fuel dissolver in reprocessing plant for LWR fuels	Array of LWR fuel rods immersed in U-Pu nitrate solution		Requested by S. Mitake (NUPEC)
A17	Validation of the calculation system to be used for the dissolver	LWR UO_2 and MOX fuel rods immersed in fuel solution with poison	TCA or PNL	Requested by T. Natsume (MHI)
A18	Verification of a peak in the multiplication factor in partial moderation	Array of LWR fuel rods with various water density		Requested by S. Mitake (NUPEC)
A19	Understanding criticality accidents induced by local disturbances in multi-unit systems like SF storage pond	Fuel units separated by several materials	TCA	Proposed by T. Suzaki (JAERI)

Abbreviations (SF: Spent Fuel, FP: Fission Product, MA: Minor Actinide, BUC: Burnup Credit)

Table 2. Reactivity worth measurements proposed and requested in Japan

No.	Objective	Fuel	Exp. Facilities	Note
B1	Effective use of BUC in SF management from high burnup LWRs involving MOX fuel, and FBRs in near future	FPs and MAs	TCA	Proposed by T. Suzaki (JAERI)
B2	Rationalisation of the evaluation conditions such as calculational hypotheses to simulate reflection or moderation	Minor objectives (e.g. piping, supports, pots) inserted in the reflector position	NUCEF with some modifications	Requested by K. Itahara *et al*. (MHI)
B3	Evaluation of absorber effect			Requested by Y. Naito (JAERI)
B4	Benchmark data for calculation codes	MOX fuel	EOLE (France), VENUS (Belgium), TCA	Requested by I Mitsuhashi (Toshiba)

Abbreviations (SF: Spent Fuel, FP: Fission Product, MA: Minor Actinide, BUC: Burnup Credit)

Table 3. Subcritical experiments proposed and requested in Japan

No.	Objective	Fuel	Exp. Facilities	Note
C1	Rational management due to the reduction of design margin for the nuclear fuel cycle facilities; development of subcriticality monitoring system	Enriched U or Pu pins	DCA	Proposed by N. Aihara (PNC)
C2	Development of subcriticality monitor	Uranium rods and solution	NUCEF with some modifications	Requested by Y. Naito (JAERI)
C3	Accurate evaluation of subcriticality			Requested by K. Sakurai (JAERI)
C4	Criticality safety diagnosis by prompt γ-ray measurement		TCA, NUCEF	Requested by T. Suzaki (JAERI)

Table 4. Supercritical experiments requested in Japan

No.	Objective	Fuel	Exp. Facilities	Note
D1	Development of alarm system, accident analysis	LEU powder or solid and solution, MOX powder or solid and solution		Requested by Y. Naito (JAERI)

Abbreviations (LEU: Low-enriched uranium)

Annex

Questionnaire on Experimental Needs in Criticality Safety

NAME:
AFFILIATION:

Q1 Considering your past experience, do you think that you have any needs in criticality experiment(s)?

A1 YES/NO*

The following questions only concern those who answered YES to Q1. If more than one need exists, please fill in one form per need.

Q2 What is the need about?

A2

Q3 What is it for?

A3

Q4 Will the need be fulfilled by the established programme?

A4 YES/NO*

Q5 If A4 is YES, please specify the programme.
If A4 is NO, what kind of experiment would it be?

A5

Q6 Would the experiment be realised at present facilities? If there are any, please specify. Otherwise, please describe the facility at which the experiment would be conducted.

A6

* Please circle your answer.

Thank you for your collaboration

SESSION III

REVIEW OF DATA FROM EXPERIMENTS

Available Internationally

THE INTERNATIONAL CRITICALITY SAFETY BENCHMARK EVALUATION PROJECT (ICSBEP)

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Abstract

The International Criticality Safety Benchmark Evaluation Project (ICSBEP) was organised to identify and evaluate a comprehensive set of critical-experiment benchmark data. The evaluation process includes verifying the data, by reviewing reports and other documentation and by talking with experimenters, compiling the data into a standardised format, and performing calculations with standard criticality safety codes. The evaluation process concludes with the formal documentation of the work into a handbook of verified benchmark critical data, which can be used by criticality safety analysts as input for validations of their calculational techniques.

The International Criticality Safety Benchmark Evaluation Project (ICSBEP)

In October of 1992, the United States (US) Department of Energy (DOE) initiated a project that was designed to:

1. Identify and evaluate a comprehensive set of critical-experiment benchmark data;
2. Verify the data, to the extent possible, 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;
3. Compile the data into a standardised 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.

A Working Group was formed that was initially comprised of criticality safety experts from the Idaho National Engineering Laboratory (INEL) and Los Alamos National Laboratory (LANL). The Working Group was soon expanded to include participants from the Hanford Site, Lawrence Livermore National Laboratory (LLNL), the Oak Ridge Y-12 Plant, the Rocky Flat Plant (RFP), and the Savannah River Site (SRS).

In 1994, an International Criticality Safety Data Exchange component was added to the project. Representatives from the United Kingdom, Japan, the Russian Federation, France, and Hungary joined the project, and scientists from other countries expressed an interest. In December of 1994, the Organisation for Economic Co-operation and Development/Nuclear Energy Agency's (OECD/NEA) Nuclear Science Bureau voted to include the project as an official activity of the OECD/NEA. Thus, the project is now known as the International Criticality Safety Benchmark Evaluation Project (ICSBEP).

Documentation of the work performed by the ICSBEP will be an "International Handbook of Evaluated Criticality Safety Benchmark Experiments". Benchmark specifications in the handbook may then be used by criticality safety analysts to perform the necessary validations of their calculational techniques. The document will also be a useful tool for cross section evaluation groups and nuclear criticality safety training organisations. Upon completion, the handbook will span seven volumes:

I. Plutonium Systems

II. Highly Enriched Uranium Systems

III. Intermediate and Mixed Enrichment Uranium Systems

IV. Low Enriched Uranium Systems

V. Uranium-233 Systems

VI. Mixed Plutonium-Uranium Systems

VII. Special Isotope Systems

Each of the seven volumes is divided into four major sections: metal systems, compound systems, solution systems, and miscellaneous systems. Metal, compound, and miscellaneous systems are subdivided into fast, intermediate, and thermal systems. Only the thermal subdivision is applicable to solution systems.

Each evaluation in the handbook is organised in a standardised format that includes the following elements:

- **Identification number**. Each evaluation (which may contain several similar experiments) has a unique identifier, which indicates its category and sub-category under the organisation of the handbook.

- **Experiment overview and detailed description of materials and geometry** (Section 1.0). The information in this section is based on published reports, logbooks, discussions with experimenters, and memos or other records provided by the experimenters. The physical location of the experiment with respect to large structures (apparatus, concrete walls) is noted. The surroundings of experiments in which neutron leakage is a predominate characteristic are described in more detail than those of well-reflected experiments. Inconsistencies and uncertainties in data are noted. The method of determining the critical condition is stated, as well as relevant reactivity information.

- **Evaluation of experimental data** (Section 2.0). The acceptability of the experiment for use as a benchmark critical experiment is justified here. The effects of missing, uncertain, and inconsistent data on k_{eff} are discussed and, if practical, quantified by performing sensitivity studies. If experimental data are judged to be acceptable for use as criticality safety benchmark data, justification for accepting the data is given. If all or part of the data are found to be unacceptable for use as benchmark data, this fact is noted. The evaluation process for unacceptable experiments is terminated at this point.

  In some cases, rather than rejecting an experiment, measurement tolerances and missing or uncertain data are replaced by an estimated uncertainty in k_{eff}. The uncertainty in k_{eff} may be estimated by reported experimental reactivities or by calculations. The combined effect of several uncertainties is estimated by squaring, adding, and then taking the square root of the separately determined _k values.

  In some cases, simplifications to the benchmark model were necessary or desirable. In these cases, a “benchmark-model k_{eff}” that is slightly different from 1.00 or that has a small additional uncertainty may be quoted.

  A few subcritical experiments are included. However, only those with values of neutron multiplication that are sufficiently high to enable the subcritical k_{eff} value to be appropriately approximated by 1-1/M were accepted.

- **Benchmark specifications** (Section 3.0). Benchmark specifications refer to the data that are necessary to construct the recommended calculational model of the critical system. A concise description of the calculational model is given. Simplifications and approximations made to geometric configurations or material compositions are described and justified. All constituents of the materials used in the experiment description are included, or a justification for omitting them is provided. Temperature data are provided. Tables of all required dimensions and of material atom densities are included. Tables are subdivided into core, structural materials, and reflector materials subheadings.

- **Results of sample calculations** (Section 4.0). Calculated results obtained with the benchmark specification data (Section 3.0) are tabulated in this section. Results are provided for the standard set of codes and cross section data of the country in which the evaluation was performed. Results from other countries follow the evaluator's results in the order in which they were provided to the ICSBEP Working Group. Descriptions of the calculations, including input listings, are given in Appendix A.

Nuclear constants are taken from the 14th edition of General Electric's *Nuclides and Isotopes.*

Note that the evaluations are not to be considered as validations of neutronics codes. Although the calculated results may show that a code package gives good results under a certain set of circumstances (hardware and input options), the criticality safety analyst must still demonstrate that the particular code package, input options, and computer hardware used for a particular criticality safety analysis will give valid results. This is done by choosing a critical experiment composed of materials and geometry similar to those in the analysis and showing that a calculation with the particular neutronics code package, input options, and computer hardware will produce a k_{eff} close to 1.0. These evaluations will provide the experimental data for validation calculations.

Each experiment evaluation included in the handbook has undergone a thorough peer review process beginning with an internal review by the evaluator's organisation. The internal reviewer verified:

1. The accuracy of the descriptive information given in the evaluation (Section 1.0) by comparison with original documentation (published and unpublished),

2. That the benchmark specification (Section 3.0) can be derived from the descriptive information (Section 1.0) given in the evaluation,

3. The completeness of the benchmark specification (Section 3.0),

4. The results (Section 4.0) and conclusions (Section 2.0),

5. Adherence to format.

In addition, each experiment has undergone an independent peer review by another working group member at a different institute or facility. Starting with the evaluator's submittal in the appropriate format, the independent peer reviewer verified:

1. That the benchmark specification (Section 3.0) can be derived from the descriptive information (Section 1.0) given in the evaluation,

2. The completeness of the benchmark specification (Section 3.0),

3. The results (Section 4.0) and conclusions (Section 2.0),

4. Adherence to format.

A third review by the Working Group verified that the benchmark specification and the conclusions were adequately supported.

Often, engineering judgement is required in order to develop a reasonable model of an experimental apparatus, to supply information that is not provided in the experimental documentation, or to resolve discrepancies in the information that is provided. In such instances, inclusion of benchmark specification data in the handbook means that the Criticality Safety Benchmark Evaluation Working Group is in consensus with any engineering judgements that were required in deriving the specification.

The first formal distribution of the handbook began in May of 1995. The handbook is organised into seven volumes; however, at the present time, Volume VII, Special Isotope Systems, contains no entries. The remaining six volumes contain 46 evaluations with benchmark specifications for 376 critical or near critical configurations. Experiments that were found unacceptable for use as criticality safety benchmark experiments are discussed in these evaluations; however, benchmark specifications are not derived for such experiments. A total of 101 experimental configurations are categorised as unacceptable for use as criticality safety benchmark experiments.

Additional evaluations are in progress and will be published and distributed periodically. Interested parties may contribute internally reviewed evaluations for consideration by the ICSBEP Working Group for inclusion into the handbook. The document is organised in a manner that allows easy inclusion of revisions and additional evaluations as they become available.

The OECD/NEA will be responsible for distribution of the document. The handbook will also be published on CD-ROM in the near future and will soon be available to participating countries on World-Wide Web.

In conclusion, as a result of the efforts expended in producing this compilation, a large portion of the tedious and redundant research and processing of critical experiment data will be eliminated. The necessary step in criticality safety analyses of validating computer codes with benchmark critical data will be greatly streamlined, and valuable criticality safety experimental data will be preserved.

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The International Criticality Safety Benchmark Evaluation Project – Addendum

The International Handbook of Evaluated Criticality Safety Benchmark Experiments was published in March of 1995. The handbook is organised into seven volumes; however, at the present time, Volume VII, Special Isotope Systems, contains no entries. The remaining six volumes contain 46 evaluations with benchmark specifications for 376 critical or near critical configurations. Experiments that were found unacceptable for use as criticality safety benchmark experiments are discussed in these evaluations; however, benchmark specifications are not derived for such experiments. A total of 101 experimental configurations are categorised as unacceptable for use as criticality safety benchmark experiments.

As of September 1995 there are at least 88 additional evaluations in progress. The next publication of the handbook, scheduled for May of 1996, is expected to contain over 134 evaluations. A brief description of the expected contents of the handbook follows.

Plutonium Systems, Volume I, is expected to contain a total of 37 evaluations, comprised of 26 metal systems and 11 solution systems. There are no evaluations of compound or miscellaneous plutonium systems in progress.

Highly Enriched Uranium Systems, Volume II, is expected to contain a total of 52 evaluations, comprised of 22 metal systems, 19 solution systems, and 11 compound systems. There are no evaluations of miscellaneous, highly enriched uranium systems in progress.

Intermediate and Mixed Enrichment Uranium Systems, Volume III, is expected to contain a total of 7 evaluations, comprised of 6 metal systems and 1 compound system. There are no evaluations of solution or miscellaneous intermediate or mixed enrichment uranium systems in progress.

Low Enriched Uranium Systems, Volume IV, is expected to contain a total of 11 evaluations, comprised of 2 solution systems and 9 compound systems. There are no evaluations of metal or miscellaneous low enriched uranium systems in progress.

Uranium-233 Systems, Volume V, is expected to contain a total of 13 evaluations, comprised of 6 metal systems and 7 solution systems. (Evaluation of these solutions systems are in progress, but completion of these evaluations may not occur until after the next publication of the handbook.) There are no evaluations of compound or miscellaneous uranium-233 systems in progress.

Mixed Plutonium-Uranium Systems, Volume VI, is expected to contain a total of 8 evaluations, comprised of 4 metal systems, 3 solution systems, and 1 compound systems. There are no evaluations of miscellaneous, mixed plutonium-uranium systems in progress.

Special Isotope Systems, Volume VII, is expected to contain a total of 6 evaluations, all of which will be replacement measurements for metal systems. There are no evaluations of solution, compound, or miscellaneous special isotope systems in progress.

Evaluations of metal systems often include only a single configuration while evaluations of solution and compound systems more typically include several configurations. Most metal or solution systems include only a single fissile unit while compound systems tend to be fuel rod lattices.

The distinguishing characteristic of the numerous metal experiments is typically the composition of the reflector material. Reflector materials evaluated to date include water, tungsten, copper, thorium, steel, aluminium, normal and depleted uranium, nickel, beryllium, beryllium oxide, graphite, polyethylene, and paraffin.

There are numerous experiments for which the evaluation process has not begun. At the current level of effort, about three more years will be required before evaluation of the majority of the US experiments can be completed.

Proprietary Data

VVER REACTOR PHYSICS EXPERIMENTS

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Abstract

VVER is a PWR of Russian design. VVER-type lattices are low-enriched UO_2-H_2O lattices of hexagonal shape. These lattices have been investigated at KFKI for almost twenty years. Experiments have been performed on more than 300 different lattices to create an evaluated data set for validating computer codes. An essential portion of the results is being included in the *NEA International Handbook of Evaluated Criticality Safety Benchmark Experiments* (LEU-COMP-THERM-015). However the results other than critical eigenvalues are important for validating calculations as well. The present paper reveals some of these experimental results.

Introduction

The VVER type reactors are pressurised water reactors (PWR); they have been developed at the Kurchatov Atomic Energy Institute, in Moscow (Russia). Most of the reactor physics problems of PWRs were solved in the sixties. The majority of the PWR fuel lattices is rectangular, but the VVERs are hexagonal. This is the reason why the reactor physics problems of these lattices must be solved separately.

VVER users (Soviet Union, Finland, Czechoslovakia, Hungary, etc.) established a joint project in order to pursue the possibility of experimental investigations. This project was based on the critical assembly ZR-6, built at KFKI Budapest. Experiments were carried out during the period 1972-1990. As a result of these experiments, evaluated experimental data has been created [1,2] which can be used for validating reactor physics calculations. An essential part of these results is being included in the *NEA International Handbook of Evaluated Criticality Safety Benchmark Experiments* (LEU-COMP-THERM-015). However the results other than critical eigenvalues are important for validating reactor physics calculations as well. The aim of the present paper is to demonstrate the variety of these experimental results.

Experiments

Critical assembly ZR-6 was built in 1972, its purpose being to perform various fuel lattice measurements with the aim of validating reactor physics codes, calculating the VVER reactors. ZR-6 can be considered as the nuclear physics model of the VVER reactors. Criticality was reached by varying the moderator height. This kind of regulation allowed work without control rods inside the core.

The lattices investigated in ZR-6 are different from most of the lattices investigated elsewhere, in:

- The hexagonal shape of the lattice,
- The ability to vary the temperature (in the range 20-130°C),
- The extremely low $^{1}H/^{235}U$ ratios (the entire range is 70-540).

More than 300 lattices have been investigated. In most cases the fuel rods correspond to the rods used in VVER reactors.

The first group of investigations utilised regular lattices, in which four main parameters were varied:

- Fuel enrichment (1.6, 3.6 and 4.4 at .%),
- Lattice pitch (11.0, 12.7, 15.0 and 19.05 mm),
- Boric acid concentration in the moderator (0-7.2 g/l),
- Temperature (20-30°C).

The second group of lattices consisted of perturbed lattices, where perturbations such as:

- Absorber rods (containing boron, europium, gadolinium),
- Water holes,
- Assembly wall imitators (water, aluminium),
- Steel-water mixture reflectors,

disturb the symmetry of the fuel lattice – in most cases in what may be considered a regular manner. In choosing the lattice perturbations to be studied, the characteristic perturbations of VVER-1000 reactors were used as a means of orientation.

The following lattice characteristics were measured:

- Criticality parameters H_{cr} and $\partial\rho/\partial H$,
- Reactivity coefficients $\partial\rho/\partial T$ and $\partial\rho/\partial C_B$
- Radial and axial distributions of reaction rates (by activating foils and measuring fuel activity),
- Spectral ratios for different detector materials and lattice positions,
- Intra-macrocell distributions of reaction rates.

In some cases the usual experimental methods were employed, in other cases new methods have been developed. For example, spectral indices, characterising the epithermal energy range, were measured on the basis of the parallel irradiation technique [3], whose main advantage lies in the avoidance of the flux perturbation effect. Possibly the most important of the spectral indices is SI(Np/Ce), which gives the reaction ratio of ^{238}U capture to the fission of both uranium isotopes, normalised for a Maxwell type spectrum.

The accuracy of the results is satisfactory from the viewpoint of the accuracy requirements of the calculations to be tested. The typical experimental errors are the following:

- 1% for reaction rate distributions,
- 2 to 4% for special ratios,
- 0.1% for bucklings if their errors are propagated to errors of k_{eff},
- 3-33% for the coefficients $\partial\rho/\partial T$ and $\partial\rho/\partial C_B$, in these cases the errors of critical heights are 0.2-1.0 mm.

The experimental results were evaluated using unified (throughout the entire series of measurements) statistical criteria. The evaluation was fulfilled by the computer code RFIT [4,5], and the following confidence probabilities were used throughout the evaluations:

- 99% for testing the identity of two quantities,
- 99% for picking up outliers,
- 95% for finding asymptotic regions (according to the point drop technique).

Some of the experimental data have been used for solving particular problems, for instance the criticality behaviour of spent fuel storage pools [6].

Results

As an example for the data other than critical eigenvalues the spectral index SI(Np/Ce) was selected. The results measured for a series of lattices, differing in four basic parameters (lattice pitch, fuel enrichment, boron concentration in the moderator and temperature) are given in Table 1. Values, calculated by using the KARATE programme system [7] and the difference between calculated and measured values are given in Table 1 as well.

Table 1. Comparison of the measured and calculated values of spectral index SI(Np/Ce)

Lattice pitch (mm)/ Fuel enrichment (%)/ Boron concentration (g/l)/ Temperature (°C)	Measured (M)	Calculated (C)	M - C
11.0/3.6/0.0/20	4.415 ± 0.062	4.810	-0.395
11.0/3.6/0.0/130	4.610 ± 0.051	5.086	-0.476
11.0/3.6/1.0/20	4.687 ± 0.053	4.844	-0.157
12.7/1.6/0.0/20	2.230 ± 0.042	2.267	-0.037
12.7/3.6/0.0/20	3.407 ± 0.054	3.539	-0.132
12.7/3.6/0.0/130	3.480 ± 0.054	3.647	-0.167
12.7/3.6/5.8/20	3.603 ± 0.046	3.700	-0.097
12.7/3.6/5.8/130	3.812 ± 0.057	3.895	-0.083
12.7/4.4/0.0/20	3.721 ± 0.110	3.901	-0.180
12.7/4.4/0.0/130	3.917 ± 0.112	4.132	-0.215
15.0/1.6/0.0/20	1.662 ± 0.045	1.850	-0.188
15.0/3.6/0.0/20	2.420 ± 0.033	2.615	-0.195
15.0/3.6/4.0/20	2.586 ± 0.028	2.810	-0.224
15.0/4.4/0.0/20	2.603 ± 0.080	2.913	-0.310

One of the methods for experimental description of the neutron spectrum is the parallel irradiation technique [3], which basically irradiates activation detectors both in a neutron field to be investigated and in a reference field of thermal neutrons. Spectral index SI(Np/Ce), then, is defined as a double ratio:

$$SI(Np/Ce) = ({}^{U8}A_i/{}^{U5}A_i)/({}^{U8}A_r/{}^{U5}A_r)$$

where ${}^{U8}A_i$ and ${}^{U8}A_r$ are the activations of ${}^{238}U$ captures in the investigated and in the reference spectra respectively, ${}^{U5}A_i$ and ${}^{U5}A_r$ are analogues quantities for ${}^{235}U$ fissions. These spectral indices are very useful in assessing the validity of cell calculations. As a means of characterising the neutron spectrum the spectral index SI(Np/Ce) is very powerful, as it has the important advantage that it was measured activating only internal elements (fuel). Thus, in the case of this measurement, no flux perturbation occurs. SI(Np/Ce) is mainly characteristic for the neutron spectrum in the epithermal energy range.

The KARATE programme system [7] was developed at KFKI Atomic Energy Research Institute for the calculation of VVER-1000 reactors. Subsequently, another version of KARATE for VVER-440 reactors was developed. The KARATE programme system has been validated using evaluated experimental data, results of precise calculations and operational data of reactors. The experimental data from ZR-6 measurements played a decisive role in the validation process.

The spectral index SI(Np/Ce) was used to validate the KARATE cell calculations. The cell calculations are made by using a multi-group first flight probability approximation with a special treatment of neutron resonance absorption.

The agreement between the calculated and measured values is rather good, as can be seen from Table 1. The differences are in the order of magnitude of the experimental errors. However a clear tendency can be seen, i.e. the measured values are all lower than the calculated ones, which indicates a slight over-prediction of the spectral index SI(Np/Ce) on the part of KARATE, which probably can be explained by the resonance treatment of ${}^{238}U$.

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CRITICAL EXPERIMENT PROGRAMMES FOR FUEL SOLUTION WITH STACY AND TRACY

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Abstract

The Japan Atomic Energy Research Institute has continued to support an experimental research programme on criticality safety. To that end, two new types of critical facilities – Static Experiment Critical Facility (STACY) and Transient Experimental Critical Facility (TRACY) – have been completed at the Nuclear Fuel Cycle Safety Engineering Research Facility (NUCEF) [l].

STACY was designed to obtain fundamental criticality data of low-enriched uranium and plutonium nitrate solution which are handled in LWR fuel reprocessing plants. STACY attained initial criticality on February 23, 1995, and the first series of experiments were begun using 10 wt.% enrichment uranyl nitrate solution with a cylindrical tank.

TRACY is a critical facility created to simulate the transient behaviour of low-enriched uranyl nitrate solutions in criticality accidents. Subjects of study include variable parameters such as solution concentration, reactivity addition rate, initial source density and initial solution temperature. Feedback mechanism and total amount of energy release, spatial distribution of radiation and total fission release will be also measured in order to allow testing of the confinement system.

This report describes the main features of the experimental cores and programmes of these two critical facilities.

Introduction

Nuclear criticality safety experiments are essential for the verification of the criticality calculation code used in criticality safety design for all nuclear fuel cycle processes or equipment. Nevertheless, adequate experimental criticality data directly applicable for the evaluation of light-water reactor fuel are presently insufficient and sometimes unreasonable safety margins are required in the design of nuclear facilities.

Under the auspices of the national research programme for nuclear safety, the Japan Atomic Energy Research Institute has constructed the Nuclear Fuel Cycle Safety Engineering Research Facility (NUCEF). The following research subjects related to back-end technology in nuclear fuel cycle facilities will be investigated:

1. Nuclear criticality safety,

2. Advanced reprocessing processes and partitioning,

3. Transuranic waste treatment and disposal.

NUCEF contains two criticality experiment facilities, Static Experiment Critical Facility (STACY) and Transient Criticality Experiment Facility (TRACY), which are used to conduct nuclear criticality safety research. STACY is used to measure the criticality conditions of uranium, plutonium and their mixtures, and TRACY is used to study criticality accident phenomena with uranium solutions.

STACY

STACY is a solution-type criticality facility used for obtaining systematic criticality data concerning low-enriched uranium nitrate solutions and plutonium nitrate solutions which are handled in reprocessing plants [2,3]. STACY's main specifications are displayed in Table 1.

Table 1. Specifications of STACY

Core configuration	Operational limits & requirements
<u>Core tank:</u> *Single unit system* cylinder: diameter 21~100 cm height 1.5 m slab: thickness 10~50 cm (U) 10~35 cm (Pu) width 70 cm height 1.5 m *Two units system* cylinder: diameter 21~60 cm height 1.5 m slab: thickness 10~35 cm width 70 cm	Maximum excess reactivity: 0.8$ Maximum reactivity addition rate: 3¢/sec Reactivity shutdown margin all safety rods inserted: k_{eff}<0.985 one safety rod stuck: k_{eff}<0.995 Critical height: 40~140 cm Core temperature: Room temp. ~40°C

Table 1. Specifications of STACY (*cont.*)

Core configuration	Operational limits & requirements
Fuel: *Uranyl nitrate solution* enrichment 6, 10 wt% concentration max. 500 gU/*l* volume max. 1,100 *l* *Uranyl-plutonium nitrate* Pu enrichment 0~100 wt% concentration max. 300g(Pu+U)/*l* volume max. 1,100 *l* **Poison:** soluble, solid **Reflector:** water, concrete, SUS, etc.	Maximum power: 200W Maximum integrated power: 3 kWh/y

Many core configurations will be tested, including isolated cylindrical and slab-type tanks, two-unit interacting tanks, and heterogeneous tanks composed of a fuel rod array and a nitrate solution. In addition to changing the fuel concentration, acidity and temperature, various reflectors are used in order to measure the reactivity effects of these materials. These experimental subjects can be categorised in the following manner: a basic system of homogeneous core, a two-interaction system of homogeneous core, and a heterogeneous system composed of a fuel rod array immersed in the fuel solution.

A. Basic system

- *Criticality of low-enriched uranium solution*
  At first, criticality data regarding low-enriched uranyl nitrate solution are accumulated through the use of a single core tank. The ^{235}U enrichments are 10 wt% and 6 wt%. The critical height in the cylindrical or slab tanks with or without water reflector is measured with changing fuel concentration by dilution in a storage tank [4].

- *Criticality of Pu-U solution*
  Criticality data concerning the nitrate solution of uranium-plutonium mixture are obtained after the experiments using an uranyl nitrate solution have taken place. The plutonium concentration is a main experimental parameter. This series of experiments starts with a plutonium solution of which the ^{240}Pu isotopic composition is approximately 25%, which corresponds to the burnup of 30000 MWd/ton.

- *Reactivity effect of structural material and soluble poison*
  Structural materials encountered in the reprocessing plant such as ordinary concrete, borated concrete, polyethylene with/without neutron absorber have a reactivity effect on the fuel solution system. The reactivity effects of structural materials are important as regards the evaluation of the safety margin of the criticality safety design. From the difference between the critical height with and without structural materials, the reactivity effect is measured based on the reactivity worth of the solution height of STACY. The dependence of the reactivity on the thickness of the structural materials and the concentration of the neutron absorber will be investigated.

B. Two-interaction system

The neutron multiplication factor of a multiple-unit system becomes larger than that of a single unit by means of the neutron interaction effect. In order to study the fundamental characteristics of the neutron interaction effect, an array of two tanks are set up with STACY. The neutron interaction effect between two tanks is measured by placing two cylindrical or slab tanks in the air or water, and varying the tank geometry and distance between the two tanks. Reactivity worth from one core to another is also measured. This occurs through the use of the solution level worth method, which calculates the difference between the critical heights of a two-unit core and an isolated core. Additionally, the reactivity effects of the shielding materials placed between two cores will be gauged by altering both the thicknesses of these structural materials, as well as the content of neutron absorbers such as B_4C.

C. Heterogeneous system

In addition to the homogeneous system, criticality conditions for the heterogeneous system – composed of low-enriched uranium fuel rods immersed in the fuel solution – are obtained using a cylindrical core tank. The lattice pitch of the fuel rod array is modified in order to simulate the change in the volume ratio of fuel to moderator in the dissolving process of the reprocessing plant.

The reactivity effect of solution temperature and soluble poison will be ascertained from variations in the critical solution height; this exercise is necessary for confirming the safety margin of the criticality safety design. Kinetic parameters such as prompt neutron life time and effective delayed neutron fraction are measured by means of the pulsed neutron source and reactor noise methods.

The temperature coefficient – which effects the behaviour of transient phenomena – is measured at the main experimental core; this is achieved by determining the change in the critical solution height as the variations in temperature are applied. The variations range from room temperature to 40°C.

TRACY

As a means of better understanding the characteristics of criticality accidents concerning fissile solutions, supercritical experiments using a low-enriched uranium solution fuel will be performed with TRACY [5]. TRACY is a criticality facility conceived to perform supercriticality experiments that possess a maximum reactivity of three dollars. TRACY’s main specifications are outlined in Table 2.

The fuel used for TRACY is a l0 wt% ^{235}U enriched uranyl nitrate aqueous solution. The reactivity insertion is accomplished by:

i. Rapid withdrawal of a transient control rod from the core,

ii. Slow withdrawal of the rod, and

iii. Feeding fuel solution to the core tank from a storage tank.

The total released energy in an experiment is limited to 32 MJ (= 10^{18} fissions).

Table 2. Specifications of TRACY

Core configuration	Operational limits & requirements
Core tank: cylindrical tank with a channel for the transient rod movement diameter: 50 cm **Fuel:** *Uranyl nitrate* enrichment 10 wt% concentration max. 500 gU/*l* volume max. 700 *l* **Reflector:** bare, water, etc.	Maximum excess reactivity: static operation mode 0.8$ transient operation mode 3$ Reactivity shutdown margin all safety rods inserted: k_{eff}<0.985 one safety rod stuck: k_{eff}<0.995 Critical height: 40~100 cm Integrated energy: max. 32 MW·sec (1×10^{18} fissions) max. 230 kWh/y Peak power: max. 5000 MW Maximum pressure: 9 kg/cm^{2}g

The principal objective of these experiments is to verify the ability to contain radioactive materials in a reprocessing plant if a criticality accident occurs. Topics to be investigated experimentally are as follows.

A. Nuclear kinetics

The transient characteristics of criticality accidents will be investigated, as will the correlation between transient mechanism and total released energy with regard to the following parameters:

- Fuel composition, insertion rate of reactivity, total reactivity inserted,
- Initial neutron density, etc.

In this experiment, the time history of nuclear power, released energy, temperature, pressure, and so on will be measured. Several experimental instrumentations are currently under development. The analysis of the experimental data produced will enable the study of reactivity feedback mechanisms in criticality accidents. The kinetic analysis code is also being developed.

B. Radiation and dose

The spatial distribution and energy spectrum of neutron and gamma radiation will be measured in order to establish an evaluation method of exposure dose in a criticality accident. Several kinds of shielding materials will be used in the measurements. In addition, a functional test concerning criticality alarms will be conducted.

C. Radio-nuclides transport

The amount of radioactive materials released from the fuel solution to ventilation lines during the experiments will be measured to arrive at an understanding of the transport phenomena of gaseous fission products and aerosols in a criticality accident. To this purpose, sampling systems are installed in the TRACY core tank and ventilation lines.

The schedule of experiments for STACY and TRACY is shown in Table 3.

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Table 3. Research subjects and programme of STACY and TRACY

Research subjects	1995	1996	1997	1998	1999
STACY					
A. Basic system					
• Criticality of low-enriched U solution	UN (10wt.%)		UN (6wt.%)		
• Criticality of Pu-U solution					Pu-U
• Reactivity worth of poison and structure material		Structure		Poison	
B. Two-unit interaction system					
• Reactivity effect of neutron interaction			Interaction		Interaction
C. Heterogeneous system					
• Criticality of fuel rods in aqueous fissile solution				Dissolver	
TRACY					
Criticality accident phenomena					
A. Neutron kinetics		Reactivity feed back effect			
B. Radiation dose					Exposure evaluation
C. Radioactive nuclide transport			Fission product release		

SESSION IV

EXPERIMENTAL NEEDS NOT COVERED BY CURRENT PROGRAMMES

CRITICALITY EXPERIMENTS: RESEARCH, COLLABORATION, AND BUSINESS

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Abstract

This paper discusses the work of the Criticality Experiments Project Team, whose purpose was to develop an action plan for marketing the service of performing criticality experiments for others at Arzamas-16 and Chelyabinsk-70. This service would provide criticality experiments for industry and governments at a fraction of the cost of conducting the experiments in the United States.

Introduction

The need for new criticality safety experiments is obvious [1], yet the ability to meet this need becomes increasingly difficult. The difficulties do not arise from a lack of expertise or facilities. The primary reasons for these obstacles, particularly in the United States (US), are the cost of the experiments and the regulatory environment. A current project, the Russian-American Criticality Experiment Project, developed as a joint effort between the Russian and American governments, may help alleviate some of the difficulties. Successful completion of the project will result in criticality experiments, needed elsewhere, being conducted in Russian facilities.

The Experts' Meeting on Experimental Needs in Criticality Safety sponsored by the Organisation for Economic Co-operation and Development is an appropriate forum to discuss this undertaking, for it consists of two important aspects: economics and co-operation. The project is the result of the Fourth Russian-American Entrepreneurial Workshop on Defence Conversion.

The purpose of the Fourth Russian-American Entrepreneurial Workshop was to develop detailed action plans for 10 projects having potential commercial value that involve the Russian nuclear weapons institutes at Arzamas-16 and Chelyabinsk-70. At the workshop, teams of Russians and Americans were formed to develop action plans.

This paper discusses the work of the Criticality Experiments Project Team, whose purpose was to develop an action plan for marketing the service of performing criticality experiments for others at Arzamas-16 and Chelyabinsk-70. This service would provide criticality experiments for industry and governments at a fraction of the cost of conducting the experiments in the United States.

We hope that the plan discussed in this paper will provide a means to fund needed criticality experiments, which will permit the full utilisation of the expertise and facilities of the Russian laboratories. We feel that this Experts' Meeting on Experimental Needs in Criticality Safety will be one method of making the scientific community aware of this project, and perhaps it will engender the support and guidance of its members toward the pursual of our goal.

The Fourth Annual Russian-American Entrepreneurial Workshop

The Fourth Annual Russian-American Entrepreneurial Workshop was sponsored jointly by the United States Arms Control and Disarmament Agency, the Department of Energy and the Russian Ministry of Atomic Energy (MINATOM). It was held in Moscow in June 1995. The workshop was organised by personnel from Arzamas-16, Russia, and the Centre for International Security Affairs at Los Alamos National Laboratory in the United States.

The first workshop was in held in 1992, and meetings have followed yearly since then. The main goal of the workshops is to assist in converting a major portion of the Russian Weapons R&D complex into self-sustaining, profit-making organisations. Detailed action plans for ten projects having potential commercial value and involving the Russian nuclear weapons institutes at Arzamas-16 and Chelyabinsk-70 were developed in this workshop.

Teams were formed to develop action plans. Each team was composed of Russian and US scientists, industrial representatives and faculty members, i.e. entrepreneurs, venture capitalists and business managers. The team relative to our interests is the Criticality Experiments Project Team.

The action plan

The purpose of the Criticality Experiments Project Team was to develop an action plan for marketing the service of performing criticality experiments at Arzamas-16 and Chelyabinsk-70 for US industry. Government laboratories could also be considered as potential customers. This service would provide criticality experiments at a fraction of the cost at which they could be conducted in the United States. Arzamas-16 and Chelyabinsk-70 have well developed laboratories, thus no developmental funds would be required.

The criticality experiments and analysis for design and verification could be used for current needs such as weapons materials disposition, storage and transportation issues, nuclear fuel processing, experimental data for data bases and reactor safety and design. There is market potential based on the premise that needed experiments have not been done because costs exceed savings in the United States, and a status quo remains. An example would be the transportation of spent fuel. If experiments could demonstrate that more spent fuel could safely be placed in a cask, then considerable savings could be made; however, if the cost of the experiments exceed the savings, the experiments would not be performed. If it can be demonstrated that savings exceed the cost of the experiment, we feel that a large potential market would exist.

The competitive advantages of the Russian facilities include: fast response, cost-effectiveness, accuracy and quality assurance, regulatory compliance with MINATOM, comprehensive capability, uniqueness of the facilities, world class capability, secure environment for special materials, communication links and a nuclear materials process capability.

The history, research programmes and overview of criticality safety research at the Russian Federal Nuclear Centre Institutes of Experimental and Technical Physics have recently been presented in papers [2,3,4,5] at the Fifth International Conference on Nuclear Criticality Safety. The Russian Institutes have decades of experience performing criticality experiments. These experiments have been extremely sophisticated and accurate and have been used to investigate a wide range of fissile material configurations. Through their contributions to the International Criticality Safety Benchmark Experiments Project, they have demonstrated that they can provide documentation and quality assurance that meets or exceeds US and international standards [6]. In addition, the institutes involved in the project have excellent safety records.

We felt that it would be prudent to involve the United States Nuclear Regulatory Commission (US NRC) in the plan from the outset because criticality experiments are used to validate codes, and the NRC is the US governing body that certifies the codes. Thus, we decided to approach the NRC to participate in the project at no cost to them.

The first year of the project would consist of a criticality experiment performed as a demonstration of capabilities and feasibility. It would be partly funded by the Los Alamos Industrial Partnership Programme Office. An industrial partner with an immediate need for a criticality experiment would be sought. The industrial partner would be expected to make financial contributions; however, the cost would be only a fraction of the cost of having the experiment conducted in the United States. The experiment chosen for demonstration should be of benefit both to the NRC and the industrial partner. The Russian laboratories would also make monetary contributions. Oversight of the experiment would come from US national laboratories. A US scientist would actually participate in the experiment.

Suggestions for a demonstration experiment, should industrial support be found, could be a criticality experiment with spherical assemblies made of metal fissile material with a large cavity. The cavity would be filled with powdery materials of complex nuclide composition of low fissile material concentrations. The reactivity changes introduced by such materials will be measured. The goal of such experiments would be to obtain test data to verify criticality calculation of fuel production wastes with low fissile material concentration.

Another possible candidate for a demonstration experiment, again should industrial support be found, would be criticality experiments using alternate thin layers of ^{235}U, ^{238}U, iron and other materials. These criticality experiments simulate possible situations that may occur in accidents during fuel transportation.

Following a successful conclusion to the demonstration experiment, we believe that customers in the US and other countries will feel confident in purchasing the criticality experiment and analysis services from the Russian laboratories. Subsidies will no longer be necessary. The project will have become a self-sustaining commercial enterprise.

Conclusions

We believe this plan to be feasible. International co-operation, economic benefits and needed criticality safety experiments will be the results of a successful conclusion to the project.

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FORECAST OF CRITICALITY EXPERIMENTS AND EXPERIMENTAL PROGRAMMES NEEDED TO SUPPORT NUCLEAR OPERATIONS IN THE UNITED STATES OF AMERICA: 1994-1999

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Abstract

This forecast is generated by the Chair of the Experiment Needs Identification Workgroup (ENIWG) with input from Department of Energy and the nuclear community. One of the current concerns addressed by ENIWG was the Defense Nuclear Facilities Safety Board's Recommendation 93-2. This recommendation delineated the need for a critical experimental capability, which includes (1) a programme of general-purpose experiments, (2) improving the information base and (3) ongoing departmental programmes. The nuclear community also recognises the importance of criticality theory, which, as a stepping stone to computational analysis and safety code development, needs to be benchmarked against well-characterised critical experiments. A summary projection of the Department's needs with respect to criticality information includes (1) hands-on training, (2) criticality and nuclear data, (3) detector systems, (4) uranium- and plutonium-based reactors and (5) accident analysis. The Workgroup has evaluated, prioritised, and categorised each proposed experiment and programme. Transportation/Applications is a new category intended to cover the areas of storage, training, emergency response and standards. This category has the highest number of priority-1 experiments (nine). Facilities capable of performing experiments include the Los Alamos Critical Experiment Facility (LACEF) along with Area V at Sandia National Laboratory. The LACEF continues to house the most significant collection of critical assemblies in the Western Hemisphere. The staff of this facility and Area V are trained and certified, and documentation is current. ENIWG will continue to work with the nuclear community to identify and prioritise experiments because there is an overwhelming need for critical experiments to be performed for basic research and code validation.

Introduction

This report identifies critical experiments forecast for 1994-1999, based on the consensus of the Experiment Needs Identification Workgroup, which is sponsored by the Department of Energy's (DOE) Nuclear Criticality Technology and Safety Project. This *Forecast* is generated by the Chair of the Workgroup, with input from DOE contractors, DOE programme offices, special groups working in the area of criticality safety, DOE critical mass laboratories, and the Nuclear Regulatory Commission.

This document is considered a 'living' document and will be updated periodically. A glossary of nuclear criticality terms and a list of symbols (Appendix A), a list of criticality acronyms (page 20), and a list of ENIWG participants (page 24) can be found in the Los Alamos report LA-12683 *.

Current concerns

The Defense Nuclear Facilities Safety Board unanimously approved Recommendation 93-2 (Appendix B) which deals with "the need for critical experiment capability". The Board delineated in its Recommendation that a continuing programme of general-purpose critical experiments is necessary to insure safety in the handling and storing of fissionable material. Specifically, the Board recommends that:

- The Department of Energy should retain its programme of general-purpose critical experiments;
- This programme should normally be directed along the lines that satisfy the objectives of improving the information base, which underlies the prediction of criticality and serves in the education of the criticality engineer community;
- The results and resources of the criticality programme should be used in ongoing departmental programmes where nuclear criticality would be an important concern.

Specific experimental and programmatic responses to the DNFSB recommendation are listed in Table 1.

Also, based on the previous version of this forecast, several questions were raised concerning criticality physics and the calculational methods being used for criticality analysis. These evaluations and questions become extremely important as the DOE complex changes its mission, faces numerous weapons returns from the stockpile, and places an ever increasing importance on regulatory compliance. Because the experimental facility chosen must conduct its operations based on its financial and personnel resources, the ENIWG provides the guidance and information that are needed for the allocation of resources in the early planning of criticality experiments.

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* D. Rutheford, "Forecast of Criticality Experiments and Experimental Programs need to Support Nuclear Operations in the United States of America: 1994-1999", Los Alamos National Laboratory report LA-12683 (July 1994).

Table 1. Experiments and experimental programmes identified by ENIWG that address specific DNFSB recommendations

DNFSB recommendation	Experiments or experimental programmes that address the recommendation
"...maintain a good base of information for criticality control, covering the physical situations that will be encountered in handling and storing fissionable material ..."	104, 106, 202, 203, 302, 303, 305, 306, 402, 502g, 502h, 504, 406, and 701
"...theoretical understanding of neutron multiplication processes in critical and subcritical systems ..."	103, 105, 204, 205, 207, 208, 301, 501, 502, 502a, 502d, 502e, 502f, 502I, 503, 505, 601, 605, 605a, 609, 702, 703, and 704
"...to ensure retaining a community of individuals competent in practising the [criticality] control."	All experiments and experimental programmes, specifically 507 and 508 – training
"...experiments targeted at the major sources of discrepancy between the theory and the experiments	101, 102, 304, 606, and 707

ENIWG operations

The function of the Workgroup is to provide the criticality community with a hierarchy of experiments needed to support US DOE, HRC, and its licensees contractor operations. At the beginning of a new DOE programme or modification to an existing programme that involves fissile material, the ENIWG makes an evaluation to determine if current criticality benchmarks are adequate. If these benchmarks are found to be inadequate, a new criticality experiment may be necessary for safety and/or economic reasons. If such an experiment is indeed required, then a listing will appear in this document.

Identifying experiments and experimental programmes

Experimental programmes delineate general representations of a broad experimental need (i.e. dosimetry). Experiments are more specific in nature.

For each experiment and experimental programme identified by the Workgroup, the requester or sponsor provides a justification statement (see form in Appendix C*). This justification information is used to evaluate the need for the experiment and should (1) discuss existing criticality data (if any) and why it is deficient, (2) provide a description of the needed experiments and (3) list potential benefits.

At the beginning of each experiment and experimental programme listing, the following general information is given: (1) the DOE contractor who needs the experimental data, (2) the experiment or experimental category and (3) the application of the experiment or experimental programme.

* D. Rutheford, "Forecast of Criticality Experiments and Experimental Programs need to Support Nuclear Operations in the United States of America: 1994-1999", Los Alamos National Laboratory report LA-12683 (July 1994).

Rating experiments and experimental programmes

Experiments and experimental programmes are rated by representatives from the ENIWG who have determined the priority listing for each entry. These representatives also consider the identification of a sponsor and the extent to which such experiments will support programmatic needs or provide basic physics data.

In addition, a subcommittee of the Weapons Criticality Committe has been formed to identify the needs and priorities of nuclear safety experiments that are nuclear-weapons specific. This effort will be co-ordinated with the Workgroup.

Each experiment and experimental programme listed in the document has a *priority* listing that corresponds to one of the following:

- Maximum practical attention,
- Required for new or ongoing DOE operation, or
- Less urgent than priority (2).

The *status* ranking of each experiment and experimental programme is designated as one of the following:

- Initial request,
- Justification completed,
- Justification being prepared,
- Experimental identified,
- Anticipated need,
- Experiment in progress, or
- Experiment complete.

Note that the *status* and *priority* are different and can differ for any single experiment and experimental programme. However, every effort should be made to bring them to an equivalent level so that, for instance, the highest priority experiments should also be the ones closest to completion.

Summary listing of experiments and experimental programmes and their priorities

Table 2 lists the 58 experiments and experimental programmes that have been identified and prioritised. The 23 experiments considered highest priority (maximum practical attention) are listed in Table 3.

Table 2. Identified and prioritised experiments and experimental programmes

	Number of Priority		
Categories	*Priority 1*	*Priority 2*	*Priority 3*
Highly-enriched uranium (HEU)	2	5	0
Low-enriched uranium (LEU)	2	5	1
Plutonium (P)	4	1	0
Plutonium/Uranium fuel (PUF)	0	1	2
Transportation/Applications (T/A)	9	8	0
Baseline theoretical (BT)	5	2	4
Criticality physics (CP)	1	5	1
Total	**23**	**27**	**8**

Table 3. Highest priority experiments and experimental programmes

Category	Experiment	Experimental Programme or Experiment Title
HEU	104	Advanced neutron source
	106	TOPAZ-II reactor
LEU	206	Sheba reactivity parametrisation
	207	Sheba reactivity void coefficient
P	301	Plutonium solution in the concentration range from 8g/L to 17g/L
	303	Effectiveness of iron in plutonium storage and transport arrays
	304	Plutonium with extremely thick beryllium reflection
	305	Arrays of 3-kg Pu-metal cylinders immersed in water
T/A	501	Assessment for materials used to transport and store discrete items and weapons components
	Programme 502	Waste processing, transportation, and storage
	502c	Validation of WIPP hydrogen generation calculations
	502h	Minimum critical mass of fissile-polyethylene mixture
	502i	Criticality studies that emphasise intermediate energies
	Programme 503	Validation of criticality alarms and accident dosimetry
	Programme 504	Accident simulation and validation of accident calculations
	Programme 505	Evaluation of measurements for subcritical systems
	508	Development of a demonstration experiment
BT	601	Critical mass experiments for actinides
	606	Establishing the validity of neutron-scattering kernels
	607	Extending the standard ANSI/ANS 8.7 to moderated arrays
	608	Fission rate spectral index measurements in three assemblies
	609	Validation of calculational methodology in the intermediate energy range
CP	702	Spent fuel safety experiments (SFSX)

New transportation/applications category

This new subset of criticality experiments is intended to cover the areas of storage, transportation, waste, dosimetry alarm systems, training, emergency response, processing, and regulations and standards. The material is divided into two parts-programmes and specific experiments. The programme areas are further subdivided into specific experiments where appropriate.

It is assumed that the physical facilities of the critical mass laboratories are "User Facilities". These facilities would be maintained to support experimental capability, and are made available to experimenters. Of course, the permanent facility staff would maintain the capability to conduct experiments, or to supervise the temporary staff for particular experiments.

Training would be included as part of continuing capability. The training is divided into three parts. Training is provided to those who operate the critical experiments, which is the first part. The second part is a continuation and expansion of the nuclear-criticality-safety hands-on, two-, three-, and five-day training courses that have been provided for several years. The third type of training is an "intern-in-residence" programme to allow personnel an opportunity to gain experience in the day-to-day operation of a critical experiment facility. An important adjunct of the training programme is developing a simulator to demonstrate the characteristics of critical systems. We proposed that this development become a "catalog" item under the auspices of the DOE and that this simulator be made available to contractors and others at cost.

Programmes and experiments included in this category are identified in Table 4.

Table 4. New transportation/applications experiments and experimental programmes

Experiment 501	Assessment for material used to transport and store discrete items and weapons components	**Priority 1**
Experimental prog. 502	Waste processing, transportation, and storage	**Priority 1**
Experiment 502a	Absorption properties of waste matrices	**Priority 2**
Experiment 502b	In-situ drum stacking	**Priority 2**
Experiment 502c	Validation of WIPP hydrogen generation calculations	**Priority 1**
Experiment 502d	The in-tank precipitation (ITP) process for ^{235}U	**Priority 2**
Experiment 02e	The in-tank precipitation (ITP) process for ^{235}U + ^{239}Pu	**Priority 2**
Experiment 502f	The in-tank precipitation (ITP) process for ^{239}Pu	**Priority 2**
Experiment 502g	Determination of fissionable material concentrations in waste materials	**Priority 2**
Experiment 502h	Minimum critical mass of fissile-polyethylene mixture	**Priority 1**
Experiment 502i	Criticality studies that emphasise intermediate energies	**Priority 1**
Experimental prog. 503	Validation of criticality alarms and accident dosimetry	**Priority 1**
Experimental prog. 504	Accident simulation and validation of accident calculations	**Priority 1**
Experimental prog. 505	Evaluation of measurements for subcritical systems	**Priority 1**
Experiment 506	Safe fissile mass thresholds for an array of waste storage drums	**Priority 2**
Experimental prog. 507	Simulator development	**Priority 2**
Experiment 508	Development of a demonstration experiment	**Priority 1**

Resources and status of facilities

The current (1994) status of available critical facilities and their resources are listed below. Although several facilities have been closed, they are listed here for historical reasons. Included in the description of each facility are the:

- Core technical capabilities (that is, what assemblies, or test cells, and what materials are available for experiments),

- Current documentation (for example, SARs, TSRs, and operating procedures),

- Personnel resources.

A. LACEF

1. Core technical capabilities

The mission of the Los Alamos National Laboratory (LANL) is:

"The Los Alamos National Laboratory is dedicated to applying world-class science and technology to the nation's security and well being. The Laboratory will continue its special role in defense, particularly in nuclear weapons technology, and will increasingly use its multi-disciplinary capabilities to solve problems in the civilian sector."

S. Hecker (1993)

Operating at Pajarito Site since 1946, the Los Alamos Critical Experiments Facility (LACEF) has been actively involved in this mission. Much of the original nuclear criticality research was performed at this site, and the facility continues to house the most significant collection of critical assemblies in the Western Hemisphere. The LACEF consists of three remotely controlled laboratories, known as kivas, which are located approximately one-quarter mile from the main building that houses the individual control rooms for each kiva. The assemblies in the kivas are described below. The combination of the assemblies, a large inventory of fissile material, and structural materials makes the LACEF one of the most diversified facilities for the simulation of nuclear reactors, weapons, and process applications; it is also a resource for performing research for the nuclear community.

Assemblies

The assemblies that may be operated at LACEF (see Table 5 for those currently available) can be subdivided into four categories:

- Benchmark assemblies are stable, definable configurations containing precisely known components. They can have interchangeable or adjustable fissile cores and reflectors.

- Assembly machines are general-purpose platforms into which fissile, moderating, reflecting, and control components can be loaded for short-range study of the neutronic properties of the materials.

- Solution assemblies are specifically designed to allow critical operations with configurations containing fissile solutions.

- Experimental reactors are either cooled naturally or by self-contained heat rejection systems and may be operated for a significant time at lower-power levels.

2. Current documentation and personnel resources

The LACEF staff is trained and certified and documentation is current.

Table 5. Critical assemblies at the LACEF

ASSEMBLY	TYPE	APPLICATIONS
BIG TEN	Large, fast-spectrum, steady-state benchmark assembly	1, 2, 3, 4
COMET	General-purpose, vertical assembly machine (portable)	2, 5, 6
FLATTOP	Fast-spectrum, steady-state benchmark assembly	1, 5, 6
GODIVA IV	Fast-burst assembly (portable)	1, 2, 4, 6, 7, 8
HONEYCOMB	Large, general-purpose, horizontal assembly machine	5, 9, 10
MARS	Large, general-purpose, vertical assembly machine	3, 5, 6
PLANET	General-purpose, vertical assembly machine	2, 5, 6
SHEBA	Liquid, steady-state and burst assembly	1, 2, 4, 7, 8
SKUA	Annular-core fast-burst assembly	1, 2, 7, 8
VENUS	Large, general-purpose machine (used for solutions)	1, 4, 5, 6, 8

Applications legend:

1. Irradiation studies
2. Neutron/gamma transport effects
3. Nuclear fuel development
4. Detector development studies
5. Critical mass and separation studies
6. Criticality safety training
7. Vulnerability, lethality, and countermeasures
8. (VL&C) criticality alarm development
9. NEST & START technique development
10. Weapons safety study

B. Area V, Sandia National Laboratories – SNL

1. Core technical capabilities

Area V at Sandia National Laboratories (Albuquerque) comprises numerous research and test laboratories whose main activities centre upon research work conducted at versatile reactors and gamma-ray source facilities. The main components of Area V are the Annular Core Research reactor, the Sandia Pulse Reactor II, the Sandia Pulse Reactor III, the Gamma Irradiation Facility, the Hot Cell Laboratory (Glove Box Laboratory and Analytical Laboratory), and the Radiation Metrology Laboratory.

Assemblies

- The Annular Core Research Reactor (ACRR) is a pool-type research capable of steady-state, pulse, and tailored-transient operation. The reactor was designed to accommodate a 21,000 cm^3 experimental package in a high flux, near-uniform radiation field. In addition, it has two interchangeable, fuel-ringed external cavities, an unfuelled external cavity, and two neutron radiography facilities.

- The Sandia Pulse Reactor II (SPR-II) is a bare, fast-burst, unreflected and unmoderated-core reactor capable of pulse and limited steady-state operation. It has a small central cavity and is used primarily for narrow-pulse, high-dose-rate testing.

- The Sandia Pulse Reactor III (SPR-III) is a bare, fast-burst, unreflected and unmoderated-core reactor capable of pulse and limited steady-state operation. The primary experiment chamber is a large central cavity that extends through the core. SPR-III is used for high-neutron-fluence or pulsed, high-dose testing.

2. Current documentation and personnel resources

The SNL staff is trained and certified and documentation is current.

C. Argonne National Laboratories – West

1. Core technical capabilities

The Zero Power Physics reactor (ZPPR) is a modern, world-class critical facility capable of full-scale simulation of fast-spectrum reactors. ZPPR has the flexibility necessary to accommodate critical assemblies for a wide range of reactor types, from very small space reactors to the largest, fast reactors. The facility design makes it possible not only to perform measurements, but also to switch rapidly from one reactor to another. ZPPR's inventory of critical experimental materials is irreplaceable and immense. This is due to the cost of specialised materials for the facility and non-existent manufacturing capability.

The ZPPR facility, located at the Idaho site of Argonne National Laboratory (ANL), consists of a reactor cell, a fuel-element loading room, a control room, a materials storage building, and workshops. The reactor cell and loading room are situated under a large earthen mound that provides a stable experimental environment and effective safeguards.

2. Current documentation and personnel resources

Last active in March of 1992, the ZPPR facility is presently in non-operational standby. The documentation is not current. The staff is no longer certified and has been reduced to three personnel.

D. Hanford Laboratories

The Hanford Critical Mass Laboratory was shut down at the end of December 1988; it is no longer functional as a critical facility.

The majority of the world's safety data on criticality of plutonium-bearing solutions was from this facility.

E. Oak Ridge National Laboratory – ORNL

1. Core technical capabilities

Located on the South Boundary of Y-12, Building 9213 housed the critical facility at ORNL. The facility, which was operational between 1950-1975, contained three cells: one was equipped to perform solution critical experiments, and the other two were equipped to perform solid critical experiments on split tables.

2. Current documentation and personnel resources

The facility has been shut down. There is no trained and certified staff and no current documentation.

F. Rocky Flats

1. Core technical capabilities

The Rocky Flats Critical Mass Laboratory (CML) is currently in a standby mode. The facility is gradually being defuelled, decontaminated, and decommissioned. This process is not completed.

The CML has one test cell that is large and well equipped with versatile handling equipment. It is thick-walled and has a history of a very low leak rate from intentional over pressurisation. The interior atmosphere can be completely isolated during an experiment. These properties make the test cell ideal for the safe performance of critical experiments.

Assemblies

This test cell contains four assembly machines, two of which are a vertical split table and the "liquid-reflector apparatus". The former has never been used and cannot be operated without major repairs; the latter was dismantled in the 1980s, pending rebuilding using a more efficient design, but this has not yet occurred. The other two assemblies are still present and fully operational:

- The "horizontal split table" is a large assembly capable of being loaded to many tons. Its separation parameters can also be precisely controlled and accurately measured.

- The “Solution Base” is an assembly that is still connected to a uranium solution tank farm that contains 560 kg of high-enriched uranyl nitrate solution in 2700 L of solution. The solution is quite free of impurities and exists at an ideal acid normality. Two concentrations are housed: one is approximately the minimum-critical-volume concentration; the other is ~ 120 g/L of uranium. The uranium is enriched to about 93% ^{235}U.

2. Current documentation and personnel resources

   Documentation for this facility is not current; it has neither an SAR nor any procedure. The staff has been reduced to one person who has been a part of this facility since its construction in 1964; however, he is no longer certified. He is approaching retirement age but plans to continue living in the area and will be available if needed.

Conclusions

At the July 1993 meeting, there was broad representation from DOE contractors, DOE programme offices, research reactor facilities, and critical mass laboratories.

This group successfully prioritised the set of experiments, ongoing and new, that were submitted by the US nuclear communities and established the status of each proposed experiment.

Experimental categories

Evidence presented at this meeting shows the overwhelming need for a wide variety of critical experiments (refer to Table 1). Some conclusions that can be drawn from the information presented here include the following:

- The majority of Priority-1 experiments and experimental programmes (nine) are in the Transportation/Applications category, with the Baseline Theoretical and Plutonium categories having 5 and 4 Priority-1 experiments and experimental programmes, respectively.

  Note: Currently, there are no funded experiments in these three categories. Nor there is a facility that is currently open which is capable of performing plutonium solution experiments.

- Criticality safety training is recognised as one of the most important aspects of maintaining our technical capability.

- The new priorities for needed experiments reflect the change in the mission of the DOE and the current thinking in the nuclear community, as well as continued experiments that are recognised as supporting US processing facilities.

- A concerted effort has been made to integrate physics criteria for the Benchmark Critical Experiments document (see Appendix D[*]) into this forecast.

[*] D. Rutheford, “Forecast of Criticality Experiments and Experimental Programs need to Support Nuclear Operations in the United States of America: 1994-1999”, Los Alamos National Laboratory report LA-12683 (July 1994).

- An important activity that arose from the meeting was to create an initial draft of criteria for establishing areas of applicability (see Appendix E*).

Resources and status of facilities

Currently, there is only *one* general-purpose facility that remains open: the Los Alamos Critical Experiment Facility. Sandia National Laboratories (Albuquerque) has research reactors and the capability to perform small critical experiments in their kiva; however, there is no capability to perform solution critical experiments.

Rocky Flats CML is currently on standby status.

Future directions

There is an overwhelming need for critical experiments to be performed for basic research and code validation. The Workgroup will continue to work with the changing direction of the DOE and the nuclear community to identify experiments and prioritise them.

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* D. Rutheford, "Forecast of Criticality Experiments and Experimental Programs need to Support Nuclear Operations in the United States of America: 1994-1999", Los Alamos National Laboratory report LA-12683 (July 1994).

APPENDIX A

The Need for International Co-operation in Providing Nuclear Criticality Experiments

G. Elliott Whitesides

As the original developer of the KENO programme for criticality safety calculations, as Chair of the Working Group that produced the ANSI Standard entitled *Validation of Calculational Methods for Nuclear Criticality Safety*, and as Chair of the OECD/NEA Criticality Calculations Working Group, I have been very much involved in the computational aspects of criticality safety. In these roles I have come to recognise, perhaps more than some others, the acute dilemma that the nuclear industry faces due to the lack of nuclear critical experiments.

In the 1940s, 1950s, and the early 1960s the United States and other countries, particularly the United Kingdom, performed many nuclear critical experiments. Because there were very limited calculational capabilities at these times, most of these experiments were designed to verify the criticality safety of some particular operation involving fissile material. Therefore, the overriding consideration of the experiment was to ensure the safety of the operation and not to provide a computational benchmark.

In the early 1960s the development of the S_n computer programmes at Los Alamos and Oak Ridge, the development of the GEM code in the United Kingdom, and the development of the KENO programme at Oak Ridge gave the first significant stimulus for critical experiments designed specifically for validating complex three-dimensional calculational methods and materials neutron cross-section data. The early success of these computational methods in reproducing the experimental results led to the conclusion that computational methods would take over from experiments and that there was no need to perform additional experiments. The result was an increasing reliance upon computational methods data. Over the next 30 years, the number of experiments declined very rapidly, and most of the experimental facilities were closed.

Over this period, consensus standards, agency orders, regulations, and site procedures have formalised requirements for validating criticality methods with experimental measurements. This is the important driving function that has so sharply focused the need for more benchmark quality critical experiments.

The OECD/NEA Criticality Calculations Working Group recognises the acute need for additional experimental data. The main emphasis of the Working Group has been to compare calculations made with different calculational methods for important and difficult criticality safety evaluations. The Working Group is currently working on its fourth study. As we have progressed from one study to another we have chosen to address increasingly difficult problems. The first Working Group study progressed quite quickly because of the availability of applicable data. With each succeeding study we have had to deal with less and less experimental data. Examples include the lack of experimental data in the following areas:

- Intervening materials and configurations used in the packaging of unirradiated and irradiated fissionable materials for transport or storage;

- Fissionable material systems involving neutron interacting, high neutron leakage fissionable material units;

- Neutron reflector influences on large systems of heterogeneous fissionable material units (such as packaged waste or reactor fuel); and

- Fissionable material systems that have a predominance of fission chains initiated with intermediate neutron energies such as damp oxides of low- to moderately-enriched uranium, damp oxides of plutonium or ^{235}U, systems using large quantities of thermal, 1/v or resonance neutron absorbers.

At each of our recent Working Group meetings a common topic for discussion has been how we can encourage the generation of needed data to ensure the validity of cross-section data and computational methods.

I am convinced that this need for data can only be addressed through an organisation such as the OECD/NEA Nuclear Science Committee. For instance, we have frequently observed that organisations have been willing to share otherwise proprietary data when they recognised that by sharing the data a greater good could be accomplished.

Without some international agreement to share both the data and the cost of producing the data, I fear that we will drift into a situation where each country will have access restricted to their own data. The greater good can be realised if we can agree on some method of sharing.

The small tragedy will come if the cost of using nuclear energy is greater because of a lack of sharing. The greater tragedy will come if a nuclear criticality accident occurs because of a lack of adequate data. We must rise above our own selfish behaviour when safety is an issue. An internationally based action is needed to address this situation.

I appreciate the interest that the Nuclear Science Committee has shown in the past through their support of criticality computational studies. **I urge you to convene a Specialist meeting to address the needs for experimental criticality data.**

APPENDIX B

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Experts' Meeting on Experimental Needs in Criticality Safety
Albuquerque, New Mexico, USA, 25-26 September 1995

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APPENDIX C

Programme

OECD/NEA Experts' Meeting on Experimental Needs in Criticality Safety

Albuquerque Hilton, New Mexico, USA
25-26 September 1995

GENERAL CHAIR **R. Mike Westfall**, ORNL, USA

CO-CHAIRS **Francis Barbry**, CEA France and **Yoshitaka Naito**, JAERI, Japan

SECRETARIAT **Enrico Sartori**, OECD/Nuclear Energy Agency

I. **Introduction** (M. Westfall)

A. Approval of the agenda

B. Scope and objective of the meeting

C. Introduction of participants

II. **Criticality safety facilities description** (F. Barbry)

A. United States and Canada

- R. Paternoster, LANL, USA
  Criticality facilities at the Los Alamos National Laboratories

B. Japan

- Y. Miyoshi, T. Suzaki, JAERI Tokai-mura, Japan; S. Shiroya, Kyoto University, Japan; N. Ohtani, Power Reactor and Nuclear Fuel Development Corporation, Japan
  Review of criticality experiment facilities in Japan

C. Europe

- F. Barbry, Institut de Protection et de Sûreté, Centre d'Etudes de Valduc, France
  The criticality safety laboratory of Valduc (France): Potentiality, facilities, experimental possibilities

- A. Santamarina, CEA Cadarache, France
  The experimental reactors at CEA/DRN Cadarache for criticality safety studies

D. Russia

- N.P. Voloshin, Chelyabinsk-70 Russia
  Overview of criticality safety facilities in Russia: Programme of work and future needs

- B. Riazanov, V. Sviridov, IPPE Obninsk, Russian Federation
  Technical capabilities and nearest future programme of the critical facility of IPPE Nuclear Safety Division

- M. Nikolaev, A. Tsiboulia, IPPE Obninsk, Russian Federation
  Some needs in validation of critical safety being satisfied using preserved experimental installations

- A.Yu. Gagarinski, Y.S. Glouchkov, N.N. Ponomarev-Stepnoi, Kurchatov Institute, Russia
  A short review of critical experiments performed at the Kurchatov Institute

III. **Experimental needs covered by current programmes** (Y. Naito)

- P. Cousinou, G. Poullot, CEA Fontenay-aux-Roses, France
  Criticality safety programme of CEA: 1995-2000 and future needs

- H. McFarlane, R.W. Schaefer, S.E. Aumeier, ANL-West Idaho, USA
  Applicability of ZPR critical experiment data to criticality safety

- G. Harms, SNL, USA
  Criticality facilities and programmes at the Sandia National Laboratories

- Y. Naito, JAERI Tokai-mura, Japan
  Criticality safety research programmes in Japan

- H. Okuno, Y. Naito, JAERI Tokai-mura, Japan
  Experimental needs in criticality safety surveyed in Japan

IV. **Review of data from experiments** (J.B. Briggs)

A. Available internationally

- J.B. Briggs, V.F. Dean, L. Scott, INEL, USA
  The International Criticality Safety Benchmark Experiments Project (ICSBEP)

- Others

B. Proprietary data: Scope, objective and future availability

- Z. Szatmary, Institute of Nuclear Techniques of the Technical University of Budapest;
  I. Vidovszky, KFKI Budapest, Hungary
  VVER reactor physics experiments

- Y. Miyoshi, K. Nakajima, M. Itagaki, Y. Naito, JAERI Tokai-mura, Japan
  Criticality experiment programmes for fuel solution with STACY and TRACY

V. **Experimental needs not covered by current programmes** (R.M. Westfall)

- R.T. Perry, M. Houts, H.Casey, LANL, USA; A. Golubev, S.V. Voronstsov, Experimental Physics Institute, Russia; A.P. Vasilyev, Technical Physics Institute, Russia
  Criticality experiments: Research, collaboration, and business

- D. Rutherford, LANL, USA
  Forecast of criticality experiments and experimental programmes needed to support nuclear operations in the United States of America: 1994-1999

- Sharing of facilities

- Commitment of resources

VI. **General discussion** (R.M. Westfall)

VII. **Report** (R.M.Westfall)

A. Outline, objective, scope

B. Distribution of tasks

C. Conclusions and recommendations