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Conference

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Proceedings of the Nuclear Criticality Technology Safety Project

Williamsburg, Virginia
May 10–11, 1994

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Abstract

This document contains summaries of most of the papers presented at the 1994 Nuclear Criticality Technology Safety Project (NCTSP) meeting, which was held May 10 and 11 at Williamsburg, Va. The meeting was broken up into seven sessions, which covered the following topics: (1) Validation and Application of Calculations; (2) Relevant Experiments for Criticality Safety; (3) Experimental Facilities and Capabilities; (4) Rad-Waste and Weapons Disassembly; (5) Criticality Safety Software and Development; (6) Cricality Safety Studies at Universities; and (7) Training. The minutes and list of participants of the Critical Experiment Needs Identification Workgroup meeting, which was held on May 9 at the same venue, has been included as an appendix. A second appendix contains the names and addresses of all NCTSP meeting participants.

INTRODUCTION

On May 10 and 11, 1994, the Nuclear Criticality Technology Safety Project (NCTSP) held its third annual meeting, this time at the Ft. Magruder Inn and Conference Center in Williamsburg, Va. The conference was broken up into seven sessions that addressed the following topics:

1. Validation and Applications of Calculations
2. Relevant Experiments for Criticality Safety
3. Experimental Facilities and Capabilities
4. Rad-Waste and Weapons Disassembly
5. Criticality Safety Software and Development
6. Criticality Studies at Universities
7. Training.

The following proceedings present the summaries or full text of most of the papers given.

This meeting marked the first time that Russian scientists participated in the proceedings. Anatoly Tsiboulia, of the Institute of Physics and Power Engineering: Obninsk, presented a paper on the development of the computer code ABBN-90. Vladimir Yuferev, of the All-Russian Research Institute of Experimental Physics (Arzamas-16), was also scheduled to give a paper presenting an overview of criticality-safety work carried out at the Institute. He was also supposed to present a paper written by colleague Yevgeny Glushkov on similar work carried out at Moscow's Kurchatov Institute. Unfortunately, last-minute visa problems prevented Mr. Yuferev from attending, so Mr. Tsiboulia presented these papers for him. In all instances, he spoke to the conference through an interpreter.

Immediately prior to the meeting, on May 9, NCTSP working groups met in session at the conference center. These working groups addressed the topics of

- Physics Criteria for Benchmarks,
- Evaluation Techniques, Parametric Studies,
- Experimental Needs, and
- Rules and Regulations Standards.

The minutes and participant list of the Experimental Needs meeting are given in Appendix I. Appendix II contains a list of the names and addresses of all the NCTSP participants.

As a final note, it should be mentioned that a partial annular eclipse of the sun, which occurred in the sky over Williamsburg during the lunch hour on May 10, prevented the conference organizers from starting Session 2 (Relevant Experiments for Criticality Safety) on time. This resulted in the session's running late, which necessitated a drastic abbreviation in the session's concluding remarks given by Burton Rothleder.

AGENDA
NUCLEAR CRITICALITY TECHNOLOGY SAFETY PROJECT MEETING
MAY 10 AND 11, 1994

TUESDAY, MAY 10

Introduction and opening remarks	Burton Rothleder (DOE/EH-64)	0830-0845
Keynote speaker	Herbert J. Kouts (DNFSB)	0845-0945

Session 1:
Validation and Applications of Calculations

Co-chairs:
Blair Briggs
(INEL)
Dae Chung
(DOE)

Speaker	Topic	
Anatoly Tsiboulia (Institute of Physics and Power Engineering)	Critical Experiments Analysis by ABBN-90 Constant System	1000-1030
Cecil V. Parks (ORNL)	Calculations of k_∞ for Homogenous U-235/Metal Mixtures: Will the Real k_∞ Please Stand Up?	1030-1055
Fritz Trumble (ORNL)	Criticality Safety Benchmark Evaluation Project: Recovering the Past	1055-1120
Roger Brewer (LANL)	The Impact and Applicability of Critical Experiment Evaluations	1120-1145
Lunch	Speaker: William Vernetson, past Chair of the National Organization of Test Research and Training Reactors	1145-1315

TUESDAY, MAY 10 (cont.)

Session 2:
Relevant Experiments for Criticality Safety

Co-chairs:
Burton Rothleder
(DOE/EH-64) and
Dennis Cabrilla
(DOE/EM-431)

Speaker	Topic	
Burton Rothleder Dennis Cabrilla	Introduction: Experiment Selection Procedure by the Methodology and Experiments Subcommittee (MES) of the Nuclear Criticality Experiments Steering Committee *	1315-1330
Michaele Brady (SNL)	Spent Fuel Safety Experiments (SFSX)	1330-1350
Rene Sanchez (LANL)	Proposal for Experiments with Actinide Elements	1350-1410
Richard E. Anderson (LANL)	Validation of Calculational Methodology in the Intermediate Energy Range	1410-1430
Robert Rothe (RFP)	Plutonium Solution in Concentration Range from 8 to 17 g/L	1430-1450
Richard E. Malenfant (LANL)	A Program to Evaluate Measurements of Subcritical Systems	1450-1510
J. Blair Briggs (INEL)	Absorption Properties of Waste Matrix Materials	1530-1550
Dennis Cabrilla Hans Toffer (WHC)	Alternate Measurements of Benefit to Criticality Issues at Hanford	1550-1610
Burton Rothleder	Reduction of the Experimental Burden: Calculations as Benchmarks	1610-1630
Working Group Summaries		1630-1700
Banquet	Speaker Victor Stello	1800-2000

** See Appendix I, Attachment 3*

WEDNESDAY, MAY 11

Session 3:
Experimental Facilities and Capabilities

Chair:
Richard Malenfant
(LANL)

Speaker	Topic	
Vladimir Yuferev* (All-Russian Research Institute of Experimental Physics)	An Overview of Criticality Safety Research at the All-Russian Research Institute of Experimental Physics	0815-0845
Yevgeny Glushkov* (Kurchatov Institute)	A Short Review of Critical Experiments Performed at the Kurchatov Institute	0845-0915
Richard Malenfant (LANL)	Critical Experiments, Accident Simulation, and Dosimetry at the Los Alamos Critical Experiments Facility	0915-0945

**Papers presented by Anatoly Tsiboulia*

Session 4:
Rad-Waste and Weapons Disassembly

Co-chairs:
Adolf Garcia (ANL)
Ivon Fergus (DOE/EH-11)

Speaker	Topic	
Ron Knief (Ogden Environmental)	Criticality Analysis for Weapon Disassembly at the Pantex Plant – Part I: Bare Pits	1000-1020
Steve Payne (DOE/ALO)	Postulated Accident Scenarios in Weapons Disassembly	1020-1040
Steve Troyer (Batelle, Pantex)	Criticality Safety in High Explosives Dissolution	1040-1100
John Schlesser (LANL)	Next Generation Storage Facility	1100-1120
Jor-Shan Choi (LLNL)	Long-Term Criticality Concerns Associated with Disposition of Weapons Plutonium	1120-1140
Ron Knief	Criticality Analysis for Weapon Disassembly at the Pantex Plant – Part II: Staging	1140-1200
Lunch	Speaker: Dr. Robert Wilson (US NRC)	1200-1300

WEDNESDAY, MAY 11 (cont.)

Session 5:
Criticality Software and Development

Co-chairs:
Arthur R. Forster
(LANL)
Michael Westfall
(ORNL)

Speaker	Topic	
Roger Blomquist (ANL)	VIM – Monte Carlo Neutron Transport Code	1300-1320
Dan Hollenbach (ORNL)	KENO Developments	1320-1340
Bill Lloyd* (LLNL)	COG Developments	1340-1400
Art Forster (LANL)	Recent Developments in the Los Alamos Radiation Transport Code System	1400-1420
Mark Williams (Louisiana State University)	Energy-Pointwise Discrete Ordinates Transport Methods	1420-1500

** Paper presented by Jor-Shan Choi*

Session 6:
Criticality Safety Studies at Universities

Chair:
Robert Busch
(University of New Mexico)

Speaker	Topic	
Don Harris (Rensselaer Polytechnic Institute)	Critical Experiments with Mixed Oxide Fuel	1500-1520
David Hetrick (University of Arizona)	Student Research in Criticality Safety at the University of Arizona	1520-1540
Lee Dodds (University of Tennessee)	Criticality Safety Research at the University of Tennessee-Knoxville	1540-1600
Robert Busch (University of New Mexico)	Nuclear Criticality Research at the University of New Mexico	1600-1620

WEDNESDAY, MAY 11 (cont.)

Session 7:
Training

Chair:
Mayme Crowell
(ORISE)

Speaker	Topic	
Ava King Harvey (Y-12 Plant)	Training at the Y-12 Plant	1630-1640
Bart Woodruff (LANL)	Criticality Safety Training	1640-1650
Richard Taylor (Y-12 Plant)	Training of Nuclear Criticality Safety Engineers	1650-1700
Lee Dodds (University of Tennessee)	Nuclear Criticality Safety Course Descriptions	1700-1715
Richard Malenfant (LANL)	Criticality Safety Training at Los Alamos	1715-1730

THE IMPORTANCE OF EXPERIMENTAL INFORMATION IN CRITICALITY RESEARCH

H. J. Kouts
Defense Nuclear Facility Safety Board

About 30 years ago, when I was a member of the Advisory Committee on Reactor Safeguards, a frightful thought occurred to me. It was that at some distant future time there would be nuclear power plants, but there would be no people who really understood the neutron physics of chain-reacting systems. Reactors would be designed by cookbook methods, using procedures written by people who themselves had received their instruction from books written by other people. The thought was frightful because of the implications for the safety of the reactors. To be sure, neutron physics and the protection against power excursions underlie only part of the safety of nuclear plants, but that part is very important. And I am not comfortable with the thought that there might be no individuals associated with the safety of these plants who had developed the kind of insight into the behavior of chain-reacting systems that comes from taking systems of this kind to critical under a variety of situations. Likewise, as my attention has shifted in recent years to safety in the defense nuclear arena, I have the same frightening thought concerning nuclear weapons in the future.

I have faith that the era of nuclear power is not drawing to an end, that the Luddites who oppose all advances brought by high technology will in due time be defeated by reason and the reality of a world hungry for electricity produced even when the sun does not shine and the wind does not blow. And I am unable to visualize a future world without nuclear weapons, if that world contains more than a single country. I do not believe that any major nuclear-weapons power will ever place itself in the position where it would become defenseless in the face of discovery that some other country had not played by the same rules of disarmament.

So, I am convinced that it is important to make sure that there always continues to exist a cadre of research scientists who knows criticality as something more than what happens in running a reactor simulator, or what is found by solving an eigenvalue problem, or running a computer code like KENO, or a weapons design code.

What has been the origin of my personal feeling of concern regarding this matter? It is a result of a long-time background in a world in which experimental studies in criticality abounded. Such studies were carried out in numerous facilities of the Atomic Energy Commission (AEC). Simply to list them is a source of comfort in respect to a widespread diffusion of understanding of the behavior of neutron chain-reacting systems.

The principal centers for development of data on criticality were Los Alamos, where, in my time, Hugh Paxton and his coworkers developed so much understanding, particularly of metal systems, and Oak Ridge, where Dixon Callihan and his associates did so much work on uranium

systems of many kinds. But there were many other places where important experimental work was carried out. At Rocky Flats, a group under Schuske generated information on criticality of plutonium metal systems, important to the safety of handling components of nuclear weapons. At Hanford, Duane Clayton's group developed wider understanding of the criticality of plutonium systems, especially solutions. Of course, at Argonne East and Argonne West there were numerous critical experiments directed to reactor design, ranging from those pertinent to the first Nautilus reactor core to the basic design experiments for the Savannah River reactors and numerous basic studies and design experiments for fast reactor cores and breeder assemblies containing oralloy and plutonium fissile elements. Other submarine design experiments were done at the Bettis and Knolls Laboratories. We should not forget the important basic water-reactor studies at Bettis under Dan Klein, and the early studies at Knolls on beryllium-moderated intermediate-neutron-energy critical systems.

We can continue at greater length: there were flexible critical experiments at Savannah River for improved understanding of the physics of heavy-water-moderated reactors and for design of production reactor loadings. At what is now the Idaho National Engineering Laboratory, there were basic studies in the RMF and the ARMF facilities, as well as critical experiments to ensure safe loading of the test reactors operated at that site. At Livermore there were basic experiments in weapon design and weapon safety that only ended after a plutonium fire shut down the experimental facility. Critical experiments were conducted at Sandia. The Air Force ran a critical experiment facility at its Plum Brook facility. Critical experiments were conducted at Hanford in the design of graphite-moderated reactors, leading up to design of the N-Reactor. And there were important reactor design facilities operated by Westinghouse at Walt Mills, by General Electric at Vallecitos, by Babcock and Wilcox near Lynchburg, and by Combustion Engineering near Hartford. I know that I have slighted some important areas that I have just not recalled or that I was not aware of.

But I do have to add to the list the richly varied array of critical experiments that were done by my talented group of experimenters at Brookhaven. And you will have to forgive me if I mention this work in somewhat more detail, as it does underlie the importance that I personally assign to the actual experience of conducting critical experiments.

At Brookhaven we did basic exponential and critical experiments with slightly enriched uranium, ^{233}U/thorium, and plutonium/uranium systems–most often with light water as the neutron moderator, all in order to produce general reactor physics data. But we also did experiments on graphite-bismuth systems in connection with the design of a liquid-fueled reactor, and we ran series of neutron physics studies for design of a number of research reactors, including the Brookhaven High-Flux Beam Reactor, the Brookhaven Medical Research Reactor, the reactor for the Aberdeen Proving Ground, and several university reactors. In designing the High-Flux Beam Reactor, we ran a very large number of critical experiments; I believe it must have been well over a thousand criticalities. Our program ended with several fast reactor critical assemblies performed with a fuel of thin uranium-aluminum alloy foil, for the purpose of evaluating heterogeneous effects in fast critical assemblies.

During this period when critical experiment facilities were abundant and very active, the community of those engaged in the experiments was large and strongly interacting. Data and techniques were widely shared. Individuals made frequent visits to each others' facilities. We at Brookhaven had a joint program with Bettis that involved use by both facilities of the same slightly enriched uranium fuel elements and permitted interchange of experimental data and analytical methods. This program established the experimental data base underlying design of the Shippingport Reactor and successor light-water reactors. We sent some of the fuel that we had used to MIT for use in exponential experiments using heavy water. We sent some to the SPERT facility in Idaho for use in the second SPERT destructive test. In some of our experiments, we used fuel made at Fernald. In others, we used fuel made at Oak Ridge, at Los Alamos, at Babcock and Wilcox, at Nuclear Metals in Cambridge, Massachusetts. We sent data to a number of places for a variety of uses: the interactive program with Bettis that I have just mentioned; to Hanford for use in ensuring safety in dissolver operation at the PUREX facility; to the California Research Corporation for use in designing the target for E. O. Lawrence's Materials Test Assembly, which was to be an accelerator used to produce plutonium; to Savannah River; and to commercial facilities for use in development of reactor design codes. These interactions are simply examples of those that I knew firsthand and that involved research at Brookhaven, which was not even one of the major sites for criticality studies. No doubt those who were engaged in programs at the major sites could relate even richer stories of accomplishment and interaction.

Of course, the abundance of research in the days whose history I am repeating was served by a much greater freedom of action than is found now. There was much less formality associated with funding, and there was much less external safety review. Again a note from the Brookhaven program to illustrate the point: in the course of our conduct of exponential experiments, we built three source reactors each having a maximum power of 100 kilowatts without the need to seek authorization or approval from the AEC. We did write safety analysis reports on each, and we subjected each to a safety review by our local safety committee, which was a very high-class group, but the review ended there. I doubt that we could have done nearly as much research if we had worked in today's climate. Tolerance for error was higher in the past, and I will have more to say about that in a few minutes.

But first, I want to relate some stories from the past that illustrate the importance of good understanding of the physics of neutron chain-reacting systems. Most of these stories are not written down anywhere that I know of. Some may even be apocryphal to some extent. Some are slight in content, and some are amusing to a degree. But all are related to experimental experience in criticality.

The first story concerns the first post-war production reactor built at Hanford, which, I believe, was the H-Reactor. I simply repeat what I was told afterwards by someone who had access to information on the event. This reactor was designed by a new crew--Fermi ,Wigner, Weisskopf, and coworkers having long departed that scene. The new crew decided that they would use better reactor theory than had been used in design of the earlier reactors and better neutron data that had been developed in the interim. They calculated the expected number of

channels for operation and fortunately, following the example that had been set by Fermi prior to his discovery of xenon poisoning, built into the graphite structure and the cooling capability a generous excess of channels above the number estimated as necessary. It developed that the extra channels were needed. I was also told that this was the first of the production reactors that used commercial steel plate for the forms containing the poured concrete shielding. For this purpose, the earlier reactors used excess steel armor plate that had been set aside for battleships that had been sunk at Pearl Harbor. This armor plate came in odd shapes and had numerous holes for fasteners. It had been necessary to make design drawings showing how the plate was to be reshaped and the holes filled for the new purpose. The shield for the new reactor started from these very drawings. The commercial plate was cut to fit the original shapes as shown in the drawings, and the holes were drilled to match. They were then recut as the drawings showed, the holes were filled, and the forms were erected and the shield was poured. I have no firsthand knowledge of these early Hanford stories, and they may be apocryphal to some extent, but I only relate to you anecdotal information as I received it.

My second story is better established because I did hear it from the principal. It illustrates the triumph of insight over bad theory. Irving Kaplan, of whom I am sure you have all heard and some know, had left Brookhaven, where he had been the physics designer of the Brookhaven Graphite Research Reactor (GRR), which was the first nuclear reactor built for purely peaceful research. Irving went to MIT to join the faculty being assembled by Manson Benedict in nuclear engineering. On a visit about a year later he told me that one of the problems he had assigned his class was to calculate the k_∞ of the Brookhaven GRR. Now at the time, the reactor was fueled with natural uranium slugs identical to those used in the eight Hanford production reactors: 1.1 inches in diameter and about four inches long. These were placed end-to-end in aluminum cladding, so that each fuel channel contained two composite elements about 12 feet long. Irving expected some straightforward use of the four-factor formula.

One of the students brought in his results in the form of a thick sheaf of calculations. Irving went directly to the bottom line of the calculation, where the result was stated as something like "$k_\infty = 9$." Irving said simply, "No." The student was outraged: "What do you mean? You haven't even looked at my calculation. See, I used multigroup theory, and it's all laid out." Irving said "No" again. "But I used a computer, and it can't be wrong," the student replied. This is certainly a straightforward example of the value of *insight* into the neutron physics of a system, and how it can defeat bad science. It illustrates what I call a "sanity check," which is a simple practice of asking whether an answer makes sense in the context of all that is known about the subject of the question.

A third story involves another operation at Hanford in the early 1950s. This was also told to me by an individual who knew it firsthand. As I said earlier, some of the Brookhaven water-lattice criticality data was used at Hanford in connection with dissolver safety. It was also used for ensuring criticality safety of irradiated fuel slugs stored in buckets underwater in spent-fuel storage pools. Both of these uses were highly conservative because the criticality data were based on the assumption of regular arrays of fuel in a water moderator, whereas the storage was

under conditions far from optimum for achieving criticality. A Hanford visitor told me that the conservatism had been a source of vexation to one of the technicians involved in storage. He had heard that it should really be possible to place many more spent-fuel slugs in a bucket than the rules allowed. So he was found testing that point, moving into an already fully loaded bucket additional slugs from a neighboring bucket. The technician was fired, but the maximum loading in buckets was increased to reflect his experimental finding. Another triumph of experiment over idealized theory.

Now a story about Rocky Flats. Schuske's group conducted experiments to establish the safety of handling and storage of plutonium in process and after the formation into weapons components manufactured at the Flats. At one point, during the intensive buildup of the weapons stockpile during the Cold War, the vaults were becoming rather full of plutonium components, and the question was raised as to whether safety of storage might be compromised by neutron moderation in the bodies of individuals working in the vaults. So an experiment was run. The count rate was measured with the vault empty of people, with one person in the vault, with two people in the vault, etc. The results were plotted as an inverse multiplication curve, as in an ordinary approach to critical.

This story was somewhat ruined by new information on this experiment that I received not long ago from Tom McLaughlin when I mentioned the experiment to him. It turned out that not only did he already know about it, he had a copy of the original report. In fact, the count rate in the vault was reduced as additional people entered. The neutron density was affected more by the neutron *absorption* of the additional bodies than by the neutron *moderation*. What is the moral of this story? I guess it is that even the best insight from long experience with criticality needs testing experimentally. I wonder what theoretical calculations would have predicted?

And a final story in this sequence. This concerns experiments at Livermore during the period before the fire shut down the critical experiment facility at that site. Again, the story was told to me by a participant. Concern arose as to effects of heightened neutron reflection during handling of one particular plutonium assembly by experimental personnel. Someone in the experimental group established that reflection by a human hand could be reproduced by the use of a pork chop. So, in this case, a reciprocal multiplication curve was plotted as a function of the number of pork chops piled on the assembly. In this case, I was told that the familiar form of an approach to critical was seen, though I never saw the curve myself. Nor was I told of the subsequent fate of the pork chops, and whether they formed the basis of someone's dinner.

These are a few examples of the importance of experimental information on criticality, where theory, or even intuition, has not been adequate. Of course, the examples refer back to times when theory was not as advanced as it is now, when neutron data were not as well established, and when powerful digital computers were not available to take advantage of detail in calculational methods. But the advance in capability over the years is being matched by growth in difficulty of problems, as attention shifts from design of simply connected reactor cores of elements in regular arrays and relatively simple geometries encountered in weapons designs to

complicated problems of arrays of storage regular containers generated during cleanup of facilities that are now surplus from defense activities.

I said earlier that I would say some things about how the formality of research in criticality has changed over the years. Why can't we still build small source reactors without long processes to get programmatic and financial approval, and environmental impact studies and prolonged safety review?

Part of the reason, of course, is that bureaucracy grows with time, and administrative arteriosclerosis sets in. Things naturally become more difficult as time passes. I remember going to see Dixie Lee Ray a few months after she had left the AEC and had become the First Assistant Secretary of State for Oceans, Atmosphere, and Environment. She told me that shortly after she joined the State Department she had needed to take on a certain former high official of the AEC as an advisor. She managed to get a consultant contract through for him in a week's intense effort. Whereupon, she said, the State Department formed a committee to find out how she had done that so quickly, and to close down that process so it could never be done again. So bureaucracy is part of the problem. But it is not the full problem.

I am afraid that there are two diametrically opposed sets of arguments that can be made regarding the need for formality–and here I narrow the coverage of the discussion to mean safety in critical experiments. One argument is from the standpoint of logic and reason. The other is a societal one that pays attention more to effect and popular reaction.

First, the logical argument. A very large number of critical experiments have been done in the United States. There have been critical experiment accidents, a fair number, in fact–perhaps one or two dozen, depending on the definition of a critical experiment accident. But following the accidents that occurred during the Manhattan Project, there have been no fatalities in these accidents. The reason is that the experiments have been done with care and under conditions such that if an undesired excursion did take place, injury to nearby individuals would be very unlikely. The design of experimental facilities takes advantage of distance and shielding.

This realization has caused some to feel that the level of safety that seeks no accidents at all may be too stringent. Some years ago, one well-known practitioner in critical experiments said to me that if you never have an accident you are probably being too careful. I shall not say who said this to me. He meant that the balance between programmatic and safety needs was probably not optimized right in such a case. There can be some truth in this view, strictly from the standpoint of logic.

But the institutional arguments on the other side will clearly win. We live in a world now where the slightest departure from the normal in nuclear matters is a cause for hysterical reaction. Such stories receive the widest possible circulation, in the most lurid prose, and the readers, who know so little about nuclear matters, are frightened by even the inconsequential. Accidental criticality would be regarded by most people as equivalent to the detonation of a nuclear weapon.

The prevalent requirement for formality in critical experiments recognizes this, and seeks to avoid accidents of whatever magnitude and consequence.

I am afraid that the free and easy days will not return.

But it is necessary to maintain active programs in critical experiments under the prevailing rules, to maintain as respectable the number of individuals who understand from firsthand experience the physics of chainreacting neutron systems.

I am deeply pleased to have made the opening remarks at a meeting dedicated to answering this.

Session 1: Validation and Applications of Calculations

CRITICAL EXPERIMENTS ANALYSIS BY ABBN-90 CONSTANT SYSTEM

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G. Manturov, and M. Semenov
Institute of Physics and Power Engineering
Obninsk, Kaluga Region, Russia

ABSTRACT

The ABBN-90 is a new version of the well-known Russian group-constant system ABBN. Included constants were calculated based on files of evaluated nuclear data from the BROND-2, ENDF/B-VI, and JENDL-3 libraries. The ABBN-90 is intended for the calculation of different types of nuclear reactors and radiation shielding. Calculations of criticality safety and reactivity accidents are also provided by using this constant set. Validation of the ABBN-90 set was made by using a computerized bank of evaluated critical experiments. This bank includes the results of experiments conducted in Russia and abroad of compact spherical assemblies with different reflectors, fast critical assemblies, and fuel/water-solution criticalities. This report presents the results of the calculational analysis of the whole collection of critical experiments. All calculations were produced with the ABBN-90 group-constant system. Revealed discrepancies between experimental and calculational results and their possible reasons are discussed. The codes and archives INDECS system is also described. This system includes three computerized banks: LEMEX, which consists of evaluated experiments and their calculational results; LSENS, which consists of sensitivity coefficients; and LUND, which consitsts of group-constant covariance matrices. The INDECS system permits us to estimate the accuracy of neutronics calculations. A discussion of the reliability of such estimations is finally presented.

1. INTRODUCTION

I would like to present some work carried out over 10 years at the Institute of Physics and Power Engineering (IPPE) constant laboratory. The aim of this work was to construct a group-constant set that could satisfy the needs of fast breeder designers and also be suitable for calculating normal and accident situations for other types of reactors.

This group-constant set is based on evaluated neutron data files that represent a sufficiently large amount of group data. The set was validated on the basis of a representative collection of macroscopic experiments, which were adopted by the designers of neutronic calculation codes. We already had some experience in the field of group-constant set preparation for reactor and shielding calculation, as is shown in Fig. 1*, which summarizes the development of the ABBN

* *Ed. note: Because of the large number of figures in this article, they have been placed at the end of the paper.*

group-constant set. The ABBN-78 group-constant set is already used as a standard system for fast reactor calculations in Russia. In this year, the ABBN-93 group-constant set will be certified as the standard and then given the widest possible dissemination.

2. ABBN-93 CONSTANT SYSTEM

The scheme of ABBN-93 constant system is shown in Fig. 2. The subsystems are

MICRO – used for group constant calculations.
CONSYST-2 – used for preparing group constants for reactor and shielding calculations.
MACRO – used for validation of group-constant set on the basis of integral and macroscopic experiments.
INDECS – used for accuracy estimation and for group-constant adjustment.

A. MICRO Subsystem

The MICRO subsystem is shown in Fig. 3. The library of files of evaluated neutron data (FOND) includes data files selected from the libraries shown. Some corrections were made to the files before their inclusion in FOND. The BROND-2, ENDF/B-VI, and JENDL-3 libraries are widely known. LIPAR, developed by L. P. Abagyan of the Kurchatov Institute, is the library of cross sections in the thermal and resolved resonance regions used for WWER-type* reactor calculations in Kurchatov Institute. The majority of group-constant calculations were carried out using both GRUCON and NJOY codes to ensure greater reliability of results.

The characteristics of the ABBN-93 group-constant set are as follows:

1. Traditional 26-group approximation for the energy scale is used. For the more important nuclides, 299 representative groups are used, 72 of which lie in the region of thermalization.
2. The specifications of the group constants has been broadened (Fig. 4).
3. Data are presented in a format suitable for easy computer or human analysis (Fig. 5).
4. ABBN-93 group-constant set has been verified by a wider set of macroexperiments than ABBN-78.

B. CONSYST Code Subsystem

CONSYST code (Fig. 6) is used for preparing group constants in various formats and in reactor and shielding calculation codes. The constants for transport calculations are prepared in the ANISN format. The calculations in diffusion and P1 approximations are provided in the ARAMACO format.

While preparing group-constant resonance, CONSYST takes into account self-shielding,

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* *Ed. note: Water-water power reactor - the equivalent of the pressurized water reactor.*

while it estimates integral neutron spectra of zones in B2 approximation. Then macro- and microscopic constants are collapsed in the necessary numbers of broad groups. Resonance self-shielding may be taken into account also by the subgroup (optimized multiband) method.

C. MACRO Subsystem and INDECS System

A schematic diagram of INDECS (improvement of neutron data on the basis of experiments on critical systems) is shown in Fig. 7. It comprises the following:

LEMEX – Library of evaluated macroscopic experiments contains experimental data measured on many critical assemblies and reactors as well as integral experiments data. Results of calculations are also included in LEMEX. Uncertainties of all these values are also incorporated.

LTASK – Library of descriptions of calculation models of experiments (Library of TASKs).

LUND – Library of uncertainties of neutron data contains neutron data covariance matrix in group representation (Fig. 8).

LSENS – Library of sensitivity contains sensitivity of measured values and important physical values of selected test models to the neutron constants.

CORE – The code for the estimation of calculational uncertainties and the adjustment of constants.

3. EVALUATION OF MACROSCOPIC EXPERIMENTS

Main experiments included in LEMEX are shown in Fig. 9. The objectives of the experiments are as follows:

- to evaluate the macroscopic experiment means;
- to construct calculational model of this experiment (the model has to be a simple enough for precise calculation methods);
- to introduce corrections into the experimental data in order to add them to the conditions of the calculation model;
- to evaluate the methodical inaccuracies of these corrections and to take them into account in estimating data uncertainties side by side with experimental uncertainties; and
- to introduce all the data into the appropriate data bank.

4. RESULTS OF VALIDATION

A comparison of the results obtained using ABBN-90 with other codes for the criticality of uranium and plutonium spheres with uranium reflectors is shown in Fig. 10a.

1. Criticality of the lead-reflected and copper-reflected uranium and plutonium spheres is shown in Fig. 10b. These experiments were performed at Arzamas-16.
2. The calculations in subgroup approximations are necessary for iron-reflected spheres (Fig. 10c). The results are essentially decreased in applying transport cross-sections averaged on first harmonic (Σ_1) of flux. Application of zero harmonic (Σ_0) also decreases the result.

3. BFS and Ermine critical assemblies with k_{∞}=~1 are used for verification of uranium and plutonium neutron data (Fig. 11a). Analogous experiments to verify the neutron data of thorium and structural materials were carried out on KBR assembly (Fig. 11b). During the evaluation, we observed that the uncertainties of corrections to k may be rather essential (to 1%). It is necessary to pay a great attention to evaluation of such experiments.
4. Criticalities of fast critical assemblies with uranium and plutonium fuel are described within part of percent (Fig. 12).
5. Criticalities of aqueous solutions of uranium and plutonium salts are also described rather well. This verification was performed by the Kurchatov Institute.

Special experiments on cross-section ratios and central reactivity worth ratios have been used for ABBN verification (Figs. 13a-13d). Some discrepancies were found:

1. Fission rates and reactivity worth of Pu-240 are overestimated.
2. Reactivity worth of Pu-241 is overestimated (especially in the assemblies with hard spectra). Fission rates are described well.
3. Reactivity of B-10 is underestimated about 6% for all assemblies.
4. The U-235 fission spectrum we used, which was close to that adopted in ENDF/B-VI, differs significantly from the data of differential measurement evaluations (Figs. 14a and 14b). Difference of these spectra in SCHERZO-5.56 leads to 0.8% difference in k.
5. Systematic discrepancies in the criticality of small water-uranium systems were found. These discrepancies can be removed by reducing the hydrogen cross section about 1.5% in the fast energy range (Fig. 15).

5. CONCLUSION

As can be seen from the material presented, the agreement between calculation and experiment data is very good. The adjustment of the constants on the basis of these experiments does not lead to significant displacements. But group-constant covariance matrices are quite different. The last matrix leads to the smaller calculation inaccuracies.

As an example, let us consider the calculated inaccuracy of k_{eff} for the BN-800 reactor, designed for actinide transmutation (Fig. 16). To reduce the discrepancy between the calculational errors and the target accuracy, macroscopic experiments sensitive to inelastic scattering on minor actinides must be completed. Such experiments with Np-237 are planned on the BFS assembly, but work has been delayed because of financial difficulties.

Until ABBN-93 is certified by a government commission, the previous version of ABBN group-constant set will continue to be used for fast and thermal reactor calculations and for calculations concerning nuclear safety.

ABBN-64
- 26 groups
- about 40 nuclides
- self-shielding factors in resolved and unresolved resonance regions f(σ,T)
- only transport constants

ABBN-72
- new data for fission nuclides
- using subgroup parameters

ABBN-78
- 28 groups
- about 90 materials
- gamma constants
- testing and correction, some data based on integral experiments

ABBN-90
- 28 groups as basic group structure
- about 200 nuclides
- self-shielding factors and subgroup parameters
- multigroup data (300 groups) for nuclides H,B,C,O,N,Na,Fe,Cr,Ni,U5,U8,Pu39,Pu40,Pu41
- constants for calculation of photon sources
- constants for photon transport calculation
- other constants
- convenience in use data

↓

ABBN-93

- based on Nuclear Data Files only
- testing data in integral experiments
- correction of Files if needed

Figure 1. Historical development of ABBN.

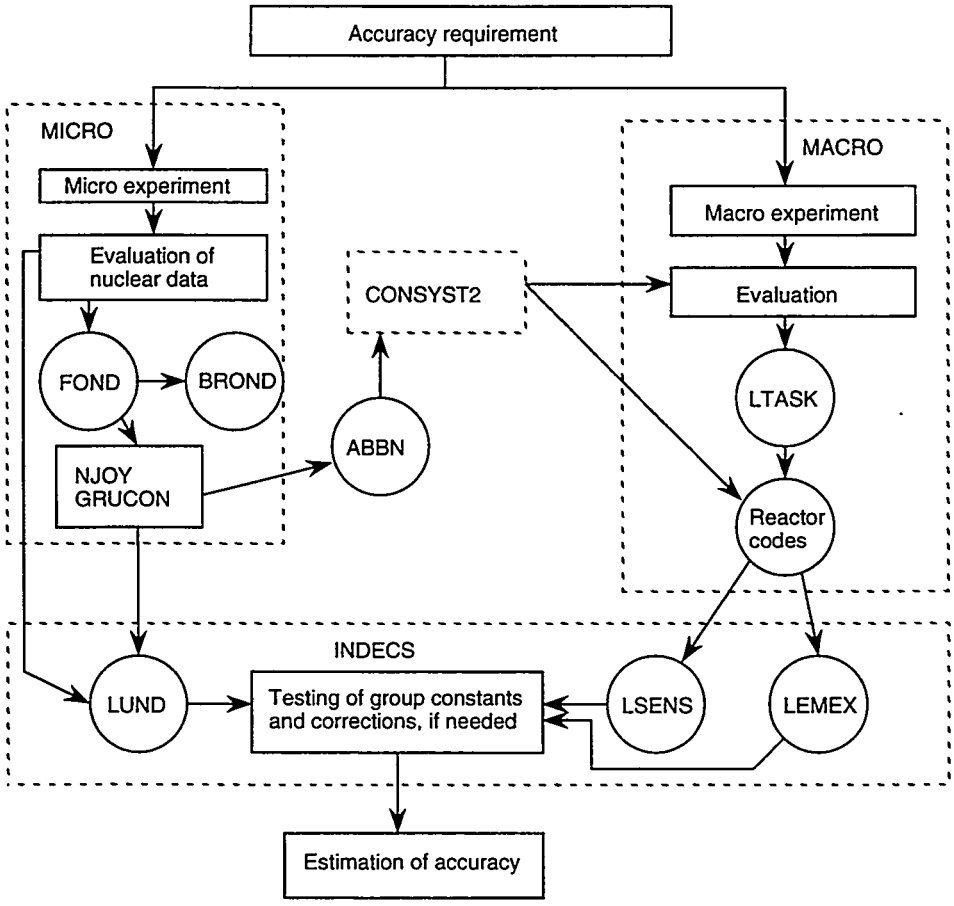
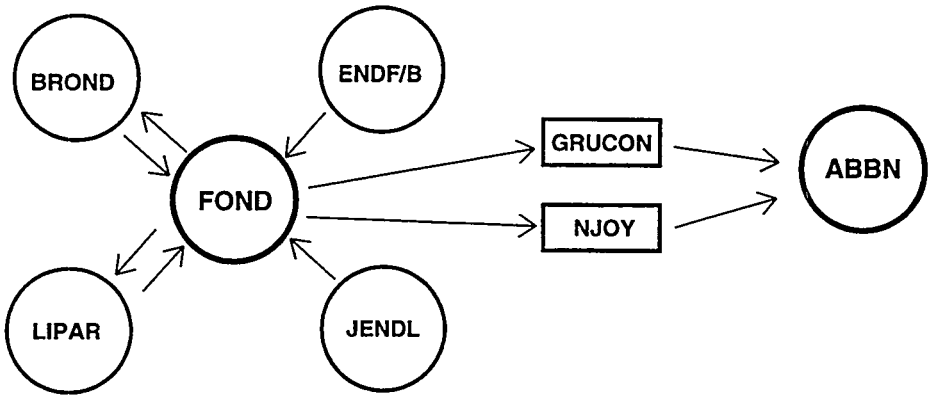


Figure 2. Schematic diagram of ABBN-93.



FOND ≠ *BROND*
- *FOND* contains more materials than *BROND*;
- some materials from *BROND* are not included in *FOND*;
- *all files* from *FOND* were processed by *GRUCON* and some by *NJOY* also.

LIPAR – resonance parameter library below 1 keV;
- the basic isotopes were included in *ABBN*.

GRUCON
- our own development (Sinitsa V.);
- possibility of local energy problem solution;
- various method of subgroup calculation;
- analysis of transmission experiment results;
- file editing.

NJOY
- reliability (many users);
- completeness (not only neutron cross-section).

Figure 3. ABBN-93 MICRO subsystem.

ABBN-93 contains:

- main neutron constants for transport calculations (total, capture, fission, etc.)
- inelastic scattering transfer matrices for two angular momenta L=0,1
- elastic scattering transfer matrices for six angular momenta L=0,1,2,3,4,5
- hydrogen scattering data for angular momenta L=0,1,2,...20
- resonance self-shielding factors of total, capture, fission, and elastic cross sections for T=300K and 16 standard dilutions
- Doppler increments of self-shielding factors for T=300-900K and T=900-2100K
- subgroup resonance structure parameters (mutual, consistent with f-factors)
- multigroup data for the most important nuclides
- neutron reaction cross sections (which are available in the range up to 20 MeV)
- delay neutron data
- thermal neutron data (G-factors of Westcott and scattering matrices for some materials)
- data for calculation of energy releasing in different reactions (KERMA-factors)
- photon production in neutron reactions
- photon interaction cross sections
- fission product yields
- decay data of radionuclides produced in neutron reactions
- decay photon spectra of radionuclides
- covariance matrices of the most important reactions

Figure 4. Group-constant factors included in ABBN-93.

```
*
* revision: 1; 0 group cross sections was corrected
*             corresponding with U238V78 data.
*           2) Self Shielding Factor table (MF=4,5) was
*             obtained from subgroup parameters table.
*             In 17 group doppler increment was corrected
*             same as in BNAB78.
*           3) 12, 13, and 14 group constant set (MF=1 MT=0)
*             was taken from multygroup data.
*
NAM=U238     BIB=FOND  MF=  1  MT=  0  AWR=2.36006E+02
             LT =  28  LC= 10  LS= 10  LF = (I4,E7.0,2E9.0,E7.0,5E6.0)
*                          92-U -238
*                    MAIN NEUTRON CONSTANTS
* NG  total   capture  fission  elast  inel  mult    nu     mu    ksi
*
  -1   5.80    0.0027   1.1892   2.92 1.696 2.147 4.538 .8615 .0012
   0   5.77    0.0033   0.9992   2.87 1.906 1.797 4.087 .8277 .0015
*
   1   6.46    0.0056   0.9424   3.49 2.021 1.369 3.513 .7866 .0018
   2   7.55    0.0107   0.5733   4.33 2.638 1.001 3.115 .7673 .0020
   3   7.76    0.0206   0.5380   4.24 2.965 1.000 2.811 .7363 .0022
   4   7.12    0.0489   0.4651   3.84 2.769 1.000 2.642 .5510 .0038
*
   5   7.11    0.1099   0.0396   4.66 2.301 1.000 2.546 .4506 .0046
   6   8.14    0.1154   0.0011   6.33 1.697 1.000 2.493 .3577 .0054
   7   9.79    0.1185   0.0001   8.46 1.211 1.000 2.434 .2201 .0063
   8  11.35    0.1500   0.0000  10.44 0.756 1.000 2.397 .1209 .0074
*
   9  12.54    0.2424   0.0000  11.99 0.303 1.000 2.378 .0586 .0080
  10  13.63    0.4335   0.0000  13.19 0.000 1.000 2.369 .0248 .0083
  11  14.88    0.6107           14.27               2.364 .0100 .0084
*
  12  16.60     .8646           15.74               2.362 .0028 .0084
  13  19.86    1.2458           18.61               2.361 .0028 .0084
  14  21.84    1.8521           19.98               2.361 .0028 .0084
*
  15  22.28    3.3105           18.96               2.361 .0028 .0084
  16  21.88    4.5296           17.35               2.361 .0028 .0084
  17  89.09   20.2163           68.88               2.360 .0028 .0084
*
  18  41.00   16.5577           24.44               2.360 .0028 .0084
  19 143.40   54.1026           89.30               2.360 .0028 .0084
  20 126.47   83.8098           42.66               2.360 .0028 .0084
*
  21 189.36  169.7841           19.58               2.360 .0028 .0084
  22   8.87    0.6553            8.21               2.360 .0028 .0084
  23   9.14    0 4814            8.66               2.360 .0028 .0084
*
  24   9.39    0.5938            8.80               2.360 .0028 .0084
  25   9.67    0.8155            8.85               2.360 .0028 .0084
  26  11.61    2.7100            8.90               2.360 .0028 .0084
*
```

Figure 5a. An example of the format of ABBN-93 (continued in Figs. 5b, 5c, and 5d).

```
NAM=U238      BIB=FOND  MF=  3  MT=  0  AWR=2.36006E+02
              LT =  14  LC=  7  LS=  7  LF = (I4,6E11.0)
*                     ANGULAR MOMENTA OF ELASTIC TRASITIONS
*                     FROM GROUP g TO THE SAME GROUP
* G/L        0           1           2           3           4           5
*
  -1     0.9375      0.8141      0.6931      0.5782      0.4889      0.4205
   0     0.9861      0.8252      0.6878      0.5850      0.4937      0.4030
*
   1     0.9885      0.7849      0.6469      0.5521      0.4552      0.3502
   2     0.9928      0.7664      0.6255      0.5181      0.4056      0.2937
   3     0.9933      0.7363      0.5801      0.4494      0.3302      0.2098
   4     0.9919      0.5516      0.3649      0.2935      0.1716      0.0552
*
   5     0.9902      0.4516      0.2572      0.1529      0.0642      0.0019
   6     0.9899      0.3593      0.1539      0.0490      0.0178      -.0033
   7     0.9884      0.2228      0.0680      0.0064      0.0039      -.0038
   8     0.9881      0.1242      0.0240      -.0024      0.0034      -.0023
*
   9     0.9888      0.0621      0.0087      -.0015      0.0019      -.0003
  10     0.9889      0.0284      0.0033      -.0006      0.0005      -.0000
  11     0.9886      0.0137      0.0008      -.0002      0.0001      -.0000
  12     0.9890      0.0065      0.0000      -.0000      -.0000      0.0000
*
NAM=U238      BIB=FOND  MF=  3  MT=  1  AWR=2.36006E+02
              LT =  14  LC=  7  LS=  7  LF = (I4,6E11.0)
*                     ANGULAR MOMENTA OF ELASTIC TRASITIONS
*                     FROM GROUP g TO THE NEXT GROUP
* G/L        0           1           2           3           4           5
*
  -1     0.0625      0.0474      0.0415      0.0358      0.0297      0.0255
   0     0.0139      0.0025      0.0002      0.0008      0.0010      0.0003
*
   1     0.0115      0.0017      0.0010      0.0016      0.0015      0.0013
   2     0.0072      0.0009      0.0008      0.0008      0.0008      0.0007
   3     0.0067      -.0000      0.0006      0.0004      0.0004      -.0000
   4     0.0081      -.0006      -.0004      0.0006      -.0000      -.0005
*
   5     0.0098      -.0010      -.0000      -.0003      0.0000      -.0003
   6     0.0101      -.0016      -.0003      -.0004      0.0000      -.0001
   7     0.0116      -.0027      -.0004      -.0003      0.0001      -.0001
   8     0.0119      -.0033      -.0003      -.0001      0.0001      -.0000
*
   9     0.0112      -.0035      -.0001      -.0000      0.0000      0.0000
  10     0.0111      -.0036      -.0000      0.0000      0.0000      0.0000
  11     0.0114      -.0037      -.0000      0.0000      0.0000      0.0000
  12     0.0110      -.0037      -.0000      0.0000      -.0000      -.0000
*
```

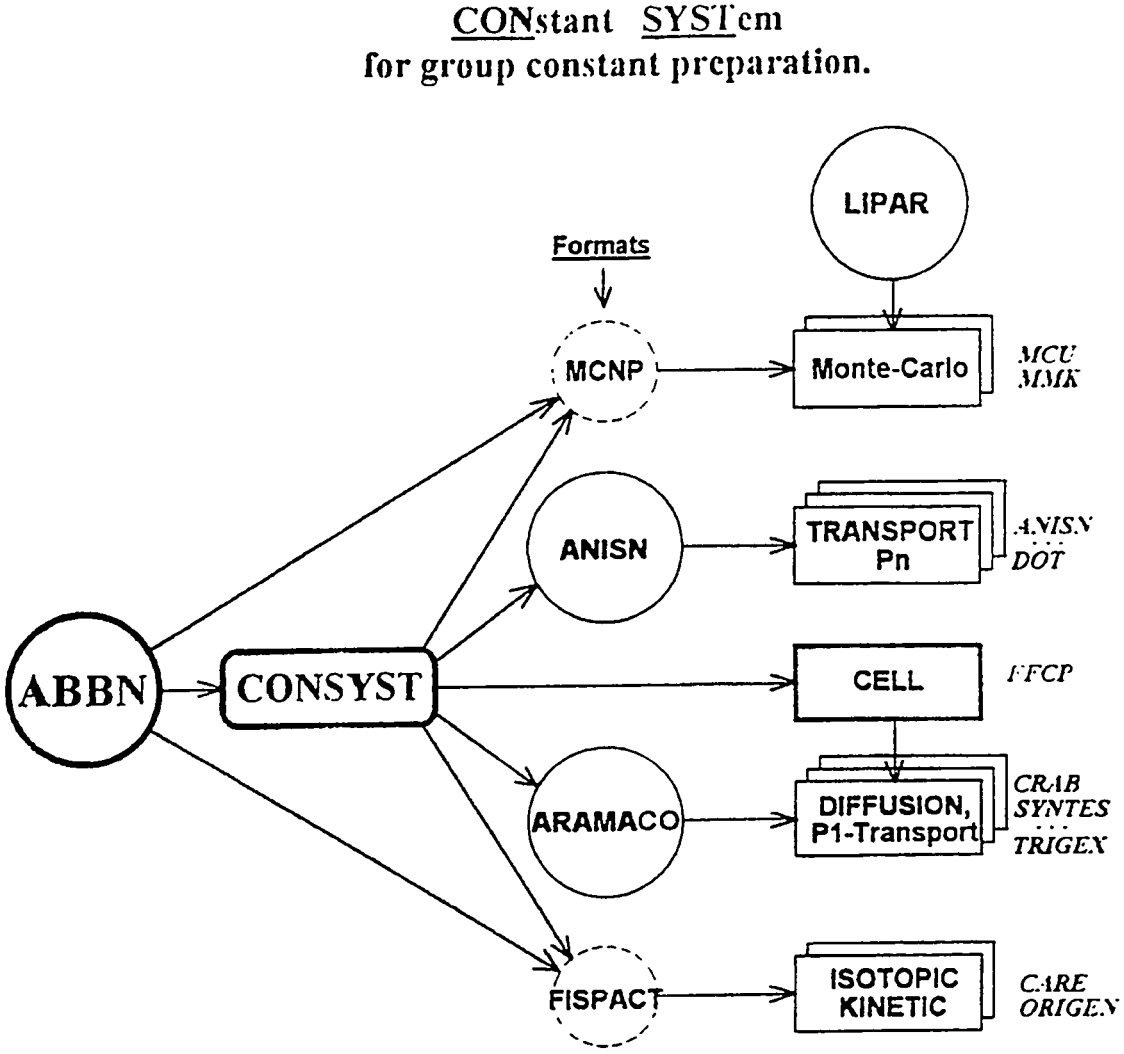
Figure 5b. Example of ABBN-93 format.

```
NAM=U238      BIB=FOND  MF=  4  MT=  1  AWR=2.36006E+02
              LT =  14  LC= 18  LS= 18  LF = (I4,I2,1X,16I4)
*                    RESONANCE SELFSHIELDING FACTORS
*                    TRANSPORT
*  G N    0   .1  ...  ...   1  ...  ...   10 ... ... 100 215 465 1000...*10
   8 0  955 955 956 956 958 960 964 970 976 981 985 986 987 988 988 988
   9 0  917 917 917 918 920 924 932 942 955 967 975 980 982 983 984 984
  10 0  880 881 881 882 885 890 900 916 938 961 978 988 994 997 998 999
  11 0  797 797 798 799 802 807 818 838 871 911 946 971 985 993 996 998
  12 0  704 705 705 706 707 711 718 734 766 816 876 927 962 981 991 995
  13 0  574 575 575 576 578 581 588 600 622 661 722 801 876 931 965 983
  14 0  205 224 243 277 325 372 411 445 485 535 597 672 761 847 913 955
  15 0  349 353 357 365 376 391 407 424 448 482 530 599 691 792 877 935
  16 0  480 480 481 484 487 494 502 513 527 546 575 619 684 766 849 915
  17 1   72  81  89  99 110 121 133 148 175 223 301 416 565 717 839 916
  18 1  203 208 212 218 225 234 244 260 285 319 368 452 584 730 847 922
  19 1   60  62  64  67  72  78  86  95 108 127 155 197 273 410 590 757
  20 1   63  63  63  64  65  66  68  74  86 105 134 185 289 459 650 804
  21 1   56  57  58  59  61  65  72  82  96 118 156 224 350 529 709 842
*
NAM=U238      BIB=FOND  MF=  4  MT=102  AWR=2.36006E+02
              LT =  14  LC= 18  LS= 18  LF = (I4,I2,1X,16I4)
*                    CAPTURE
*  G N    0   .1  ...  ...   1  ...  ...   10 ... ... 100 215 465 1000...*10
   8 0  984 984 984 985 985 986 988 991 994 996 998 999 999 999 999 999
   9 0  963 963 964 964 965 968 972 978 985 991 995 997 998 999 999 999
  10 0  929 929 930 931 932 936 943 954 967 980 989 994 997 998 999 999
  11 0  853 853 854 855 858 864 876 894 920 947 969 984 992 996 998 999
  12 0  730 730 731 733 736 743 756 781 820 870 918 954 976 988 994 997
  13 0  569 569 570 571 574 579 590 611 648 707 782 857 918 956 978 989
  14 0  267 270 274 280 290 306 332 374 437 520 619 726 825 900 947 974
  15 0  170 171 172 174 178 186 203 234 286 364 469 596 729 838 913 956
  16 0  126 126 127 128 131 136 147 168 205 267 357 476 616 750 855 924
  17 1   42  45  48  52  60  74  98 136 196 283 402 547 697 820 904 953
  18 1   47  48  49  52  59  71  94 131 188 267 377 520 677 810 898 949
  19 1   33  34  35  36  39  45  54  70  95 135 195 287 422 586 742 858
  20 1   18  18  19  20  23  29  39  57  87 134 206 318 474 646 792 890
  21 1   27  28  29  32  37  45  59  81 115 171 256 381 542 704 832 913
*
NAM=U238      BIB=FOND  MF=  4  MT=  2  AWR=2.36006E+02
              LT =  14  LC= 18  LS= 18  LF = (I4,I2,1X,16I4)
*                    ELASTIC
*  G N    0   .1  ...  ...   1  ...  ...   10 ... ... 100 215 465 1000...*10
   8 0  980 980 981 981 982 983 985 989 992 995 997 998 999 999 999 999
   9 0  961 961 961 962 963 966 970 977 984 990 995 997 998 999 999 999
  10 0  928 928 928 929 931 935 942 953 967 979 989 994 997 998 999 999
  11 0  862 862 863 864 867 873 883 901 925 951 972 985 993 996 998 999
  12 0  770 770 771 772 775 781 793 814 848 891 931 962 980 990 995 997
  13 0  638 638 639 640 643 648 659 678 711 761 822 883 933 964 982 991
```

Figure 5c. Example of ABBN-93 format.

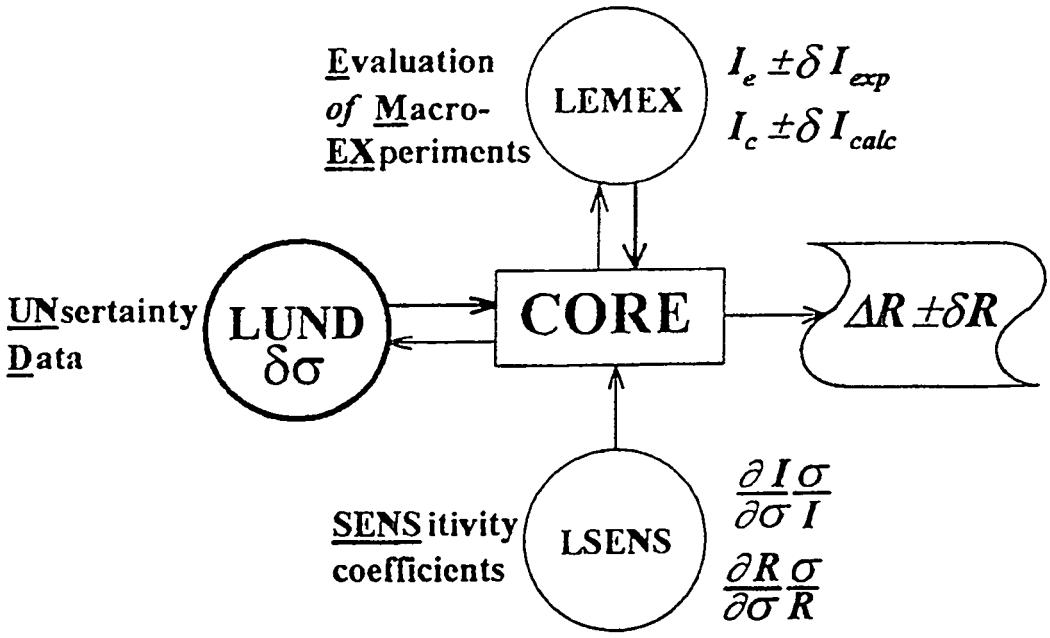
```
NAM=U238      BIB=FOND  MF=  6  MT=  0  AWR=2.36006E+02
              LT =  47  LC=  5  LS=  5  LF = (I4,E5.0,9E7.0)
*                    SUBGROUP PARAMETERS
* G share    capt   elast    fis
*
   9.3511 1.1732 1.2425 0.0
    .6489 0 9063 0.8688 0.0
  10.3567 1 3065 1.3758 0.0
    .6433 0.8301 0.7916 0.0
  11.1058 2.3591 2.4065 0.0
    .8607 0.8695 0.8518 0.0
    .0335 0.0600 0.3667 0.0
  12.0384 4.7587 5.1930 0.0
    .3874 1.5140 1.1184 0.0
    .5696 0.4021 0.6440 0.0
    .0046 0.3615 0.1105 0.0
  13.0247 8.9924 10.124 0.0
    .1361 3.3327 1.8650 0.0
    .8241 0.3889 0.6002 0.0
    .0151 0.2540 0.0961 0.0
  14.0198 15.058 16.941 0.0
    .0650 7.6782 2.8316 0.0
    .8920 0.2241 0.5374 0.0
    .0232 0.1254 0.0503 0.0
  15.0196 24.915 19.069 0.0
    .0581 7.0192 2.8172 0.0
    .8801 0.1134 0.5192 0.0
    .0422 0.0970 0.1334 0.0
  16.0116 49.981 25.209 0.0
    .0276 12.268 2.7856 0.0
    .7857 0.0933 0.7149 0.0
    .1751 0.0477 0.3940 0.0
  17.0213 37.891 34.171 0.0
    .0701 2.3471 1.8306 0.0
    .8161 0.0298 0.1732 0.0
    .0925 0.0443 0.0267 0.0
  18.0093 86.755 59.505 0.0
    .0191 7.9674 3.3303 0.0
    .7437 0.0484 0.4555 0.0
    .2279 0.0220 0.1943 0.0
  19.0142 62.687 56.127 0.0
    .0472 1.7148 1.7974 0.0
    .5564 0.0375 0.1737 0.0
    .3822 0.0210 0.0563 0.0
  20.0143 59.091 50.288 0.0
    .0182 6.8134 5.8890 0.0
    .0553 0.3659 0.4617 0.0
    .9122 0.0118 0.1624 0.0
  21.0272 30.739 17.956 0.0
    .0392 3.1657 2.0815 0.0
    .1900 0.1627 0.6167 0.0
    .7436 0.0120 0.4207 0.0
```

Figure 5d. Example of ABBN-93 format.



ARAMACO - **NGN=26 group**
ANISN - **NGN=1 - 300, NGG=15 group**

Figure 6. Schematic diagram of CONSYST subsystem.



Covariance matrix:

WAS	**- ABBN-78**	**12 groups,**	**60 reactions**
NOW	**- ABBN-90**	**28 groups,**	**500 reactions**

- **is not obtained from ENDF/B file but from consideration of experimental conditions and theoretical parameterization;**
- **is widely used in uncertainty analysis of reactor physics calculations;**
- **28 groups are quite sufficient;**
- **all ENDF/B-VI MF=3 data are processed by NJOY but are not analyzed;**
- **results obtained on the basis of covariance matrix should be interpreted with the care taking common sense into account.**

Figure 7. Schematic diagram of INDECS subsystem.

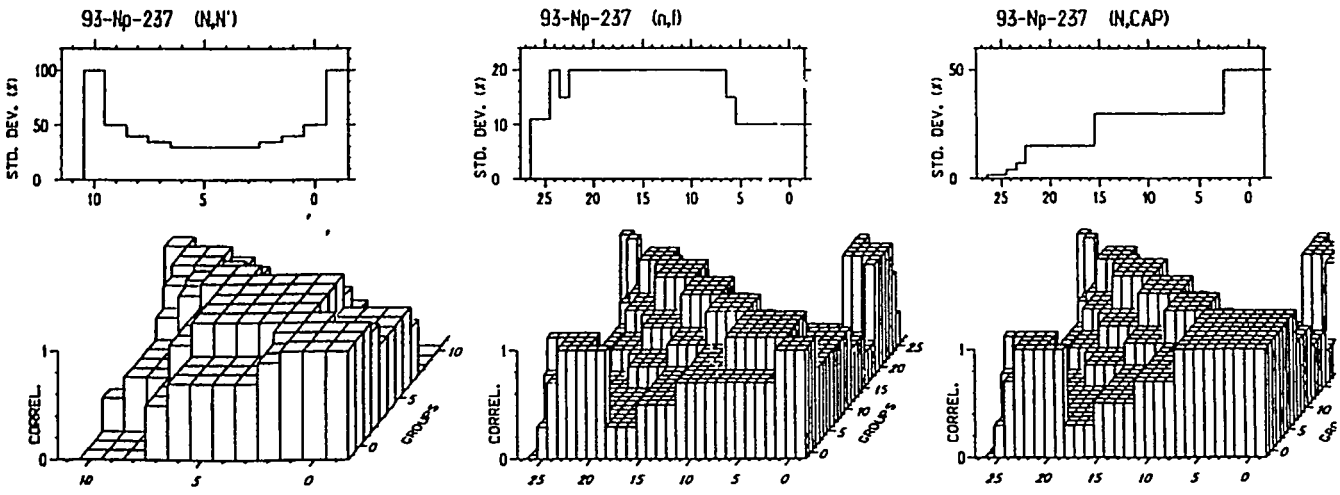


Figure 8. LUND covariance matrices for Np-237.

A. INTEGRAL EXPERIMENTS

- Neutron transmission through large thicknesses (resonance self-shielding testing);
- Sphere transmission of fission neutrons (measurement of escape cross section under fission threshold of U-238);
- Average cross section measurements in fission neutron spectra of U-235 and Cf-252.

B. MACROSCOPIC EXPERIMENTS

- Criticality of uranium and plutonium spheres with and without reflectors;
- Fast critical uranium and plutonium assemblies;
- Assemblies with central $k_{\infty} = 1$ regions of uranium, thorium, and structural material compositions;
- Measurements of fission product and actinide central reactivity coefficients in assemblies with different spectra;
- Measurements of actinide capture cross sections in BN-350 reactor using small sample transmutation technique;
- Criticality of aqueous solutions of uranium and plutonium;
- Criticality of H_2O-moderated uranium lattices.

Figure 9. Experiments included in LEMEX library.

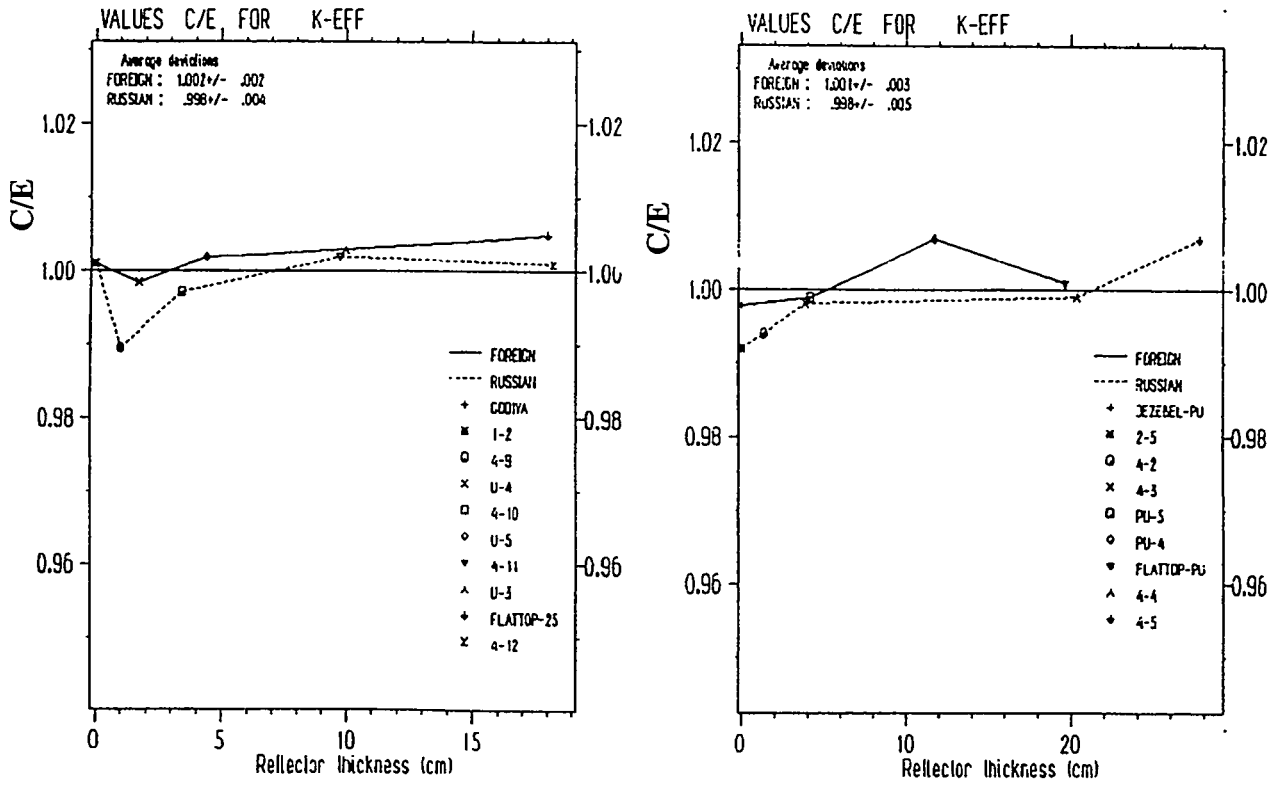


Figure 10a. C/E values for k_{eff} of uranium (left) and plutonium (right) assemblies. The dotted line (Russian) represents ABBN-90 results.

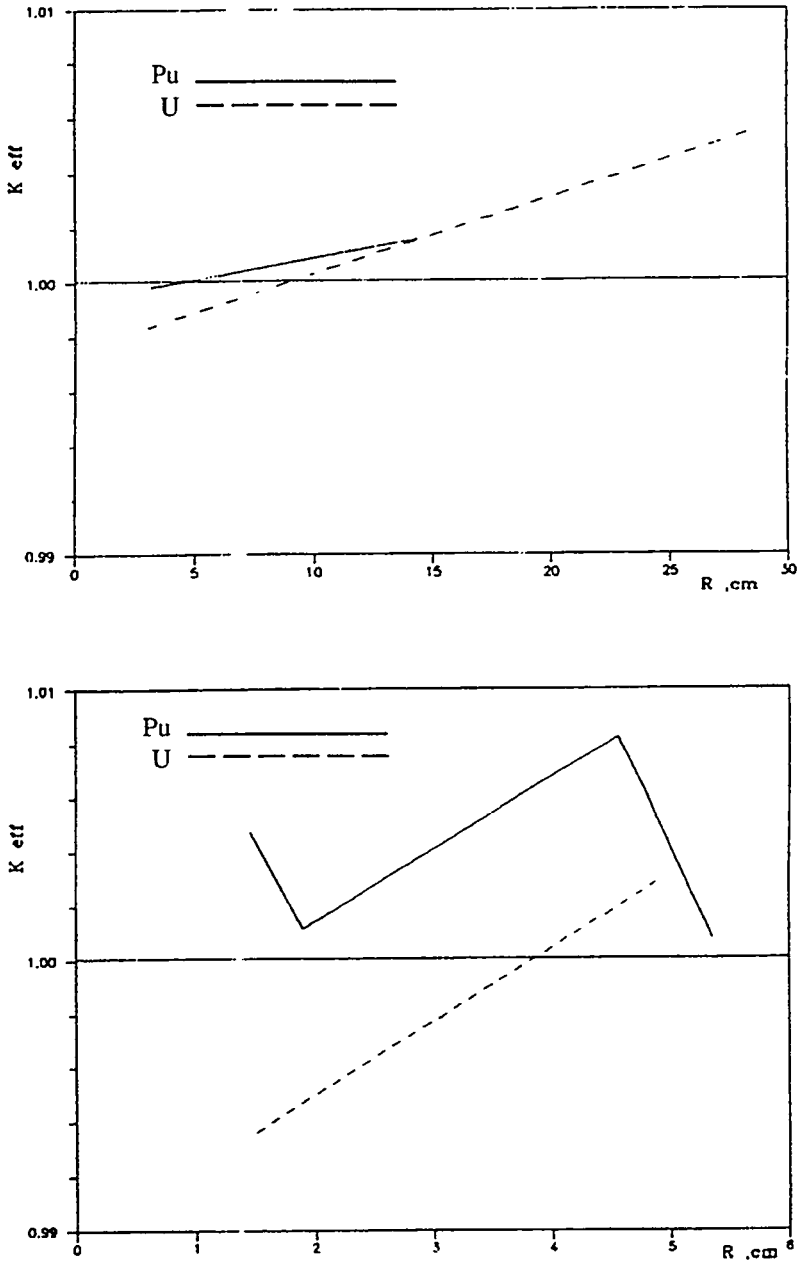


Figure 10b. C/E values for k_{eff} of uranium and plutonium spheres with Pb (top) and Cu (bottom) reflectors.

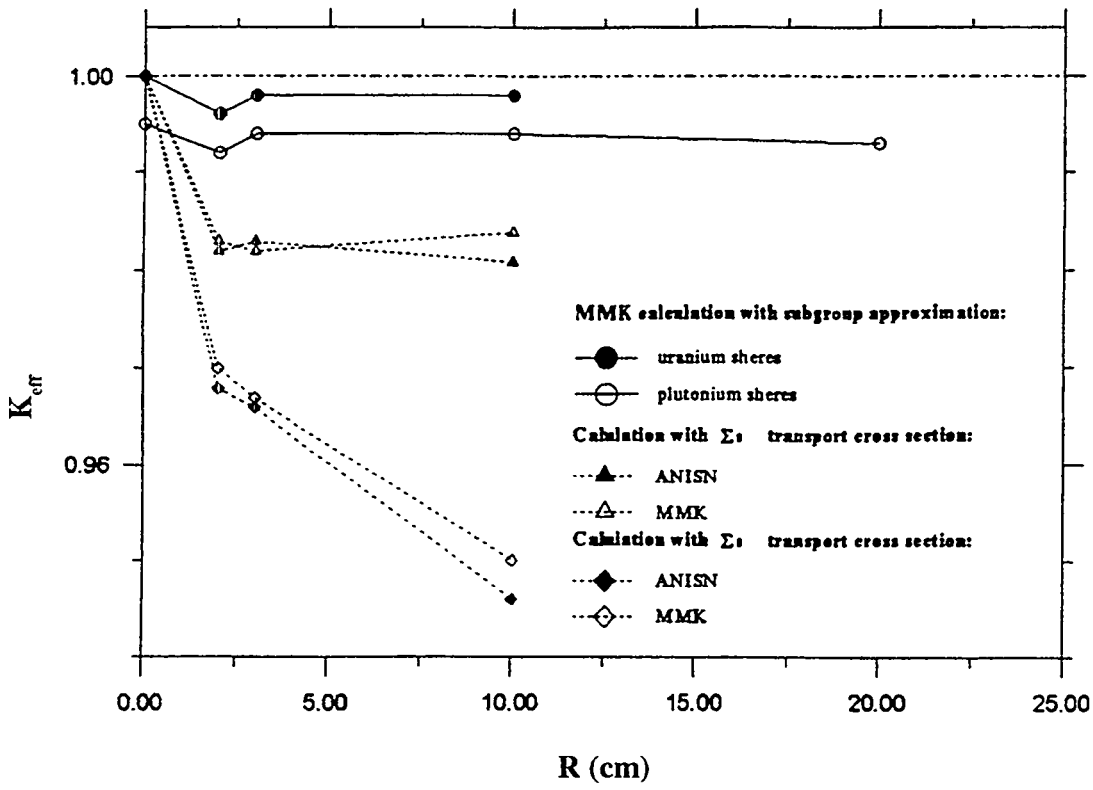


Figure 10c. K_{eff} C/E values of uranium and plutonium spheres with Fe reflectors.

K_∞^+ of BFS and ERMINE Assemblies

Assembly	Absorption per one neutron of source					$\int_{0\,Kev}^{10\,Kev} \varphi(E)dE$	K_∞^+
	^{238}U	^{235}U	^{239}Pu	^{240}Pu	Constr	%	C/E
SCHERZO-5.56	66	34	—	—	—	<1	1.003
BFS-35	66	35	—	—	<1	<1	0.993
BFS-33	58	32	—	—	2	9	0.998
BFS-38	70	2	25	<1	1	1	1.002
BFS-41	64	2	32	<1	2	10	1.000
BFS-42	60	2	34	1	2	15	1.001
OP-10 (ERMINE)	64	3	31	1	2	11	1.008
OP-11 (ERMINE)	61	2	30	1	2	5	1.020
OP-40 (ERMINE)	67	3	20	16	3	11	1.003
OP-41 (ERMINE)	64	2	19	7	2	5	1.010
OP-50 (ERMINE)	64	3	26	7	3	11	1.004

Fig. 11a. Verification of neutron data using BFS and ERMINE assemblies.

K_∞^+ of KBR Assemblies

Assembly	Absorption per one neutron of source							$\int_{0\,Kev}^{10\,Kev} \varphi(E)dE$	K_∞^+
	^{238}U	^{235}U	Fe	Cr	Ni	Mn	^{232}Th(Mo)	%	C/E
KBR-7	2	56	2	<1	41	<1	—	14	0.996
KBR-9	1	56	17	7	6	5	—	17	1.008
KBR-10	1	55	14	6	6	3	(11)	14	0.994
KBR-11	32	50	7	3	3	1	—	12	.0995
KBR-12	33	52	10	6	<1	1	—	11	.0994
KBR-13	26	51	3	14	1	<1	—	9	0.985*(1.01)
KBR-15	1	55	4	31	1	1	—	10	1.030*(1.12)
KBR-18	2	46	2	—	—	—	51	2	1.020
KBR-19	2	49	2	—	—	—	49	10	1.017
KBR-20	2	53	2	—	—	—	45	26	.990

* $f_c(\sigma_\theta)$ – from ABBN-78

Fig. 11b. Verification of neutron data using KBR assemblies.

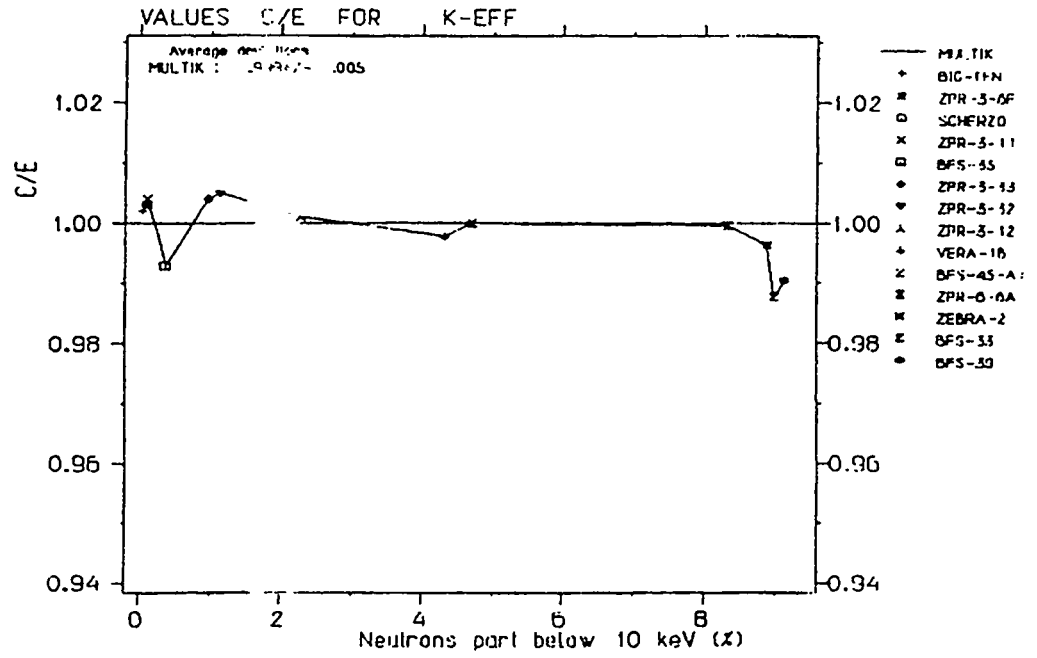


Figure 12a. C/E values for k_{eff} of uranium assemblies.

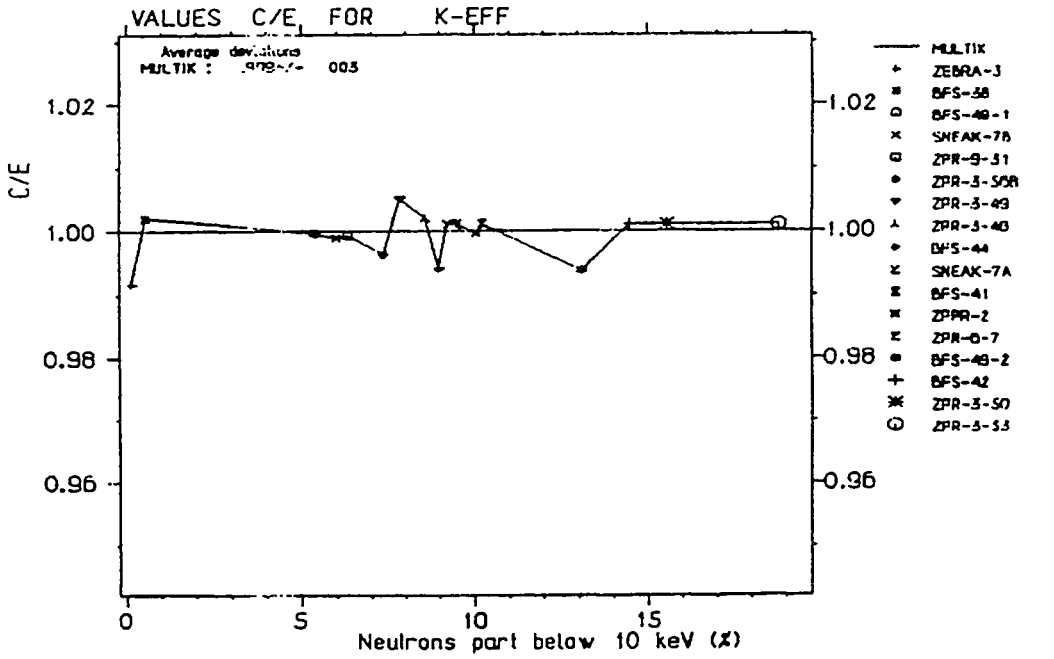


Figure 12b. C/E values for k_{eff} of plutonium assemblies.

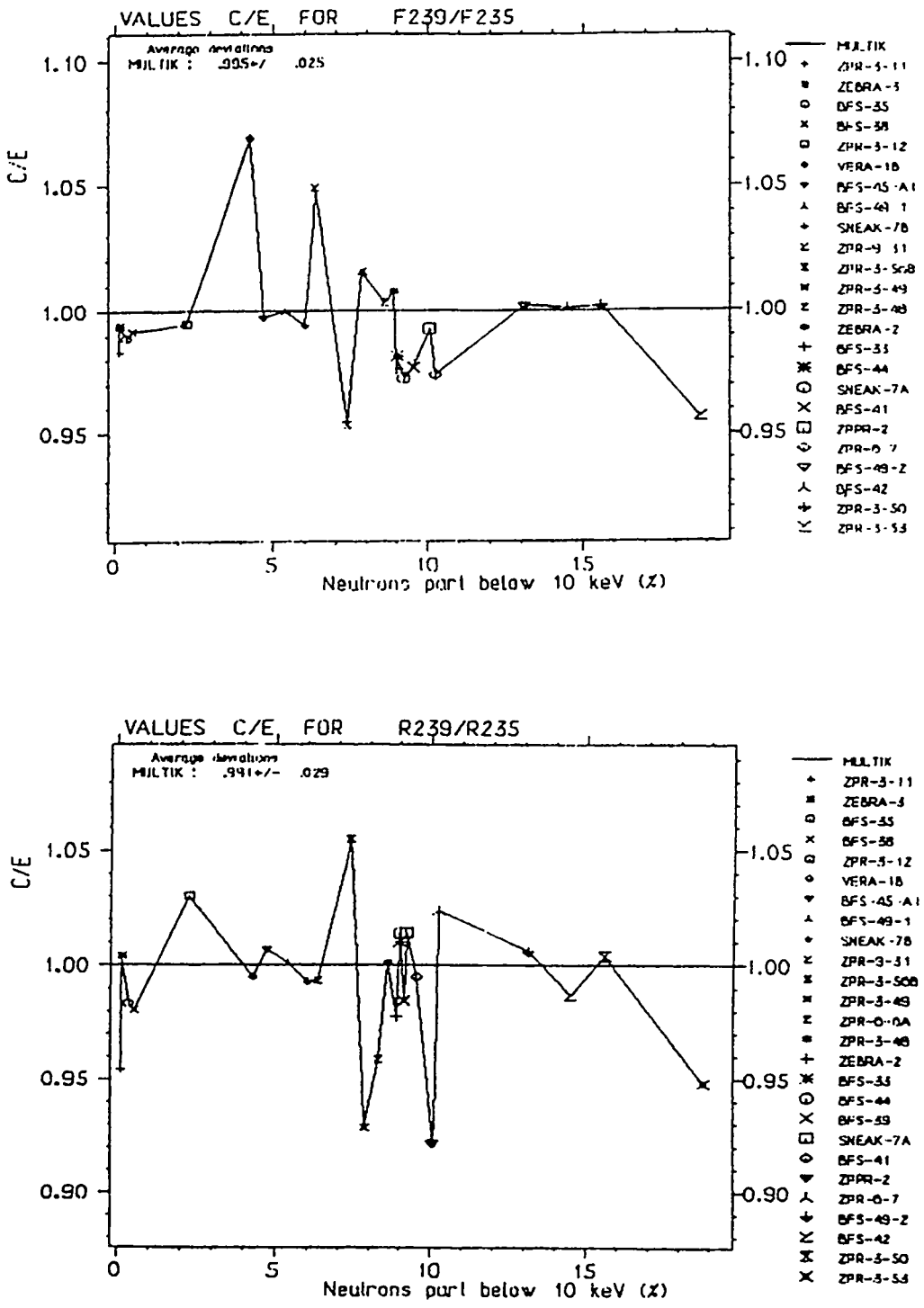


Figure 13a. C/E values for F239/F235 (above) and R239/R235 (below).

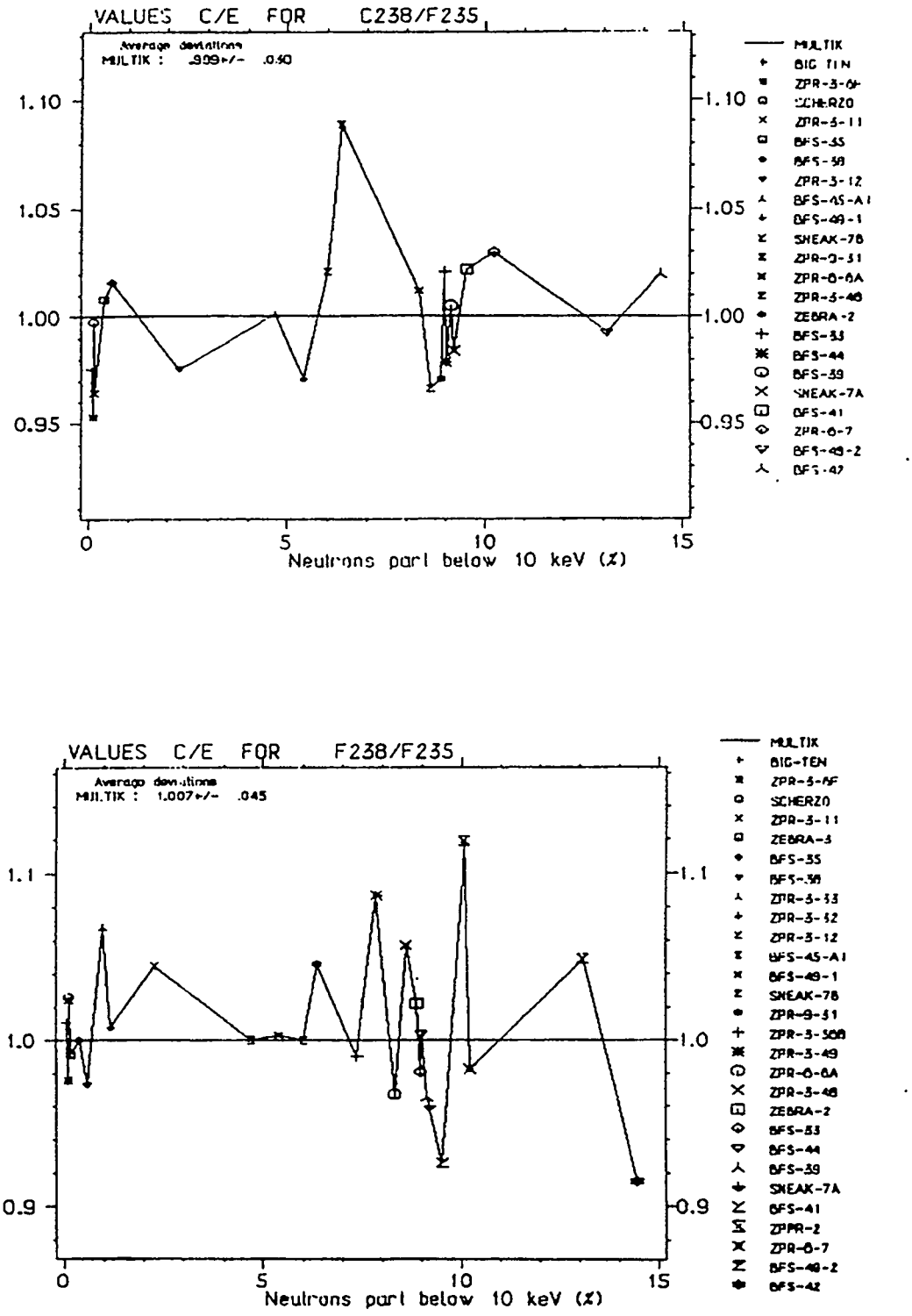


Figure 13b. C/E values for C238/F235 (above) and F238/F235 (below).

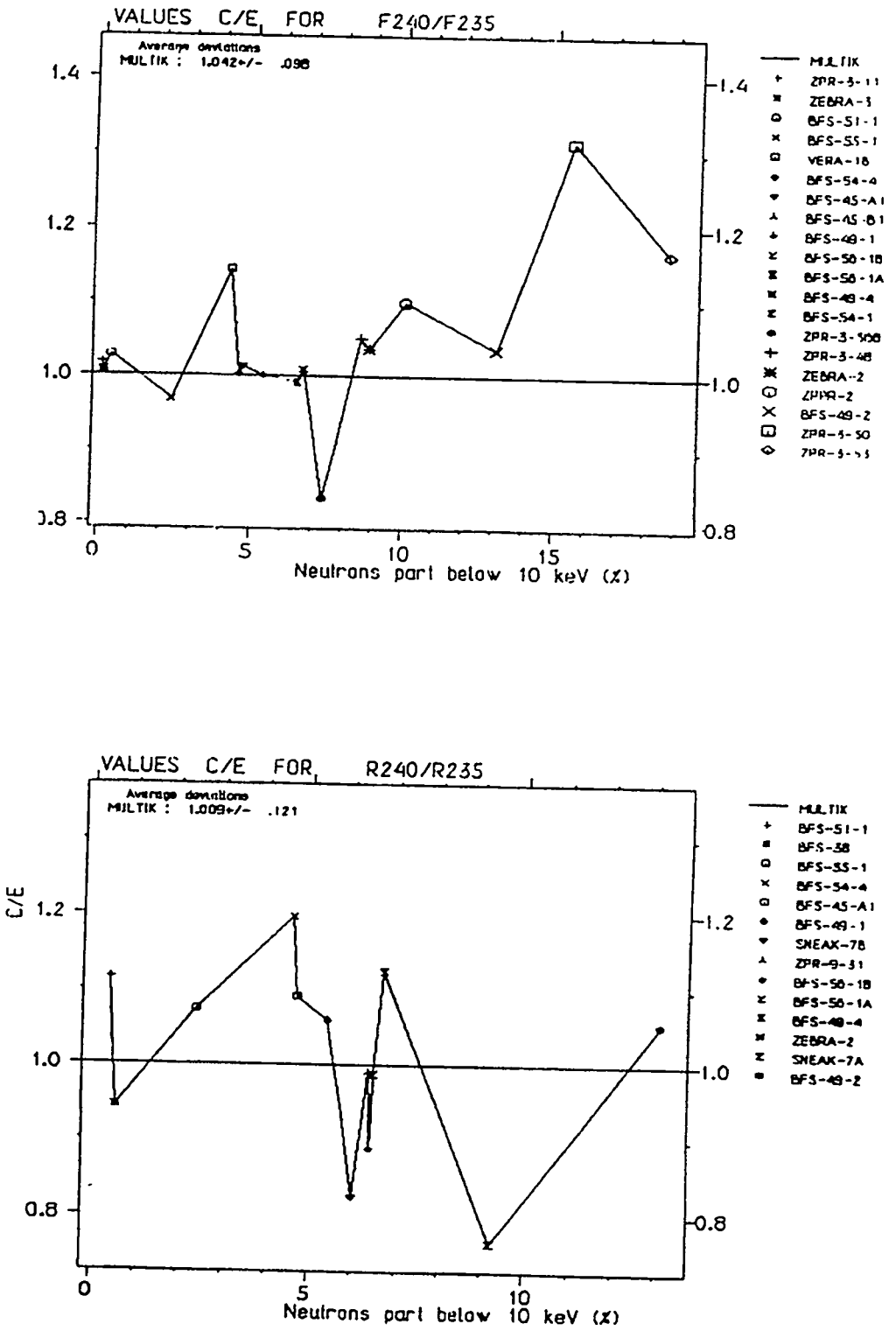


Figure 13c. C/E values for F240/F235 (above) and R240/R235 (below).

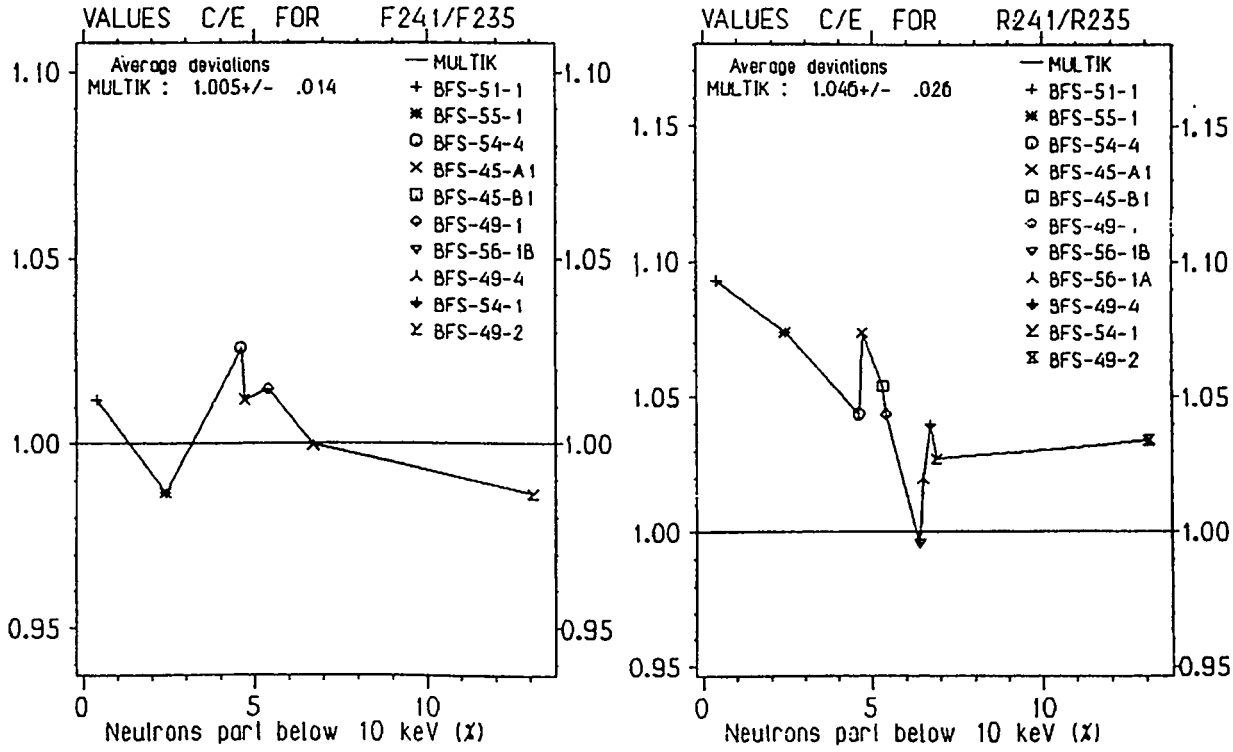


Figure 13d. C/E values for F241/F235 (left) and R241/R235 (right).

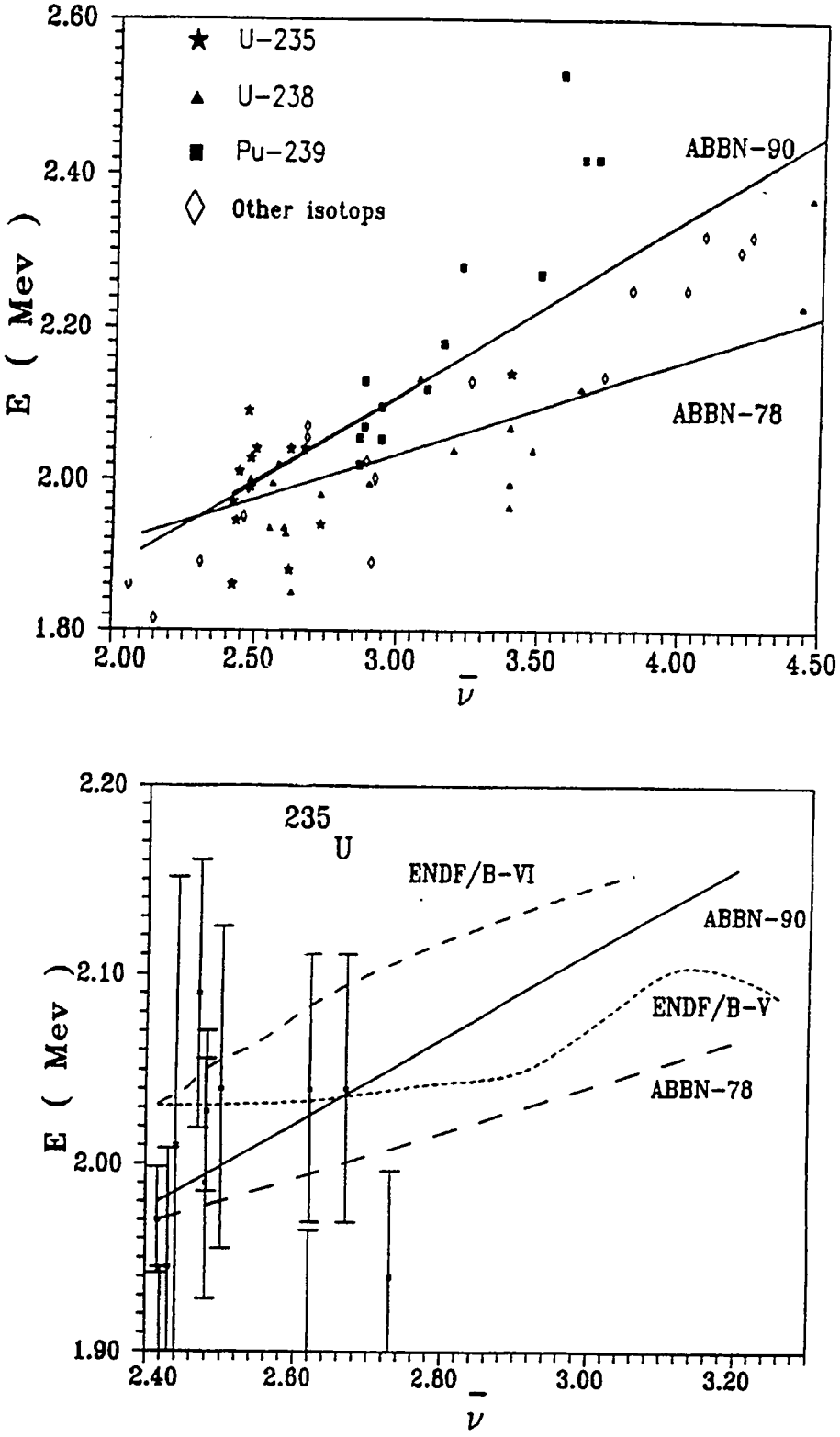


Figure 14a. Energy dependence of average neutron per fission.

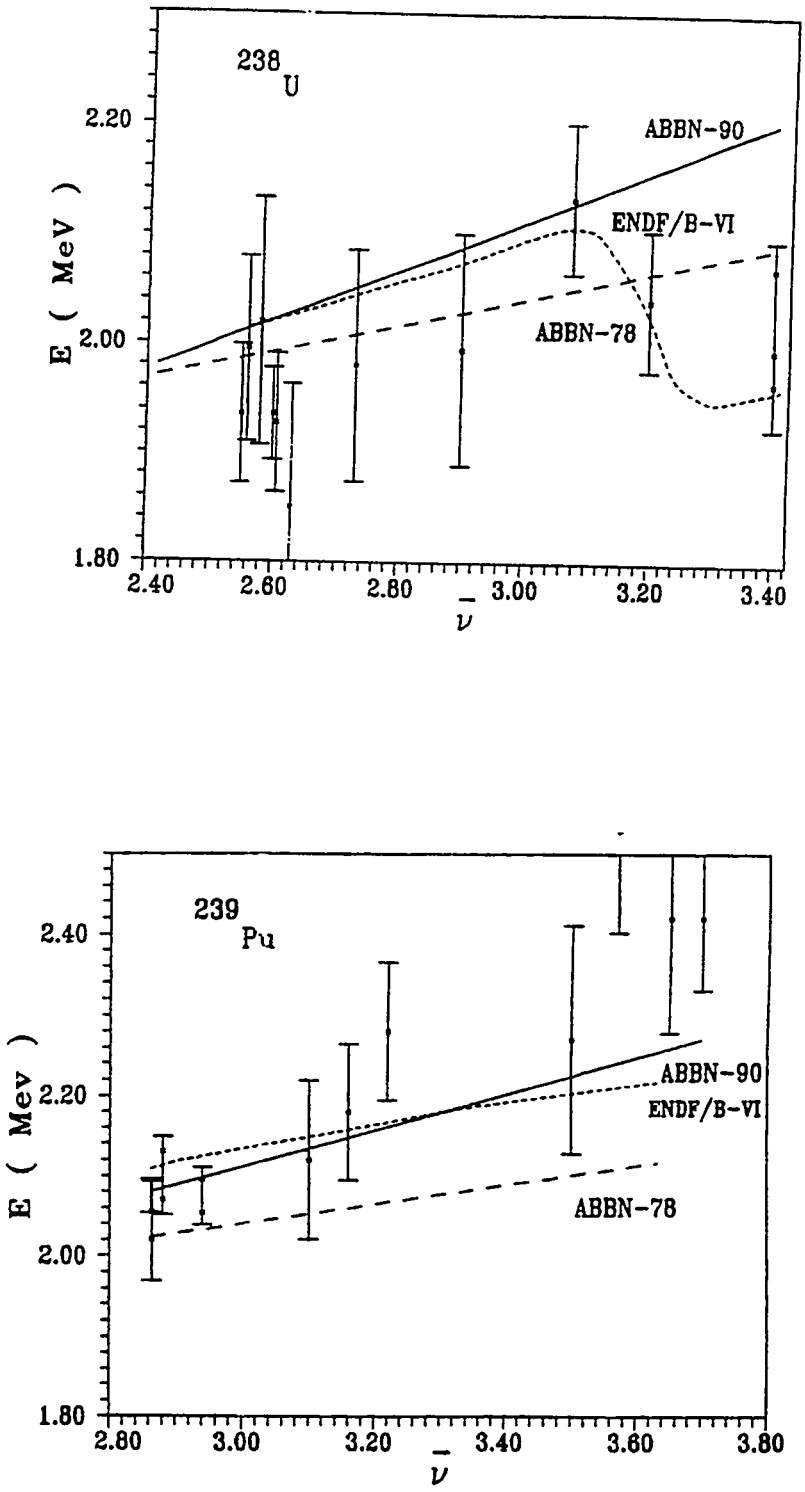


Figure 14b. Energy dependence of average neutron per fission.

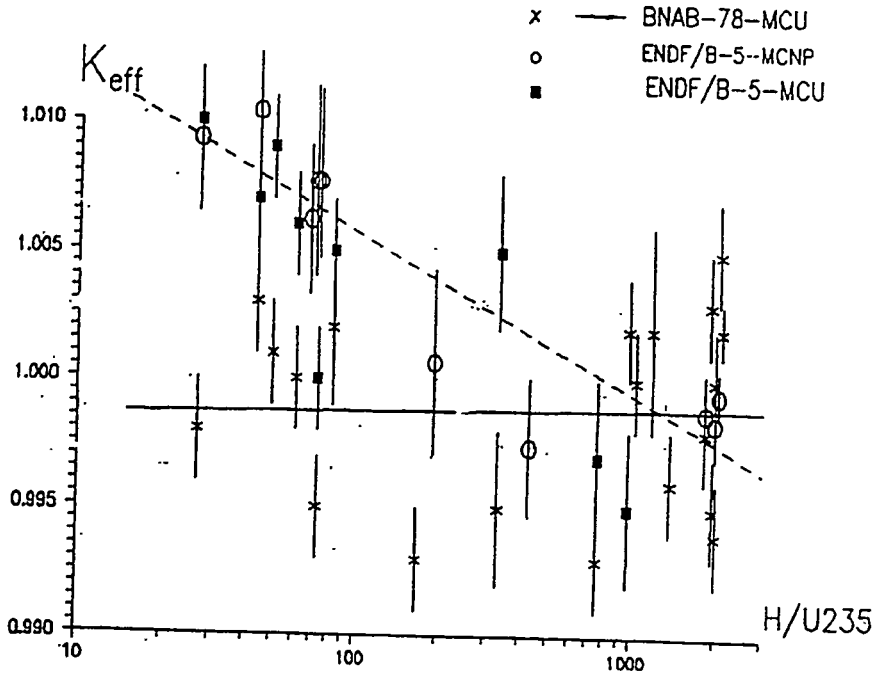


Figure 15. Influence of hydrogen cross section on criticality of enriched uranium solutions.

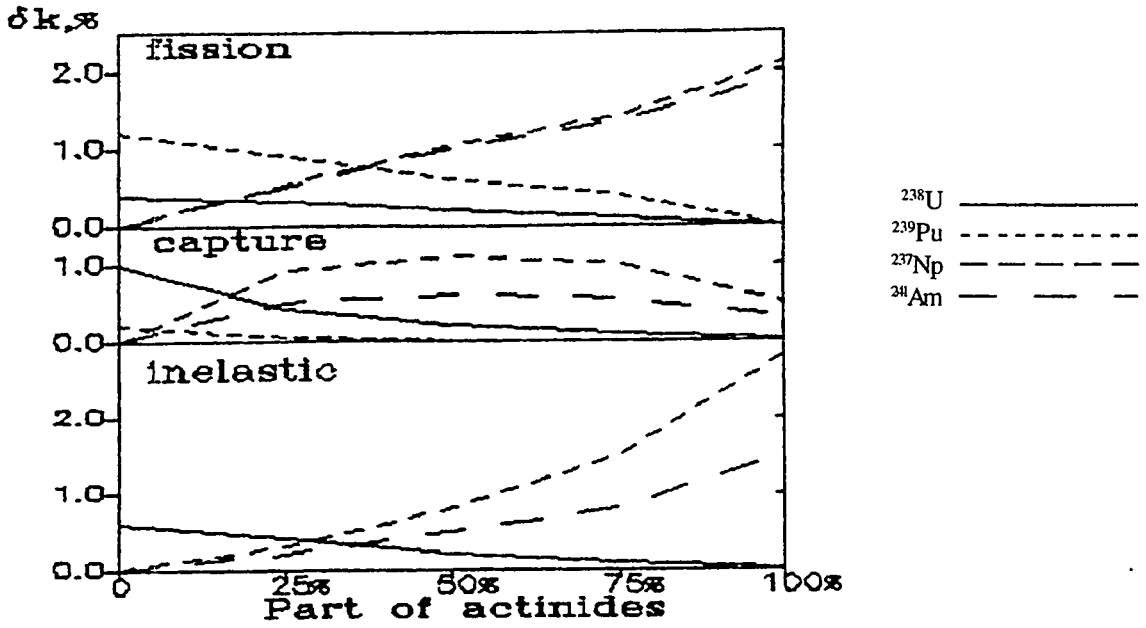
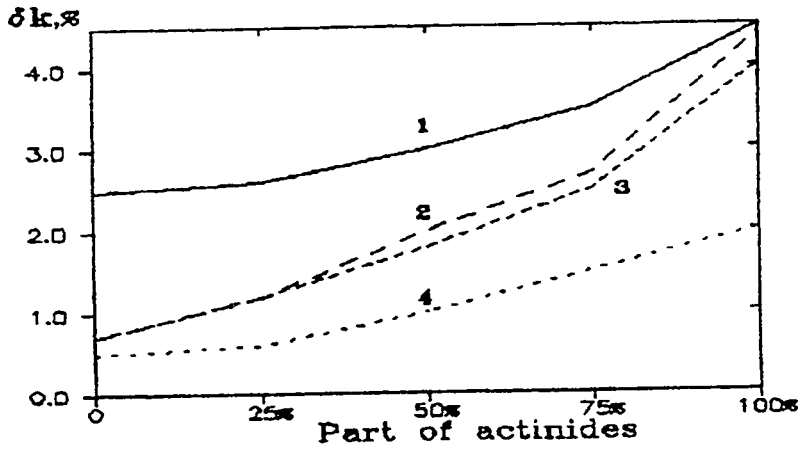


Figure 16a. Main sources of k_{eff} uncertainty of MA actinides burner.



1 – not taking into account the result of macroexperiments
2 – taking into account of macroexperiments
3 – taking into account these results and additional BN-350 and BFS experiments
4 – required accuracy (target accuracy)

Figure 16b. K_{eff} calculational accuracy of the BN-800 type core as MA actinides burner

CALCULATION OF K_∞ FOR HOMOGENEOUS ^{235}U METAL MIXTURES: WILL THE REAL K_∞ PLEASE STAND UP?

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The recent article in the winter 1993 issue of the *Criticality Safety Quarterly* entitled "k_∞ for Certain Metals Mixed with ^{235}U" raised some important questions about the adequacy of calculational methods and the nuclear data that they use. The emphasis placed in the article was on the differences between calculations using groupwise cross sections and continuous-energy cross sections, with the suggestion that the continuous-energy cross sections gave more accurate results. It has been demonstrated through years of validation experience, calculational intercomparison, and general use that properly prepared multigroup cross sections provide excellent agreement with critical experimental data and results using continuous-energy cross-section data. The ^{235}U/metal mixtures described in the reference article are dry, fast systems with several unusual characteristics that, to our knowledge, have never been investigated via critical experiments. In the reference article, the majority of multigroup libraries used in the analysis were developed for well-moderated thermal systems. Thus, the trap was easily set for assuming the continuous-energy results were "correct." The problem is more fundamental than "do group cross sections compare with continuous-energy cross sections?" The fundamental problem is "in the absence of relevant critical experiments for validation of the code and cross sections, what is the real system multiplication factor – i.e., how does one establish a calculational bias?" The answer is the bias cannot be established without relevant critical experiments. (The degree of relevancy or range of applicability is also an issue.) The ^{235}U/metal systems addressed in the article are a severe test of both group cross-section and point cross-section methodology, as will be discussed in the presentation.

Table 1 presents the results of the reference article. The range of calculated k_∞ values for these systems significantly exceeds the margin of subcriticality which is usually applied to calculations. The sources of discrepancy between the SCALE 218- and 27-group ENDF/B-IV results and the MCNP results have been determined. In studying these systems and the published discrepancies, several deficiencies have been identified in the use of ENDF data to generate group cross sections, and in the methodology used in SCALE to generate "quasi-system-independent" broad-group libraries. Each deficiency has been identified and will be addressed in the presentation. Most of the deficiencies have been addressed in the new SCALE 238-group library based on ENDF/B-V data. The study has also identified that the MCNP cross sections for zirconium are inadequate and that the MCNP default cross section for iron does not appear to represent ENDF/B-V data. The study has also identified that the ENDF/B-V evaluation of the aluminum capture cross section is poor in that the evaluation does not adequately represent the resolved resonance structure. This poor representation results in about a 10% negative bias when compared with a more correct representation of the aluminum resonance structure. The identification of the inadequacy in the ENDF/B-V aluminum capture cross section has led to the identi-

fication of several nuclides that are carried as nonresonance nuclides in ENDF/B-V but have significant resonance structure which, in some cases, has not been adequately represented in the ENDF point representation of the cross sections.

Table 2 presents the results of additional calculations performed using several different codes and cross sections on the ^{235}U/metal mixtures. Based on our study of the nuclear data and the codes that use the data, an estimate of the "correct answer" will be discussed in the presentation, together with proposed reasons for the remaining discrepancies calculated by the various codes and data.

Table 2. Additional K_{eff} values for mixtures of Al, Fe, and Zr with ^{235}U.

Code and cross sections	Al/^{235}U = 2470	Fe/^{235}U = 320	Zr/^{235}U = 103
SCALE 238-group ENDF/B-V	0.9802 (1.1176)[b]	1.1446	1.1070
Vitamin E 174-group ENDF/B-V	1.003	1.081	1.0679
VIM[c] continuous ENDF/B-V	0.9983 ± 0.0006	1.0877 ± 0.0007	1.1068 ± 0.0009
MC^{2}[c] ultra fine ENDFB-V	0.9738	1.0965	1.0972
MONK[d] point energy UKNDL	1.0627 ± 0.0008	1.0762 ± 0.0012	1.1205 ± 0.0010
MCNP continuous:			
ENDF/B-V	0.9983 ± 0.0016	1.0895 ± 0.0017[e]	1.0048 ± 0.0025
Default iron	-	0.9901 ± 0.0021[f]	-
ENDF/B-VI Zr	-	-	1.1034 ± 0.0012[g]
Vitamin B6 199-group ENDF/B-VI	1.0071	1.1085	1.1077

[a]Calculations performed by ORNL-NEAS, except as noted.
[b]ORNL-NEAS aluminum evaluation by R. Q. Wright.
[c]Results provided by R. Blomquist, Argonne National Laboratory.
[d]Results provided by Nigel Smith, Atomic Energy Authority Technology Winfrith.
[e]Iron cross section 26000.50c from file endf5p.
[f]Default iron cross section 26000.55c from file rmccs.
[g]Result provided by Art Forster, Los Alamos National Laboratory.

The conclusion from this work is that the criticality safety community must continue to encourage nuclear data development and the understanding of the physics of neutron interactions in light of new and ever-changing applications.

CRITICALITY SAFETY BENCHMARK EVALUATION PROJECT: RECOVERING THE PAST

E. F. Trumble
Westinghouse Savannah River Company

In these times of shrinking budgets and staffing reductions, efficiency is essential. This is especially true in the criticality realm, which has traditionally been understaffed even in good economic times. One of the most time-consuming and tedious tasks is that of code validation. The Criticality Safety Benchmark Evaluation Working Group (CSBEWG) was established to provide a source of evaluated criticality safety experiments that have been placed in an easy-to-use format.

The process used by the CSBEWG to perform an evaluation consists of a review of the experiment, the verification of all input data, the compilation of this data into a standardized format, the running of "sample" calculations with standard criticality codes, and the documentation of this data. Because of the large number of code options and large number of operating systems under which these codes run, these calculations do not represent a validation of these codes, only a check that the experiment description is sufficient to develop code input. The use of these evaluations should significantly speed up the validation process by freeing the individual sites from having to research experiment reports, make and justify assumptions, and develop their own benchmark descriptions.

The first draft of the evaluated experiments was published as a DOE handbook in September of 1993. By October 1994, a second draft containing over 40 evaluations covering about 500 experiments will be published. The first formal publication is expected by the end of March 1995.

Another goal of the CSBEWG is to make a last search to determine if any experiments that may have been "lost," or which contained discrepancies that kept them from being used, can be resurrected. One set of experiments for which this was successful was the Livermore Plutonium Button Array experiments carried out in the late 1960s. Twenty-eight experiments were performed on a split-bed table using 3-kg cylinders of highly enriched plutonium. Of these experiments, four were taken to delayed critical. Two of these critical experiments contained geometric discrepancies which had historically disallowed their use as benchmark experiments.

Discussions with experimenters, review of the facility logbooks and unpublished reports, and the recovery of drawings of components led to the resolution of these discrepancies to the point that these two experiments have been accepted by the working group as benchmark quality. This is important, as these experiments provide a unique set of data involving arrays of highly enriched plutonium, moderated by a mock high explosive, and reflected by polyethylene (to simulate personnel). These experiments have current interest as the complex moves to handle the increased need for plutonium storage from weapons returns and for Complex 21.

THE IMPACT AND APPLICABILITY OF CRITICAL EXPERIMENT EVALUATIONS

R. Brewer
Los Alamos National Laboratory

Currently an effort is underway to evaluate critical experiments which were performed in the past. This effort is led by the Criticality Safety Benchmark Evaluation Working Group, which is composed of representatives from many of the Department of Energy nuclear facilities.

The evaluation effort is basically intended for use by criticality safety engineers to verify their calculations. However, it can be used for identification of data which may need investigation by other working groups, such as the Cross Section Evaluation Working Group and/or the Experimental Needs Working Group.

The evaluation process involves a significant amount of engineering judgment on the part of the evaluator. Normally, the evaluation begins with an examination of the documentation. The documentation consists of any or all of the following: logbooks, notebooks, reports, journals, and meeting transactions. The completeness and accuracy of the written documentation is observed, (i.e., check to ensure that all of the sources agree in the stated parameters). The documentation is examined to assess the calculational reproducibility. The reproducibility applies to the ability to calculate the critical experiment based upon the information given and the ability to assess the accuracy of corrections which were made.

The accepted benchmark critical experiments are to be used as a standard for verification and validation purposes. The effort as a whole is designed to make available critical experiments which may not be widely available or known. Also, the effort records the various critical experiments as a comprehensive document and locates valuable reference materials.

The most visible aspect of the project is the usability. Criticality engineers can take the document off the shelf and easily find a critical experiment whose specifications match the system which they are analyzing. In the past, the Nuclear Criticality Information Service (NCIS) provided this service, but the NCIS no longer exists. Even when the NCIS was in existence, it only referenced critical experiments which were documented in American Nuclear Society meeting transactions and journals. There are numerous critical experiments which do not appear in either of these references.

An aside to this project is the location and preservation of the older reference materials. Many of the older references stored in the archives at Los Alamos appear to be cataloged under the name of the person who submitted the materials to the archives, and nearly all of the material is classified. In fairness to the personnel at the Los Alamos archives, they have a large amount of material with very little expertise in technical matters or current classification policies.

The impacts of the effort should be self-evident. Critical experiments are documented for future use, and their associated references are preserved and identified. Experiments which provide the foundations for recent or present critical experiments are identified. As stated previously, the evaluations should identify data, methods, or models which need improvement.

In summary, the Criticality Safety Benchmark Project will provide a comprehensive reference document which will be extremely useful to the criticality safety community. The final document could be used as an educational tool for future students of criticality safety and for future critical experimentalists. The project can serve to promote international cooperation in the area of criticality safety. Gaps in our critical experimental database may be filled in by using foreign critical experiment data. The effort can also be seen as preserving documents which have historical significance. In the case of many of the early experiments, such notables as Enrico Fermi and Richard Feynman provided their guidance and expertise.

Session 2: Relevant Experiments for Criticality Safety

PROPOSAL FOR EXPERIMENTS WITH ACTINIDE ELEMENTS

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Background for the Proposed Work

It is well known that actinides with even numbers of neutrons, for example $_{93}Np^{237}$, $_{94}Pu^{238}$, $_{94}Pu^{240}$, $_{95}Am^{241}$, $_{95}Am^{243}$, and $_{96}Cm^{244}$, can most probably be made critical with fast neutrons. Computer calculations and replacement measurement techniques have predicted that critical masses for these elements may be in kilogram quantities. Nonetheless, no direct measurements have been performed to estimate the critical masses for these elements. Similarly, actinides with odd numbers of neutrons, such as $_{94}Pu^{241}$, $_{95}Am^{242m}$, $_{96}Cm^{243}$, and $_{96}Cm^{245}$, among others, can more likely be made critical because they exhibit high fission cross sections at low neutron energies. Analytical studies indicate that when these elements are mixed and reflected with water, their critical masses may be in gram quantities. However, no experiments have been performed to confirm these results.

To address this lack of knowledge, we have completed an analytical study where critical masses for some of these elements were calculated with the Monte Carlo Neutron Photon (MCNP) Transport computer code. For each case, a total of 300,000 source histories was run and continuous-energy cross-section data used. For those actinide elements with even numbers of neutrons, the computer model consisted of a sphere which was assumed to contain any such nuclides in a metal form. This sphere was surrounded by a reflector which was assumed, in different cases, to be beryllium, steel, or water. On the other hand, for those actinides with odd numbers of neutrons, the computer model consisted of a sphere in which any of those odd-neutron nuclides was assumed to be in idealized metal-water mixtures. The sphere was surrounded by a 20-cm-thick water or beryllium reflector. The computer models are shown in Figs. 1 and 2.

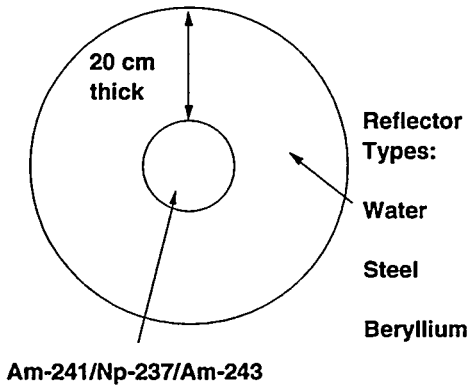

Figure 1. Computer model used for actinide elements with even numbers of neutrons.

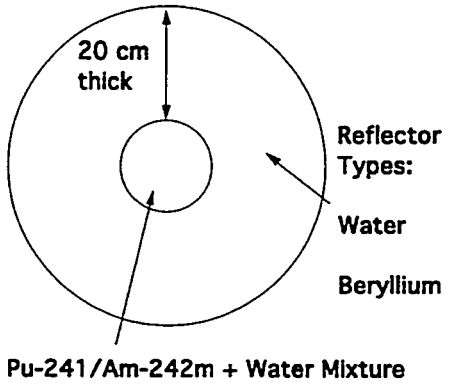

Figure 2. Computer model used for actinide elements with odd numbers of neutrons.

Table 1 shows the critical masses obtained in this analytical study for some of the actinide elements. In addition, critical masses that have been deduced from indirect data from reactivity coefficient measurements are presented in this table. It is important to point out that for those actinide elements with even numbers of neutrons, there are significant uncertainties in the critical masses predicted by computational calculations compared to those estimated by the data from the reactivity coefficient measurements as seen in Table 1 and reported in Ref. 1. No experiments have been performed involving actinide elements with odd numbers of neutrons. Therefore, we strongly believe that an experimental program for actinides should be established so that we can address the inadequacies seen in Table 1 and be able to benchmark our computational calculations against well-characterized experiments.

The present proposal outlines the experiments that will be performed to determine the critical masses of some actinide elements with odd and even numbers of neutrons. In addition, this proposal describes the machines and equipment that will be used during the performance of these experiments.

Table 1. Critical masses for actinide elements.

ODD-NEUTRON NUCLIDES

$_{94}Pu^{241}$

Computational Results					
Metal (kg)			Solution (g)		
Type of Reflector	Total Pu-241 Mass (kg)	k_{eff}	Type of Reflector	Total Pu-241 Mass (g)	k_{eff}
Bare	13.0	1.005 ± 0.0014	Water	270.0	1.000 ± 0.0016
Water	5.7	0.994 ± 0.0016	Beryllium	105.0	0.995 ± 0.0017
Beryllium	3.0	0.999 ± 0.0019	—	—	—

$_{95}Am^{242m}$

Computational Results					
Metal (kg)			Solution (g)		
Type of Reflector	Total Am-242m Mass (kg)	k_{eff}	Type of Reflector	Total Am-242m Mass (g)	k_{eff}
Bare	9.0	0.991 ± 0.0012	Water	20.0	0.999 ± 0.0020
Water	3.25	1.007 ± 0.0020	Beryllium	6.6	0.995 ± 0.0018
Beryllium	1.55	0.997 ± 0.0021	—	—	—

Table 1 (continued)

EVEN-NEUTRON NUCLIDES

$_{93}Np^{237}$

Computational Results			Indirect Experimental Measurements	
Type of Reflector	**Total Np-237 Mass (kg)**	**k_{eff}**	**Type of Reflector**	**Critical Mass (kg)**
Bare	56.5	0.993 ± 0.0013	Bare	88
Steel	33.0	0.990 ± 0.0014	Steel	55
Water	51.1	0.993 ± 0.0013	Water	83
Beryllium	33.0	0.960 ± 0.0013	Beryllium	N/A
Nat. Uranium	33.0	0.997 ± 0.0014	Nat. Uranium	N/A

$_{94}Pu^{242}$

Computational Results			Indirect Experimental Measurements	
Type of Reflector	**Total Pu-242 Mass (kg)**	**k_{eff}**	**Type of Reflector**	**Critical Mass (kg)**
Bare	85.0	0.994 ± 0.0014	Bare	90
Steel	80.0	0.997 ± 0.0014	Steel	84
Water	50.0	1.000 ± 0.0018	Water	56
Beryllium	60.0	0.995 ± 0.0016	Beryllium	N/A

$_{95}Am^{241}$

Computational Results			Indirect Experimental Measurements	
Type of Reflector	**Total Am-241 Mass (kg)**	**k_{eff}**	**Type of Reflector**	**Critical Mass (kg)**
Bare	110.0	0.999 ± 0.0015	Bare	58
Steel	62.0	0.995 ± 0.0018	Steel	34
Water	95.0	0.993 ± 0.0017	Water	51
Beryllium	80.0	0.995 ± 0.0016	Beryllium	N/A

$_{95}Am^{243}$

Computational Results			Indirect Experimental Measurements	
Type of Reflector	**Total Am-243 Mass (kg)**	**k_{eff}**	**Type of Reflector**	**Critical Mass (kg)**
Bare	150.0	0.993 ± 0.0014	Bare	N/A
Steel	95.0	0.996 ± 0.0017	Steel	N/A
Water	140.0	0.997 ± 0.0019	Water	N/A
Beryllium	110.0	0.995 ± 0.0019	Beryllium	N/A

Scientific and Technical Impact

We expect to verify experimentally, by direct or indirect means, the critical masses of such actinide elements as $_{93}Np^{237}$, $_{95}Am^{241}$, $_{95}Am^{243}$, $_{95}Am^{242m}$, and $_{96}Cm^{243}$, among others. The results of these experiments will benchmark the computer calculations presented above, as well as confirm the actinide element mass limits reported in the standard ANSI/ANS-8.15-1981, "Nuclear Criticality Control of Special Actinide Elements."

R&D Approach

To determine the critical masses of these elements, two major options are available. If enough material is available, the first option is to perform the experiments on the Comet or Planet universal assembly machines at the Los Alamos Critical Experiments Facility. These machines consist of an upper platform with a square/circular hole in the middle. An upper reflector, such as beryllium or steel, will be stacked on the upper platform. Beneath this platform, carried on a hydraulic cylinder, there is a platen on which the fissile material is placed.

In this experiment, the fuel will consist of two hollow nesting spheres of a given actinide element, to allow for fuel additions, and a small Pu-Be neutron source. The fuel and bottom reflector will be placed on the platen (see Fig. 3). The platen will then be raised into the upper reflector by remote control to complete the final assembly. BF_3 neutron detectors will monitor

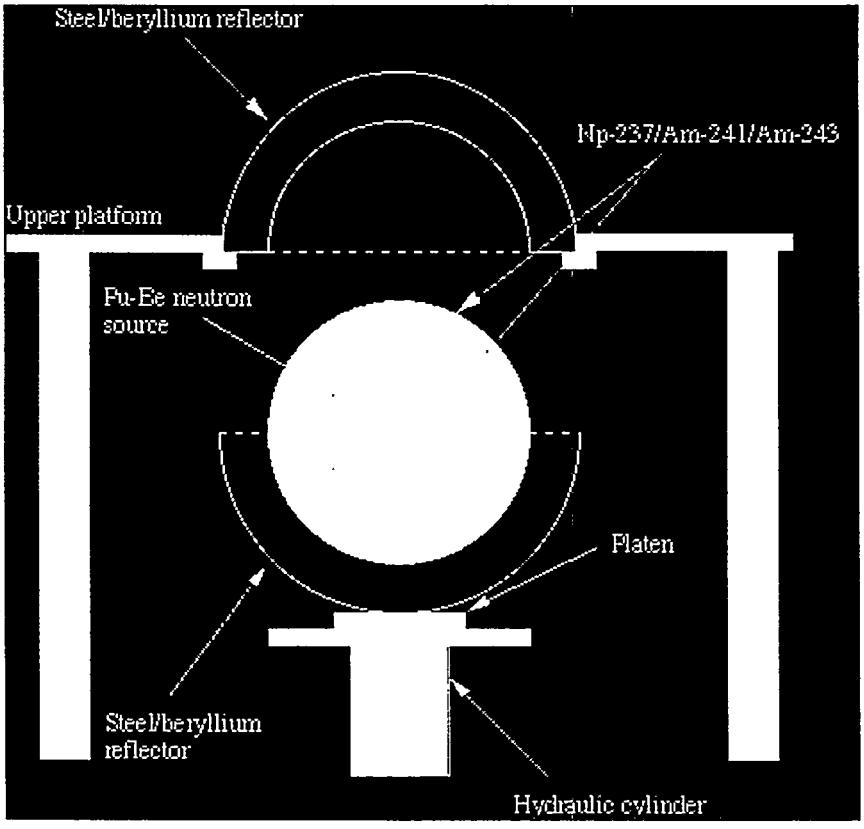

Figure 3. Experimental set-up at the Comet/Planet assembly machine.

the neutron population of the assembly. The approach to critical will be followed by a plot of the reciprocal multiplication as a function of fuel additions, where the linear extrapolation of the last two points will yield the estimated critical mass.

If not enough material is available, the second option is to perform small sample replacement measurements in one of our existing critical assemblies, such as Godiva. The experimental technique consists of operating a critical assembly in the delayed-critical condition with a void in the center. The void is then filled with an actinide element or plutonium sample which will disturb the equilibrium between neutron production and neutron loss. This perturbation will be then restored to equilibrium by means of a calibrated control rod. It is well known that when a critical assembly operates at high enough power (50 watts) levels, the Δk_{eff} ratio between plutonium and the actinide element is proportional to the ratio of their microscopic production cross sections as seen in Eq. 1:

$$\frac{\Delta k(\text{Pu})}{\Delta k(\text{Actinide Element})} \cong \frac{\sigma_p(\text{Pu})}{\sigma_p(\text{Actinide Element})} \tag{1}$$

We can then solve for σ_p(Actinide Element) and compute the macroscopic production cross section Σ_p(actinide element). The critical mass of a bare metal sphere of an actinide element may be estimated by means of the one-group, extrapolated-end-point theory equation

$$(r_c + \frac{0.71}{\Sigma_p + \Sigma_{tr}})\Sigma_{tr} = \tan\left(\frac{r_c(\Sigma_p + \Sigma_{tr}) + 0.71}{\pi}\right)$$

where r_c is the critical radius and Σ_{tr} is the macroscopic transport cross section.[2]

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PLUTONIUM SOLUTION IN CONCENTRATION RANGE FROM 8 TO 17 G/LITER

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Critical experiments are a national necessity to ensure safety in the application of computational methods to nuclear criticality safety throughout the nuclear community. This is so despite the fact that they are expensive. The nation has had only a few facilities capable of performing these experiments and only a few decades in which to collect the data. Naturally, the first experiments selected – when faced with the broad spectrum of much-needed studies – would be those which are arbitrarily judged "most important" and those which happen to be easy to perform. These criteria prompted the criticality safety industry to study first many plutonium and uranium metal configurations as well as systems of fissile liquids having the same elements. High-concentration solutions near the minimum-critical-volume concentration were chosen because they are the most reactive and constitute a criticality safety "envelope." All these programs were urgent in a fledgling industry and relatively easy to do.

The importance of these fundamental studies in no way diminishes the continued need, in the 1990s, for an extensive set of still-significant experiments. These experiments have been carefully collected in recent months, and a detailed list of relevant experiments still needed for criticality safety is now published.[1] One particular program described in Ref. 1 is the subject of this paper. It is labeled as "Experiment 301," and this paper attempts to present the reasons why this study is felt to be important.

If all these much-needed criticality experiments are costly, plutonium studies are especially so because of additional concerns introduced by that nuclear material. Very few experiments involving low concentrations of plutonium solutions have ever been done in the past 40 years for the reasons given above. Yet, these experiments have increased in importance in recent years with the industry's shift in emphasis.

Low-concentration solutions are produced by the act of rinsing out a tank which once had contained high-concentration plutonium solution. The latter is characteristic of an operating production plant. The former approximates conditions found as a tank may have been taken out of service or is in the process of being taken out of service. One important safety point is that criticality is almost just as credible – and certainly just as undesirable – in a tank of low concentration plutonium solution as in a tank of rich feed. Only the detailed parameters of the criticality accident might be slightly different.

Low concentrations would also be encountered in acts of washing contaminated pieces of equipment, metal sheets, plastic, and rubber components. Unknown pockets of plutonium compounds, for example, might be broken loose from hidden recesses of, say, an unused lathe and be quickly dispersed as a low-concentration plutonium solution. These same pockets of compounds

could be unleashed unintentionally if a fire sprinkler system were activated, flooding the equipment and generating the same low-concentration plutonium solution.

All these scenarios are characteristic of "waste." They are also relevant because of the current trend toward the decontamination and decommissioning of many of the nation's nuclear facilities. High- and intermediate-concentration solutions are not likely to be encountered in the coming decade. The nuclear industry seems to be moving into an era of shutting down and deactivating its nuclear facilities; and this suggests low-concentration solutions.

Another historical argument also supports the need for this proposed study, as well as a great many other experiments – many of which the naive observer might think completed. Critical experiments performed in the early decades were generally well done and adequate for the needs of the day. They provided the data used to build the nation's nuclear capability. They were empirical data points which addressed specific needs. They were never designed to provide the rigorous level of detail required in today's precise computer applications. Material compositions were not described in sufficient detail. Geometries were often given as nominal values of a commercial item rather than an actual measurement of the component described. Finally, all descriptions usually ended in the published reports with inadequate descriptions of the room and the environment in which the experiment was performed. The consequence of this point is that even some of the experiments thought to be documented for certain computer validation applications may really need to be performed again.

One specific aspect of the above argument concerns the relative importance between the elements plutonium and hydrogen in describing a fissile solution. Modern laboratory methods can measure the plutonium content of a solution to between + 0.5 and + 1.0%. Unfortunately, equally modern analysis techniques cannot measure the hydrogen content of a solution with anywhere near the same accuracy. Some claim an uncertainty of + 5% in that parameter. A study of the relative importance of these two elements to the reactivity of a fissile solution reveals a surprising result. The hydrogen content is three times more important in calculating the neutron reproduction factor of a solution system than the plutonium concentration. Thus, the uncertainty in the hydrogen content of an experimental liquid is 10 to 15 times more important than the uncertainty in the plutonium.

The number of critical experiments in the area of low plutonium concentrations is pitifully small. This paucity is worsened by two factors. There is a wide gap in concentration between the few experiments performed in the upper part of this range of low concentrations and the one experimental program performed at less than 10 g/L.[2] This limiting critical concentration is too important a parameter to have the entire nation rely on just one (albeit well-done) experiment without any corroborating experimental evidence. This criticism is especially valid in light of the second factor. The second problem is that the criticality parameters one would choose to evaluate as a function of concentration vary greatly, non-linearly, and non-uniformly in this range. That is, the curves exhibit a great deal of structure. The critical mass of a plutonium-solution sphere decreases about a factor of 10 between the study of Ref. 2 and, say, 20 g/L! The minimum

critical concentration reported in the literature varies between 7.2 and 8 g/L. The actual value of this parameter is of vital importance because concentration controls will be used during decontamination and decommissioning operations to preclude criticality.

One final dilemma in this low plutonium concentration range concerns the actual bias which might exist between calculational methods and the few data points available. The text of Experiment 301 of Ref. 1 says: "Validation of computer codes at 9 g/L and above 17 g/L appear to give contradictory results with a computational bias appearing to become strongly negative below 20 g/L." This statement can be neither denied nor supported by this author; and no source of the quoted position is known. The data of Ref. 2 displayed a strong positive bias, according to very recent results obtained by one specific calculational package. At the same time, an equally recent publication also presents a strong positive bias between the few experimental data points of a different calculational package.[3] This recent comparison would refute the quote; but the calculational packages used to obtain this finding do differ. So, in summary, the origin of the quote in Experiment 301 is not known and no data exists with which to confirm or deny it.

The national standard ANSI/ANS 8.1 requires that the trend in the bias be established when using computational techniques without further data to determine precisely the trend in the bias and its magnitude. Criticality calculations in support of this deactivation mode will not comply with DOE orders to abide by this standard's requirements; and the actual margin of safety will be undetermined.

Further experiments, specifically in the plutonium solution concentration range up to 20 g/L, would greatly improve the nation's confidence in nuclear criticality safety as it moves more and more toward the decontamination and decommissioning of its once-vast number of nuclear facilities.

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ABSORPTION PROPERTIES OF WASTE MATRIX MATERIALS

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Fissile material in waste is frequently encountered in decontamination and decommissioning efforts, found in process sludge and settling tanks, *in situ* vitrification, and during waste remediation efforts (including waste storage, retrieval, characterization, volume reduction, and stabilization). The safety envelope for many of these operations can be effectively defined by characterizing the waste matrices by major "non-poison" components. By neglecting impurities, the inclusion of which almost always results in a decrease in reactivity, conservative representations of waste matrices can be obtained.

Some of the more predominant waste matrix materials of interest are:

Al_2O_3	Cellulose	Graphite	NaCl
CaCl	Concrete	Metals	Polyethylene
CaO	Fe_2O_3	MgO	SiO_2

With the exception of NaCl, CaCl and Fe_2O_3, these materials are among the more reactive materials that are present in waste. The limiting critical fissile concentration in most of these materials is less than the limiting critical concentration in some of the more traditional and well-known materials: water and polyethylene. Calculated limiting critical concentration values for some of these materials are:

Matrix Material	**Limiting Critical Concentration** (g Pu/L matrix material)
Al_2O_3	2.7
CaO	—
Cellulose	2.5
Concrete	3.9
Graphite	0.1
Metals (Al)	6.5
MgO	1.0
SiO_2	1.0

The limiting critical concentrations in water and polyethylene are about 7.5 and 8.3 g Pu/L, respectively.

Using codes and cross-section data that are typically used for criticality safety analyses, large differences (as much as 10%) in calculated k_{eff} values are obtained for systems that contain significant quantities of these materials by simply changing cross-section data sets. In order to demonstrate the safety of waste streams containing large quantities of these materials, experimental results to compare with calculational results are needed to resolve these differences and to establish realistic biases.

These experiments further two objectives set forth in DNFSB Recommendation 93-2:

- The prediction of the critical state of a system by methods that use theory must be benchmarked against good and well-characterized critical experiments, and

- Experiments should be targeted at the major sources of discrepancy between theory and experiments.

ALTERNATE MEASUREMENTS OF BENEFIT TO CRITICALITY ISSUES AT HANFORD

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ABSTRACT

Measurements in a critical mass facility, such as the Los Alamos Critical Experiments Facility, could be performed on simulated nuclear waste materials that would provide important critical mass information and concurrently provide a calibration of alternate measurement techniques. In addition to criticality information, the measurements could also provide a better assessment of diluent material/neutron interaction cross sections.

At Hanford, large quantities of fissionable materials are dispersed in tanks, fuel storage pools, and in solid waste. Although the fissionable materials are well diluted by a variety of neutron-moderating and -absorbing substances, it is difficult to assess the margin of subcriticality. A number of measurement approaches are proposed that will either help determine fissionable material concentrations, distributions, or provide a direct measure of subcriticality. The methods under consideration involve passive neutron counting, active neutron measurements, pulse neutron applications, neutron noise analyses, and cover gas evaluations. Active neutron measurements can also provide insight into the determination of neutron absorber concentrations. Efforts are underway to test some of the methods in actual waste tank environments and geometries. It is important that these methods be tested and calibrated in a critical mass facility.

The information derived from these critical mass measurements would not only be of value to the Hanford Site but throughout the industry where criticality concerns are related to safe handling of waste materials. The "Forecast of Criticality Experiments and Experimental Programs Needed to Support Nuclear Operations in the United States of America: 1994-1999" identifies a series of experiments – 502, 502a through 502i – that describe needed criticality measurements associated with nuclear waste handling.

BACKGROUND

The formal implementation plan for recommendation 93-2 of the Defense Nuclear Facilities Safety Board recognizes that a special-purpose experiments program operating within a general-

purpose criticality facility is an ingredient of an effective criticality safety program. For environmental restoration and waste management activities in which criticality safety issues come into play, the question is raised: what exactly is the relationship between such a facility and those issues? This is a valid question that ought to be carefully addressed if effective use is to be made of a general-purpose criticality facility. Clearly, the answer depends on (a) what precisely is the nature and extent of the criticality issues involved and (b) how exactly does a critical facility contribute to the resolution of these issues? Assuming that answers to those questions have been provided, an adjunct to those questions might well be: are there situations in which a criticality facility can be used in an optimal way so that the issues in question can be resolved in the most effective manner? One situation in which this last question comes into play is the use of the Los Alamos Critical Experiments Facility (LACEF). Use of the LACEF might be optimized if it could be applied in conjunction with alternative measurement approaches at Hanford and other DOE sites in a way that the resolution of criticality issues at those sites would benefit the most.

Large quantities of nuclear waste at different U.S. Department of Energy (DOE) production sites contain fissionable materials. In many instances the concentration and distribution of fissionable material in the waste, as well as the chemical composition, are not well known. The lack of such information makes it difficult to implement nuclear criticality safety controls. Chemical and isotopic analyses of waste are a possible, but a very costly, option. More expeditious approaches to waste nuclear criticality control could be based on *in situ* measurements estimating fissionable material and neutron-absorber concentrations. Such measurements need to be performed with instrumentation that can be calibrated in simulated materials in environments and mockups that lend themselves to computer modeling. To establish margins of subcriticality, parametric measurements are needed in simulant materials that look at neutron multiplication values as a function of fissionable material or nuclear diluent and absorber concentrations. A facility, such as a critical mass laboratory, is required for these measurements.

The *in situ* measurements are of specific importance to Hanford. Two major unreviewed safety questions (USQ) – one in the tank farms and the other in the KE Basin – have been related to nuclear criticality limits compliance issues. Although the tank farm criticality USQ has already been resolved, with applicable calibrated measurement techniques in place, the resolution of these USQs could have been expedited or prevented the USQ declaration.

Over the 42 years of nuclear materials production, significant quantities of plutonium and uranium ended up in waste streams. The 177 waste tanks at Hanford are estimated to contain as much as 1,000 kg of plutonium and significantly more uranium. In addition to the waste tanks, plutonium can be found in cribs, solid waste, spent fuel, and sludges in spent fuel pools. The concentration of the plutonium and the geometric configuration, moderation, and control by neutron absorbers is not well known.

Although the waste tanks were shown to be safe in the current configuration, based on analyses, future processing of tank wastes will perturb existing conditions. Monitoring and surveillance instrumentation need to be in place to demonstrate that criticality margins are not

compromised. Because the tank waste subcriticality depends in part on neutron absorbing diluents, their presence, along with the impact of diluents on critical masses, needs to be known. Measurements of subcritical and critical configurations with substantial amounts of SiO_2, MgO, CaO_2, and nitrogen compounds would be highly desirable. Correlation of the measurements with computer codes would resolve cross-section ambiguities. Alternate measurement techniques, such as the application of pulse neutron systems, could provide very useful insights into diluent and absorber concentrations, in addition to fissionable material concentrations.

MEASUREMENT APPROACHES

There are very limited critical mass data for plutonium and uranium as chemical compounds and as part of a waste matrix. Critical mass measurements are needed for prototypic waste material and a geometry, both of which can be readily replicated with computational tools and cross-section sets. As critical mass measurements are performed, alternate techniques for monitoring subcritical conditions could be calibrated. Eventually, such alternate techniques will provide *in situ* surveillance for compliance to nuclear criticality limits. The experiments should initially consider simple geometries, such as cylindrical tanks filled with homogeneous materials. The tanks would have limited access for measurement devices both on the outside and internal to the tank. Support facilities should have the capability to build up gradually fissionable material concentrations within the tank and add in nuclear diluents and absorbers.

Fissionable materials can be detected as a result of the fission process and transmutations. Characteristic gamma rays, alpha particles, neutrons, and gaseous fission products that are given off by the nuclear processes can serve as detection indicators. If the gamma-ray background radiation is strong, such as in a typical waste tank (10 - 600 rad/h), neutron-based methods should be more viable.

Because of the high gamma-ray background, the concepts considered are mainly neutron based. Most are differential measurements in that they will tend to characterize materials in a narrow cylinder around the observation well or interrogation port. They will, however, provide very useful data on distribution gradients in the axial direction. The concepts considered involve passive neutron counting devices, such as track recorders and activation foils; active systems using neutron sources and detectors; and systems consisting of a pulsed neutron generator and detectors.

In addition to the differential measurements, integral measurements are being considered that quantify certain gaseous fission products being generated in the waste tanks. Gases such as helium, krypton, and xenon would be characteristic of total transuranic or fissionable material content. Certain other gases, such as radon, could provide information on uranium contents.

Fission track recorders consisting of highly enriched uranium foils imbedded between sheets of mica would record fission product tracks following neutron-induced fissions in the foils. The

tracks can be counted and related to a neutron density. Such methods were successfully used in the Three Mile Island core recovery efforts. The advantages of fission track recorders are that they require no special electronics, can be readily deployed, and can remain in place over extended time periods. The tracks in the mica become a permanent record. The disadvantages are that neutrons are detected that arise either from (α,n) reactions or fissions and that special computerized optical equipment is needed to read the mica sheets.

Metallic foils such as copper, indium, zinc, and magnesium are very similar to fission track recorders except they may have short saturation times and, again, require some highly specialized equipment for counting. With the right combination of foils, it may be possible to discriminate neutrons coming from fissions versus (α,n) reactions.

Use of neutron counters such as BF_3 or ^3He can be employed for passive neutron detection. The advantage of the detectors is they can provide instantaneous readout and can be readily repositioned to obtain scans of neutron distributions. With filters on the tubes, some discrimination as to neutron origin is possible.

Active neutron sources such as ^{252}Cf or pulse neutron generators can be employed to measure low concentrations of fissionable materials and also provide some assessments of the subcriticality of test configuration. ^{252}Cf-source-driven neutron noise analyses techniques have been employed by J. T. Mihalczo to measure neutron multiplication factors as low as 0.3. Active methods can resolve the origin of neutrons and even provide insights into the concentrations of other materials, such as moderators or neutron-absorbing diluents in the matrix. Disadvantages of such systems are the cost, the size of equipment, and the fact that measurements cover localized regions.

Another measurement concept that could provide useful information is gas analyses for helium and fission products. Alpha-particle decay in nuclear waste produces helium gas, while spontaneous and induced fissions create noble gas fission products. The gases produced by fissionable material in a matrix will tend to migrate through the material and collect in air spaces. Sampling of the air for these special constituents will tell if the system is generating gases in excess of natural background. An excess can be correlated to a fissionable material content. A gas measurement would represent an integral fissionable material content of the system. High-precision mass spectrometers, as well as radioactive material detectors, are needed for measurements. Some disadvantages of this method are the requirement for very sophisticated gas analyses and control over the airflow in the air space.

Neutron activation and detection of characteristic gamma rays is a viable measurement alternative, provided a background gamma-ray field does not exist. In the Hanford Site waste tanks, dose rates of hundreds to 1,000 rad/h are encountered, therefore, gamma-ray-based techniques may be less successful.

EXPERIMENTAL MEASUREMENTS

The objective of alternate measurement techniques is to detect very low concentrations and distributions of fissionable materials in nuclear waste. A typical range would be 0.05 g/L to 3 g/L. The range is below subcritical measurement capabilities. However, if the concentration in tank waste at these low levels can be determined, an assessment of the total fissionable inventory in the tank can be made. Current plans call for intercomparison of measurement techniques in the tanks supplemented with some criticality computations and chemical sample analyses. What is needed is a well-characterized system in a controlled environment of a critical mass facility such as the LACEF. An experiment in a tank containing a simulated waste matrix is envisioned where the concentration of fissionable material is gradually built up. At each concentration change, measurements with passive and active neutron systems are made. As the system neutron multiplication increases, gas analysis would be attempted. Noise analysis (pulse neutron) would track the approach to critical and provide a much needed intercomparison between methods. Perturbations on the basic experiment could include diluent material changes and subsequent approach to critical. In such a measurement, multiple important objectives could be met to help provide much needed benchmark and calibration data to nuclear waste criticality problems.

More complicated systems are envisioned for subsequent measurements that would require mockups of distribution gradients, fissile material layering, positive reactivity insertions due to mixing, and accounting for localized concentrations of fissile materials. The experimental data, calibration of methods, and calculations based on such information should help in demonstrating compliance with DOE Order 5480.24, especially in the area of risk determination.

CONCLUSIONS

It is important that a nuclear criticality laboratory be available to provide measurements for nuclear waste, a multicomponent chemical system about which little is known. Simultaneous with the critical mass measurement during the approach to critical, a number of *in situ* measurement techniques can be calibrated and intercompared. Perturbations on the waste composition could also furnish insights into cross-section uncertainties of diluents in waste materials and their impact on the neutron multiplication constant. Significant benefit to a broad spectrum of sites could be derived from such an experimental program in terms of enhanced nuclear criticality safety implementation and possible cost avoidance.

Session 3: Experimental Facilities and Capabilities

AN OVERVIEW OF CRITICALITY SAFETY RESEARCH AT THE ALL-RUSSIAN RESEARCH INSTITUTE OF EXPERIMENTAL PHYSICS

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ABSTRACT

This paper presents a summary of experimental and calculational activities conducted at VNIIEF from the late 1940s to now to study the critical conditions of systems as part of a nuclear safety program.

1. INTRODUCTION

Criticality safety is a very important in operations involving fissile materials. Generally, criticality safety problems are addressed by calculations and experimental data taken from critical systems.

Since late '40s, the All-Russian Research Institute of Experimental Physics (VNIIEF) has carried out R&D efforts under the scientific direction of Y. B. Khariton to experimentally study critical systems of fissile materials and to develop calculational tools for critical parameters. Basically, the research focused on the characteristics of spherical symmetrical assemblies that contained metal fissile materials. Virtually no experiments were performed on fissile materials in solution or those mixed with moderators. The primary purpose of this R&D program was to obtain data to ensure the safe handling of metal fissile material at industrial facilities and to gain experimental knowledge of various multiplication-system characteristics for neutron data testing and verification.

2. EXPERIMENTAL INVESTIGATIONS

Experiments to characterize systems containing fissile materials (FM) were carried out on a dedicated assembly machine known as the bench.[1] This bench was capable of bringing together assemblies as heavy as 2 metric tons and up to 1 m in diameter. The bench assembles a system in two parts, each being subcritical. The upper part is fixed, while the lower is vertically movable. Mechanically, these parts can be brought together at velocities of 1 mm/s, 0.1 mm/s, or 0.01 mm/s, thus achieving criticality.

The bench is located in an experimental hall whose dimensions are 12 m by 10 m by 8 m. The walls surrounding the bench are concrete and vary in thickness from 1 to 3 m. The bench is

operated remotely from a control panel outside the experimental hall. There is also an automated control and safety system (CSS) to supervise bench function.

A neutron counting system is used to monitor the operation. Three parallel measurement channels are used to monitor neutron flux over the multiplication process. There is a safety feature that measures the leakage neutrons from the system and core temperature. When the neutron flux or temperature goes beyond permissible values, the CSS removes power from an electromagnet, thereby causing the lower part of the assembly to descend.

The bench incorporated a package of experimental equipment supported by computerized data-processing capabilities. This package can provide measurements of the following multiplication-system parameters (errors equivalent to 1σ):

- multiplication factor for source neutrons, with error less than 1%;
- reactivity ρ/β with the accuracy of 1% to 3% in the range $-3 < \rho/\beta < 0.6$ [$\rho = (k_{eff}-1)/k_{eff}$, β - delayed neutrons fractions];
- reactivity disturbance Δk_{eff} by microsamples of various materials with the accuracy about $10^{-7}\ \Delta k_{eff}$;
- decay constant for instantaneous neutrons near critical state ($-3 < \rho/\beta < 0.2$) with the accuracy of 1% to 4%;
- distribution of fission in the core radius with the accuracy about 3%;
- spectral indices (σ_x/σ_f^5) with the error below 1%.

The reactivity or multiplication-factor measurements taken by the test equipment help determine quite accurately the critical conditions of the system in question. Shifting from a real critical system characterized by density gradients, gaps, and more reflection added by engineering equipment to an idealized one of a uniform, simple geometry that does not allow for extra reflection was achieved by applying a limited number of corrections. As estimated, these corrections would result in a prediction accuracy of 0.3% in k_{eff} for critical systems.

Over four decades, VNIIEF scientists have investigated, to varying degrees, about 1000 multiplication systems. About a half of these were carried out to achieve critical conditions. Others were used to measure multiplication factors or reactivity and thereby establish the system's subcriticality level for the purposes of solving safety problems. By tentative estimates, about 200 of these multiplication systems had their critical states measured quite accurately and so could be presented as benchmarks.

For most test assemblies, the core was shaped like a ball or layered sphere, or like a hemisphere or layered hemisphere. The fissile materials to be considered included various enrichments of U-235—90%, 75%, 36%, and 10%—and also U-235(90%)+Mo(9%), Pu(α), Pu(δ), and U-233.

The 30 materials most frequently encountered in production plants were looked at as reflectors for the core. These included water, polyethylene, copper, carbon, aluminum, iron, beryllium, beryllium oxide, U-238, natural uranium, concrete, lead, tungsten, nickel, molybdenum, titanium, B_4C, zirconium, Al_2O_3, CCl_4, plexiglass, cadmium+polyethylene, B_4C+polyethylene, oil, etc. Inert materials were placed outside and inside the core, and also in between the layers of fissile material. We once undertook an experimental program to determine the most efficient reflector material used in production. We found that as an external reflector 20-cm-thick polyethylene was the most efficient, and as an internal reflector used to fill the spaces inside the assembly, 2.5-cm-thick polyethylene was the most efficient.

Almost all experimental investigations were performed on a single isolated assembly. Virtually no experimental work was done to study spatial arrays of several multiplication-system assemblies. But a few experiments were conducted to address the neutron interactions between two U-235(90%) spheres.[2]

The reason that the interaction of spatially arranged multiplication systems was given so little consideration was due to the safety procedures adopted for the storage and transportation of fissile materials.[3] The method adopted to reduce neutron interactions between the neighboring assemblies was to place neutron absorbers between them rather than enlarging the distance between them or limiting the number of assemblies that might be placed in an array (this approach would have required array studies). Moreover, the amount of absorber in each assembly's container was selected such that the array would be kept subcritical for any number of assemblies arranged randomly in the space. This meant that is was possible for workers to operate freely in these arrays without any restrictions in terms of criticality safety. This approach also made it possible to abandon any experimental investigation of arrays. We only had to specify the required amount of absorber, a mass based on experiments with a single assembly having an additional reflector around it.

The technique was based on the following idea. If the assembly contains fissile material and an absorber (given a neutron source at the center) surrounded by inert material, the neutron flux from the system will be unchanged (if the absorber is large enough) or decrease (if the absorber is more than required) to compensate for neutron multiplication in fissile material. The data from a large series of experiments using this approach were taken as a basis to give recommendations on the specifications for the "protective containers" that were implemented in the industry.[3]

Also, VNIIEF has performed over the past 40 years a broad range of integrated experiments (as many as 300) that are commonly used to calibrate the multigroup and elementary constants libraries applicable in neutron calculation codes.

These integrated experiments for multiplication systems include the following data:

- characterization of critical systems (about 200 assemblies);
- reactivity factors for material microsamples at various multiplication system points;

- spectral index values; and
- decay constant values for instantaneous neutrons.

3. NUMERICAL STUDIES

The complex and diverse geometries of multiplication systems found in industry make any calculation for criticality safety for a given situation unique. Until late '50s, there were no standard approaches to the problems for complex-shaped systems. For each individual case, ways had to be found to idealize a real system. This approach would allow us to evaluate the critical parameters using techniques developed for spherical symmetrical systems. The first step to overcome this problem was to develop a simple evaluation technique for the critical conditions of arbitrarily shaped bodies having their materials density arbitrarily distributed. This approach was based on an approximation of Peierls integral equation, which is single-group in energy.[4,5] Later, this technique underwent improvement in accuracy and versatility.[6] With the improved technique, the critical parameters of bodies having both arbitrary shapes and arbitrary material distribution through the volume could be found from a set of spherical-sector calculations.

Even more difficult to solve are issues involving the interaction of subcritical systems, issues that are raised in the transportation and storage of fissile materials. For very rough criticality estimation of such systems, a numerical technique was developed in late '50s based on the idea that these systems would neutronically multiply the way neutrons multiply microscopically in a body of fissile material. But in this case we would assume that the interaction centers were not nuclei but fissile material bodies that made up a spatial system. While this assumption is fundamentally correct, the criticality calculations for these systems must make use of realistic elementary constants (i.e., nuclear cross sections), which are then scaled up to their macroscopic parameters.[5]

In addition to these single-group calculations, which are primarily intended for routine criticality safety evaluations, VNIIEF has also been developing multigroup numerical techniques for the neutron transport kinetic equation and the Monte Carlo method. Currently, a number of multigroup numerical codes, each using a different computational scheme, are functioning to calculate the critical parameters of systems in l-D and 2-D geometries. Although these codes may use any number of energy groups, 8-group and 26-group calculations have become most commonly accepted. Normally, the multigroup constants are verified against integrated experimental data before they are used.

Today, we find the Monte Carlo method the most preferable and effective to calculate critical parameters for criticality safety purposes. Thus, virtually all calculations are currently performed using the VNIIEF-developed code implementing this statistical testing technique. With these numerical techniques, calculations can be made for systems having a 3-D geometry, including arrays with systems randomly positioned in space. To make the description of the geometry of periodically structured systems more simple, mirrorlike reflecting surfaces are used.

The numerical techniques take into account all the major physical phenomena involved in the neutron interactions with substance, including neutron thermalization. The elementary interaction cross section as a function of energy can be linearly interpolated between the nodes. Different cross-sectional constant libraries can also be used with these techniques, including the commonly known ENDF. There is virtually no limitation on the energy points where the cross sections can be specified. Therefore, the resulting description of the cross-section resonance structures is rather good.

Criticality safety calculations have also been widely undertaken using the BAS system of constants developed by All-Russian Research Institute of Theoretical Physics at Chelyabinsk-70.[7] This system of elementary constants has been verified against a long line of experimental data. The constant error in k_{eff} calculations for a broad range of multiplication systems is within 1% or 2%. To illustrate, the table below summarizes k_{eff} calculations with BAS and ENDF/B-V constants for experimental critical assemblies.[8]

K_{eff} Values for Experimental Critical Systems
(figures in parentheses show the statistical calculation error of 1σ)

Core	Reflector	BAS	ENDF/B-V
Ball, Pu(δ), ρ=15.6 g/cm^3, 4.5% Pu^{240} (table 32/8)	none	1.001 (0.2%)	—
Ball, U(93.71), (table 29/8)	none	0.993 (0.3%)	0.991 (0.3%)
Ball, Pu(δ), ρ=15.6 g/cm^3, 4.9% Pu^{240} (table 32/8)	$U_{nat.}$ ρ=18.9 g/cm^3 t=10 cm	0.996 (0.3%)	—
Ball, U(93.9), (table 29/8)	$U_{nat.}$ ρ=19 g/cm^3 t=10 cm	0.996 (0.4%)	0.999 (0.5%)

Generally, the VNIIEF-operated code has capabilities similar to those of the Los Alamos MCNP code. The VNIIEF code is also used to validate safe conditions for fissile materials storage and transportation. For this purpose, k_{eff} calculations are made for a mesh that represents the multiplication system as it is enclosed by a surface to reflect neutrons.[9] This surface is used to generate angular and spatial distributions of the neutrons incident upon such a system. In some cases, this description may prove accurate. For instance, for a rectangular spatial array with identical systems at its nodes, the mesh will be right parallelepiped.

4. CONCLUSIONS

An overview has been presented of the VNIIEF's program in experimental research and numerical studies for criticality safety. The experimental research includes the development and operation of the bench machine to investigate systems with fissile materials at near-critical states. The research also included the conduct of numerous (about 1000) experiments on multiplication systems containing metal fissile materials in order to characterize critical systems.

The fissile materials used by the experiments were U-235(90%), U-235(75%), U-235(36%), Pu(α), Pu(δ), and U-233. About 30 inert materials were also used as external reflectors or internal fillings of the assemblies. Most test assemblies were either spherical or hemispherical in geometry.

About 300 various parameters have been obtained from these investigations to characterize critical systems, and they have been used to verify and qualify neutron constants. These are data on

- critical system characteristics (about 200 systems),
- reactivity factors,
- spectral indexes, and
- decay constants of instantaneous neutrons.

The codes have been developed for neutron calculations using multigroup techniques for transport equations and the Monte Carlo method. Multigroup constants (normally, 8- or 26-group) may help find neutron characteristics for 1-D and 2-D systems.

With the Monte Carlo method, calculations can be made for arbitrary 2-D and 3-D systems and also spatially arranged multiplication systems. The calculations account for all the phenomena involved in neutron interactions with substance, while continuously specifying the interaction constants in terms of the neutron energy. The Monte Carlo calculations may use different neutron constant libraries. The BAS library used now has been calibrated against integrated experimental data and allows k_{eff} calculations for small-size metal-cored systems with an accuracy 1% to 2% or better.

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A SHORT REVIEW OF CRITICAL EXPERIMENTS PERFORMED AT THE KURCHATOV INSTITUTE

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SUMMARY

Since the 1950s, the Institute of Atomic Energy (now the Russian Research Center Kurchatov Institute) has investigated nuclear reactors intended for various purposes. Our investigations necessitated the building of critical assemblies.[1] Of course, their number, design, and purpose have changed with time. A summary of the present state of these assemblies is given in Attachment 1.

In the area of power reactors, our work took several paths. Critical experiments on water-moderated assemblies began in the 1950s and have been conducted until now. During this period, we studied reactor systems in the enrichment range from natural uranium to about 6% U-235 for a wide range of water/uranium ratios. The generalized data on the first stage of experimental investigations into this area of commercial water-type power reactor physics (including multiplication parameters for lattices of various enrichments) are provided in Ref. 2.

Our program to investigate water-water reactor (WWER)* lattices was expanded extensively with the use of critical assemblies in Hungary (ZR-6)[3] and Czechoslovakia (LR-0).[4] We used the ZR-6 assembly primarily for precision measurements of a set of reactor parameters (material parameter, spectral indices, reactivity coefficients, etc.) in the enrichment range of 1.6% to 4.4% and temperature range of 20° to 130°C. We used the LR-0 to investigate systems with burnable poisons (boron, gadolinium). It should be noted that we used the ZR-6 assembly to thoroughly investigate the effect of technology uncertainties (material composition, core geometry, etc.) on various neutronics parameters.

An extensive set of critical experiments was performed on the uranium-graphite RBMK assembly, which represented a physical model of this power reactor. Although the assembly's design called for a fixed lattice spacing, we were able to compensate for this by using fuel enriched up to 2.4% and by varying the graphite content in the reactor cell and the amount of water in the channels. Some of the results of this work were published in Ref. 5.

However, the broadest range of our studies focused on small nuclear power systems intended for decentralized power generation, transportation, space, etc. We built critical assemblies moderated by water and zirconium hydride whose fuel elements ranged in enrichment from 5% to

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* *Ed. note: Water-water power reactor - the equivalent of the pressurized water reactor.*

95% U-235. In these assemblies the ratio between hydrogen and U-235 varied from 25 to 1000, while the temperature ranged from 20° to 300°C. A brief description of these critical experiments is given in Attachment 2. It should be pointed out that some of these experiments had very simple geometries and well-described composition, which allowed them to be classified as benchmark experiments.[6,7]

Along with these experimental programs, much of the effort at the Kurchatov Institute has been devoted to theoretical and calculational work to develop codes designed to research nuclear criticality safety issues for multiplication systems.

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ATTACHMENT 1
KURCHATOV INSTITUTE CRITICAL ASSEMBLIES

No.	Name	Type	Purpose	Power W
1.	SF-1	Uranium-water reactor prototype	Reactor core investigation at moderator temperature up to 300°C and pressure up to 200 kgf/cm^2	100
2.	SF-3	Uranium-water reactor prototype	Reactor core investigation at moderator temperature up to 90°C	100
3.	SF-5	Uranium-zirconium hydride critical assembly	Reactor core investigation at temperature up to 300°C	100
4.	SF-57	Uranium-water reactor prototype	Reactor core investigation at moderator temperature up to 90°C	100
5.	Kvant	Uranium-water reactor prototype	Irradiation of samples, moderator temperature up to 300°C	1000
6.	Delta	Uranium-water reactor prototype	Reactor core investigation at moderator temperature up to 90°C	100
7.	Narciss	Heterogeneous critical assembly with highly enriched uranium, hydride moderator, beryllium reflector	Reactor core investigation at ambient conditions	10
8.	Astra	Uranium-graphite core with enriched uranium, graphite reflectors	Reactor core investigation at ambient conditions	100
9.	Grog	Uranium-graphite core with graphite reflector	Reactor core investigation at ambient conditions	100
10.	UG	Physical models of uranium-graphite reactors, coolant simulator in the core channels - water, graphite reflector	Investigation of uranium-graphite channel reactor physics	
11.	RBMK	Uranium-graphite channel assembly. Experiments with and without water in reactor channels.	Investigation of RBMK-type reactor core at ambient conditions.	25
12.	Ephir-2M	Uranium-water reactor prototype	Reactor core investigation at ambient conditions	100
13.	Mayak	Uranium-water research reactor prototype	Investigation of the pulsed mode of reactor operation	10
14.	B-1000	Prototype of water-water reactor with fuel elements enriched in U-235 up to 4.4%	WWER reactor core investigation at ambient conditions	200
15.	P	Prototype of water-water reactor with fuel elements enriched in U-235 up to 4.4%, 6.5%, 10%	WWER reactor core investigation at ambient conditions	200
16.	MR reactor physical model	Physical model of research reactor MR	MR reactor core investigation	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 H_{cr} and $d\rho/d$H is based on measurement of reactivity ρ 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 H_{cr} and $d\rho/d$H is based on measurement of reactivity ρ 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 H_{cr} and $d\rho/d$H is based on measurement of reactivity ρ 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 water gap with/ without aluminum tubes, lattice spacing 12.7 mm. The number of experiments – over 30.

The determination of criticality parameters H_{cr} and $d\rho/d$H is based on measurement of reactivity ρ as a function of moderator height H near the critical level.

Power distribution is determined by 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 boric acid.

LEU-COMP-THERM-(005-007)

Critical experiments on assemblies of WWER B-1000-type hexagonal bundles (7, 19, 31) with various UO_2 fuel enrichments (2.0, 3.0, 3.3, 3.6, and 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 enrichments (3.6 and 4.4%) and Gd concentrations (2.0 and 6.0%). Critical experiments on the assemblies of WWER-type hexagonal bundles and boron-containing shroud tubes—mockups of the hermetically sealed spent fuel storage.

These experiments were carried out on LR-0 zero-power facility in Czechoslovakia.

LEU-COMP-THERM-(008-010)

The uranium-graphite facility is a physical model of the RBMK power reactor. Graphite bricks of the facility are made in the form of right-angle prisms with outer dimensions of 4.50 by 4.50 by 4.10 mm. 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.

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 "modernized" 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, were 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.

LEU-COMP-THERM-011

Uranium dioxide rod, water-moderated lattice is used. Cylindrical fuel rods 85.6 cm long consist of UO_2 clad in stainless steel. Those rods 0.51 cm in diameter are enriched to 10%; those rods enriched to 7.5% are 0.5 cm in diameter. A variety of critical assemblies with hexagonal lattices, having spacing 0.7, 0.8, 1.22, 1.4, 1.83, 1.852 cm, were studied at room temperature. Critical experiments for the lattices with fuel rods enriched to 10% (and spacings of 0.7, 1.4 and 1.852 cm) were carried out in the temperature range of 20-300°C. All these lattices, which had no 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.

In addition, a set of critical experiments for square lattices with these fuel rods was performed. The lattices included some with variable spacing as well as assemblies consisting of two separate parts with water gap between them.

IEU-COMP-THERM-001

Critical assemblies consisting of 19 air-cooled hexagonal fuel blocks with carbon and water reflectors 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 ZrH_x having hydrogen/zirconium ratio of 1.88. These experiments were performed in the temperature range of 20-240°C. Thermal neutron flux distributions and β_{eff} were measured.

HEU-COMP-THERM-001

In these experiments we used four fuel rod types, differing only in uranium concentration. The fuel block consists of UO_2 (80% enrichment), mixed and pressed with aluminum powder so that uranium concentration in each of the next fuel-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 - 7 mm diameter, 60 mm height. 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 stainless steel cladding (13 mm diameter) and could be dismantled.

Two series of experiments were carried out with above fuel rods:

1. Critical assemblies made of the fuel rods arranged in circular water lattices with the range of average hydrogen/uranium-235 concentration ratio of 50 to 600. One of the fuel-block-type experiments was performed in the vessel a with pressure of 120 atm in the temperature range 20-250°C. Three others were performed in the temperature range of 20-90°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 Cd from the outside around the circumference. Critical state for each assembly was attained by adjusting the height of the B_4C rods. All these experiments were performed at room temperature only. Besides the critical number of fuel rods, detailed fission density distributions were measured.

HEU-COMP-THERM-002

These were metal rod, hexagonal, water-moderated lattices. They consisted of rods 0.47 by 0.47 cm square and 90 cm long. The fuel was U-Zr alloy (20 wt% U, 80 wt% Zr) 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 had no control rods and were fully reflected. Their perimeters were made as close to circular as possible.

HEU-COMP-INTER-003

Assembly is a small heterogeneous, intermediate reactor, fueled 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 with poison segments.

Several benchmark critical experiments have been performed on intermediate critical assembly for the nuclear criticality safety program.

Session 4: Rad-Waste and Weapons Disassembly

CRITICALITY ANALYSIS FOR WEAPON DISASSEMBLY AT THE PANTEX PLANT – PART I: BARE PITS

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Investigations have been performed on the potential for a critical excursion in weapon assembly and dismantlement at the Pantex Plant. Pits and pits in containers were modeled in their most reactive configuration to estimate the effective multiplication factor (k_{eff}) for a number of postulated scenarios. Results are reported for calculations performed for Pantex safety analyses reports, radiological hazards assessments, and a current study to justify the criticality alarm exemption. Analyses for bare pits (i.e., those that are uncontained in this context) are reported in this paper, while pits in containers are considered in a later paper in this session ("Criticality Analysis for Weapon Disassembly at the Pantex Plant – Part II: Staging").

The basic models represent the most reactive configurations. A generic pit, defined to be an unclassified composite of the largest mass and smallest volume that might be anticipated, is modeled as a spherical shell of plutonium with a 6-in. outside diameter and an assumed mass of 6.5 kg. The model ignores the presence of cladding and the presence of any appurtenances.

The criticality calculations were performed with the KENO and MCNP code packages in versions developed for IBM-286- to -486-class personal computers. Validation for each code version and each specific computer system was performed consistent with the guidance of American Nuclear Society Standard ANSI/ANS-8.1 by computing k_{eff} values for benchmark experiments. The lowest calculated value of k_{eff} for the benchmark cases minus three times the standard deviation was established at 0.96. Thus, a subcritical multiplication factor for a given configuration is considered to be established when the computed k_{eff} plus two standard deviations is less than 0.96.

The computer model for the pit incorporated additional simplifying assumptions. Because the exact composition of plutonium varies in the higher-mass-number isotopes due to production variations and decay, the generic pit shell is assumed to be entirely fissile ^{239}Pu, which produces the most reactive configurations.

Bare pits (that is, those not in a container or device) are handled only in robust assembly/disassembly cells or bays where they are removed from or mated to explosives or subjected to testing. Scenarios were developed based on administrative limits and actual or potential physical conditions that could exist in these facilities. Operational guidance limits the number of pits out of containers. In one particular cell, three pits may be subjected to testing at the same time. Even here, however, the testing is accomplished in three separate cubicles, making it unlikely that all three pits would be in a single location. However, proximity is not impossible.

The calculations for generic pits showed that a 2 x 2 rectangular planar array and a three-unit hexagonal-packed array may be critical if each pit is cracked open, water-flooded, and fully (≥30 cm) water reflected. A cubic array 2 x 2 x 2 may be critical if fully water reflected. These cases all represent extreme situations that are not considered credible given these operating constraints that are in place:

1. a distance of 3 ft must physically separate pits (less distance if specific tools are being used),
2. no more than three pits may be out of their containers in a single cell at one time (and these pits are required to be in different physical areas for specific tests), and
3. no procedure may be undertaken that could permit the introduction of water into a pit cavity (e.g., crack a pit and submerge it in water).

From the standpoint of what might conceivably happen during operations (even though it requires multiple violations of procedures), the analysis indicates that reasonable configurations (three units or even four units in a dry, planar array) are substantially subcritical.

Thus, it is concluded that a criticality event involving bare pits is not credible. This is attributable in large measure to the solid form of the material (i.e., metallic shells) and, in this instance, to the operational practices and controls.

Another scenario considered a single flooded unit which would increase in reactivity if the plutonium dissolved, e.g., by corrosion in the water. Units larger than the generic pit used in this analysis could become critical under these circumstances. However, long dissolution times and a requirement for agitation to sustain a quasi-solution form make the scenario extremely unlikely.

Finally, the study predicts the following configurations to be subcritical (i.e., $k_{eff} + 2\sigma < 0.96$) at the Pantex Plant:

- Six generic pits in close-packed, body-centered cubic array, dry and unreflected or concrete reflected on the bottom.
- Six modified generic pits with a diameter close to 7 inches, water-moderated and fully reflected (≥30-cm) on top and concrete-reflected on the bottom in either: (1) a body-centered cubic array, or (2) a "4+2" array (i.e., a 2 x 2 cubic array with two more units in the gaps on top).
- Seven generic pits, water-moderated and fully reflected (≥ 30-cm) on top and concrete-reflected on the bottom in a close-packed hexagonal array. Moderation by water of less than full density (i.e., partial "mist" moderation) does not lead to an anomalous increase in the multiplication factor.
- An infinite X-Y planar array of generic pits, bare or on concrete, and in close contact (i.e., "skin-to skin").
- An infinite X-Y planar array of generic pits water-moderated and fully reflected (i.e., with ≥ 30 cm) on top and concrete-reflected on the bottom and with a minimum separation ≥ 6.5 cm (i.e., 2.6 in. or an 8.6-in. center-to-center spacing).

- An infinite X-Y planar array of generic pits moderated and reflected by > 30 cm of partial-density water (maximum multiplication occurs between 20% and 50% of full density) on top and reflected on the bottom by concrete with a minimum separation of nearly 10 cm (i.e., about 4 in. or a 10-in. center-to-center spacing)
  (NOTE: The subject of reactivity effects caused by spacing and by full- and partial-density moderation and reflection is addressed by S. Payne in the next paper in this session.)

A final scenario is based on a new operation that has been planned. An assembly is sprayed with dimethyl sulfoxide (DMSO) fluid to remove chemical high explosive from a pit. The prospect that fissile material from a damaged pit could be dissolved (or suspended) in the DMSO and lead to a critical excursion has been evaluated thoroughly and judged to be incredible. (The subsequent paper in this session by S. Troyer describes the applicable criticality safety evaluation.)

Thus, configurations involving bare pits that are even close to realistic are subcritical by a substantial amount (and actually significantly more so than the calculations show, due to the conservative model assumptions, e.g., pits all ^{239}Pu isotopic composition, large mass, and small volume). Thus, it is concluded that a critical configuration involving the bare pits is not credible.

POSTULATED ACCIDENT SCENARIOS IN WEAPONS DISASSEMBLY

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Three postulated accident scenarios for weapons disassembly have been investigated. The first deals with a tetrahedral configuration of four generic pits; the second, an infinite planar array of generic pits with varying interstitial water density; and the third, a spherical shell with internal mass suspension in water varying the size and mass of the shell. Calculations supporting this analysis were performed using the Monte Carlo Neutron Photon transport code MCNP4A on a SUN SPARCstation. The generic pit is a pure Pu-239 shell, 6.5 kg in mass and 6 in. outside diameter (o.d.), with a solid density of 19.86 g/cm^3.

The analysis for the first scenario was performed by setting up a three-dimensional, close-packed (i.e., surface-to-surface contact) configuration of three generic pits. All the pits had internal voids and thick (>30 cm) water reflection. The pits were then separated from the center point of the tetrahedron in 0.5-cm increments and the k_{eff} was calculated. The pits were moved apart until the surface-to-surface separation was approximately the pit diameter. There is an inconclusive indication that pits at small (near a tenth of pit diameters) separations may be slightly more reactive than at the close-packed configuration. More resolution in the 0-2 cm regime is needed to more completely understand this scenario.

The second scenario analysis follows from the first. Infinite planar arrays of 6.0-kg and 6.5-kg, 6-in. o.d. pits were analyzed with varying interstitial water density. In two cases concrete reflection on the bottom surface was used with thick water reflection above (>60 cm) to account for the change in water density. The third case used water in all directions surrounding the array. A peak in k_{eff} was noticed in the 20-30% water-density regime. This peak was 15-20% higher than the full water density k_{eff}. It was also noted that concrete reflection on one side of the array contributed significantly to the overall reactivity of the system over the purely water reflection.

The third analysis looked at internal flooding and subsequent suspension of Pu internally in a single, thick water-reflected generic pit. Analysis of typical stockpile items indicates that maximum reactivity is obtained when 10% of the total pit mass is internally suspended in the internal water. For this scenario, Pu masses of 6.0 kg, 6.5 kg, and 7.0 kg were used. These masses - which started as solid spheres - were expanded until 12 in. o.d. was reached. A constant 10% of the mass was uniformly dispersed in the internal water. The remaining Pu shell was maintained at constant solid density. Generally, for outside diameters between ~4 and 8 in., the configuration is subcritical. At outside diameters less than 4 in. fast neutrons are the main contributors to supercriticality while above moderation plays a greater role.

Future studies will focus on higher resolution of small pit separation regimes and snapshots of hydrodynamic processes of water/plutonium mixtures.

CRITICALITY SAFETY IN HIGH EXPLOSIVES DISSOLUTION

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INTRODUCTION

In the fall of 1992, an incident occurred at the Pantex Plant in which the cladding around a fissile material component (pit) cracked. This incident resulted in small amount of Pu contamination in the process tooling and in the water used in the process. This event occurred after a "thermal shock" process. Prior to this incident, the nuclear explosive assembly was submerged in liquid nitrogen and then allowed to warm, causing the high explosives (HE) to crack and come off the pit. The small amounts of residual HE were then manually removed by running warm water over the pit in a 2-in. deep tray on a work table. This tray drained into a non-criticality, safe-geometry drain container. The crack was observed during this manual cleaning process. Normally, the pit would have then been immersed in a "soak tank" of dimethyl sulfoxide (DMSO) to remove the small amounts of residual HE and adhesives. However, due to the crack incident, the process was stopped. The dimensions of this soak tank were 1 by 2 by 1 ft.

CONCERNS RESULTING FROM THE "CRACKER"

DOE 5480.24, "Nuclear Criticality Safety," requires that criticality alarm systems or criticality monitoring systems be installed for operations in which an inadvertent criticality is presented as credible in a DOE-approved safety analysis report (SAR). These systems are not installed in Pantex facilities, since the probability of such an event was judged to be less than 10^{-6} per year (incredible).

Prior to the cracker incident, the design laboratory's position was that the probability of a pit cracking was less than 10^{-6} per year. However, the review of this incident led to the conclusion that this event was credible.

As a result of the change in position with respect to the probability of a crack, additional attention was given to criticality safety. In particular, since the crack provided a release path for a fissile Pu/water mixture into a large-diameter drain container, this process was considered unacceptable.

The step in the original process that followed the manual cleaning involved soaking the pits in a DMSO vat (soak tank) for several hours to remove any residual adhesives that might be on the pit. The soak tank measured 1 by 2 by 1 ft. Hence, this geometry has now been determined to be undesirable.

Although the cracker event did not result in any significant contamination or personnel exposures, concerns pertaining to the potential for recurrence led to the conclusion that the current process was unacceptable.

REDESIGN CONCEPTS

In the design meetings that followed the clean-up process, the following options were considered with respect to methods for restart of the dismantlement process:

- Continue with thermal shock using a new drain container and no soak process. This recommendation would have required the least design effort but would have increased extremity doses to the technicians. Also, since the thermal shock process was now judged to be undesirable, this recommendation was considered unacceptable.
- A "total soak" process, in which the nuclear explosive is soaked in a volume-controlled tank with soluble and fixed neutron-absorbing materials and "isolation zones" to control external reflection. This recommendation was not used because of concerns related to maintaining the isolation zone and ensuring the continuous presence of soluble neutron-absorbing materials. Another factor involved in rejecting this option was the amount of time that would be required for full dissolution of the HE from the unit.
- A "shower head," or solvent spray system, in which the nuclear explosive assembly is sprayed with the solvent, using a closed-loop, limited-volume, and controlled-geometry spray system. This design was selected by Mason & Hanger as the preferred method, since it provided potentially faster HE removal, reduced the need for handling the unit, allowed the use of larger DMSO volumes, and eliminated the need for neutron-absorbing materials.

DISSOLUTION TOOLING DESIGN CONSIDERATIONS

- Criticality alarm/monitoring systems are not installed in the Pantex bays and cells, since an inadvertent criticality has always been considered an incredible event. The cracker incident led to questions about the validity of this position with respect to this process. Hence, the tooling designs were redeveloped not only to ensure a sufficient margin of safety, but to ensure that an inadvertent criticality was still incredible.
- As of the date of the design, no corrosion data was available for Pu immersed in DMSO. The design lab was requested to perform experiments to ascertain this data, using pure and "wet" DMSO (up to 20% water content), and DMSO with dissolved HE. Problems with approval to start the experiment have significantly delayed these experiments. To date, this data is still unavailable. Hence, the design analysis had to be performed using extremely conservative, worst-case assumptions.
- The computer code used in the Mason & Hanger analysis was MCNP. Although it is an industry standard package, it had not been benchmarked and validated in place. A criti-

cality specialist in DOE/ALO/SPD provided independent review of the analysis, and also performed confirmatory computer runs to validate the computer runs included in the analysis.

DISSOLUTION TOOLING DESIGN FEATURES

The tooling design that resulted from the various project meetings and reviews (Fig. 1) contained the following features:

- Use of 2-in. slab geometry. The tank incorporates the use of a 2-in. slab geometry in the dissolution vessel tank, equipped with redundant overflow drains. The unit is placed above the DMSO volume, and the spray heads surround the unit to provide full coverage. The DMSO sprays onto the unit and then falls to the bottom of the tank, which is controlled to less than 2-in.depth.
- The redundant drains feed a 2-in. slab tank below, which is used to capture overflow, and also is equipped with dual overflow drains which drain to the floor. The main dissolution tank drains are designed to overflow at DMSO depths of less than 2 in.
- The two slab tanks are separated by an aluminum heating platen, and a total separation distance of approximately 18 in., which is close to the characteristic length of the tanks themselves. Hence the tanks are very near the point of complete isolation from each other. Also, since the platen has water passages drilled through it, the water contained in the platen provides additional isolation of the two tanks. This platen is designed as a completely separate component such that a platen leak cannot result in water introduction to either slab tank.
- A filtered pump exhaust-capture system was added to the design to capture DMSO that could be forced out the pump exhaust following a spray pump failure. Following such a failure, contaminated DMSO would potentially be pumped into the spray pump exhaust, but would be captured in the safe geometry of the exhaust system. This exhaust feeds into a 6-in. diameter cylinder on the floor that provides a criticality-safe geometry for this fluid.
- A slight negative pressure is drawn on the inside of the dissolution tank to minimize operator exposures to airborne DMSO/Pu. The vacuum system that supplies the negative pressure exhausts to the building stack. Contaminated material are filtered out of this effluent with a charcoal filter and dual HEPA filters mounted on the dissolution cart.
- Six-in. diameter cylindrical drain containers have been built for use in the removal of the potentially contaminated DMSO from the dissolution tank. These containers are designed with 32-in. diameter "stand-offs" to ensure sufficient separation under all postulated conditions.

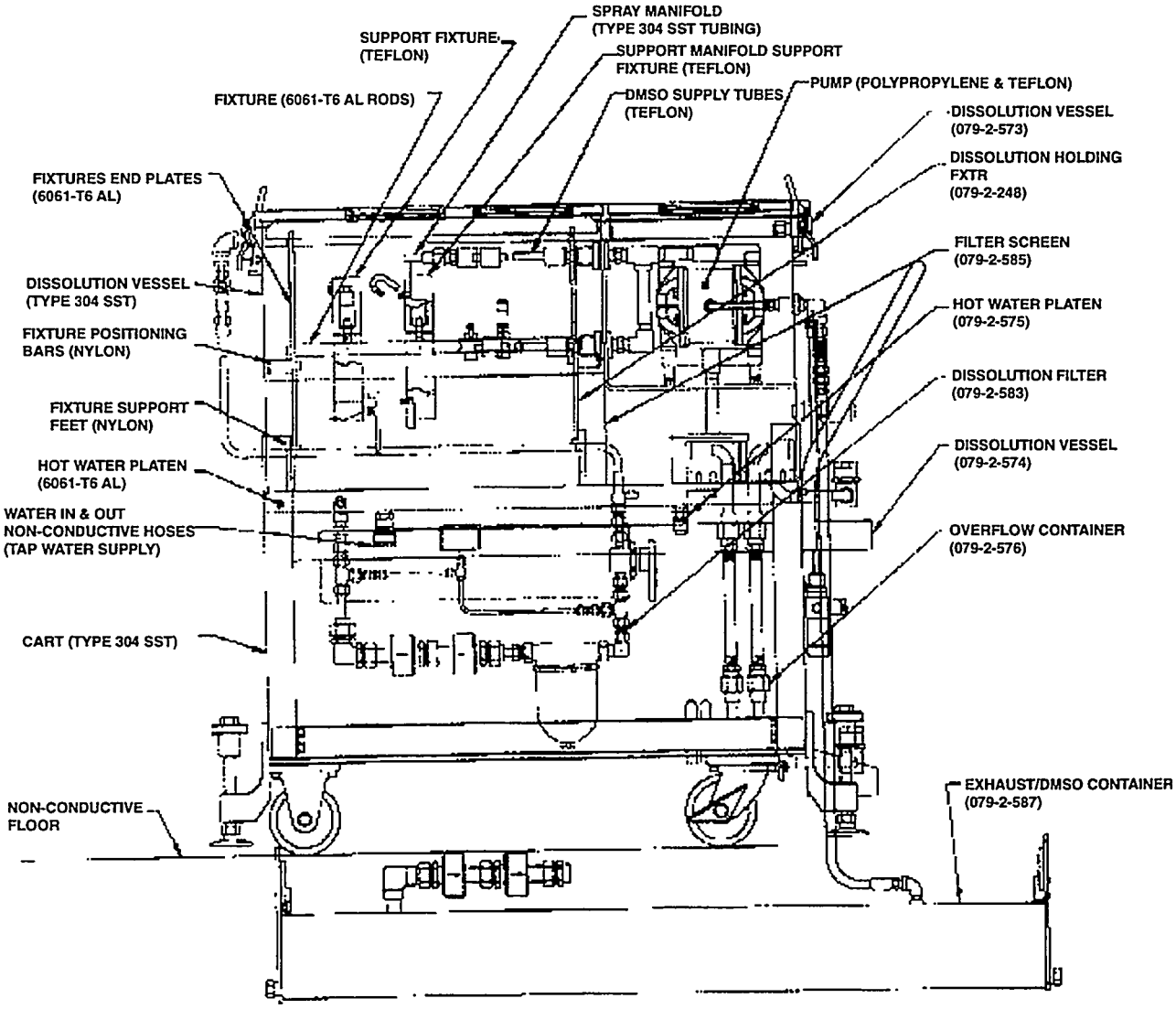

Figure 1. New design of dissolution assembly.

ANALYSIS

The analysis of the dissolution tooling was developed as a formally documented, stand-alone analysis, with each of the references, data sources, and assumptions described in detail. Computer code outputs and independent review comments and resolutions were also provided as attachments to the analysis text.

- Due to the number of uncertainties associated with characterizing the source, the analysis was performed using very conservative assumptions:
  - The pit was assumed to be the generic "SAR" pit used in the analyses for Pantex Safety Analysis Reports (SARs). This pit has a mass of 6.5 kg, and was assumed to be completely dissolved in the DMSO.
  - Fluid levels well in excess of the height of the drains were analyzed. These heights also exceeded the maximum height that would occur if the DMSO were "double batched" after a component failure and the failure of both overflow drains.
  - Various densities of external water reflection were considered, up to full-density water.
  - The fissile material was conservatively assumed to be pure ^{239}Pu, since the less reactive materials present in the materials are considered to be replaced with ^{239}Pu.
- The computer models were executed using MCNP.
- Confirmatory runs were performed by the independent reviewer, since the Pantex version of MCNP had not yet been benchmarked "in place."
- The analysis demonstrated that, even under scenarios which were clearly incredible, the tooling designed would remain subcritical.

CONCLUSIONS

The final tooling design developed for this process incorporates a number of safety features and provides a simple, self-contained, low-maintenance method of HE removal for nuclear explosive dismantlement. The "hands-off" design significantly reduces the amount of handling time and consequently provides dose reductions consistent with the ALARA concept. Also, the filtering systems provided on the pump exhaust system and vacuum return lines ensure that releases of potentially contaminated DMSO vapors to the work area and the environment are minimized. The analyses demonstrate that the tooling design will remain subcritical under normal, abnormal, and credible accident scenarios.

NEXT GENERATION STORAGE FACILITY

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Los Alamos National Laboratory

ABSTRACT

With diminishing requirements for plutonium, a substantial quantity of this material requires special handling and, ultimately, long-term storage. To meet this objective, we at Los Alamos have been involved in the design of a storage facility with the goal of providing storage capabilities for this and other nuclear materials. This paper presents preliminary basic design data, not for the structure and physical plant but for the container and arrays which might be configured within the facility, with strong emphasis on criticality safety features.

INTRODUCTION

In order to provide preliminary estimates relating to criticality safety for a long-term plutonium storage facility, Monte Carlo calculations were used to determine neutron multiplication factors (k_{eff}) of several storage configurations. Abnormal (upset) conditions, other than water flooding, were not considered at this time. The configurations investigated were finite arrays of containers with metallic plutonium and uranium in various shapes, such as spheres, cylindrical annuli, and discs.

Some of the features of the storage container design and loading, the storage bay configuration(s), and facility design-basis accidents have not been finalized. Therefore, it was necessary to perform several evaluations to determine the sensitivity of neutron multiplication factors to variations in storage container contents.

The purpose of these evaluations is to provide criticality safety guidance for a plutonium storage facility as design parameters become finalized. It will be an ongoing, iterative effort until the design is finalized and the facility becomes operational.

FISSILE MATERIAL CONFIGURATION

These evaluations were centered around two types of fissile material which may be stored in the facility. They are plutonium (95 wt% ^{239}Pu and 5 wt% ^{240}Pu with a theoretical density of 19.86 g/cm^3), and highly enriched uranium (93 wt% ^{235}U , 7 wt% ^{238}U at a theoretical density of 18.8 g/cm^3). Although variations of these two materials may be stored, for the sake of these calculations, only these two material types were evaluated. In addition, no non-fissile materials

were assumed to be present in the primary storage container, thus, reducing the effects of moderation in the storage container and array.

CONTAINER DESIGN

Although the container design has not been finalized, the model considered was on the order of 6 in. o.d. with a height of approximately 12 in. This container could serve (1) as the inner containment in a shipping container and (2) as the primary storage container in the facility. The container would be used by those facilities which need to ship material to the storage facility, and it will have to conform to quality assurance requirements during packaging. After receiving and unloading the shipping package at the storage facility, the container could be placed in the storage array without any additional and unnecessary handling.

The container, as shown in Fig. 1, is made up of two commercially available pipe end-caps separated by a length of pipe, also commercially available. The material of construction is stainless steel and the wall thickness is on the order of ~0.25 in. (nominally schedule 40).

After the material is inserted, the end-caps will be seal welded in an inert atmosphere. X-ray and leak detection can be used to confirm the quality of the seal welds. Calorimetry and other non-destructive analysis methods will ensure the proper container mass loading.

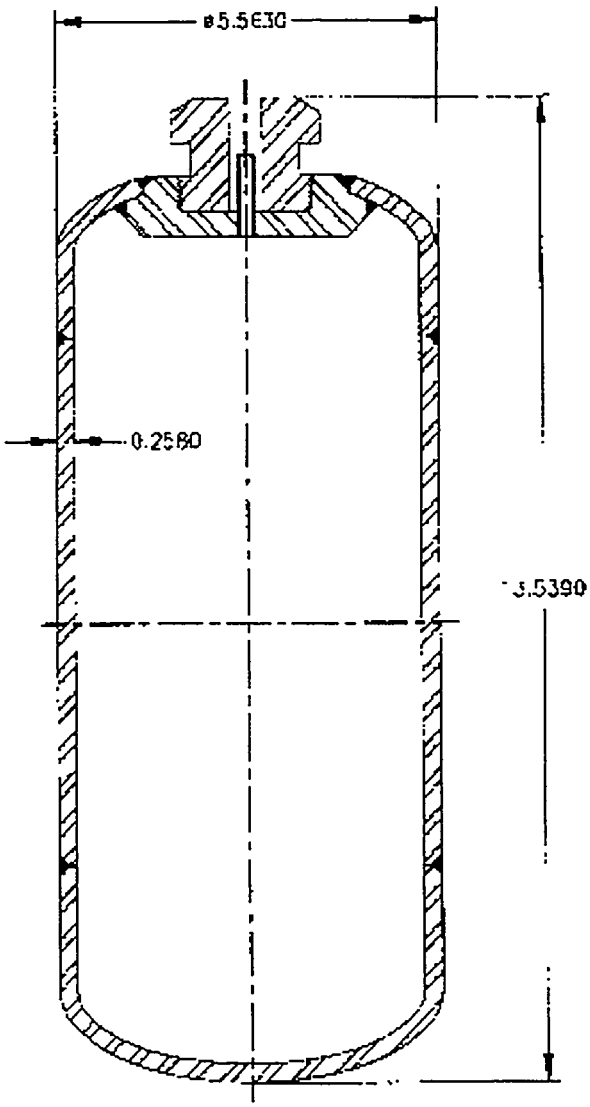

Figure 1. Primary storage container.

ARRAY DESCRIPTION

The array of storage containers was initially specified to fit within a floor area of 13 x 127 ft with a height not to exceed 10 ft. The floor area restriction will allow peripheral and periodic inspection of the array, as appropriate, while the height limitation is to provide sufficient space in the storage cell above the array for mechanical equipment such as gantry crane(s).

The array was first developed on the assumption of 18-in. center-to-center spacing requirements and then filling the floor area with as many storage locations as possible. The height restriction, coupled with the nominal height of the individual storage container and its spacing requirements, resulted in approximately 7,000 storage locations per storage array. Figure 2 is an illustration of the two arrays configured in the storage cell.

CALCULATIONAL METHOD

Two calculational methods were employed for these initial criticality calculations. The first was KENO-V.a, which utilized the Hansen-Roach 16-group cross sections.[1] The second was MCNP, which utilizes ENDF/B continuous-energy cross-section data.[2] The two codes were used as a 'validation' for the systems being analyzed since there is little critical experiment data for storage arrays approaching the dimensions given above. In addition, while the American National Standard Institute's Guide for Nuclear Criticality Safety in the Storage of Fissile Materials[3] provides storage criteria and limits for air-spaced units, this data was not directly usable for this effort. Therefore, the two Monte Carlo codes were employed.

Each case was executed until ~100,000 neutron histories were analyzed which resulted in a standard deviation of less than 0.003 for any case.

CALCULATIONAL MODEL

The description of the storage container, as given above, was the building block in the storage array. The storage array was then built on a conceptual design provided by Paul Smith, of

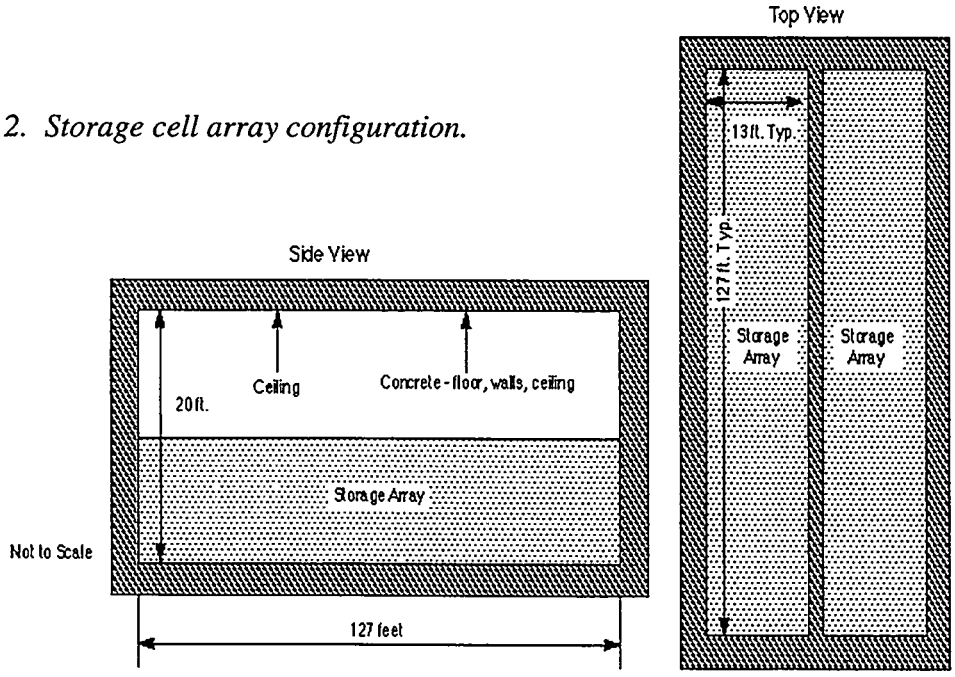

Figure 2. Storage cell array configuration.

Merrick, Inc., in which the storage locations were made up of carbon steel 'shrouds.' A shroud is 18-in. square with a right-circular hollow core of approximately 14 in. This provides not only for storage of larger items such as unirradiated fuel rods, but also establishes the center-to-center spacing requirements in the array. Figure 3 illustrates the carbon steel shroud.

The fissile materials previously described were modeled in the storage container in several shapes, namely, sphere, annuli of different dimensions and therefore different masses, rod, and button (right circular cylinder). The first attempt was to determine the viability of these shapes in each of the storage locations. There was no mixing of the several shapes. Figures 4 through 9 are illustrations of the material types and shapes modeled.

Only one upset condition was considered, that in which the array became flooded with full-density water. The flooded condition did not exist inside the individual storage containers. That is, the water only occupied the space outside of the primary container (Fig. 1) and inside the 14-in. cavity in the steel shroud. Moderation caused by water sprays of varying density has not been evaluated yet.

CALCULATIONAL RESULTS

The calculational results for the several material and array configurations modeled are listed in Table 1. There is parallel and fairly close agreement between KENO and MCNP for the cases analyzed. Since this was only a conceptual analysis, the relatively small discrepancies in the results have not been investigated.

The configuration with the solid sphere is the most reactive with the average (of KENO and MCNP) (unflooded) array multiplication factor = 1.04. As the shape of the material was changed to increase the surface area of the individual unit (for example, to an annular cylinder) the array multiplication factor decreases.

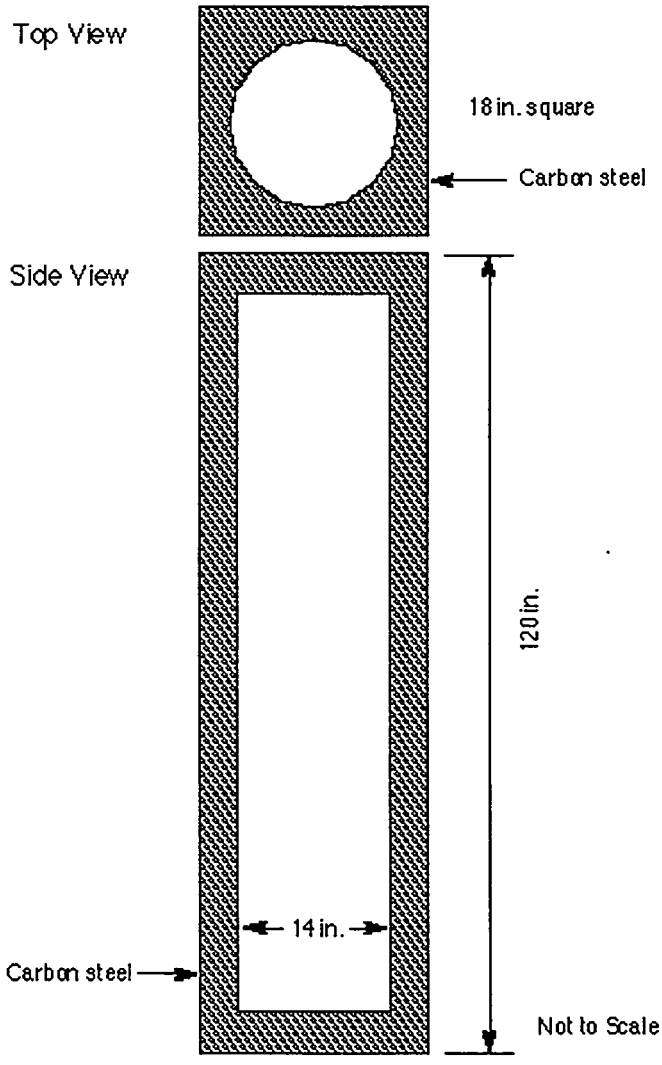

Figure 3. Carbon steel shroud.

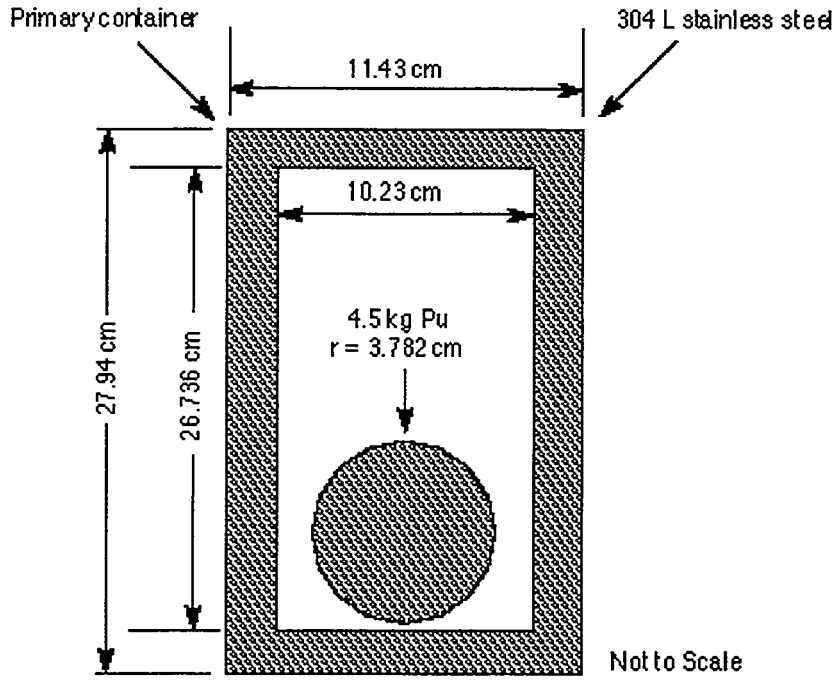

Figure 4. Primary storage container with 4.5 kg Pu as sphere.

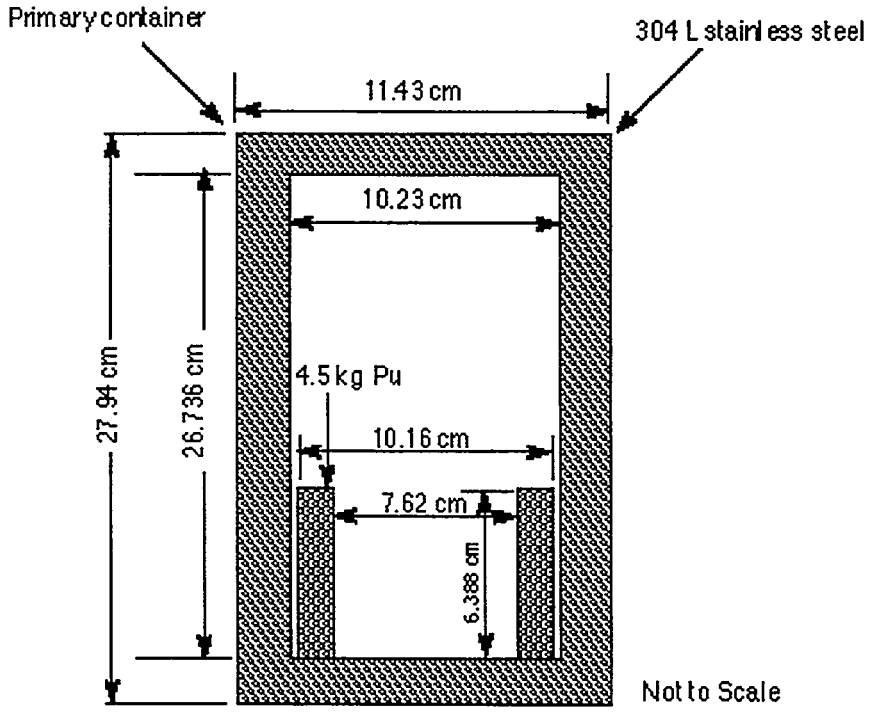

Figure 5. Primary storage container with 4.5 kg Pu as annular cylinder.

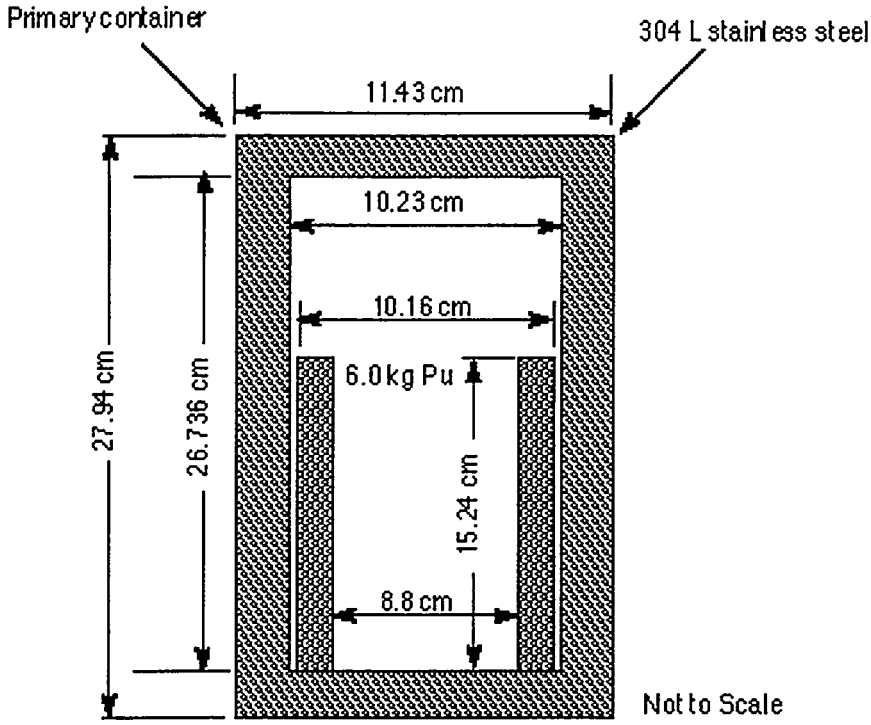

Figure 6. Primary storage container with 6.0 kg Pu as annular cylinder.

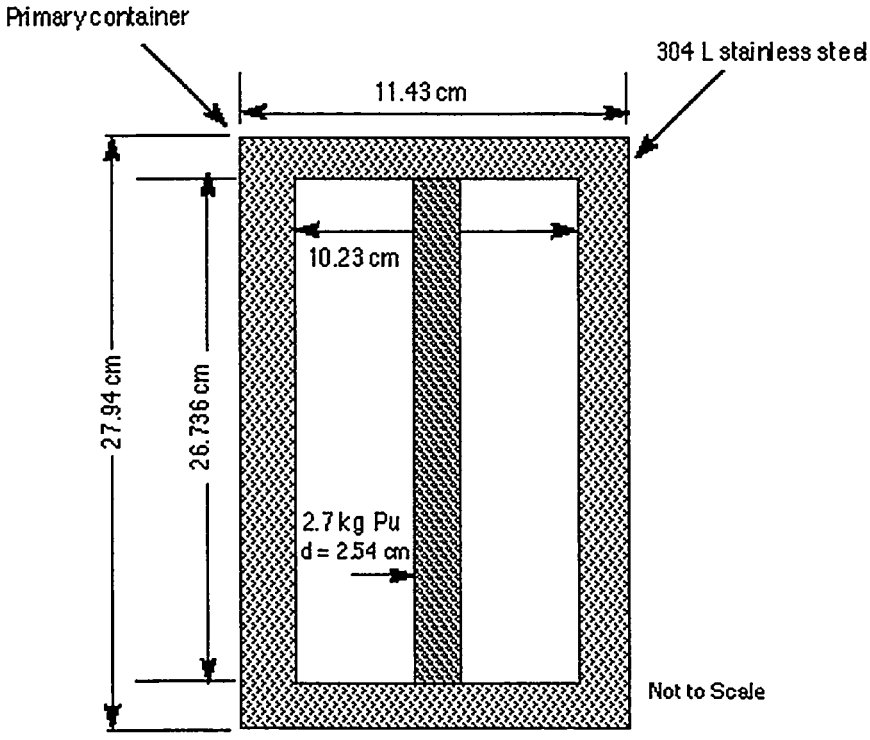

Figure 7. Primary storage container with 2.7 kg Pu as rod.

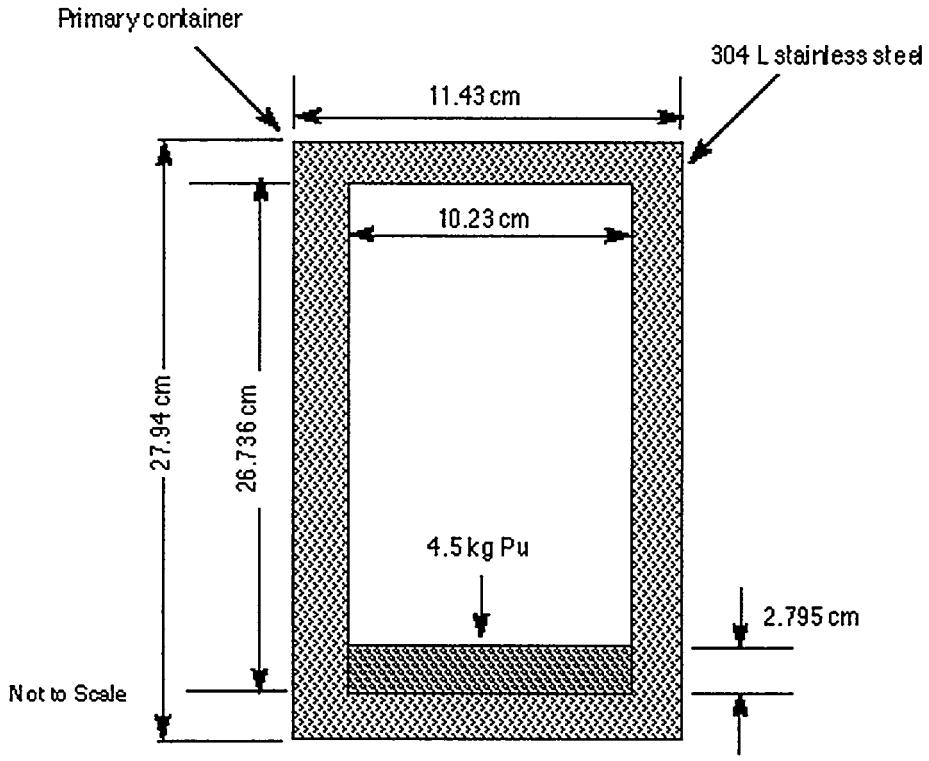

Figure 8. Primary storage container with 4.5 kg Pu as right circular cylinder.

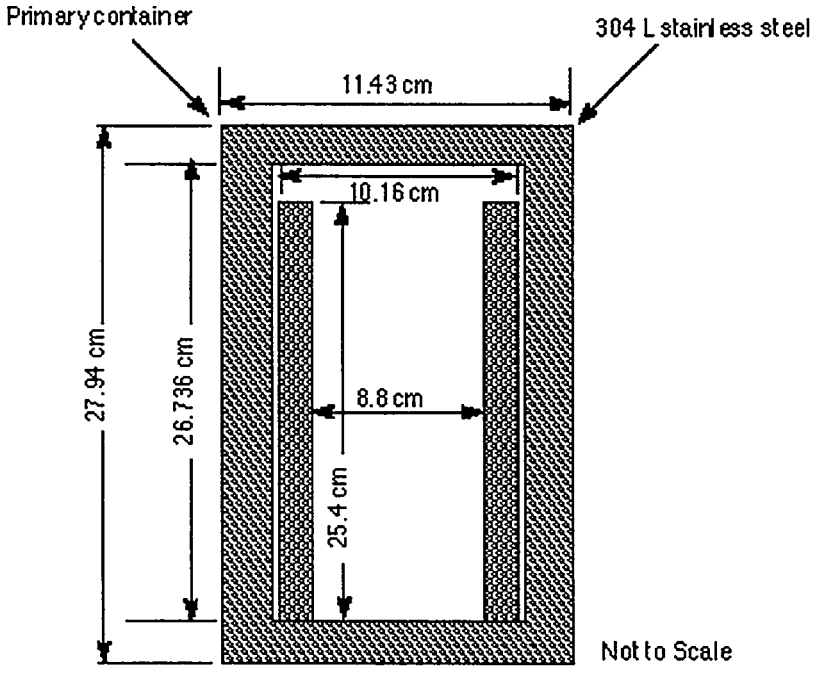

Figure 9. Primary storage container with 10.0 kg U(93) as annular cylinder.

Table I. Array Multiplication Factors

Fissile Material Mass and Shape	Array Size	Config.*	Multiplication Factor, k_{eff} MCNP	KENO-V.a
4.5 kg Pu Sphere	8x83x11	Bare	1.02	1.07
		Flooded	0.88	0.88
	16x83x11	Bare	1.04	1.07
		Flooded	0.88	0.88
4.5 kg Pu Annulus	8x83x11	Bare	0.84	0.88
		Flooded	0.68	0.67
	16x83x11	Bare	0.85	0.88
		Flooded	0.68	0.67
6.0 kg Pu Annulus	8x83x11	Bare	0.95	1.00
		Flooded	0.75	0.73
	16x83x11	Bare	0.97	1.00
		Flooded	0.74	0.74
2.7 kg Pu Rod	8x83x11	Bare	0.65	0.69
		Flooded	0.52	0.52
	16x83x11	Bare	0.66	0.69
		Flooded	0.52	0.53
4.5 kg Pu Button	8x83x11	Bare	0.95	1.01
		Flooded	0.81	0.80
	16x83x11	Bare	0.97	1.01
		Flooded	0.81	0.80
6.0 kg Pu Annulus (15.24 cm tall)	8x83x11	Bare	0.87	0.92
		Flooded	0.66	0.66
	16x83x11	Bare	0.89	0.92
		Flooded	0.67	0.65
10 kg HEU Annulus	8x83x11	Bare	0.83	0.86
		Flooded	0.61	0.60
	16x83x11	Bare	0.83	0.90
		Flooded	0.62	0.60

* Flooded condition placed water in the space between the primary container and the steel shroud.

SUMMARY AND CONCLUSIONS

The storage of fissionable material, namely plutonium, in a primary storage container has been analyzed to evaluate criticality safety of the facility. The Monte Carlo codes KENO-V.a and MCNP 4A were used. The calculations performed suggest that the storage array will be subcritical during normal operating conditions and will remain subcritical under conditions of array flooding. However, processing of metal into extended shapes, particularly a thin-walled right-circular annulus will enhance criticality safety concerns and/or provide for significantly increased storage capacity.

Also noteworthy is the nearly 100% neutronic isolation of individual containers provided by the massive steel present. This is demonstrated by the two array sizes having similar multiplication factors. As would be expected, flooding of the shroud that places several inches of water between neighboring units indeed decouples the units 100% and lowers the array multiplication factor.

It is important to emphasize that the results of evaluations presented herein are preliminary in scope, and should not be relied upon to justify a storage facility of any design. Calculations have not been performed for all types of fissionable materials which may become candidates for storage. Additional, in-depth analyses need to be performed as the facility design evolves and as fissile materials to be stored are better characterized.

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1. L. M. Petrie and N. F. Landers, *KENO-V.a, An Improved Monte Carlo Criticality Program with Supergrouping,* NUREG/CR/200, Volume 2, Section F11, ORNL/NUREG/CSD-2/V1/R.

2. J. F. Briesmeister, editor, *MCNP - A General Monte Carlo Code for Neutron and Photon Transport, Version 4a,* Los Alamos National Laboratory document LA-12625-M, December 1994.

3. American National Standard Institute, *Guide for Nuclear Criticality Safety in the Storage of Fissile Materials,* ANSI/ANS-8.7-1975; R1987 [N16.5], American Nuclear Society, La Grange Park, Illinois.

LONG-TERM CRITICALITY CONCERNS ASSOCIATED WITH DISPOSITION OF WEAPONS PLUTONIUM

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SUMMARY

A substantial inventory of excess separated plutonium will result from the ongoing and planned dismantlement of U.S. and Russian nuclear weapons. This excess plutonium, as suggested by the National Academy of Sciences (NAS) study,[1] could be disposed of by irradiating it into reactor-grade spent fuel, or immobilizing it in molten glass (or syn-rock) along with defense high-level waste, followed by burying it in a geologic repository. Another option mentioned by the NAS study is to bury the plutonium in 4-km-deep boreholes sealed with bentonite clay and concrete. Virtually, all these plutonium-disposition options will generate one or more final disposal waste forms, which require the evaluation for long-term criticality safety concerns.

The long-term criticality safety concerns arise because the fissile content (i.e., Pu-239 and its decay daughter U-235) in these Pu-dispositioned waste forms is, in general, higher than that in a reference spent-UO_2 fuel. MOx spent fuel could contain 3 to 4 wt% of reactor-grade plutonium, compared to only 0.9 wt% plutonium in the reference spent-UO_2 fuel. For the immobilization and the deep borehole options to be economically viable, a plutonium content of 3 to 7 wt% would be required.[2] If dissolution and leaching of these disposal waste forms in a geologic repository or borehole could not be ruled out, then in some future time (tens of thousand years) when the waste forms were deteriorated by the intruding groundwater, the water could mix with the long-lived fissionable materials to form into a critical system. If the critical system is self-sustaining, the fission products produced could be readily available for dissolution and release out to the accessible environment and adversely affect public health and safety.

Another criticality safety concern arises during the process of vitrification or immobilization of weapon-grade plutonium into the chemical matrix. Plutonium causes concern because it tends to form pockets of locally concentrated material in glass, which could lead to a criticality problem in the glass melter. The design and operation of the melter has to ensure that a criticality event would not occur.

The study proposed here is to evaluate the long-term criticality safety concerns for disposition of fissionable material in a geologic setting. Issues to be addressed include the fundamental limitations on waste-form disposal suitability based on criticality safety considerations; the identification of worst-case geo-chemical conditions and waste-form geometries which present the most concern for long-term criticality; suggestions of technical fixes for such concerns; and the evaluation of the criticality-safety design for the melting/mixing vessels.

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1. National Academy of Sciences, Committee on International Security and Arms Control, "Management and Disposition of Excess Weapons Plutonium," National Academy Press, 1994.

2. Bette Hileman, "U.S. and Russia Face Urgent Decisions on Weapons Plutonium," Chemical & Engineering News, June 13, 1994.

CRITICALITY ANALYSIS FOR WEAPON DISASSEMBLY AT THE PANTEX PLANT - PART II: STAGING

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Investigation of the potential for a critical excursion at the Pantex Plant was extended (from that reported in Part I) with examination of pit staging. Here, the basic model represents the most reactive configurations of pits in the containers used for handling and staging at the Pantex Plant. The generic pit model considered previously (i.e, a 6.5-kg, 6.0-in.-diameter ^{239}Pu shell) was used again. The container was assumed to have the smallest volume, thus providing for the maximum numbers to be staged within a vault or magazine. For modeling purposes, the Rocky Flats AL-R8 package was selected. The smallest container of this series is the 2030, nominally a container with a 20-in. diameter and a 30-in. height. Actually, the 2030 is an 18-gauge (0.048-in. thickness) steel can with a minimum inside diameter of 18.12 in. and an inside height of 27.21 in. This container is lined with Celotex and contains a minimum of other hardware. The pit is assumed to be centered in the container in a holding fixture.

The maximum number of AL-R8 containers that may be staged in a given facility generally is subject to administrative limits (e.g., in assembly/dismantlement cells and bays) and/or shelving and other hardware constraints. The ultimate limit is established by the physical volume of the facility. Several bounding physical arrangements of the containers were modeled for normal, abnormal, and severe accident conditions. The specific scenarios are discussed below.

The KENO and MCNP computer models (developed and validated as described in the earlier paper) for the pit-and-container combinations incorporated additional simplifying assumptions. The outer shell of the AL-R8 2030 container is assumed to be a right circular cylinder with a steel wall of uniform thickness and without perturbations (i.e., rings, bolts, clamps are neglected). The modest amount of refractory fiber insulation in the drum over the pressure relief vent is modeled by an equivalent amount of Celotex.

Scenarios were based on administrative limits and actual or potential physical conditions that could exist in the various facilities. Pits that are not involved in assembly/disassembly operations, subject to testing, or installed in explosive devices are staged in vaults or magazines in approved storage containers. These storage facilities have nominal capacities ranging from 154 to 440 AL-R8 containers. Arrays include containers in close-packed planar arrays on the floor or shelving, multi-level concentric rings on shelves, and in multi-unit pallets (which provide for both spacing and ease of handling).

The calculations predicted that all of the following configurations are subcritical:

- a single AL-R8 2030 staging container.
- arrays of bare, undamaged AL-R8 2030 staging containers stacked upright to as much as four-high in a close-packed or palletized infinite planar array. Water flooding of the interstitial space between containers actually reduces the multiplication factor. Flooding the pit and container with water isolates the units (as shown by the fact that a single such container and an infinite array thereof have essentially the same multiplication factor). These results (and a parametric study of low-density interspersed water moderation) indicate that the presence of water or other moderating material among the intact containers decreases, rather than increases, multiplication.
- finite arrays up to 16 x 16 x 10 in unmoderated close-packed configurations, which is considerably more containers than the maximum number that could be placed into any of the staging facilities.
- arrays of severely compressed (i.e., by 1/3 in diameter, corresponding to over 50% in volume) containers stacked upright in a one-high infinite planar array or in a large 16 x 3 x 7 tumbled array, both tightly reflected by thick concrete.
- intact DP20 containers in any number or geometric arrangement.

Additionally, an infinite X-Y-Z dry array was predicted to be only slightly above the subcritical limit. Accounting for the material composition of, and the spacing provided by, pallets would be likely to produce a subcritical value. Further, it is shown that interstitial water moderation up to and including full flooding would lead to a less reactive configuration.

Essentially all of the configuration of pits in AL-R8 containers that were modeled are subcritical by a substantial amount (and actually significantly more so due to the conservative model assumptions, e.g., pits all ^{239}Pu isotopic composition, large mass, and small volume; small volume containers). Thus, it is concluded that a critical configuration involving pits (as reported in the earlier paper) and the pit/container combinations (in this paper) is not credible. Overall, this conclusion is primarily attributable to the solid form of the fissile material, i.e., metallic shells, and to the ruggedness of the containers.

Session 5: Criticality Safety Software and Development

VIM — MONTE CARLO NEUTRON TRANSPORT CODE
(Viewgraphs)

R. Blomquist
Argonne National Laboratory

VIM – PRIMARY FEATURES

- Primarily reactor neutronics → easy reaction rate, balance, and cross-section edits
- Also neutron or photon shielding
- Detailed continuous-energy physics data:
  - probability tables in unresolved resonance range
  - pointwise data in resolved resonance range
  - thermal data processing from modified FLANGE-II
  - 118 nuclides/materials
  - neutron data from ENDF/B-V (or -IV)
  - photon interaction data (from MCPLIB)

- Flexible geometry:
  - combinatorial geometry
  - lattices of hexagonal or rectangular combinatorial cells
  - plate lattice (ZPPR criticals)
  - infinite medium
  - SRS supercell (periodic)

VIM ESTIMATES – ALL CALCULATIONS

- k_{eff}, including optimal combination of pairs of track length, collision, and analog estimates
- Group reaction rates and cross sections by region:
  - macroscopic & isotopic
  - universal tallying helps code speed

- Group scalar fluxes by region
- Total leakage, absorption, production, fission
- χ(E), leakage(E):
  - 1-σ error estimates for all quantities estimated

OPTIONAL ESTIMATES

- Volume-integrated group net currents by region
- Scattering matrices (microscopic)
- Integral reaction rate ratios, e.g., c^8/f^9, f^8/f^9

VIM – NEWER CAPABILITIES

- Group-to-group transfer tallies:
  - inelastic, n,2n, and P_L (or μ-binned) elastic scattering
  - $\bar{\mu}_{gg}$
  - event splitting for variance reduction
  - isotopic cross sections produced
  - error estimates corrected for correlation with scattering rate

- Multigroup calculations, with cross sections from ISOTXS, CASMO output, or ASCII input
- k_{eff} variance assessment:
  - lag-1 generation k_{eff} correlation
  - corrects estimated error

- Elevated temperature data:
  - some actinides up to 2000K
  - some coolants

VIM APPLICATIONS

- ZPPR criticals: k_{eff}, reaction rate distributions, detector fluxes
- EBR-II: core physics, nodal methods benchmarking using interface currents; intermediate sodium activation
- NPR HWR & MHTGR core physics, moderator temperature reactivity coefficients, cross section preparation
- General geometry collision probability method benchmarking
- Reduced Enrichment Research & Training Reactor core physics
- IFR Fuel Cycle Facility criticality hazards assessment
- Intense Pulsed Neutron Source: criticality, power densities from subcritical multiplication, moderator fluxes, counter-diversion analysis
- Space reactor shielding
- TMI-2 ex-core detector response to downcomer water level
- Boron Neutron Capture Therapy flux calculations
- Multigroup cross section generation
- CASMO benchmarking at Studsvik of America

VIM CODING

- FORTRAN 77, except for a few lines (dynamic memory allocation & timing) → portability. In production on Suns & IBM RS6000s. Has run on Cray, CDC, IBM 3084.
- > 3 times faster than MCNP, ~>5 times faster for reactor calculations

- Parallelized for distributed memory Multiple Instruction Multiple Data machines, i.e., RANetwork, IBM SPl:
  - work partitioned by tasks consisting of tens to thousands of histories
  - user control of task size: large tasks reduce message passing; but small tasks provide natural load balancing
  - scalable performance for up 10 processors on RANetwork

- Extensive input checks; lost-particle coordinates and direction shown
- Quality:
  - exhaustive benchmarking vs experiment and other codes
  - under configuration control
  - routine short test problem stream for code modifications not affecting random walks, long benchmark tests for more extensive code changes

- Documentation:
  - user's guide
  - validation bibliography
  - extensive internal comments

VIM BOUNDARY CONDITIONS

- Combinatorial geometry:
  - vacuum
  - white reflection
  - specular reflection
  - Savannah River supercell periodicity

- Repeating lattice geometries:
  - vacuum or specular on various combinations of x, y, z, and hex surfaces
  - periodic on various combinations of x, y, z, and hex surfaces

Sample Detail

	U-235	U-238	Pu-239	Ni
Smooth Data Points	4392	15083	5156	12954
Unresolved Resonance Probability Tables	137	36	94	0
Elastic Angular Distribution Tables	19	29	21	84
Inelastic Angular Distribution Tables	42	121	213	488
Secondary Energy Tables	77	817	4771	643
KBytes	146	291	190	479

VIM Nuclear Data Materials List
Fissionable Nuclides

Isotope	300	560	1000	1500	2000	other
Pu-238	x					
Pu-239	x	x	x			
Pu-240	x	x	x			
Pu-241	x	x	x			
Pu-242	x	x	x			
U-233	x	x	x			
U-234	x	x	x			
U-235	x	x	x	x	x	
U-236	x	x	x			
U-238	x	x	x	x	x	800
Th-232	x	x	x	x	x	
Np-237	x					
Am-241	x					
Am-243	x					
Pa-233	x					
Cm-244	x					
UO_2	thermal x	x	x			

All data ENDF/B-V; 300K and 1000K data available for ENDF/B-IV

VIM Nuclear Data Materials List
Coolants and Moderators

Isotope		Temperatures (K)					
Na-23		300					
He-4		300					
Be-9		300					
K		300					
H_2O	thermal	300	390	560			
D_2O	thermal	300	341	390	438		
C_6H_6	thermal	300					
Be Crystal	thermal	300					
BeO	thermal	300					
graphite	thermal	300	900	1000	1200	1500	2000
CH_2	thermal	300					
ZrH	thermal	300					

All data ENDF/B-V; 300K data available from Version IV

VIM Nuclear Data Materials List
Structure, Absorber, etc.

Cr	Ni	Fe	A-127	Hf-174	Hf-l74	Hf-l76	Hf-l77
Hf-l78	Hf-l79	HF-180	O-16	C-12	Mo	Mn55	B-10
B-ll	Ta-l81	Cu	H-l	Pb	Bi-209	Ti	Si
Li-6	Li-7	N-14	Au-197	Mg	Sm-l49	Eu-151	Eu-153
He-3	H-2	Ca	V	Co-59	F-19	Cd	Cd-ll3
In-ll3	In-ll5	W-182	W-183	W-184	W-186	Gd-l55	Gd-l52
Gd-l54	Gd-l56	Gd-l58	Gd-l60	Ag-l07	Ag-l09	Cs-133	Nb-93
Gd-l57	Xe-l35	Eu-152	Tb-l59	Eu-154	Re-l85	Re-l87	Rh-l03
Ta-l82	Tc-99	Dy-l64	Lu-175	Ba	Ga	Zr-90	Zr-91
Zr-92	Zr-94	Zr-96	Dy-l60	Dy-l61	Dy-l62	Dy-l63	Er-l67

All at 300K, some also at higher temperatures
All data ENDF/B-V; ENDF/B-IV also available

k_{eff} Comparisons: U Metal Criticals

Critical	VIM σ		MCNP[1] σ		SCALE[1] σ	
SIMP.1	0.97628	0.00079	0.9779	0.0020	0.98366	0.00283
SIMP.1 (Revised)	0.99569	0.00091				
SIMP.2 (H_2O)	1.00087	0.00106	0.9980	0.0024	1.00410	0.00398
SIMP.3 (graphite)	0.99906	0.00083	1.0013	0.0024	0.99967	0.00275
SIMP.4	0.99553	0.00067	0.9933	0.0018	1.00329	0.00272
SIMP.5 (H_2O)	0.99494	0.00089	0.9933	0.0018	1.01183	0.00383
SIMP.6 (graphite)	0.99467	0.00082	1.0002	0.0025	1.01626	0.00343
SIMP.7	0.99174	0.00065	0.9905	0.0022	0.99761	0.00310
SIMP.8	0.99490	0.00069				
SIMP.9	0.99273	0.00067	0.9964	0.0019	0.99246	0.00306
SIMP.10	0.99507	0.00084	0.9966	0.0019	0.99322	0.00287
SIMP.11	0.99604	0.00080	0.9938	0.0021	1.00236	0.00313
SIMP.12	0.99581	0.00069	0.9953	0.0020	1.00263	0.00288
MIH.20 (poly)	0.99689	0.00090	0.9927	0.0023	1.00221	0.00291
MIH.53 (graphite)	1.00076	0.00065	1.0001	0.0022	1.00663	0.00295
MIH.59 (graphite)	0.99755	0.00084	0.9996	0.0026	1.01693	0.00322
ARRAY.2	0.99698	0.00072	0.9982	0.0020	1.00209	0.00309
A.12 (paraffin)	1.00600	0.00090	1.0085	0.0028	1.01750	0.00351
A.51	0.99246	0.00140	0.9946	0.0020	1.00160	0.00279
ROT.2 (H_2O/concr)	1.00469	0.00098	1.0094	0.0035	1.00995	0.00379

[1] *Validation of MCNP, A Comparison with SCALE*, by C. Crawford and B. M. Palmer, WINCO-1110, October, 1992.

VIM k_{eff} for Various Criticals

Critical	k_{eff} (σ)	$(k_{eff} - 1)/\sigma$
Jezebel	1.0008 (0.0014)	0
Flattop-EU	1.0072 (0.0061)	1
Flattop-Pu	1.0040 (0.0040)	1
Godiva	0.9972 (0.0007)	4
Jemima(12)	0.9969 (0.0051)	0
Jemima(37)	0.9944 (0.0037)	1
Jemima(53)	0.9943 (0.0025)	2
LTR-II-A	1.0008 (0.0020)	0
IPNS-01	1.0028 (0.0024)	1
IPNS-02	1.0018 (0.0020)	0
IPNS-03	1.0030 (0.0038)	0
ORR	1.0043 (0.0024)	1

RETALLY

- VIM tally postprocessor
- Allows for retrospective tally statistical processing:
  - energy group collapse
  - sum (or average) over unions of regions
  - skip early batches
- Invokes VIM statistical edit package which produces regular VIM edits
- Input produced by VIM, requiring minimal modification

KEFCODE

- VIM k_{eff} postprocessor
- Allows for retrospective k_{eff} statistical processing:
  - skipping early batches
  - aggregate sequential batches
  - skip later batches

XSEDIT

- VIM material file editing program
- ASCII-to-binary
- Binary-to-ASCII
- Binary or ASCII to formatted print
- DISSPLA plot of pointwise data

FILEONE AND BANDIT

- Library collection and preparation for a VIM problem library
- Energy bands (supergroups) for memory conservation
- Up to 40 isotopes in a library

PICTURE

- Line printer zone or composition layouts
- Planar snapshots

SABRINA

- Color renderings of 3-D geometries
- Cuts allow viewing internals of geometry

LOCAL PLOTTING CODE

- Limited to a few body types which define reactor lattices

KENO DEVELOPMENTS

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Oak Ridge National Laboratory

The series of KENO multigroup, criticality-safety transport codes has been used continuously for over 25 years. During this time KENO has progressed from its original form as a small, highly specific code to the general-purpose criticality code embodied in the latest version, KENO-V.a. Development and maintenance of KENO-V.a is an ongoing effort. Additionally, KENO-V.a is an integral part of the SCALE package, which is also being continually updated and improved. The modifications and developments over the past year relating to the following areas are addressed in this presentation: (1) modifications to KENO-V.a, (2) development of KENO-VI, (3) modifications to the CSAS4 sequence of the SCALE package, and (4) future work on KENO-related programs.

No significant development work has been done on KENO-V.a. The majority of effort here has been devoted to maintenance. A shortcoming involving the use of holes has been remedied. Previously, tangent or touching holes could produce incorrect results. This shortcoming has been removed by reworking the hole-crossing algorithm in subroutines TRACK and CROS. Updated versions of these subroutines will be included in SCALE 4.3 when it is released.

A new version of KENO, called KENO-VI, has been developed and should be ready for release through the Radiation Shielding Information Center in the fall of this year. KENO-VI has all the abilities of KENO-V.a with a more general geometry package. KENO-VI is capable of representing any system that can be modeled by using sets of quadratic equations. A set of 13 geometry shapes is available in KENO-VI. Other shapes can be constructed using sets of quadratic equations. These shapes can be rotated and/or translated to any orientation and position. In addition to rectangular-pitched arrays, triangular-pitched arrays can now be explicitly modeled. The use of array boundaries enables arrays to completely fill regions whose boundaries do not coincide with those of the array. A SCALE version of KENO-VI will be released with SCALE 4.3.

A new search type is currently being developed in SCALE to allow CSAS4 to do a concentration search on a mixture component. The search iterates through the modules BONAMI-S, NITAWL-II, XSDRNPM-S, KENO-V.a, and MODIFY, updating the cross sections in each pass. Plans are underway to similarly modify the PITCH and DIMENSION searches to update the cross sections at the beginning of each pass.

Development work on the series of KENO codes continues. Work has already begun on developing a continuous-energy version of KENO-V.a. Development of continuous-energy cross sections for use in this version of KENO-V.a has also begun. A state-of-the-art graphics package is to be added to KENO-V.a. Plans are being developed to interface an existing graphics package with KENO-V.a that is capable of creating interactive 2-D slices and rotatable images of a system.

Development and maintenance of the series of KENO criticality safety codes are ongoing tasks. The above-mentioned enhancements will be incorporated into KENO as manpower and funding allow.

COG DEVELOPMENTS

W. R. Lloyd
Lawrence Livermore National Laboratory

COG is a Lawrence Livermore National Laboratory (LLNL) Monte Carlo computer code that runs on Hewlett-Packard and SUN workstations. It solves the Boltzmann equation for transporting neutrons and photons. It uses pointwise cross sections from the ENDF/B-V library exactly as evaluators present the data. It solves deep penetration (shielding) and nuclear criticality problems.

COG geometry descriptions use either analytic surfaces up to the fourth order or pseudo-surfaces described by boxes, finite cylinders, and topographic surfaces. The geometry can be verified by perspective sector, or material pictures in black-and-white, or in color.

The computer-aided design software Pro/ENGINEER from Parametric Technology, Inc., is combined with the LLNL code Pro/COG to produce geometry input in the proper format for COG.

Three critical experiments for low-enriched fuel rods in water were calculated with COG and the pointwise cross-section set ENDF/B-V. Some characteristics of these critical experiments are presented in Table 1 below.

Table 1. Characteristics of the Critical Experiments.

TRX-1 & -2 WAPD-TM-931 (1970).	BAW 1484-7 (1979) Experiment XIII.
Al Clad 0.387"POD 0.453"ROD 48"long.	1728 Rods 3x3 Bundles 14x14 Rods/Bundle
1.291 w/o U-235 in U.	Al Clad 0.405"POD 0.475"ROD 60" long.
-TRX-1 0.711" Triangular Pitch 763 rods.	2.46 w/o U-235 in U in UO2.
-TRX-2 0.856" Triangular Pitch 577 rods.	1.6 w/o Boron in Boral Plates between bundles.

Some results of these calculations are presented in Table 2 below. They are compared with results using the KENO-V.a code taken from Ref. 1.

These results compare well with these three experiments, and they are within the range of the KENO-V.a results. Benchmarking work for COG against critical experiments is continuing.

Table 2. Benchmarking Calculation Results.

	WAPD-TM-93 (1970)				BAW 1484-7 (1979)	
	TRX-1		TRX-2		Experiment XIII	
	K_{eff}	1 σ	K_{eff}	1 σ	K_{eff}	1 σ
COG, ENDF/B-V	0.9981	.0036	0.9961	.0028	0.9952	0.0033
KENO-V.a, 27GROUPDF4	0.9831	.0032	0.9873	.0030	0.9793	0.0038
123GROUPGMTH	1.0028	.0032	0.9935	.0031	1.0008	0.0041
218GROUPNDF4	0.9761	.0038	0.981	+.0035	0.9789	0.0045

REFERENCE

1. W. R. Lloyd, "Determination and Application of Bias Values in the Criticality Evaluation of Storage Cask Designs," UCID-21830, Lawrence Livermore National Laboratory, Livermore, California, January, 1990.

RECENT DEVELOPMENTS IN THE LOS ALAMOS RADIATION TRANSPORT CODE SYSTEM

R. A. Forster and K. Parsons
Los Alamos National Laboratory

The Los Alamos Radiation Transport Code System (LARTCS) integrates the DANT (Diffusion Accelerated Neutral Transport) discrete ordinates codes with the MCNP (Monte Carlo N-Particle) code. Both codes have a long history of research, development, and application. Since the solution methods of discrete ordinates and Monte Carlo are complementary in a number of ways, the LARTCS is a flexible and powerful tool for solving criticality and fixed-source problems.

The LARTCS is being developed under the umbrella of a graphical user interface (GUI) for problem setup and analysis. This interface simplifies the input and reduces the opportunities for incorrect user problem specifications. The GUI has been under development for over a year and will allow the simultaneous development of both DANT and MCNP input descriptions. The GUI has been tested and analyzed by about a dozen Group X-6 staff. The present version of the GUI is named JUSTINE and will allow the user to set up, view, rotate, and zoom in on geometries in both 3-D solid and two 2-D cut-plane views simultaneously. The software will be portable and will not need any special expensive graphics hardware. Group X-6 anticipates a prototype GUI will be available for customer testing in November 1994.

There has been progress in the DANT system for criticality applications. TWODANT/GQ (for generalized quadrilateral) is available in X-Y or R-Z geometries. This new capability makes it possible to represent nearly any 2-D geometry because the mesh cells can have arbitrary quadrilateral shapes.

The TWODANT and THREEDANT code modules can be linked to a mesh-generation code called FRAC-IN-THE-BOX. FRAC accepts nested region body input and applies a user-specified mesh to the geometry. An interface file is produced which is then read by either TWODANT or THREEDANT. Hence, X-Y or X-Y-Z orthogonal mesh models of almost any combination of nested bodies can be generated. Cells with more than one material are homogenized, but the cell material masses are preserved.

A new iteration scheme that saves considerable computer time for criticality safety problems has been implemented into the DANT system. The normally tight convergence for all the pointwise fluxes can be relaxed for criticality applications. Criticality results for k_{eff} and the fission distribution can now be obtained 20 to 50% faster with no loss in numerical accuracy in the k_{eff} result. There are also new mass and neutron production edits, as well as new print and cross-section file name options.

A DANT System Criticality Tutorial was held at the 1993 San Francisco ANS meeting. About 60 people attended the day-long session on the DANT system, the GUI, and applications. All modules, including THREEDANT, were run on scientific workstations. An early prototype of the GUI was also demonstrated. A total of 28 demo copies of the DANT system (except TWODANT/GQ) were distributed to interested attendees. Discussions about the DANT system are ongoing with interested users concerning code availability and different computer platforms.

MCNP Version 4A was released to RSIC on 10/1/93. The primary focus was on code quality. Any bug found in MCNP can result in a $4 cash award if it really is a bug and has not been found before. The test set of problems has been substantially enhanced to test more combinations of features. The new laws required by ENDF/B-VI physics have been incorporated and tested. The LANL release of the ENDF/B-VI library is expected in the November 1994 time frame. Sixteen-group Hansen-Roach data will also be available at that time.

A new focus put into MCNP4A is on assisting the user in determining if the calculated Monte Carlo results are statistically correct. MCNP now checks criticality problems to determine if all cells with fissionable material have produced at least one fission source point during the calculation. A warning message is produced on the new k_{eff} summary page if all cells have not been sampled. Each of the three MCNP k_{eff} estimators—collision, absorption, and track length—are checked to determine if the batch values appear to be normally distributed at the 99% confidence level. This is the expected result for a converged spatial fission source. If the batch values for all three appear not to be normally distributed, final k_{eff} confidence intervals are not printed in the MCNP output. The first and second active halves of the problem are compared to see if both the mean and estimated standard deviation appear to be the same. If not, a warning is printed. The k_{eff} results are also calculated for a worst-case analysis of each of the three largest k_{eff} *s* occurring on the next cycle. This is useful for assessing an upper confidence interval based on the k_{eff} *s* sampled so far.

New MCNP4A tally assessment features involve the relative variance of the variance and the empirical history score probability density function. Both have been incorporated into 4A and are used to analyze the statistical convergence of tally results.

MCNP4A currently runs on many computing platforms, including Cray, VAX, HP, Sun, IBM 6000, DEC, Silicon Graphics, and IBM PCs and clones. MCNP4A can use PVM to distribute one problem to several workstations. An installation package was developed to make it very easy to install and test MCNP. MCNP4A timing studies are presented by Hendricks and Brockoff in the April 1994 issue of *Nuclear Science and Engineering*.

In addition to three 1991/1992 Los Alamos National Laboratory (LANL) MCNP Benchmark Reports, new MCNP criticality documentation is available, or soon will be. The completely rewritten 4A manual contains new or enhanced documentation about MCNP criticality calculations and the new checks. A new criticality primer for MCNP is nearing completion. This work has been done with Chuck Harmon and Bob Busch (University of New Mexico). This primer

(see Bob Busch's summary for comments on the primer in Session 6) will be used in upcoming MCNP criticality courses. A new 120-page Los Alamos report by Urbatsch et al. on the three combined k_{eff} estimators used in MCNP is being finished. WINCO has published a four-volume set of MCNP comparisons with SCALE, and INEL has performed an MCNP analysis of the Foehn Experiment.

MCNP training classes have been presented throughout the past year on various topics, including introductory MCNP, variance reduction, and criticality. The next MCNP criticality class in Los Alamos will be August 8-12, 1994. Please contact Judi Briesmeister (jfb@lanl.gov) for more information. Courses have been presented to the AECL in Toronto and Winnipeg (1993) and in Sweden (April 1994). Future courses are scheduled for Tokyo, Japan (1994), and Stuttgart, Germany (1995). On-site courses for the LARTCS can be arranged with LANL on request.

In the future, LANL will have to make a distinction between paying customers and nonpaying users. This is required because of increased budgetary restrictions. Our intent is to make production versions of our codes available from RSIC. Intermediate versions, hotline help, newsletters, classes, new feature developments, and applications will be available only to our customers. LANL is presently formulating an agreement for organizations who wish to join our LARTCS Customer's Group. Relatively small contributions from a large number of organizations will enable LANL to continue to support and develop our codes and data bases, as well as to assist our customers in obtaining the best numerical solutions possible. LARTCS work-in-progress includes requests from the LANL criticality group ESH-6, finishing the GUI, completing a CRADA with Schlumberger-Doll Research, solving various applications problems, presenting training classes, and performing validation calculations.

Anyone interested in information about X-6 should contact the X-6 Group Leader, Bob Little (rcl@lanl.gov). Information on the DANT system can be obtained from Deputy Group Leader Brad Clark (bac@lanl.gov), Kent Parsons (dkp@lanl.gov), Forrest Brinkley (fwb@lanl.gov), and Ray Alcouffe (rea@lanl.gov). MCNP information is available from Monte Carlo Team Leader John Hendricks (mcnp@lanl.gov), Judi Briesmeister (jfb@lanl.gov), Art Forster (raf@lanl.gov), and Gregg McKinney (gwm@lanl.gov). Other persons to contact are the Nuclear and Atomic Data Team Leader Bob Clark (rehc@lanl.gov) and the Graphics Team Leader Stephen Lee (srlee@lanl.gov).

ENERGY-POINTWISE DISCRETE ORDINATES TRANSPORT METHODS

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An accurate determination of the space-dependent flux spectrum throughout an array of fissionable components is one of the most important and basic quantities required in criticality safety analysis. Knowledge of the detailed energy spectrum within the various fissionable and absorber components is needed to determine realistic reaction rates and resonance-shielded multigroup cross sections for subsequent criticality analysis performed with multigroup codes such as KENO. Due to the presence of resonance materials such as uranium and plutonium, the energy spectrum generally exhibits very complex, fine-structure effects within the resolved resonance range that will vary spatially from region to region. Although pointwise Monte Carlo codes such as MCNP can in theory accurately include these effects directly in the transport calculation, multigroup codes such as KENO and all deterministic codes must rely on properly averaged multigroup cross sections to reflect the impact of resonance self-shielding. The great difficulty involved with determining the complicated behavior of the flux spectrum has led to the use of rather simplistic approximations for averaging multigroup cross sections. For instance, equivalence theory and the narrow resonance approximation are inherent in the widely used Bondarenko approach, and the old Nordheim integral method assumes isolated resonances and is limited to only a single absorber component surrounded by moderator. These two methods are currently utilized in the SCALE system to self-shield multigroup cross sections for criticality calculations. Errors introduced into the problem-dependent, self-shielded cross sections by the approximations will propagate into errors in the calculated value of the multiplication factor. Hence, there is strong motivation to develop a more rigorous approach to obtain accurate problem-dependent spectra for multigroup cross section generation.

A new one-dimensional code called "CENTRM" has been developed that computes a detailed, space-dependent flux spectrum in a *pointwise-energy representation* within the resolved resonance range, coupled to a fine-group multigroup calculation above and below the pointwise range. The code uses discrete-ordinates transport theory with an arbitrary angular quadrature order and a Legendre expansion of scattering anisotropy up to P7 for moderator materials and up to P3 for heavy nuclides. The elastic scattering source moments in the pointwise range are evaluated with a new, efficient algorithm called a "sub-moment expansion" developed for s-wave center-of-mass scatter kernels. Pointwise nuclear data is rigorously processed from ENDF/B into a specially formatted CENTRM file, and multigroup data for the non-pointwise range can be obtained from any desired "Working Library" generated by the AMPX code system. For example, the criticality safety libraries in the SCALE system can be used directly in CENTRM.

The CENTRM program provides unprecedented capability to deterministically compute full energy range, space-dependent angular flux spectra in one-dimensional geometries, rigorously accounting for resonance fine-structure and scattering anisotropy effects. The code will become a

component in the SCALE system to improve the computation of self-shielded cross sections used in criticality safety calculations, thereby enhancing the accuracy of such codes as KENO.

Several applications to lattices of low-enriched fuel rods are discussed at the workshop presentation. In these examples, an energy mesh of approximately 15,000-20,000 energy points is used in the flux calculation, with an S8 quadrature and P3 scattering. It is shown that the CENTRM-produced multigroup cross sections give critical eigenvalues that agree within about 0.15% of MCNP calculations. Comparisons of CENTRM results to critical benchmark measurements also show good agreement but suggest that the U-238 capture data in ENDF/B-VI predicts more resonance capture than the experiment.

Session 6: Criticality Safety Studies at Universities

CRITICAL EXPERIMENTS WITH MIXED OXIDE FUEL

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One alternative for the disposal of excess (~100 MT) weapons-grade plutonium (<7 wt% Pu-240) is to burn it as mixed fuel in power reactors (PWRs).[1] The plutonium remaining in discharge fuel would be denatured by increased Pu-240 content (>20 wt% Pu-240) resulting from long residence times. The increased cost from the introduction of plutonium into the fuel cycle would be partially offset by the sale of electricity. Early studies of the use of plutonium in PWRs showed the advisability of a number of modifications in plant design and operation. Several considerations which relate to core physics and safety are (a) higher fissile-to-fertile ratios, (b) lower beta effective, and (c) enhanced use of burnable poisons. Recent studies emphasize the use of distributed Er_2O_3 burnable poison, an important effect of which is to make the temperature coefficient of reactivity more negative.[2] This change occurs because the negative effect of the twin capture resonances in Er-167 at 0.5 eV cancel the positive effect of the 0.3-eV fission-capture resonance in Pu-239.

It is prudent to back up core physics analyses with critical experiment measurements of power shapes, coefficients of reactivity, and critical states. Such analyses[2] for the proposed System 80+ plutonium burner were benchmarked by comparison with results from the Saxton,[3] WREC,[4] and Rensselaer Polytechnic Institute (RPI)[5,6] borated and unborated critical experiments. The Saxton experiments used fuel with relevant fuel composition (6.6 wt% PuO_2 + $U_{nat}O_2$, 90.5 wt% Pu-239 + 8.5 wt% Pu-240) and the RPI experiments used normal enrichment UO_2 fuel with relevant Er_2O_3 concentrations. No critical experiments have yet been conducted for fuel with weapons-grade plutonium and Er_2O_3 together, at various dissolved boron levels, and for specific fuel assemblies such as the ABBCE fuel assembly with its five large water holes. Here we examine the technical considerations involved in carrying out such experiments at the RPI Reactor Critical Facility (RCF). The topics dealt with are the core, the measurements, safety, security, radiological matters, and licensing. It is concluded that the experiments are feasible at RPI.

A representative core could consist of an ABBCE 16x16 fuel assembly surrounded by a 4.81wt% enriched UO_2 driver lattice of SPERT(FI) fuel pins, all in 1/8 core symmetry. All pins would be 6.75wt% Pu in HM + depleted UO_2 at 0.2 wt% tails, 93.5 wt% Pu-239 + 6.5 wt% Pu-240 and normal diameter.[2] Core support, water treatment, control, and instrumentation would be normal.[5] The experiments would be conventional as follows:

a. approach to critical,
b. control rod worths,

c. isothermal temperature coefficient of reactivity,
d. fuel pin worth,
e. void coefficient of reactivity,
f. pin-wise power shape, and
g. absolute power calibration, all at various boric acid levels in the water up to about 300 ppm.

Some of these experiments are carried out solely to satisfy Tech Spec requirements as startup measurements to verifying pre-calculated safety parameters. The control rods are fully withdrawn in the experiments after (b), so all measurements are done on rising periods. The entire campaign of experiments is estimated to involve about 100 periods performed in one calendar month. The total energy production in the campaign would be about 25 W per fuel pin, so the fuel is essentially unchanged. There is negligible fission product inventory at any time, and after a few days the radiation from the pin will decay back to previous levels.

RCF security must be upgraded to Category 1 in accord with 10CFR73.60.[7] Two or more round-the-clock guards are required with adequate training and drills. Required modifications to security hardware include (a) three-strand wire on the top of the security fence, (b) enhanced motion sensors, and (c) bullet-resistant glass on the guard building. The radiological safety requirements at RPI meet or exceed the requirements of 10CFR20. The only upgraded hardware for radiological safety would be better alpha monitoring sensors. The Emergency Procedures should be modified to include ruptured PuO_2 fuel pins. It is anticipated that no information security would be required.

Document submittals would include

a. Amendments to License CX-22 and technical specifications,
b. Amendments to security plan and procedures (10 CFR 73 App C),
c. Modifications to the Safety Analysis Report to note the presence of Pu (no change in the design basis accident, safety limits, or consequences are required),
d. Modified emergency procedures.

In summary, critical experiments at the RCF on weapons-grade plutonium mixed-oxide fuel assemblies appear to be technically and administratively feasible. They would be of appropriate quality and at relatively little cost.[9]

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STUDENT RESEARCH IN CRITICALITY SAFETY AT THE UNIVERSITY OF ARIZONA

D. L. Hetrick
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This is a brief progress report on four student projects at the University of Arizona:

1. simulations of power pulses in aqueous solutions,
2. the effect of assembly shape on the expansion coefficient of reactivity for solutions,
3. some reactivity computations for SHEBA, and
4. computations in support of the French experiment to measure temperature coefficients of dilute plutonium solutions.

The contributing students are Robert Kimpland (now a post-doctoral fellow at Los Alamos National Laboratory), Drew Kornreich (a doctoral student and DOE fellow at the university), and Sung Lee (candidate for a master's degree at the university).

1. Kimpland's dissertation was completed in the summer of 1993. His two-dimensional model for solution excursions shows improvements over previous one-region models. Expansion reactivity coefficients from TWODANT computations may now be used in computations without empirical adjustments. A second improvement is that the computed results for the delayed-neutron tail are closer to experimental data. Thirdly, the pressure-time curves are broader than before (closer to experimental data).

2. Simulation of criticality accidents requires knowledge of shutdown coefficients. The volume expansion contribution to shutdown is a function of assembly shape as well as composition (more important for tall, thin cylinders and less important for squat cylindrical shapes). TWODANT computations have been performed for uranium solutions (various enrichments) and for plutonium solutions, all for various fuel concentrations and aspect ratios. The results may be correlated by simple one-speed diffusion models. The goal is to present these correlations in a form suitable for use in accident predictions that do not require transport theory calculations.

3. We computed critical heights for the Los Alamos Critical Experiments Facility SHEBA assembly, both as it was suspended above its concrete-lined well and when lowered into the well. We used an extremely simple model (a bare cylinder of solution without any structure). We computed a decrease in critical height of 0.74 cm, or alternatively a reactivity increase of 65 cents (a sensitivity of 88 cents/cm). These results are within a factor of two of the preliminary measurements. More refined calculations are needed.

4. Experimental measurements of temperature coefficients in a dilute plutonium solution are planned at Valduc, France. The assembly is a water-reflected cylinder of radius 34 cm and reflector thickness 31 cm. Our computations employed a 69-group model for the spectral part of the temperature coefficient. A typical result (15 g/liter of Pu, 80 percent Pu-239, critical height 76.5 cm) shows expansion feedback of -0.0156 $/°C, spectral feedback +0.0670 $/°C, and net feedback +0.0514 $/°C. The proposed experiment therefore appears to be feasible, but its performance will require care.

CRITICALITY SAFETY RESEARCH AT THE UNIVERSITY OF TENNESSEE-KNOXVILLE

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During the past year at the University of Tennessee-Knoxville, graduate students, faculty, and a visiting scientist from Japan have worked on seven different research projects in the area of nuclear criticality safety. These projects are listed below along with the primary beneficiary of each project (i.e., the customer) which is indicated in parentheses:

1. Analysis of a Hypothetical Criticality Accident in a UF_6 Freezer-Sublimer (Portsmouth)
2. Shutdown Mechanisms for a Hypothetical Criticality Accident involving HEU Powder (Y-12)
3. Analysis of a Hypothetical Criticality Accident in a Waste Super-Compactor (Rocky Flats)
4. Criticality Safety Evaluation of the ^{233}U Inventory at ORNL using ENDF/B-V Cross Sections (ORNL)
5. An Update of a Slide Rule for Estimating Criticality Accident Dose Information (NRC/ ORNL)
6. Space-Dependent Kinetics Analysis of a Hypothetical Criticality Accident Iinvolving an Array of Bottles Containing UO_2F_2 (K-25)
7. KENO-V.a Code Development on a Parallel Computer (ORNL)

The first five projects listed above will be described in detail in papers *presented by students* at the national ANS meeting in New Orleans, LA, in June 1994. Preliminary results for project No. 6 showing power versus time are presented in Fig. 1 in order to illustrate results for one of our projects. The transient is for a ramp perturbation of 0.5 \$/s in a seven-bottle array of aqueous U (4.98%) O_2F_2. The results indicate that space-time effects are significant beyond $t \cong 70$ s while a simple point kinetics model appears adequate prior to $t \cong 70$ s. These results were obtained with a new code which combines neutronics from the PAD[1] code and thermal-hydraulics from the SKINATH-AR code.[2]

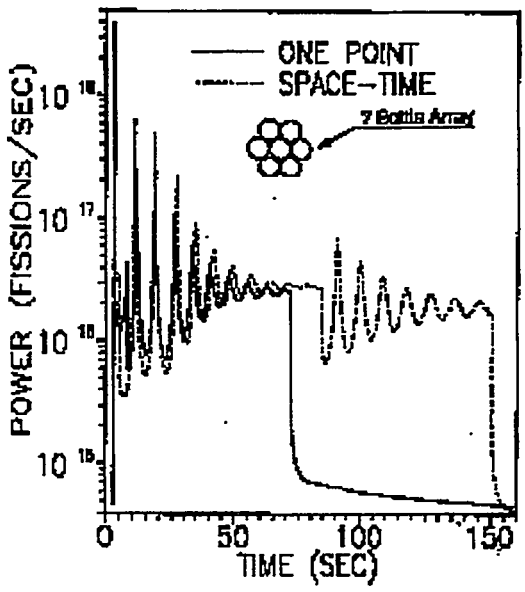

Figure 1. Power vs time (0.5\$/s, 7-bottle array.

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NUCLEAR CRITICALITY RESEARCH AT THE UNIVERSITY OF NEW MEXICO

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INTRODUCTION

Two projects recently undertaken at the University of New Mexico are worthy of note. The university's Chemical and Nuclear Engineering Department has just completed the final draft of a primer for MCNP4A, which it plans to publish soon. The primer was written to help an analyst who has little experience with the MCNP code to perform criticality safety analyses. In addition, the department has carried out a series of approach-to-critical experiments on the SHEBA-II, a UO_2F_2 solution critical assembly at Los Alamos National Laboratory. The results obtained differed slightly from what was predicted by the TWODANT code.

Criticality Calculations with MCNP: A Primer

With the closure of many experimental facilities, the nuclear criticality safety analyst increasingly is required to rely on computer calculations to identify safe limits for the handling and storage of fissile materials. However, in many cases, the analyst has little experience with the specific codes available at his/her facility. This primer will help the analyst understand and use the MCNP Monte Carlo code, Version 4A, for nuclear criticality safety analyses. It assumes that the analyst has a college education in a technical field. There is no assumption the reader is familiar with Monte Carlo codes in general or with MCNP in particular. Appendix A gives an introduction to Monte Carlo techniques. The primer is designed to teach by example, with each example illustrating two or three features of MCNP that are useful in criticality analyses.

Beginning with a "Quickstart" chapter, the primer gives an overview of the basic requirements for MCNP input and allows the reader to run a simple criticality problem with MCNP. This chapter is not designed to explain either the input or the MCNP options in detail; but rather it introduces basic concepts that are further explained in following chapters. Each chapter begins with a list of basic objectives that identify the goal of the chapter and a list of the individual MCNP features that are covered in detail in the unique chapter example problems. It is expected that on completion of the primer the reader will be comfortable using MCNP in criticality calculations and will be capable of handling 80 to 90% of the situations that normally arise in a facility. The primer provides a set of basic input files that can be selectively modified to fit the particular problem at hand.

Although much of the information required to do an analysis is provided in the primer, there is no substitute for understanding a particular problem and the theory of neutron interactions. The

MCNP code is capable only of analyzing the problem as it is specified; it will not necessarily identify inaccurate modeling of the geometry, nor will it know when the wrong material has been specified. Remember that a single calculation of k_{eff} and its associated confidence interval with MCNP or any other code is meaningless without an understanding of the context of the problem, the quality of the solution, and a reasonable idea of what the result should be.

The primer provides a starting point for the criticality analyst using MCNP. Complete descriptions are provided in the MCNP manual. Although the primer is self-contained, it is intended as a companion volume to the MCNP manual.[1] The primer provides specific examples of using MCNP for criticality analyses while the manual provides information on the use of MCNP in all aspects of particle transport calculations. The primer also contains a number of appendices that give the user additional general information on Monte Carlo techniques, the default cross sections available in MCNP, surface descriptions, and other reference data. This information is provided in appendices, so it is hoped that the reader finds the primer useful and easy to read. As with most manuals, users will get the most out of it if they start with Chapter One.

SHEBA-II: APPROACH TO CRITICAL

The approach-to-critical experiment yielded critical heights that were extremely close to SHEBA-II's actual critical height for all three cases examined (Table I). Modeling the system using TWODANT predicted larger values than the system needed to reach a critical state in all three configurations and failed to register the reflective nature of the concrete crypt that is seen in the actual values as SHEBA-II is placed in it. Above ground, the system was critical at 43.72 cm, while below ground it reached critical at 42.40 cm; however, TWODANT runs predicted higher values of 44.2 cm and 44.0 cm, respectively.

The UO_2F_2 fuel solution is worth more when SHEBA-II is in the concrete crypt than when it is above ground (0.43 \$/cm versus 0.50 \$ and 0.474 \$/cm, respectively). This increase in worth is likely due to the reflection of neutrons back into the system from the concrete surrounding it in the pit. The decrease in worth–when the polyethylene lid is placed on top of the pit–of about 0.25 \$/cm corresponds to the slightly larger solution height needed for the system to be critical in this configuration and could be the result of fission product buildup or temperature increase.

Table I. Summary of critical heights and solution worths obtained during this analysis of SHEBA-II.

System Configuration	Exp. Estimate	TWODANT	Actual	Solution Worth (\$/cm)
Above Ground	43.5±0.2 cm	44.2 (0.17\$/cm)	43.72 cm	0.4367
In Crypt Without Lid	42.25±0.15 cm	44.0 (0.35\$/cm)	42.46 cm	0.5
In Crypt With Lid	42.5±0.10 cm	44.0 (0.33\$/cm)	42.52 cm	0.474

From these results, it is evident that the approach-to-critical procedure is a valuable and quite accurate method for determining the amount of fissile material needed for a system to reach critical. TWODANT is a useful tool in predicting the behavior of a system as fuel material is added but fails to predict the actual critical height accurately. Perhaps adding more of the SHEBA II systems structure, such as its fuel tanks, would improve the accuracy of the TWODANT model, or else three-dimensional transport codes, such as MCNP, might be predictors of the critical height by allowing the evaluation of a more realistic system model.

REFERENCE

1. J. F. Briesmeister, editor, *MCNP - A General Monte Carlo Code for Neutron and Photon Transport, Version 4a*, Los Alamos National Laboratory document LA-12625-M, December 1994.

Session 7: Training

TRAINING AT THE Y-12 PLANT

A. Harvey
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We regret that a summary of Ms. Harvey's presentation could not be made available for these proceedings.

Editor's note: Ms. Harvey's presentation was to be based on a videotape used to train workers in criticality safety at the Oak Ridge Y-12 Plant. As she explained to the conference, the videotape took as its jumping-off point a 1958 criticality accident at the plant. Although the videotape was not classified, she said, her supervisors nonetheless forbade its showing to the NCTSP Workshop. She said she deeply regretted their decision. Other participants in the conference openly echoed her feelings.

CRITICALITY SAFETY TRAINING

S. K. Woodruff
Los Alamos National Laboratory

Summary

Criticality safety training is an important element of the Plutonium Facility safety program at Los Alamos National Laboratory. Training consists of student self-study handbooks and hands-on performance-based training in a mock-up laboratory containing gloveboxes, trolley conveyor system, and self-monitoring instruments. A 10-minute video tape and lecture is presented to describe how training in this area is conducted.

TRAINING OF NUCLEAR CRITICALITY SAFETY ENGINEERS

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Historically, new entrants to the practice of nuclear criticality safety have learned their job primarily by on-the-job training (OJT), often by association with an experienced nuclear criticality safety engineers who probably also learned their jobs by OJT. Typically, the new entrants learned what they needed to know to solve a particular problem and then accumulated experience as more problems were solved. Because more formalism will likely be required in the future, a site-specific analysis of the nuclear criticality safety engineer job was performed and is being used to develop training classes for newer engineers.

The analysis indicated that there are four major components:

1. analysis - assessment of fissile material activities to establish limits and conditions;
2. surveillance - examination of fissile material activities for adherence to established limits and conditions;
3. business practices or administration - integration of the results of analysis with facility operations, e.g., procedures postings, training, how things are supposed to be done; and
4. emergency preparedness - nuclear criticality accident alarm system and emergency responses.

The analysis component was further subdivided into process analysis, accident analysis, and transportation analysis. At this time, the process analysis component is of most interest. By repeatedly asking the question "What does a nuclear criticality safety engineer need to know to do process analysis?," 10 subject-matter areas were identified as candidates for class development, as shown in Fig. 1.

Seven classes have been prepared and delivered to the target audience of newer nuclear criticality safety engineers. These classes address the subject matter areas of basic nuclear criticality concepts, compilations of critical data, and part of basic subcritical limits guides shown in Fig. 1. Response to the training approach has generally been favorable, and the students seem to genuinely appreciate an emphasis on the practical.

The job content analysis has emphasized that nuclear criticality safety, like any other specialized field, has a set of basic information which is not readily recognized by new entrants. The training classes developed from the results of the job content analysis have demonstrated that the specialized information can be successfully delivered to new entrants.

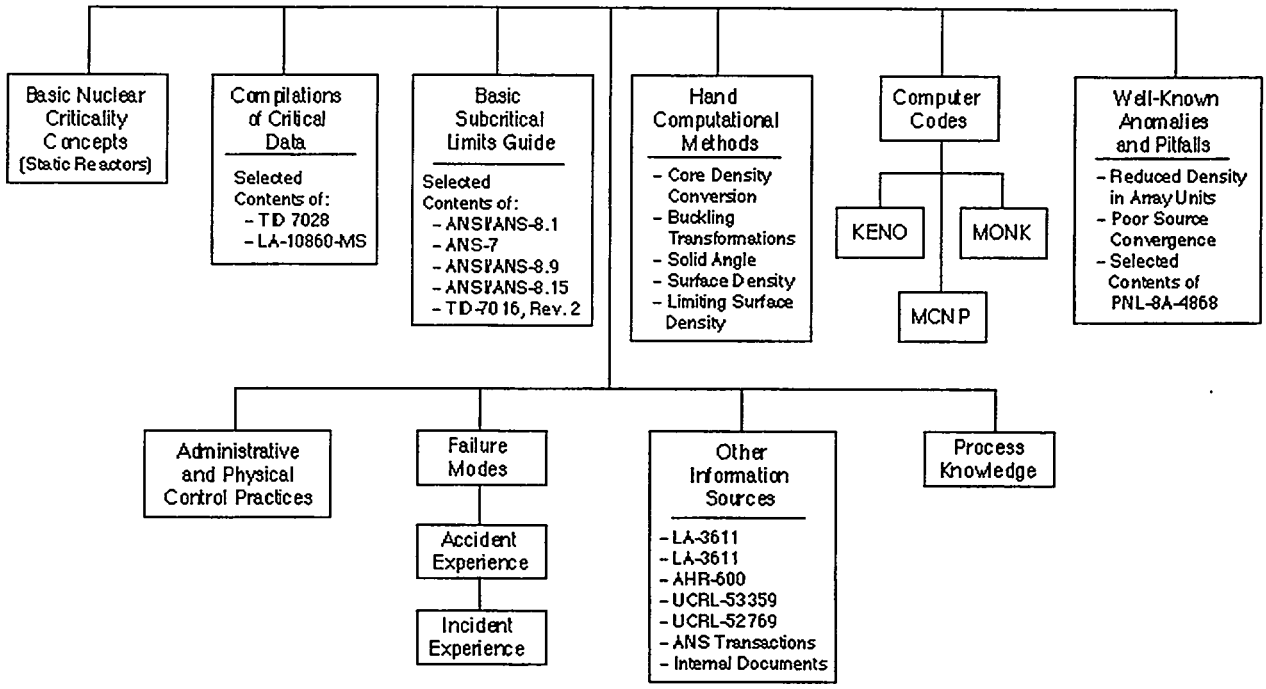

Figure 1. Nuclear Criticality Safety Engineer Process Analysis Job.

NUCLEAR CRITICALITY SAFETY COURSE DESCRIPTIONS

H. L. Dodds

The following two classes are given in the Nuclear Engineering Department at the University of Tennessee-Knoxville.

NE 421: Introduction to Nuclear Criticality Safety

Fundamentals of nuclear criticality safety; criticality accidents; safety standards; overview of experiments, computational methods, and applications.

Text: *Nuclear Criticality Safety - Theory and Practice*, by R. A. Knief, The American Nuclear Society, 1986.

Credit: 3 semester hours

Prerequisites: NE 301, NE 302

NE 543: Selected Topics in Nuclear Criticality Safety

Criticality safety computational and experimental methods for enrichment, fabrication, storage, reprocessing, and transport applications; overview of safety practices and regulatory requirements.

Text: Handout notes provided by instructors plus NE 421 text (by R. A. Knief)

Credit: 3 semester hours

Prerequisites: NE 421

Appendix I

MEETING MINUTES
CRITICAL EXPERIMENT NEEDS IDENTIFICATION WORKGROUP (ENIWIG)
May 9, 1994
1:00 to 2:45 p.m.

Chair:	Debbie Rutherford
Vice Chair:	Richard Taylor
Secretary:	Ernie Elliott

Meeting was convened at 1:05 PM; attendance list is given as Attachment 1.

Ms. Rutherford welcomed the participants and presented an outline of topics to be discussed during the course of the meeting. She requested that if additional needed critical experiments had been identified that they should be listed on the appropriate form and given to her at the meeting or sent in later. The main points of the presentation were as follows (a copy of the presentation is given as Attachment 2).

Summary of Forecast Document

Ms. Rutherford gave a quick overview of the experiments presented in "Forecast of Criticality Experiments and Experimental Programs Needed to Support Nuclear Operation in the United States of America: 1994-1999" (Los Alamos National Laboratory document LA-12638), listing the number of proposed experiments in each of the categories as well as the current prioritization for each. The experiments listed in the document and ranking of experiments within the particular categories were the subject of an extensive meeting in Golden, Colorado, in July 1993, so this particular subject was not readdressed. Questions were raised about the number of experiments that could be performed in a given year. Consensus was that about three experimental series could be performed per year, although this number would vary greatly according to the number of individual critical assemblies that needed to be constructed. Some experimental series may involve only a few (5-10) assemblies, whereas others may require hundreds. Another issue raised concerned the DNFSB Recommendation 93-2. This recommendation addresses not only actual performance of critical experiments as a priority but also maintenance of the capability (personnel, facilities, etc.) to conduct experiments. An additional consideration regarding future capability is conduct of critical experiments for currently unforeseen and specialized situations, with radioisotope production given as an example.

Status of Critical Mass Laboratories

Ms. Rutherford moved on to the status of critical experiment facilities around the country, beginning with the Los Alamos Critical Experiments Facility (LACEF).

LACEF - One training class has been conducted since the beginning of the fiscal year and some experiments are also being performed. LACEF is reported to be operating well, considering the

prevalent regulatory environment. SHEBA, Comet, Flattop, and Big Ten have operated recently. No operations with Skua and Godiva are planned, since Kiva 3 is undergoing restoration at the present time. The importation of LEU fuel pins to the Pajarito Lab (LACEF) is being encouraged with potential applications including burn-up credit experiments and LEU fuel-pin array criticals.

Rocky Flats Plant Critical Mass Laboratory - The facility is operable in that the equipment is still in working condition, but it is dead from a regulatory perspective. Shipment of highly enriched uranyl nitrate (HEUNH) solution has lost funding recently and is not being currently pursued. Rocky Flats is switching from Defense Programs to Environmental Management moneys and this has led to uncertainty about disposition of HEUNH. Storage of this material is technically sound for the long term but is out-of-date procedurally.

Other facilities and comments - Beattis and KAPL (unpressurized) critical facilities have been shut down. Rensselaer Polytechnic Institute (RPI) may be available to do some experiments. The Russians and French may be contacted concerning the feasibility of contracting some work. LACEF has been approached by the Navy for some experiments. Part of the Sandia National Laboratories CX machine is also being shipped to LACEF.

DOE Response to DNFSB Recommendation 93-2

Burt Rothleder of DOE provided this information. Mr. Rothleder stated that a Nuclear Criticality Experiments Steering Committee (NCESC) had been formed under Defense Programs. It consists of two subcommittees: (1) Training and (2) Methodology and Experiments (MES). The task MES has undertaken is to extract from LA-12683 a short list of experiments to initially fund and expand the LA-12683 write-up to be more specific. This list is presented as Attachment 3. He also suggested that perhaps analytical work (calculations) could replace the need for some of the experiments proposed in LA-12683. Mr. Rothleder stated that the steering committee is dependent on the experiment needs working group for direction and information. He said that the existence of LA-12683 had given the steering committee essentially a one-year head start in their work. Otherwise, a similar document would *have to* have been produced by DOE. Ms. Rutherford distributed the DOE "short list" of experiments at the beginning of this meeting.

A question was raised about when the decision would be made by DOE concerning experiment funding. Mr. Rothleder responded that the decision should be made by FY 1995. The current source of this funding is unknown, but that the force of safety and economics will eventually lead to funding. Details of experiment selection by DOE (the "short list") will be given during the NCTSP meeting tomorrow (5/10/94). A request was made for more information on the training subcommittee and when training would commence. Mr. Rothleder said that the subcommittee was formed from many components within DOE and that an appeal for funding on a temporary basis had been made. Dick Malenfant added that one training class had been held at LACEF in February 1994. Funds allocated for that particular training course ($50,000) have been spent in conducting the course and associated facility upgrades. He also said that Tom McLaughlin has proposed holding one class per month, depending on the availability of funds.

Reaffirm/Redraft ENIWIG Charter
(The current charter for ENIWIG is listed in Appendix F of LA-12683).

Ms. Rutherford led the members of the working group through the different paragraphs of the charter. Discussion began and continued for quite a while concerning the Purpose and Scope sections of the charter. Comments made by attendees indicated that both sections should be made as generic as possible to include all parties that have interest in experiments that would provide more data for application to criticality safety. The other sections (membership, responsibilities, etc.) required only minor corrections. It was agreed that the Purpose and Scope would be re-drafted in light of comments from the membership and be distributed for comment at the NCTSP meeting on 5/10/94. The newly drafted charter is presented as Attachment 4.

The meeting was adjourned at 2:43 PM.

Respectfully submitted,

Debbie Rutherford, Chair — 5/23/94 Date

Richard Taylor, Vice Chair — 5/19/94 Date

Ernie Elliott, Secretary — 5/18/94 Date

ATTACHMENT 1
EXPERIMENTAL NEEDS IDENTIFICATION WORKGROUP ATTENDEE LIST

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Fort Magruder Inn
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ATTACHMENT 2

Agenda - *Experimental Needs Identification Workgroup*

- **Welcome and Introductions**
- **Summary of "Forecast" Document - LA-12683**
- **Brief Status of Critical Mass Laboratories**
- **Status of Critical Experiments**
- **DOE Response to DFNSB 93-2**
- **Reaffirm/Redraft the ENIWIG Charter**
- **Call for New Experiments and Experimental Programs**
- **Next Scheduled Meeting**
- **Conclusions**

Experiments and Experimental Programs Identified by ENWIG That Address DNFSB Recommendations

DNFSB Recommendation	Experiments or Experimental Programs
". . . maintain a good base of infomation 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,
". . . to ensure retaining a community of individuals competent in practicing the [criticality] control."	505, 601, 605, 605a, 609, 702, 703, and 704 All experiments and experimental programs, 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

Identified and Prioritized Experiments and Experimental Programs

Categories		Number of Priority		
		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 (58)		23	27	8

Highest Priority Experiments and Experimental Programs

Category	Experiment	Experimental Program or Experiment Title
HEU	104	Advanced Neutron Source
	106	TOPAZ-II Reactor
LEU	206	SHEBA Reactivity Parameterization
	207	SHEBA Reactivity Void Coefficient
P	301	Plutonium Solution in the Concentration Range from 8 g/L to 17 g/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

cont.

Highest Priority Experiments and Experimental Programs (cont.)

Category	Experiment	Experimental Program or Experiment Title
T/A	501	Assessment for Materials Used to Transport and Store Discrete Items and Weapons Components
	Prog. 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 Emphasize Intermediate Energies
	Prog. 503	Validation of Criticality Alarms and Accident Dosimetry
	Prog. 504	Accident Simulation and Validation of Accident Calculations
T/A	Prog. 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 of Calculational Methodology in the Intermediate Energy Range
CP	702	Spent Fuel Safety Experiments (SFSX)

ATTACHMENT 3
EXPERIMENT RATING SYSTEM

1Exp = ill-defined subcriticality margin: rating = 8;
2Exp = uncertain protection by well-defined subcriticality margin: rating = 5;
3Exp = discrepant validation of subcriticality margin: rating = 3;
4Exp = criticality safety enhancement through economic gain: rating = 2;
5Exp = enhancement of criticality safety knowledge base: rating = 1;
6Exp = economic gain, independently: rating = 0;
Undecided (U) or Independent of the rating system (I).

The Exp ratings may be multiple, except for those of the 1Exp and 2Exp categories since these categories are mutually exclusive. Since multiple ratings can allow an experiment with a set of lower category ratings (e.g., 3Exp+4Exp+5Exp) to outscore an experiment with a single 5Exp rating if a 1,2,3,4,5 rating system were used, a Fibonacci series will be used to set the ratings (i.e., 1, 2, 1+2=3, 2+3=5, 3+5=8).

The following experiments (in LA-12638) are Project-dependent-only (Proj-do): 104, 106, 201, 202, 204, 305, 401, and 402. The priorities for these experiments are driven by an "engine" different from that driving the remaining experiments. All other experiments are Project-independent (Proj-ind). Proj-ind experiments, however, include two subclasses: Process-dependent (Proc-dep) — 203, 302, 502d, 502e, 502f; and Machine-dependent (Mac-dep) — 105, 206, 207, 502c, and 608. These two subclasses should not be used as discriminators unless specific Process or Machine requirements so warrant.

Experiment 305 should be changed from Pu (300-Series) to HEU (100-Series).
Experiment 201 should be changed from Leu (200-Series) to HEU (100-Series).

Exemplary categorization by Burt Rothleder:

1Exp = ill-defined subcriticality margin: rating = 8;
104, 105, 106, 201, 202, 204, 207, 301, 401, 402, 505, 601, 605a, 609, 701, 702.
2Exp = uncertain protection by well-defined subcriticality margin: rating = 5;
101, 102, 103, 203, 205, 302, 303, 502a, 502d, 502e, 502f, 502g, 502h, 502i, 503, 504, 506, 602, 607.
3Exp = discrepant validation of subcriticality margin: rating = 3;
101, 102, 502a, 605b, 606, 707.
4Exp = criticality safety enhancement through economic gain: rating = 2;
203, 303, 501, 502, 504, 702.
5Exp = enhancement of criticality safety knowledge base: rating = 1;
103, 105, 202, 203, 204, 205, 207, 208, 301, 501, 502, 502a, 502b, 502c, 502d, 502e, 502f, 502g, 502h, 502i, 503, 505, 506, 601, 602, 603, 604, 605, 605a, 605b, 606, 607, 608, 609, 701, 702, 703, 704, 705, 706.

6Exp = economic gain, independently: rating = 0;
502b, 502c.

Undecided (U) or Independent of the rating system (I): rating = 0;
304(I), 305(U), 306(I), 403(U), 507(I), 508(I).

ATTACHMENT 4
CHARTER
Experiment Needs Identification Workgroup
Nuclear Criticality Technology and Safety Project

I. Purpose

The purpose of the Experiment Needs Identification Workgroup is to:

- Identify new criticality experiments and experimental programs needed to support U.S. nuclear facilities.
- Serve as the national focal point for experiment and experimental programs requests.
- Publish a list of the experiment and programmatic needs identified.

II. Scope

The workgroup will identify and prioritize criticality experiments and experimental programs needed to ensure:

- The safe operations of new activities and revisions to existing activities involving fissionable materials in U.S. facilities.
- Criticality safety training.
- Criticality safety with respect to standards and regulations.
- Resolution of criticality physics problems.
- Advancement of criticality safety technology.

III. Membership

Membership will be from personnel or organizations with a vested interest in nuclear criticality safety.

IV. Responsibilities

- The Chair coordinates Workgroup activities.
- The Vice Chair serves in the absence of the Chair.
- The Secretary prepares and distributes meeting minutes.
- The Workgroup reports to DOE through the NCTSP.
- Members attend Workgroup meetings, contribute to the Workgroup report, identify experiment and experimental program needs, prepare programmatic and experiment justification statements, will participate on a voluntary basis, elect a Chair, Vice-Chair, and Secretary.

V. Report

A report listing the identified and prioritized experiments and experimental programs will be sponsored and published through funding from the Nuclear Criticality Technology and Safety Project.

VI. Meetings

The Workgroup will meet at least annually.

This draft Charter for the Experiment Needs Identification Workgroup was reviewed and affirmed at the workgroup meeting on May 10, 1994.

D. A. Rutherford, Chair

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