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DICE: Database for the International Criticality Safety Benchmark Evaluation Program Handbook

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Abstract—*The 2002 edition of the "International Handbook of Evaluated Criticality Safety Benchmark Experiments" (ICSBEP Handbook) spans more than 26000 pages and contains 330 evaluations with benchmark specifications for 2881 critical or near-critical configurations. With such a large content, it became evident that the users needed more than a broad and qualitative classification of experiments to make efficient use of the ICSBEP Handbook. This paper describes the features of Database for the International Handbook of Evaluated Criticality Safety Benchmark Experiments (DICE), which is a database for the ICSBEP Handbook. The DICE program contains a relational database loaded with selected information from each configuration and a users' interface that enables one to query the database and to extract specific parameters. Summary descriptions of each experimental configuration can also be obtained. In addition, plotting capabilities provide the means of comparing neutron spectra and sensitivity coefficients for a set of configurations.*

I. INTRODUCTION

Criticality safety studies are concerned with nonreactor configurations of the nuclear fuel cycle, such as fuel manufacturing and storage, transportation, and reprocessing of spent fuel. As for other areas of reactor physics, criticality safety analysts rely heavily on computer codes and nuclear data libraries for the prediction of the criticality parameters. Although these tools have become highly sophisticated, they still contain approximations, and the associated nuclear data are subject to uncertainties. A natural question is thus: How reliable are these computational tools for criticality safety predictions? Part of the answer can, in principle, be given by sensitivity analysis. However, in many cases, the number of parameters that are subject to uncertainties (basically, the whole set of nuclear data) is so large that this approach is not workable. Another way to address this question is to design an experimental program that represents a mockup of the envisaged design. In this framework, measurements of the most important parameters can be performed. Computer programs can then be used to calculate the experimental configuration, and the comparison of calculated and experimental quantities

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provides insight into the reliability of the tools used for design studies.

This process, known as validation or benchmarking, must be carried out for different applications and for any combination of programs and application libraries used in the design study. The role of experiments is crucial in criticality safety, at least for the following two reasons:

1. The configurations encountered cover a wide range of fuel materials, moderation conditions, and reflection and interaction environments. The validation process should cover all the intended application areas in order to correctly evaluate the prediction accuracy for any specific configuration.

2. The parameters calculated with these tools are used in the design and operation of facilities that must be safe vis-à-vis the risk of fission chain excursion. Unlike reactors, these facilities do not include a means of precisely monitoring the multiplication factor. In the early phases of nuclear fuel cycle development, criticality experiments were primarily designed to help accumulate basic critical data on the main media that were transposed into industrial scale. They also helped to gain insights into the neutronics and the chemistry of such media.

To reflect the variety of configurations, the International Criticality Safety Benchmark Evaluation Project (ICSBEP) has defined a classification scheme. The experiments are categorized according to three criteria:

1. *fissile media:* high-, intermediate-, or low-enriched uranium; plutonium; mixed uranium and plutonium; and special isotopes
2. *fuel form:* metal, compound, solution, or miscellaneous (usually mixtures of the other three)
3. *spectra:* thermal, intermediate, fast, or a mixture of the other three.

However, with the increasing number of experiments, this broad and qualitative classification becomes of limited use. Furthermore, besides the fissile media, other parameters have a significant impact on criticality configurations. The most important among them are probably moderation, reflection, and the presence of strong neutron absorbers. In general, any configuration can be represented as a function of parameters or, mathematically speaking, as a point in a multidimensional space. To gain confidence in the calculation tool used for the study of a particular system, the criticality analyst needs to know how well these tools perform in the calculation of experiments with characteristics similar to a particular application.

Several methods (see Refs. 1, 2, and 3) have been developed over the past 10 yr to answer the following question: Are the experiments representative of the actual configuration under study? In multidimensional space representation, the term "representative" indicates that the experiments and the actual configuration are reasonably close to each other. It became important to develop the means to help the ICSBEP Handbook user navigate in this space and extract quantitative information, e.g., to select experiments for a given fissile media and a given range of moderation ratios. Multiparameter searches using both numerical values and keywords are also important (e.g., experiments with water-moderated low-enriched uranium lattices and with a range of soluble boron concentration in the moderator). Furthermore, the ICSBEP Handbook is heavily used in nuclear data validation studies. More detailed information is thus required in this context (e.g., reaction rates for specific nuclides, neutron spectra) in order to be able to perform the analysis of the discrepancy between calculated and measured results as a function of important parameters.

II. WHAT IS DICE?

The user of the ICSBEP Handbook is primarily interested in experimental data that can be used for the validation of computation tools used for specific applications. Thus, the user normally has a set of features that the expected configurations should contain. Depending on the design being studied, these features can concern any combination of physical and chemical characteristics (geometry type, fuel composition, reflection conditions, etc.) and neutronics parameters (significant contribution of a specific isotope to the neutron balance, emphasis on an energy interval, etc.). The ICSBEP Handbook already has a structured content that helps the user target the experiments with the desired specifications, such as fuel, physical form, and an indication of the neutron spectrum. In addition, the CD-ROM contains an index that allows word searches of the entire ICSBEP Handbook. However, with an increasing number of experimental configurations and the necessity to perform multiple criteria searches, using both numerical fields and keywords, these basic features become insufficient. In general, the advantage of using a relational database becomes evident when the information has a complex structure and when the needed retrieval capabilities are based on multiple search criteria.

A relational database was created and loaded with information directly extracted from the ICSBEP Handbook or independently calculated. A users' interface was designed to allow the interrogation of the database. The database and the corresponding users' interface are referred to as DICE, which is an abbreviation of Database for the International Handbook of Evaluated Criticality Safety Benchmark Experiments. Figure 1 shows an overall layout of the software with the search options on the left-hand-side panel. When several search criteria concern a given category of data, they are grouped into themes (general items, fuel, moderation, geometry,

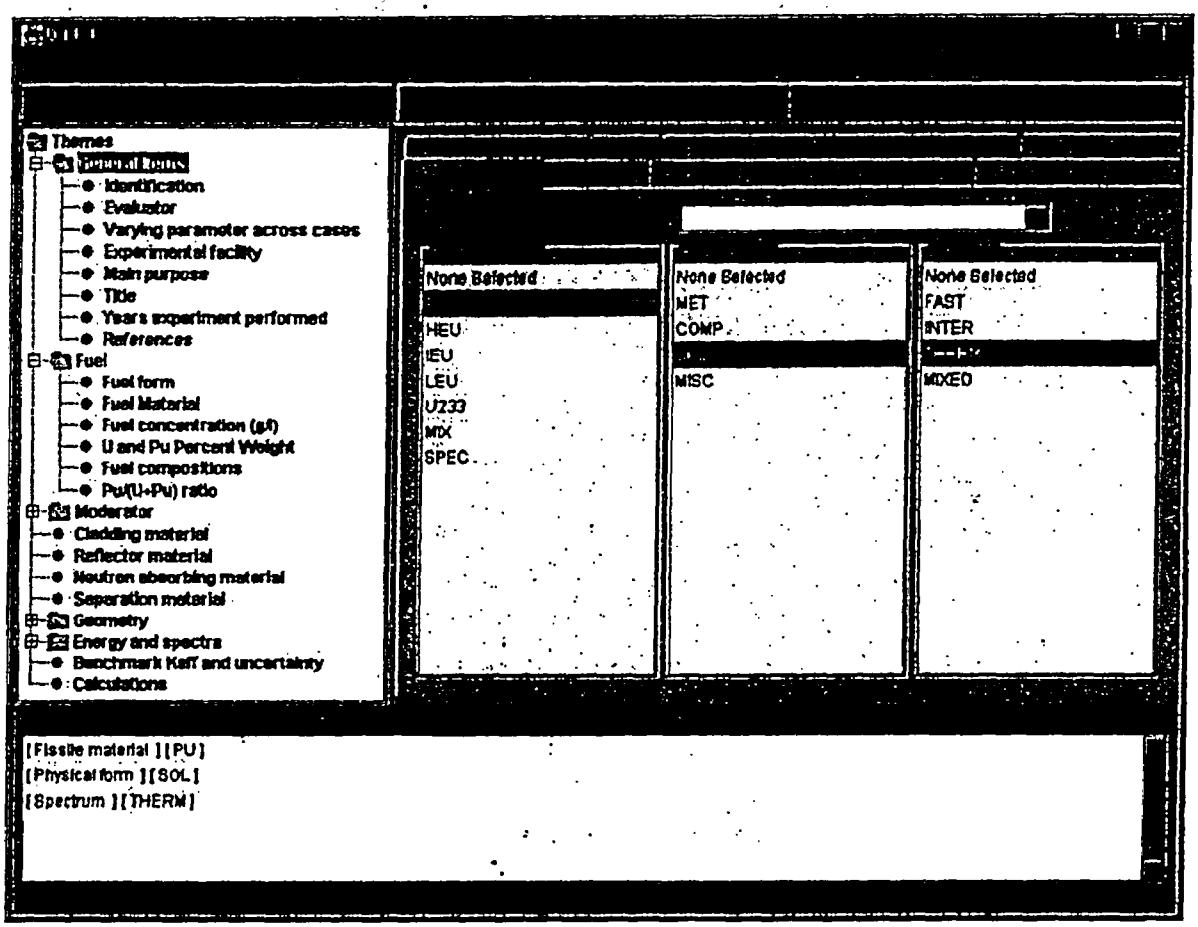


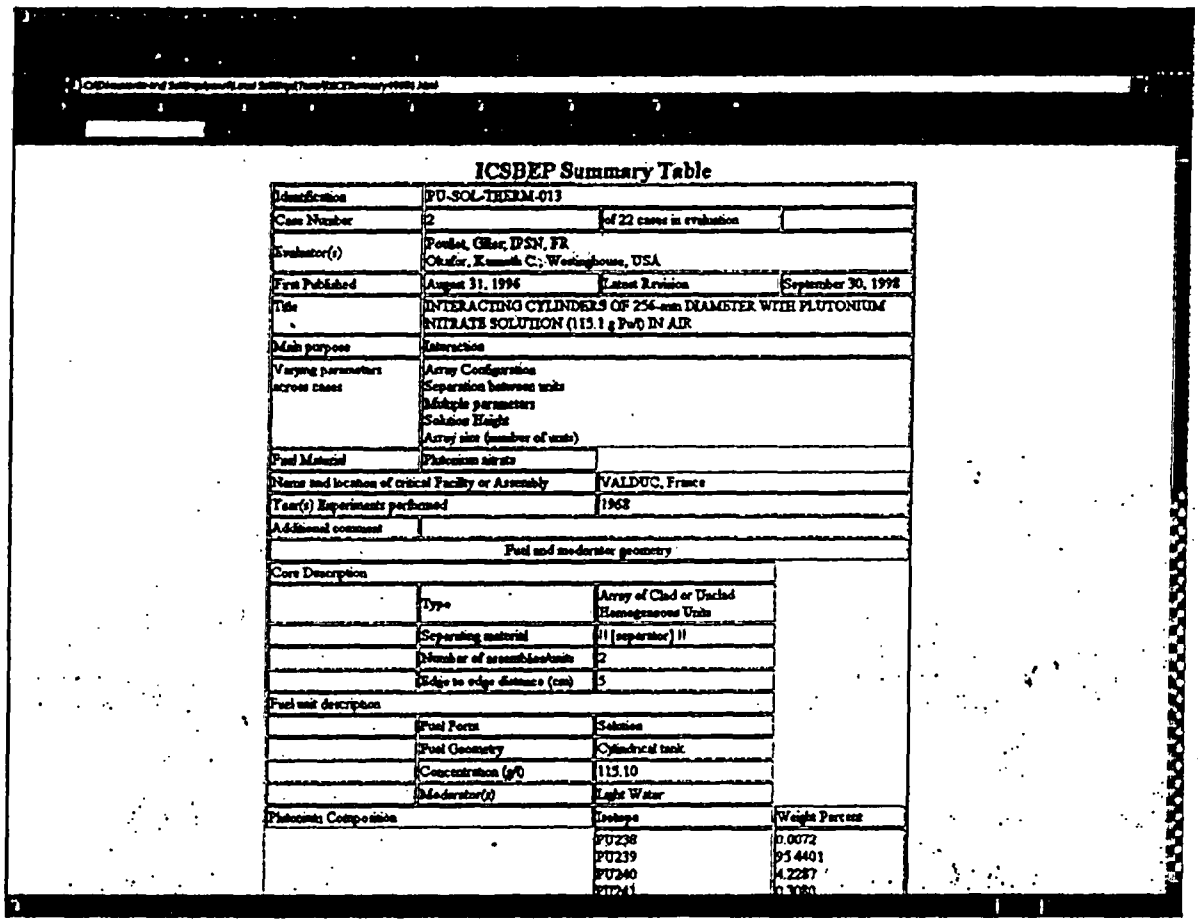
Fig. 1. General view of search criteria available in the DICE program.

spectra). Other criteria, such as reflector material, cladding, neutron-absorbing material, separator material, benchmark k_{eff} and uncertainty, and calculations, are separately accessible.

Although the search criteria allow the user to narrow down the number of experiments of interest, the search result does not always lead to a unique experiment. In many cases, the user needs to investigate more closely specific parameters for all the retrieved experiments (e.g., variation range of fuel concentration for solutions, moderation-to-fuel-volume ratio in lattices, etc.). This may also be useful for parametric analysis, such as the determination of trends in calculation results or for studying correlation between physical parameters. The DICE program enables the user to extract all the parameters in a tabular format and store this structured information onto a disk for further use with a plotting package.

Another feature available in DICE is the production of a summary table for each experiment. Each table summarizes the experimental configuration using the parameters stored in the database (see Fig. 2). The main characteristics of the experiments are thus displayed in a simple form, allowing the user to view them at a glance without navigating through each section of an evaluation. Some of the entries are common to all the configurations contained within the same evaluation (evaluators, publication and revision dates, title; main purpose of the experimental program, varying parameters across cases, fuel material, name of the laboratory where the experiments were performed, dates the experiments were performed, and the references). Other types of information are specific to each case and concern the description of the geometry, composition, moderation and reflection characteristics, spectra data, benchmark k_{eff} and associated uncertainty, and sample calculations.

The ICSBEP Project has pioneered a very rigorous experiment evaluation process, which leads to high-quality benchmarks. Furthermore, significant efforts are made to determine experimental uncertainties. This is why the ICSBEP Handbook is now considered to be the primary source of integral experiments for nuclear data validation studies for thermal and fast reactors and for criticality safety applications. In most cases, validation studies include the analysis of discrepancies between



ICSBEP Summary Table

Identification	PU-SOL-THERM-013		
Case Number	2	of 22 cases in evaluation	
Evaluator(s)	Poulet, Gilles; IPSN, FR Okrent, Kenneth C.; Westinghouse, USA		
First Published	August 31, 1996	Latest Revision	September 30, 1998
Title	INTERACTING CYLINDERS OF 256-mm DIAMETER WITH PLUTONIUM NITRATE SOLUTION (115.1 g Pu/l) IN AIR		
Main purpose	Interaction		
Varying parameters across cases	Array Configuration Separation between units Multiple parameters Solution Height Array size (number of units)		
Fuel Material	Plutonium nitrate		
Name and location of critical Facility or Assembly		VALDUC, France	
Year(s) Experiments performed		1968	
Additional comment			
Fuel and moderator geometry			
Core Description			
	Type	Array of Clad or Unclad Homogeneous Units	
	Separating material	H [separator] H	
	Number of assemblies/units	2	
	Edge to edge distance (cm)	5	
Fuel unit description			
	Fuel Form	Solution	
	Fuel Geometry	Cylindrical tank	
	Concentration (g/l)	115.10	
	Moderator(s)	Light Water	
Plutonium Composition		Isotope	Weight Percent
		PU238	0.0072
		PU239	95.4401
		PU240	4.2287
		PU241	0.3080

Fig. 2. Example of experiment summary generated with the DICE program.

calculation and experimental results and the derivation of trends for the most significant nuclear processes. To derive these trends, a fine representation of the neutron balance in the system is required along with the energy dependence of the main reaction rates. The DICE program offers the possibility of displaying graphical representations of neutron spectra within the fuel and gives a detailed neutron balance for different regions. The spectra (flux, fission, capture, and $(n,2n)$ reaction rates) are available in a 299-group energy mesh. They were generated using the KENO-V.a and KENO-VI codes of the Oak Ridge National Laboratory SCALE code system[4] and the Russian Monte Carlo code MMK-KENO and ABBN-93 (Ref. 5) 299-energy-group cross-section data.

The importance of a nuclear process can be expressed in terms of sensitivity coefficients, defined as the relative change in the k_{eff} for a given configuration, subject to a 1% change in a particular nuclear data set for a certain nuclide in a given energy group. The most important parameters in the energy range of fission applications are capture, fission, elastic scattering, inelastic scattering, nu-bar (average number of neutrons emitted per fission) and mu-bar (average cosine of scattered neutrons). They are currently available in DICE in a 30-group energy structure for a selected set of configurations (HEU-SOL-THERM). Figure 5 gives an example of the energy dependence of the sensitivity of the k_{eff} to the capture and fission cross sections for two highly enriched uranium solution experiments.

III. DICE FEATURES THROUGH EXAMPLES

The choice of the selection criteria is a nontrivial problem that must be solved by the user in each specific case. This requires the DICE program to be flexible and comprehensive. A wide range of parameters that characterize the configurations contained within the ICSBEP Handbook simplify the selection of search criteria. The sensitivity coefficient for cross sections is of special significance, as this is especially informative in nuclear data validation.

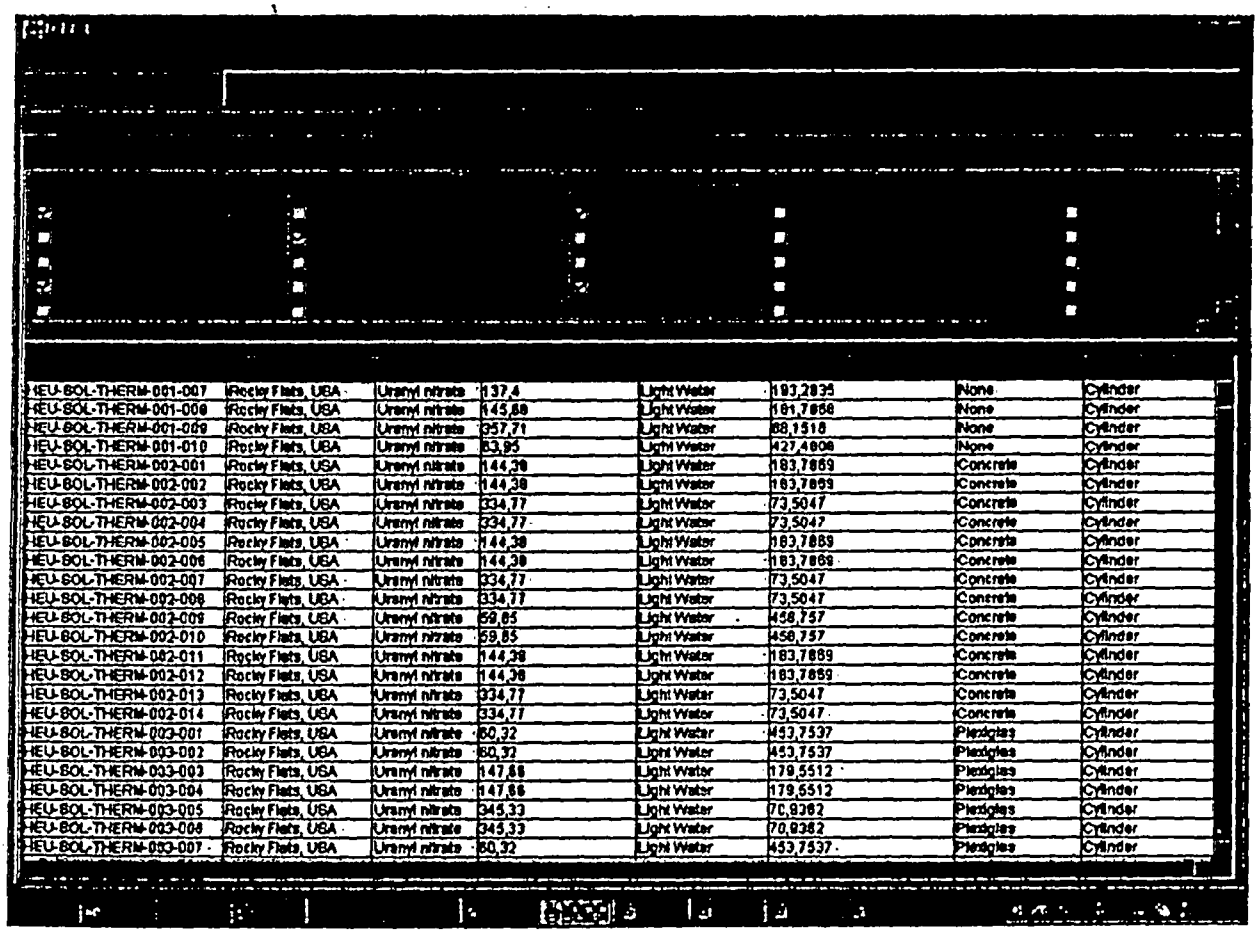
III.A. Example 1

In this example, we will search through the ICSBEP Handbook for experiments that can be used to validate calculations made for the determination of the minimal critical mass of highly enriched uranium solutions. A simple search on the experiment identifier (HEU-SOL-THERM) results in 432 configurations. Other characteristics of the experiment can be used to narrow down the query. Figure 3 provides an example of parameters that can be used for a more detailed search: fuel material, fuel concentration, moderator material, reflector material, and fuel unit geometry. The ranges of these parameters for the selected experiments are given in Table I. The primary selection list contains experiments that are not directly applicable to our specific application, such as the presence of concrete reflectors or the use of heavy water moderator. More criteria can be used to restrict the search (moderator material set to "light water," absence of neutron absorbing material, etc.).

The minimum critical mass of uranium-water systems is obtained in a water-reflected spherical geometry. These criteria can also be used to further restrict the search. The number of experiments satisfying these criteria is reduced to 16 (as compared to the 432 initial configurations).

One can impose restrictions on the fuel concentrations, knowing that the minimum critical mass is achieved for concentrations between 30 and 120 g/ℓ. Adding these criteria results in the final list of experiments presented in Table II.

Besides the physicochemical characteristics of materials present in each configuration, the DICE program offers a set of neutronic parameters that can be used for more detailed analyses. This includes neutron spectra in fuel regions, energy distributions of various reaction rates, and sensitivity coefficients of the multiplication factor to cross sections. Figure 4 shows a comparison of the neutron flux and the fission reaction rates in two of the configurations studied in this paragraph. Figure 5



HEU-SOL-THERM-001-007	Rocky Flats, USA	Uranyl nitrate	137,4	Light Water	193,2835	None	Cylinder
HEU-SOL-THERM-001-008	Rocky Flats, USA	Uranyl nitrate	145,88	Light Water	181,7858	None	Cylinder
HEU-SOL-THERM-001-009	Rocky Flats, USA	Uranyl nitrate	957,71	Light Water	68,1518	None	Cylinder
HEU-SOL-THERM-001-010	Rocky Flats, USA	Uranyl nitrate	83,95	Light Water	427,4808	None	Cylinder
HEU-SOL-THERM-002-001	Rocky Flats, USA	Uranyl nitrate	144,38	Light Water	183,7869	Concrete	Cylinder
HEU-SOL-THERM-002-002	Rocky Flats, USA	Uranyl nitrate	144,38	Light Water	183,7869	Concrete	Cylinder
HEU-SOL-THERM-002-003	Rocky Flats, USA	Uranyl nitrate	334,77	Light Water	73,5047	Concrete	Cylinder
HEU-SOL-THERM-002-004	Rocky Flats, USA	Uranyl nitrate	334,77	Light Water	73,5047	Concrete	Cylinder
HEU-SOL-THERM-002-005	Rocky Flats, USA	Uranyl nitrate	144,38	Light Water	183,7869	Concrete	Cylinder
HEU-SOL-THERM-002-006	Rocky Flats, USA	Uranyl nitrate	144,38	Light Water	183,7869	Concrete	Cylinder
HEU-SOL-THERM-002-007	Rocky Flats, USA	Uranyl nitrate	334,77	Light Water	73,5047	Concrete	Cylinder
HEU-SOL-THERM-002-008	Rocky Flats, USA	Uranyl nitrate	334,77	Light Water	73,5047	Concrete	Cylinder
HEU-SOL-THERM-002-009	Rocky Flats, USA	Uranyl nitrate	59,85	Light Water	458,757	Concrete	Cylinder
HEU-SOL-THERM-002-010	Rocky Flats, USA	Uranyl nitrate	59,85	Light Water	458,757	Concrete	Cylinder
HEU-SOL-THERM-002-011	Rocky Flats, USA	Uranyl nitrate	144,38	Light Water	183,7869	Concrete	Cylinder
HEU-SOL-THERM-002-012	Rocky Flats, USA	Uranyl nitrate	144,38	Light Water	183,7869	Concrete	Cylinder
HEU-SOL-THERM-002-013	Rocky Flats, USA	Uranyl nitrate	334,77	Light Water	73,5047	Concrete	Cylinder
HEU-SOL-THERM-002-014	Rocky Flats, USA	Uranyl nitrate	334,77	Light Water	73,5047	Concrete	Cylinder
HEU-SOL-THERM-003-001	Rocky Flats, USA	Uranyl nitrate	60,32	Light Water	453,7537	Plexiglas	Cylinder
HEU-SOL-THERM-003-002	Rocky Flats, USA	Uranyl nitrate	60,32	Light Water	453,7537	Plexiglas	Cylinder
HEU-SOL-THERM-003-003	Rocky Flats, USA	Uranyl nitrate	147,88	Light Water	179,5512	Plexiglas	Cylinder
HEU-SOL-THERM-003-004	Rocky Flats, USA	Uranyl nitrate	147,88	Light Water	179,5512	Plexiglas	Cylinder
HEU-SOL-THERM-003-005	Rocky Flats, USA	Uranyl nitrate	345,33	Light Water	70,9362	Plexiglas	Cylinder
HEU-SOL-THERM-003-006	Rocky Flats, USA	Uranyl nitrate	345,33	Light Water	70,9362	Plexiglas	Cylinder
HEU-SOL-THERM-003-007	Rocky Flats, USA	Uranyl nitrate	60,32	Light Water	453,7537	Plexiglas	Cylinder

Fig. 3. List of HEU-SOL-THERM–type experiments and selected details.

TABLE I

Range of Main Characteristics for HEU-SOL-THERM–Type Experiments

Reflector	Parameter	Fuel Material		
		Uranyl Nitrate	Uranium Fluoride	Uranium Hexafluoride
Without reflector	Fuel concentration, g/ℓ (MAX/MIN)	415/13.6	134/13.2	—
	Moderator to fissile ratio (MIN/MAX)	65.4/2050	204/2060	—
	Identification	HST-024-001/ HST-042-008	HST-043-001/ HST-020-005	—
With reflector	Fuel concentration, g/ℓ (MAX/MIN)	447/37.5	696/22.0	457.2/115.0
	Moderator to fissile ratio (MIN/MAX)	54.7/767	35.8/1272	59.0/237.13
	Identification	HST-019-001/ HST-035-001	HST-009-001/ HST-012-001	HST-039-002/ HST-039-006

TABLE II

Final List of Selected Experiments with Critical Masses of ^{235}U Close to Minimum Values

Identification	Fuel Material	Fuel Unit Geometry	Reflector Material	Fuel Concentration (g/ℓ)	^{235}U Mass (g)
HEU-SOL-THERM-010-004	Uranium fluoride	Sphere tank	Light water	112	1085
HEU-SOL-THERM-010-003	Uranium fluoride	Sphere tank	Light water	109	1063
HEU-SOL-THERM-010-002	Uranium fluoride	Sphere tank	Light water	104	1004
HEU-SOL-THERM-010-001	Uranium fluoride	Sphere tank	Light water	102	986
HEU-SOL-THERM-011-001	Uranium fluoride	Sphere tank	Light water	53.0	902
HEU-SOL-THERM-011-002	Uranium fluoride	Sphere tank	Light water	52.1	887

compares, for the same experimental configurations, the sensitivity of the multiplication factor to the fission cross section of ^{235}U.

III.B. Example 2: Application of DICE to the Validation of Nuclear Data Libraries

In this example, we will discuss, through a case study, how the DICE program can be used in the framework of nuclear data validation. We will concentrate on the particular case of arrays of mixed uranium-plutonium lattices. The selection of MIX-COMP in the experiment identifier leads to 184 configurations. In displaying some of the geometrical characteristics, one realizes that some configurations are made with fuels in the form of slabs. If one restricts the selection to experiments made with arrays of fuel pins, the number of selected experiments decreases to 151. When the purpose of the exercise is the validation of cross sections of the main nuclides (uranium, plutonium, hydrogen, oxygen), one can as a first step discard experiments with solid and soluble poisons (boron, cadmium, and hafnium). This reduces the number of experiments to 60.

Other parameters can be used to further refine the selection:

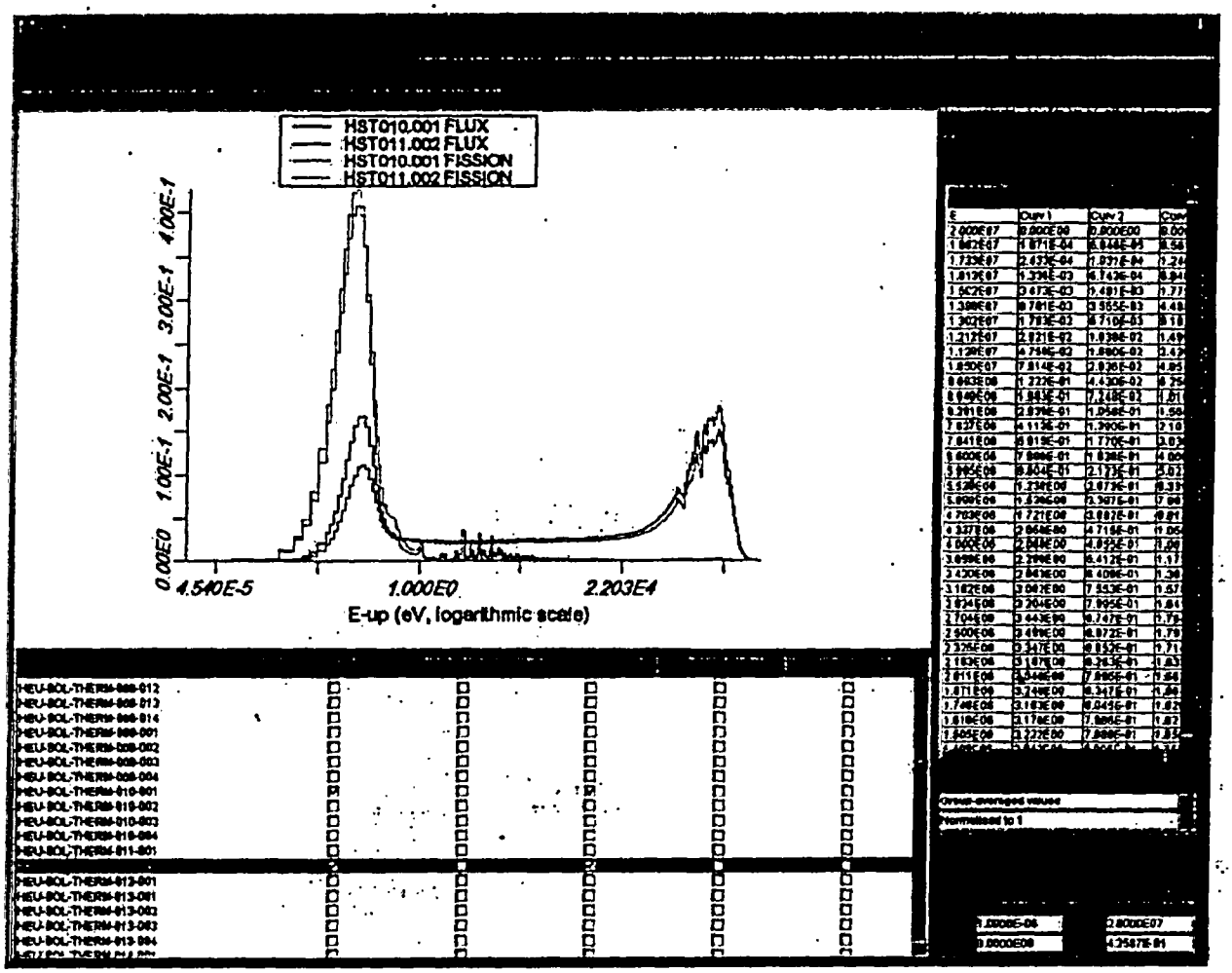


Fig. 4. Neutron spectra and fission rates for experiments HEU-SOL-THERM-011-002, HEU-SOL-THERM-010-001.

1. qualitative parameters
   a. reflector material
   b. pitch type
   c. experimental facility; it is indeed important to include experiments from different laboratories to make sure that the derived trends are not specific to a single program that can be subject to systematic experimental errors and thus to correlations between experiments (see Ref. 6).
2. quantitative parameters
   a. fuel compositions (isotopic wt%)
   b. Pu/(U + Pu) ratio
   c. moderator-to-fuel ratio
   d. moderator-to-fissile ratio
   e. pitch size
   f. energy and spectra data.

As an example, a selection of these parameters is presented in Fig. 6. Other parameters can be retrieved, such as the calculation results for a variety of codes. This enables the user to compare his or her results with those obtained by ICSBEP evaluators. This can help minimize the possibility of mistakes during the development of calculation models for experimental benchmarks.

Each of the experiments represented in the database is accompanied by the effective multiplication factors for the benchmark model and its evaluated uncertainty. These evaluated experimental values, along with the corresponding calculated values obtained by the user, allow the estimation of biases caused by

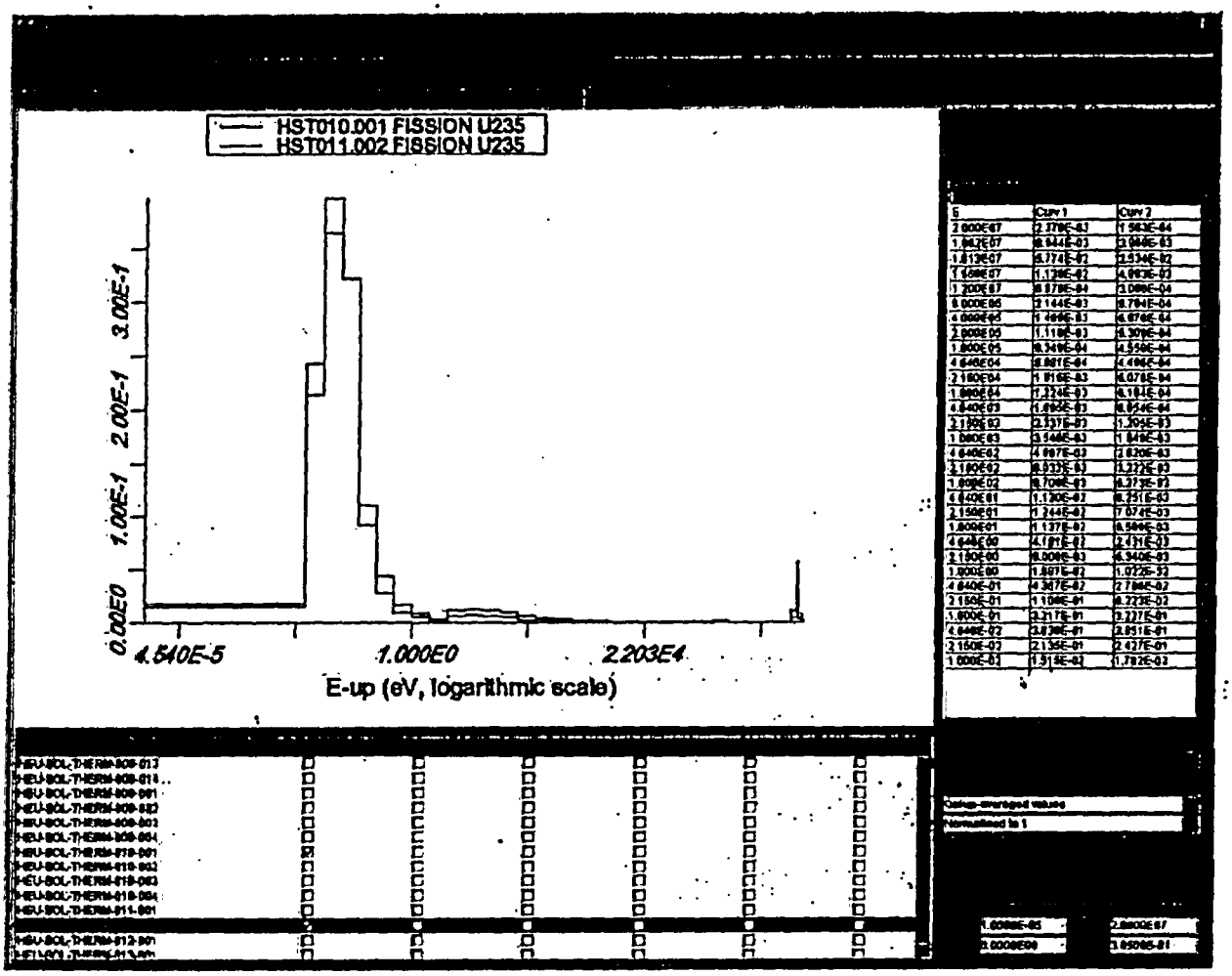


Fig. 5. The 30-group sensitivity coefficients of k_{eff} to ^{235}U fission cross-section values for experiments HEU-SOL-THERM-011-002 and HEU-SOL-THERM-010-001.

calculation methods and nuclear data for the types of systems under consideration.

IV. CONCLUSIONS

The DICE program was designed to help users of the ICSBEP Handbook identify experiments satisfying a specific set of criteria. It also enables users to generate summaries of experiments or to extract specific parameters from a set of configurations. With the possibility of viewing and extracting these data for further analysis, the DICE program further extends the capacity of the ICSBEP Handbook. While the most straightforward usage of the ICSBEP Handbook has been the modeling of experiments for validation purposes, the DICE program allows the user to easily extract systematically specific information that can be used for general criticality safety analysis. For instance, in the first example studied above, one can envisage studying the dependence of the critical mass as a function of the reflector thickness or of the core shape. The second example showed that in the case of validation studies, one can access not only the experimental results but also another set of parameters (pitch size, moderation ratio, Pu/(U + Pu) ratio, etc.) that are important for deriving trends in nuclear data validation. It is also important to note that the ICSBEP Handbook remains the primary source of criticality data. The data entered into DICE are subject to data manipulation errors and have not in all cases undergone thorough checking. Once a desired set of data has been identified with DICE, the user should always verify these data with those published in the ICSBEP Handbook.

MIX-COMP-THERM-001-001	PNL, USA	0.2237	Square	0,9525	1	0,0025	MONK	UKNDL	0,9917	0,001
MIX-COMP-THERM-001-002	PNL, USA	0.2237	Square	1,258	1	0,0026	MONK	UKNDL	0,9934	0,001
MIX-COMP-THERM-001-003	PNL, USA	0.2237	Square	1,5342	1	0,0032	MONK	UKNDL	0,9952	0,001
MIX-COMP-THERM-001-004	PNL, USA	0.2237	Square	1,905	1	0,0039	MONK	UKNDL	0,9969	0,0011
MIX-COMP-THERM-002-001	PNL, USA	0.0204	Square	1,778	1,0024	0,006	MCNP	ENDF/B-V Continuous	0,999	0,0008
MIX-COMP-THERM-002-002	PNL, USA	0.0204	Square	1,778	1,0009	0,0047	MCNP	ENDF/B-V Continuous	0,9998	0,0009
MIX-COMP-THERM-002-003	PNL, USA	0.0204	Square	2,20914	1,0042	0,0031	MCNP	ENDF/B-V Continuous	1,0031	0,0009
MIX-COMP-THERM-002-004	PNL, USA	0.0204	Square	2,20914	1,0024	0,0024	MCNP	ENDF/B-V Continuous	1,0071	0,0009
MIX-COMP-THERM-002-005	PNL, USA	0.0204	Square	2,51447	1,0039	0,0025	MCNP	ENDF/B-V Continuous	1,0061	0,0008
MIX-COMP-THERM-002-006	PNL, USA	0.0204	Square	2,51447	1,0029	0,0027	MCNP	ENDF/B-V Continuous	1,0077	0,0008
MIX-COMP-THERM-003-001	CRX, USA	0.0659	Square	1,3208	1,0028	0,0072	MCNP	ENDF/B-V Continuous	0,9983	0,0009
MIX-COMP-THERM-003-001	CRX, USA	0.0659	Square	1,3208	1,0028	0,0072	MONK	JEF-2.2 Continuous	0,9943	0,001
MIX-COMP-THERM-003-002	CRX, USA	0.0659	Square	1,4224	1,0018	0,0059	MCNP	ENDF/B-V Continuous	1,0009	0,0009
MIX-COMP-THERM-003-002	CRX, USA	0.0659	Square	1,4224	1,0019	0,0059	MONK	JEF-2.2 Continuous	0,9978	0,001
MIX-COMP-THERM-003-003	CRX, USA	0.0659	Square	1,4224	1	0,0054	MCNP	ENDF/B-V Continuous	0,9993	0,0009
MIX-COMP-THERM-003-003	CRX, USA	0.0659	Square	1,4224	1	0,0054	MONK	JEF-2.2 Continuous	0,9957	0,001
MIX-COMP-THERM-003-004	CRX, USA	0.0659	Square	1,8679	1,0037	0,0031	MCNP	ENDF/B-V Continuous	1,0042	0,001
MIX-COMP-THERM-003-004	CRX, USA	0.0659	Square	1,8679	1,0027	0,0031	MONK	JEF-2.2 Continuous	0,9979	0,001
MIX-COMP-THERM-003-005	CRX, USA	0.0659	Square	2,01158	1,0049	0,0027	MCNP	ENDF/B-V Continuous	1,0062	0,0008
MIX-COMP-THERM-003-005	CRX, USA	0.0659	Square	2,01158	1,0049	0,0027	MONK	JEF-2.2 Continuous	0,998	0,001
MIX-COMP-THERM-003-006	CRX, USA	0.0659	Square	2,8418	1	0,0023	MCNP	ENDF/B-V Continuous	1,0069	0,001
MIX-COMP-THERM-003-006	CRX, USA	0.0659	Square	2,8418	1	0,0023	MONK	JEF-2.2 Continuous	0,9974	0,001
MIX-COMP-THERM-004-001	JAERI, Japan	0.03	Square	1,825	1	0,0048	MCNP	JENDL-3.2 Continuous	0,9929	0,0011
MIX-COMP-THERM-004-001	JAERI, Japan	0.03	Square	1,825	1	0,0048	MONK	JEF-2.2 Continuous	0,9992	0,001
MIX-COMP-THERM-004-001	JAERI, Japan	0.03	Square	1,825	1	0,0048	MULTI-KENO	JENDL3.2 / MOCL137-Oro...	1,0172	0,0011
MIX-COMP-THERM-004-002	JAERI, Japan	0.0299	Square	1,825	1	0,0048	MCNP	JENDL-3.2 Continuous	0,9971	0,0011
MIX-COMP-THERM-004-002	JAERI, Japan	0.0299	Square	1,825	1	0,0048	MONK	JEF-2.2 Continuous	1,0002	0,001
MIX-COMP-THERM-004-002	JAERI, Japan	0.0299	Square	1,825	1	0,0048	MULTI-KENO	JENDL3.2 / MOCL137-Oro...	1,0149	0,001
MIX-COMP-THERM-004-003	JAERI, Japan	0.028	Square	1,825	1	0,0048	MCNP	JENDL-3.2 Continuous	0,995	0,0011
MIX-COMP-THERM-004-003	JAERI, Japan	0.028	Square	1,825	1	0,0048	MONK	JEF-2.2 Continuous	0,9998	0,001
MIX-COMP-THERM-004-003	JAERI, Japan	0.028	Square	1,825	1	0,0048	MULTI-KENO	JENDL3.2 / MOCL137-Oro...	1,017	0,001
MIX-COMP-THERM-004-004	JAERI, Japan	0.03	Square	1,958	1	0,0039	MCNP	JENDL-3.2 Continuous	0,999	0,001
MIX-COMP-THERM-004-004	JAERI, Japan	0.03	Square	1,958	1	0,0039	MONK	JEF-2.2 Continuous	0,9988	0,001
MIX-COMP-THERM-004-004	JAERI, Japan	0.03	Square	1,958	1	0,0039	MULTI-KENO	JENDL3.2 / MOCL137-Oro...	1,0092	0,001

Fig. 6. Fragment of list of selected experiments and details.

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