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Conference

Proceedings of the Nuclear Criticality Technology and Safety Project Workshop

Monterey, California
April 16–20, 1993

Los Alamos
NATIONAL LABORATORY

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Compiled by
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Abstract

This report is the proceedings of the annual Nuclear Criticality Technology & Safety Project (NCTSP) Workshop held in Monterey, California, on April 16–28, 1993. The NCTSP was sponsored by the Department of Energy and organized by the Los Alamos Critical Experiments Facility. The report is divided into six sections reflecting the sessions outlined on the workshop agenda.

Forward/Acknowledgments

This report contains the proceedings of the 1993 Nuclear Criticality Technology & Safety Project (NCTSP) Workshop, which was held in Monterey, California, from April 26-28, 1993. The Workshop was attended by 141 people and included representatives from the Department of Energy, the national laboratories, and nuclear power industry corporations.

The Workshop was divided into seven major sessions as outlined in the agenda. Several overview talks were presented in each session. In addition, eight working groups met on the first day of the Workshop for discussions in the following areas:

- Physics Criteria for Benchmarks
- Evaluation Techniques
- Experimental Needs
- Training
- Parametric Studies
- Neutron Physics Theory
- Unresolved Questions from Previous Meetings
- Rules, Regulations, and Standards

Because of budget constraints, the proceedings of this workshop contain only the summary papers of the presented talks when they were supplied by the speaker. Copies of the transparencies used in the talks are available on request to all attendees. Please contact Rene Sanchez (505-665-5343) if you would like to have a copy of transparencies for a particular talk.

For their support, commitment, and participation, we would like to thank the working group and session chairpersons: Bob Wilson, Adolf Garcia, Les Brown, Robert Seale, James Bryson, Burton Rothleder, Richard Stark, Mike Westfall, Jor-Shan Choi, Debra Rutherford, Mayme Crowell, John Pearson, Hans Toffer, John Mihalczo, and Gordon Hansen. We also wish to thank our guest speakers, Michael Lineberry and John Schinkle, for their participation in the 1993 NCTSP Workshop.

Rene G. Sanchez

Richard E. Malenfant

Organizing Committee

Agenda

Nuclear Criticality Technology Safety Project Workshop
April 26 - 28, 1993

Monday, April 26
Special Session
Chairperson Bob Wilson(NRC)/Harry Calley (DOE/HQ)
Criticality Safety With Spent Fuel High-Level Radioactive Waste, and Decontamination and Decommissioning Activities

Speaker	Topic	
Jerry Hicks (EG&G Rocky Flats)	Issues on Decontamination & Decommissioning	1800-1820
Mike Chandler (WSRC)	Savannah River Site High-Level Radioactive Waste Nuclear Criticality Safety Program	1820-1840
Terry Vail (WHC)	Nuclear Criticality Safety Issues at Hanford High-Level Radioactive Waste Tank Farms	1840-1900
Hans Toffer (WHC)	Measurements of Hanford High-Level Radioactive Waste Tank Fissile Materials	1900-1920
Todd Taylor (WINCO)	Criticality Safety Issues with a Change of Mission at the Idaho Chemical Processing Plant	1920-1940
Dick Libby (PNL)	Criticality of Nuclear Fuel During Geologic Isolation	1940-2000

Tuesday, April 27

Registration		0730 -0830
Introduction and Opening Remarks	Burton Rothleder/DOE/HQ	0830 - 0845
Welcome	Richard Malenfant/LANL/LACEF	0845 - 0900
Keynote Speaker	Richard Stark/Director Safety Policy and Analysis, US DOE, NE-74	0900 - 0945
Announcements and Break		0945 - 1000

Session I, Chairperson, Adolf Garcia, ANL
Criticality Code Development, Usage and Validation

Speaker	Topic	
Lester Petrie (ORNL)	KENO Development and Validation	1000-1020
Judith Briesmeister (LANL)	MCNP4A-What's in it for you?	1020-1040
Brad Clark (LANL)	ONEDANT, TWODANT, and THREEDANT - Application to Criticality Safety Problems	1040-1100
John T. Mihalczo- Gordon Hansen (ORNL/LANL)	Central Reactivity Worth Measurements	1100-1120
Burton M. Rothleder (DOE/HQ)	Software Verification, Validation, and Documentation	1120-1140
Blair Briggs (EG&G)	The Defense Program Sponsored Criticality Safety Benchmark Evaluation Project	1140-1155
Session Questions and Answers		11:55-1200
Lunch		1200 - 1300

Session II, Chairperson, C. Les Brown
Experimental Needs (LACEF Program Review)

Speaker	Topic	
Sol Pearlstein (DNFSB)	Defense Nuclear Facility Safety Board Recommendation 93-2, "The Need for Critical Experiment Capability"	1300-1320
Ernie Elliott (Martin Marietta)	Efforts to Validate Keno V. A Code for (1) HEU-C-H Systems and (2) HEU-C Systems at High C/U Ratios	1320-1340
Paul Felsher (EG&G Rocky Flats)	Experimental Needs for Validation Purposes from the Rocky Flats Plant Perspective	1340-1400
Hans Toffer (WHC)	Fissionable Measurement Needs in Hanford Waste Tanks	1400-1420
Mark Robinson	How More Objective Criticality Safety Analyses Can Benefit Future Waste Transportation and Storage	1420-1440

Robert Rothe (EG&G Rocky Flats)	Critical Mass Laboratory at Rocky Flats	1440-1500
Break		1500-1520
Dae Chung (DOE)/Blair Briggs (EG&G Idaho, Inc.)	Comments on DOE-DP Programs	1520-1540
Jeffrey Philbin/Richard Coats (SNL)	Operational and Safety Characterization Plans for the SPR-III Fast Burst Reactor	1540-1600
Richard Anderson (LANL)	New Experiments at the Los Alamos Critical Experiments Facility (LACEF)	1600-1620
	Minimum Critical Mass Experiments	
Rene Sanchez (LANL)	Source Jerk Measurements on Highly Subcritical Systems	1620-1640
Arnold Robba (LANL)	Radiation Accident Dosimetry and Simulation	1640-1700
Richard Malenfant (LANL)		1700-1720
Banquet	Banquet Speaker	1800 - 2000
	John Schinkle/Director, Safety Programs Division, US DOE Albuquerque Operations Office. "Managing Risk with Conduct of Operations."	

Wednesday, April 28

Announcements		0800 - 0815

Session III, Chairperson, Robert Seale, University of Arizona
Accident Analysis

Speaker	Topic	
Richard Paternoster (LANL)	Status of the Pajarito Site Safety Analysis Report	0815 - 0845
G. R. Imel (ANL)	Criticality Safety Considerations in the IFR Fuel Cycle	0845-0915
Jeffrey Philbin (SNL)	Methods for Performing Criticality Related Accident Analysis for the SNL Hot Cell Safety Analysis Report	0915-0945
Discussion time will be available after each paper and at the end of the session		0945-1000
Break		1000 - 1015

Session IV, Chairperson, James Bryson, SNL
Experimental Facilities and Measurements

Speaker	Topic	
Michaele C. Brady (SNL)	Fuel Safety Critical Experiment (FSX): A New Initiative	1015-10:35
Edward J. Parma (SNL)	A Critical Assembly Designed to Measure Neutronic Benchmarks in Support of the Space Nuclear Thermal Propulsion Program	1035-1055
Gary S. Hoovler (B&W)	Measurements in a New Compact Heterogeneous Thermal Reactor	1055-1115
Roy A. Haarman/Joseph L. Sapir (LANL)	TOPAZ-II Fuel Loading and Other Criticality Issues	1115-1135
John T. Mihalczo (ORNL)	Advance Neutron Source (ANS) Reactor Critical Experiment Needs	1135-1155
Kenneth B. Butterfield (LANL)	Godiva IV Temperature Coefficient	1155-1215
Lunch	Luncheon Speaker	
	Dr. Michael Lineberry, "New Developments in the Advanced Reactor Program."	1215-1315

Session V, Chairperson, Burton M. Rothleder, US Department of Energy
University Programs

Speaker	Topic	
David Hetrick (Univ. of Arizona)	Computer Simulation of Nuclear Power Pulses in Solutions	1315-1335
Tracy Wenz/Robert Busch (Univ. of New Mexico)	Lady Godiva & Reactivity Worth Calculations	1335-1355
Marcus Voth (Penn State Univ.)	Reconfiguring the TRIGA Core for Unique Applications	1355-1415
Donald Harris (Rensselaer Polytechnic Inst.)	Criticality Safety Experiments at the RPI Reactor Critical Facility (RCF)	
H. Lee Dodds (Univ. of Tennessee)	An Overview of the Nuclear Criticality Safety Program at the University of Tennessee	1415-1435
		1435-1455
Donald J. Dudziak/Steven G. Walters (North Carolina State Univ.)	Activities at North Carolina State University Relevant to Criticality Safety	1455-1515
Break		1515-1530

Session VI, Chairperson, Richard Stark/DOE
Open Questions From Previous Workshops
Panel Discussions

Speaker	Topic	
Robert Rothe (EG&G/Rocky Flats)	Documentation of Experiments	1530-1600
Jor-Shan Choi (LLNL)	Challenges to the Analyst to Extrapolate Data	1600-1630
Richard Stark (DOE/HQ)	Discussion and Results on DOE Order 5480.24	1630-1700

Registration Times and Working Group Session Schedule

Monday, April 26	Chairperson	Meeting Room*	Time
Registration			0730 - 0800
Physics Criteria for Benchmarks	M. Westfall	Room #1	0800 - 1000
Evaluation Techniques	J. Choi	Room #2	0800 - 10:00
Experimental Needs	C. Les Brown/ D. Rutherford	Room #3	0800 - 10:00
Training	M. Crowell	Room #1	1300 - 1500
Parametric Studies	J. Pearson/ H. Toffer	Room #2	1000 - 1200
Neutron Physics Theory	J. Mihalczo	Room #3	1000 - 1200
Lunch			1200 - 1300
Unresolved Questions from Previous Meetings	R. Malenfant	Ballroom	1300 - 1430
Break			1430 - 1445
Rules, Regulations and Standards	B. Rothleder	Ballroom	1445 - 1700

* Consult reader board adjacent to registration desk.

NCSTP 1993
CONFERENCE PROCEEDINGS

CRITICALITY SAFETY PROBLEMS IN RESTORATION, DECONTAMINATION, AND DECOMMISSIONING (HIGH-LEVEL WASTE TANKS)

ALTERNATE TECHNIQUES FOR MEASURING FISSIONABLE MATERIAL IN HANFORD WASTE TANKS

Hans Toffer
Westinghouse Hanford Company

During the 50 years of plutonium production at the Hanford Site, the 177 waste tanks have received, as part of the waste streams, some fissionable materials. Criticality limits have been established for the tanks in terms of concentration and total mass. However, it has been difficult to demonstrate compliance to these limits because of uncertainties in tank inventory numbers and the lack of sampling data. To improve information of how much fissionable materials is in the tanks and infer data about material distributions, evaluation of alternate measurement techniques concept has been initiated. The fissionable materials are mainly plutonium and uranium isotopes containing small amounts of neptunium and americium. The measurements are not only important to demonstrating compliance with criticality limits but also to support future retrieval of materials from the tanks and pretreatment prior to final disposition.

Application of measurement techniques is difficult in the tanks. Access to tank volume is limited to a few entry ports and maybe only one of these will have an uncontaminated observation well. Furthermore, the tanks contain significant quantities of fission products resulting in high gamma ray fluxes.

Fissionable materials can be detected by the transmutations and fission they undergo. Characteristic gamma rays, alpha particles, neutron and gaseous materials given off by the nuclear processes can serve as detection mechanisms.

The concepts considered are mainly neutron based. They are differential in nature in that they will tend to characterize materials in a narrow cylinder around the observation well. They will, however, provide very useful data in the axial direction. The concepts considered involve passive neutron counting devices such as track recorders and foils, active systems using neutron sources and detectors, and pulsed systems using a pulsed neutron generator and detectors.

In addition to the differential measurements, integral measurements are being considered that measure certain gases given off by the tank waste. Such 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.

To support the analysis approach considered, detailed computer models using the MCNP code have been developed. These models have proven very useful in parametric studies calculating detector responses to various design conditions.

To establish fissionable material contents in the tanks and to demonstrate compliance with limits will require combination of information from multiple sources, including past data on tanks, differential and integral measurements, sampling data, and extensive criticality analyses.

SESSION I: CRITICALITY CODE DEVELOPMENT, USAGE, AND VALIDATION

KENO DEVELOPMENT AND VALIDATION

Lester Petrie
Oak Ridge National Laboratory

This presentation is in two parts, a discussion of the differences between KENOV a geometry and KENOVI geometry and a discussion of the validation of a new cross section library. KENOVI development is not finished, and verification and a generic validation have started. It is hoped that KENOIV will be released to the public this fall through RSIC.

Besides the KENOV body types, KENOVI has several new body types, such as CONE, ELLIPSOID, HEXPRISM, etc. Since all body types in KENOVI are converted to quadratics, it is a simple operation to add other bodies if needed. KENOVI defines space in a combinational fashion using the bodies specified. This is done with MEDIA definitions, ARRAY definitions, and HOLE specifications.

At Oak Ridge, a new 238 group library has been generated from ENDF/V data. A 44 group library has been collapsed from this librsary for use in calculations involving light water reactor fuel. The fine group library was collapsed to the broad group structure using a cell average spectrum for a Westinghouse fuel pin from a 17x17 element. Relative to the SCALE 27 group structure, some fast groups were added to cover two oxygen windows, and to represent the iron window. Thermal groups were added to cover the low energy side of the thermal Maxwellian. Both libraries have been used to calculate the CSEWG fast and thermal benchmarks. For the truly fast benchmarks, both libraries agree well with other results based on NEDF/B √. For these benchmarks agreement with experiment is also good. For more intermediate spectrum benchmarks, the 238 group library still agrees well with the other results, but the 44 group library shows some deviation from these results. For the thermal benchmarks, the agreement was again generally acceptable.

The 44 group library was then further validated against 88 critical experiments. There were 60 fuel pin lattice experiments, with very good agreement achieved for all cases. There were an additional 18 thermal critical experiments analyzed. Here the overall agreement was very good, but some of the individual experiments showed somewhat larger deviations.

MCNP4A—WHAT'S IN IT FOR YOU?

Judith F. Briesmeister
Los Alamos National Laboratory

The primary focus of the MCNP computer code is quality, followed by value and then new features. The features that are new in version 4A will be discussed.

MCNP is a widely distributed general-purpose, continuous-energy, generalized geometry, time-dependent Monte Carlo computer code. It transports neutrons, photons, and electrons in either single-particle or coupled-particle mode through three-dimensional geometry and can calculate k_{eff} eigenvalues for critical systems.

Quality is the main focus of MCNP. Quality assurance includes a procedure for how problems are identified and corrected, a process of deciding what new features will be added, followed by a rigorous testing method, and a procedure for change control, version identification, and controlled release. Benchmarking calculations compare MCNP results to experimental and analytic results and calculational results from other computer codes. We are also working to meet government and industry software quality assurance standards and code validation requirements.

The second priority for MCNP is what we call value and focuses on documentation and portability. The MCNP manual is being rewritten for 4A. A primer describing use of MCNP for criticality safety problems is under development. MCNP is maintained on many computer platforms. Recently a distributed processing version of the code was demonstrated that runs simultaneously on a cluster of IBM RS/6000 workstations. Performance ten times that of a single-processor Cray-YMP was achieved when MCNP was run on 16 workstations in parallel.

MCNP4A will have several major new features and hundreds of minor improvements and bug fixes. The major new features include:

- X-Window graphics, allowing the use of XLIB, a widely available graphics library for UNIX-based workstations.

- Improved criticality calculation (KCODE) output information,[9] including (1) k_{eff} as a function of active cycles and various batch combinations, (2) fission lifetime, (3) normality checks, and (4) new plot options.

- The parallel virtual machine (PVM) software from ORNL enables multiprocessing on distributed memory computer systems.

- SABRINA particle-tracking link. MCNP4A will write out history tapes for postprocessing either by the SABRINA color graphics code to illustrate particle tracks or by the LAHET (high-energy transport) code.

- Improved source and tally capability in repeated structures and lattice geometry.

- Variance of the variance. MCNP4A will have perhaps the most sophisticated statistical package ever in a Monte Carlo computer code. In addition to the usual statistical variance, the variance of the variance, the normality of samples and rates of convergence are calculated.

- The capability to read multiple electron data sets with physics models appropriate to ranges up to 1 GeV.

A long history and wide variety of applications in key sectors of the economy, coupled with vigorous active development, make MCNP a robust Monte Carlo code with many uses. The focus first on quality, then user value such as documentation and portability, and new features as the third priority ensures a reliable and stable code for our many users in the future.

CONTROL REACTIVITY WORTH MEASUREMENTS

Gordon Hansen and John T. Mihalczo
Los Alamos National Laboratory and Oak Ridge National Laboratory

INTRODUCTION

A near spherical unreflected and unmoderated uranium (93.7) metal configuration had been assembled to delayed criticality at Los Alamos National Laboratory in the 1950s.[1] Experiments with highly-enriched-uranium metal spherical shells had also been assembled.[2] Both of these experiments have been used to estimate the unreflected and unmoderated, highly-enriched-uranium spherical critical mass. The experiments described in this paper, although originally justified for leakage spectra measurements and to investigate the use of a multiplying booster with a linear accelerator, also can provide estimates of the unreflected and unmoderated, highly-enriched-uranium spherical metal critical mass.

The uranium metal sphere was assembled in various steps to ensure that a spherical near critical configuration was ultimately achieved. This sphere was unreflected and unmoderated. The original purpose of the sphere was to measure the neutron leakage spectrum in a slightly subcritical system to validate calculational models. These measurements were performed at the LINAC at the General Atomic Company in San Diego in 1965. For these measurements, the original sphere had a 1.000-in.-diam.* radial hole within 0.350 in. of the center to allow the beam from the linear accelerator to impinge on the uranium close to the center of the sphere. In 1971 and 1972, the sphere was modified by plugging the large radial hole, reducing the radius and providing a variety of small holes for various reactor physics measurements which continued until 1975. At this time the sphere was returned to the Y-12 plant in Oak Ridge, TN. There were essentially four configurations of the sphere: two earlier ones that were not exactly spherical and two later ones in which all major sphere parts had the same radius.

A variety of measurements were performed with this uranium metal sphere in addition to the delayed critical configurations. This system was assembled in a 35 x 35 x 30-ft-high east cell of the Oak Ridge Critical Experiments Facility. This paper describes the later two configurations of the sphere for which all major parts had the same radius of curvature in detail so that this experiment can be used to validate calculational methods and cross sections.

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*Lengths are given in inches, weights in grams, and density in grams per cm^3 since they were measured and reported in these units.

SOFTWARE VERIFICATION, VALIDATION, AND DOCUMENTATION

Burton M. Rothleder
U.S. Department of Energy, NE-74

SUMMARY

Software used in DOE Programs must satisfy audit requirements imposed by DOE, and must be capable of withstanding external scrutiny provided by peer review groups and instigated by intervenors.

Software that must be governed by Software Quality Assurance (SQA) practices is that used either to justify, challenge, or support justification of reactor and reactor related design, operations, safety and consequence analysis, or environmental impact analysis.

Specified documentation is required for design, operational, safety, and environmental software in a form appropriate to the age and complexity of the software. There are two general types of software documentation: Software Development Documentation and Software User Documentation. Certain User Documentation is always required: Software Summary, User's Manual, and Programmer's Manual.

Software must be verified, i.e., purged of programming errors and requirements errors, before it is validated, i.e., determined to represent physical reality within quantified bounds.

Validation consists of two components: software development validation which examines the functionality of the software independent of its engineering context, and software engineering validation which examines the software in terms of its functionality as an engineering tool.

The engineering validation, or benchmarking, activity is continuous, going beyond the baseline point of sufficient validation after readiness for use. Benchmarking consists of the products of user experience accumulated throughout the life of the code. Continuing benchmarking products can be used for validation of previously developed software, and, to a lesser degree, for validation of newly modified software. These benchmarking products can also be used as a surrogate for verification by demonstrating the robustness of the software.

Benchmarking can be categorized as analytic, experimental, or operational. Analytic Benchmarking is performed by comparing the results of an analysis by the code being benchmarked with the same results by a more exact, higher order code. Experimental Benchmarking is performed by comparing computer code or code sequence calculations to measurements of an experimental or test configuration. Operational Benchmarking is performed by comparing the results of code system calculations to measurements made on an operating device.

Analytic computer codes that can neither be comprehensively validated nor validated at all may use expert elicitation as a substitute for validation.

Uncertainty limits must be established for each measured and calculated parameter so that calculational inaccuracies can be judged for acceptability. Such limits must reflect the acceptance criteria for safety and operations. Uncertainty limits are necessary factors in judging whether a computer code has been adequately validated and is ready for release.

Three classes of software, and, correspondingly, of documentation, are recognized: Newly Developed Software (NDS), Previously Developed Software (PDS), and Newly Modified Software (NMS). NDS and

PDS each determine opposite ends of a spectrum with internal structure consisting of a continuum of NMS with varying degrees of modification. The QA conditions of PDS and NMS are evaluated against the Software Life Cycle or its equivalent, in accordance with which NDS must be developed.

Software should be graded to establish the priority with which the software documentation should be addressed, or whether it should be addressed at all.

Software management and engineering activities must be integrated.

CRITICALITY SAFETY BENCHMARK EVALUATION PROJECT

J. Blair Briggs
Idaho National Engineering Laboratory

Every Criticality Safety Organization at Department of Energy facilities is required to compare results obtained from their calculational techniques with experimental data. The tedious process of researching benchmark critical data reported in journals, transactions, and reports is repeated over and over in an attempt to ensure criticality safety margins are accurate, and to comply with DOE orders. Since the beginning of the nuclear industry, thousands of benchmark critical experiments have been performed. However, many of these experiments were not performed with a high degree of quality assurance.

A Criticality Safety Benchmark Evaluation Working Group was established to:

1. Identify and evaluate a comprehensive set of critical benchmark data.

2. Verify the data, to the extent possible, by reviewing original and subsequently revised documentation, talking with experimenters or individuals who were associated with the experimenters or the experiment facility.

3. Compile the data into a standardized format.

4. Perform calculations of each experiment with standard criticality safety codes.

5. Formally document the work into a single source of verified benchmark critical data.

Publication of the evaluated criticality safety benchmark experiments is scheduled for October 1994. Periodic revisions to this publication will be made at appropriate intervals as the work progresses.

SESSION II: EXPERIMENTAL NEEDS (LACEF PROGRAM REVIEW)

RECOMMENDATION 93-2 TO THE SECRETARY OF ENERGY PURSUANT TO 42 U.S.C. § 2286A(5) ATOMIC ENERGY ACT OF 1954, AS AMENDED. DATED: MARCH 23, 1993

The end of the international competition in manufacture of nuclear weapons, and the transition to large scale dismantling of nuclear weapons, have generated strong pressures to reduce the defense nuclear budget and to close down many defense nuclear facilities and operations. At the same time, the development of firm plans for a Complex 21 to serve future nuclear defense needs has slowed. These trends lead to a possibility that capabilities and functions necessary for current and future needs could be terminated along with those no longer required. One of these, important for the avoidance of certain types of accidents, is support of nuclear criticality control.

Because of the importance of avoiding criticality accidents, the Board carefully follows the state of criticality control at DOE's defense nuclear facilities. This interest has been evident as Board members and staff have reviewed practices at the Pantex Plant. The Board believes it is important to maintain a good base of information for criticality control, covering the physical situations that will be encountered in handling and storing fissionable material in the future, and to ensure retaining a community of individuals competent in practicing the control.

In the course of retrenchment of its activities in recent years, the Department of Energy and its predecessor agencies have terminated use of all but one of its general purpose facilities for conducting neutron chain-reacting critical experiments with fissionable material. The research at these facilities had served programmatic purposes of diverse DOE programs as well as laying a general experimental basis for practices that ensure averting criticality accidents. The Board is informed that there is now a strong possibility that the last DOE facility capable of general purpose critical experiments will be shut down in the near future, due to lack of funding. This possibility arises because no single program of the Department has an overriding need for this remaining facility at the Los Alamos National Laboratory, and therefore no single program office is motivated to provide its financial support in this period of budget stringency. A certain complacency fed by some years of freedom from criticality accidents seems also to underlie this possibility.

The Board observes that the art and science of nuclear criticality control have three principal ingredients. The first is familiarity with factors that contribute to achieving nuclear criticality, and the physical behavior of systems at and near criticality. This familiarity is developed in individuals only through working with critical systems. It cannot be imparted solely through learning theory and using computer codes. The second is theoretical understanding of neutron multiplication processes in critical and subcritical systems leading to predictability of the critical state of a system by methods that use theory benchmarked against good and well characterized critical experiments. The third is thorough familiarity of nuclear criticality engineers with the first two factors obtained through a sound program of training that indoctrinates them in the experimental and theoretical aspects.

The Board has reviewed the status of benchmarking the theoretical methods of criticality control against existing critical experiments and has found that there are notable failures of theoretical analysis to account for the results of a number of experiments. It is not known whether this discrepancy results from inadequate nuclear data used in the analysis or from inadequate care in conducting the experiments and recording their physical features. Both factors could contribute. In addition, it seems that on the average there may be a small non-conservative bias in overall predictions of the theory. In spite of these shortcomings. conservatism in methods used to develop the limits to be applied during handling and storage of fissionable material seems to have led to adequate safety in recent years. The Board believes that in the interest of continued safety it is important to clear up the existing discrepancies, which are obstacles to confident understanding of criticality control. To do so will require conduct of further neutron chain-reacting critical experiments targeted at the major sources of discrepancy between the theory and the experiments, as well as careful analysis of the experiments.

Finally, the Board believes that there is no guarantee that the physical circumstances of handling and storage of fissionable material in the future will always be found in the realm of benchmarked theory. This point is especially important under circumstances that will exist for a number of years to come, with increasing amounts of fissionable material to be stored in a variety of chemical and physical forms. This does not appear to be an appropriate time to eliminate an ability to ensure that such activities will be free of criticality hazard. For safety purposes it will be necessary to retain the capability to perform experiments under conditions not foreseen at this time. This capability once lost would be most difficult to reproduce, and it could be approximated only at great cost and after substantial time, deterring such development even if it were needed badly.

For all the above reasons, the Board believes that continuation of an experimental program of general purpose critical experiments is necessary for continued safety in handling and storing fissionable material. It is needed to improve the basis for the methodology. It is needed as part of the process of properly educating criticality control engineers. It is needed to ensure the capability of answering criticality questions with new and previously unresearched features.

Therefore the Board recommends that:

1. The Department of Energy should retain its program of general purpose critical experiments

2. This program should normally be directed along lines satisfying the objectives of improving the information base underlying prediction of criticality, and serving in education of the community of criticality engineers.

3. The results and resources of the criticality program should be used in ongoing departmental programs where nuclear criticality would be an important concern.

John T. Conway, Chairman

VALIDATION OF KENO V.A FOR TWO TYPES OF SYSTEMS

Ernest P. Elliott
Martin Marietta Energy Systems
Oak Ridge Y-12 Plant

The Oak Ridge Y-12 Plant desired to validate KENO V.a for two general types of systems: highly-enriched uranium, carbon, and hydrogen and highly-enriched uranium and carbon at high C/U atomic ratios. These two categories of systems represent several types of contaminated recycle and waste materials encountered in the plant such as hydrocarbons, cellulose-based materials, and graphite casting molds.

A literature search was conducted to identify critical experiments that would serve as the basis for this validation effort. Any experiments considered for this validation effort had to be accurately described, include only the constituents of interest, and had to be actual critical assemblies (subcritical extrapolations were excluded). Four experimental series were chosen through this initial selection process for further evaluation: critical experiments performed in support of the Rover program, experiments performed at Los Alamos by C.C. Byers and J.C. Hoogterp, and experiments from Lawrence Livermore performed by A.J. Kirschbaum. The experiments of Mr. Hoogterp and Mr. Kirschbaum were later set aside because the definitive configuration of the critical assemblies could not be determined from reports and/or logbooks. The other two series (Rover and those performed by Dr. Byers) were used for the validation effort.

Problems such as inadequate documentation arise frequently during validation efforts since the experimenters usually were evaluating a specific plant situation of interest, not preparing data for code validation.

FISSIONABLE MATERIAL MEASUREMENT NEEDS IN HANFORD SITE WASTE TANKS

Hans Toffer
Westinghouse Hanford Company

The quantities, concentrations, and distributions of fissionable materials in the Hanford Site waste tanks are not well known. Measurements inside and outside the tanks could provide essential insights on how to establish criticality controls.

Fissionable materials have accumulated in the tanks in conjunction with waste transfers. The fissionable materials have accumulated in the tanks in conjunction with waste transfers. the fissionable materials would be chemically combined with other elements and could settle in sediment layers or be in local accumulations. From all indications, the fissionable materials concentrations are low; however, the impact of diluting materials on criticality are not well known.

A variety of measurements both inside and outside tanks are needed to establish the parameters that are important to measure criticality control. In-tank measurements such a s sampling of tank material, vertical profiles using neutronic sensitive detectors, gas measurements, and actual k_{eff} measurements could prove very useful.

As part of the initial measurement effort, tank 102-SY has been selected for possible in-tank measurements. Some preliminary estimates indicate that the tank may contain 45 kg of plutonium. Recent neutron and gamma-ray scans in that tank indicate that the tank has a low gamma-ray background and very uniform moisture distributions. In-tank neutron measurements with various techniques should prove promising in locating any vertical distributions of neutron emitting materials.

The out-of-tank measurements would require the use of a critical mass laboratory. Measurements to be performed in such a facility could establish minimum critical concentrations for plutonium, uranium, and other material mixtures, and consider the layering of fissionable and nonfissionable materials impact on criticality. The reactivity of very subcritical systems as well as criticality of pileup geometries would be of interest.

In addition to the direct criticality measurements, peripheral data would be very useful to normalizing analysis tools and calibration measurement equipment. Such measurements best performed in a critical mass laboratory would include neutron transmission, gas analyses from tanks with known contents, and calibration of neutron probes.

To resolve nuclear criticality issues associated with waste storage and processing will require both conventional criticality measurements as well as peripheral measurements that provide insights into neutron interactions with prototypic waste materials.

HOW MORE OBJECTIVE CRITICALITY SAFETY ANALYSES CAN BENEFIT FUTURE WASTE TRANSPORTATION AND STORAGE

Mark A. Robinson

EXECUTIVE SUMMARY

An examination of the limiting assumptions of criticality safety models for TRU Waste drum storage and for the TRUPACT-II transuranic waste transport vessel indicates that these assumptions are unnecessarily conservative. The criticality limits for the TRUPACT-II represent a significant constraint on processing of TRU Wastes and residues (residues being similar in substrate to transuranic wastes but contain amounts of plutonium formerly considered economically recoverable for re-use in weapons production). The TRUPACT-II limit is estimated to result in transportation costs of $850 million for disposal of Rocky Flats residues, exceeding the cost of transportation if the TRUPACT-II limits corresponded to criticality limits for infinite array drum storage by an additional $790 M. The current TRUPACT-II limit also increases risks of transportation accidents by a factor of 14 over that incurred if the TRUPACT-II limits correlated to drum storage limits. The criticality safety community needs to re-evaluate the models that form the basis for those limits.

INTRODUCTION

There are significant amounts of transuranic (TRU) waste buried and/or stored at each of the major Department of Energy (DOE) production and laboratory facilities. As of 1990, this amounted to in excess of 250,000 m^3 of waste containing at least 3000 kilograms (kg) of plutonium.[1] Perhaps more significant are the accumulated residues stored at each of the major production facilities, which are similar in substrate composition to TRU Wastes, but contain radionuclides in amounts formerly considered economically recoverable for re-use in weapons production. At Rocky Flats, for example, residue inventories amount to 3800 drums (785 m^3) containing more than 3000 kg of plutonium[2]. All residues will eventually be processed through recovery (actinide separation) and/or repackaging into TRU waste.

TRU waste and residues must be stored, treated, packaged, and shipped to the Waste Isolation Pilot Plant (WIPP) for ultimate disposal. Processing of these wastes may represent the most significant challenge and commitment of resources in the history of the Weapons Complex. Criticality modeling and establishing of plutonium Fissile Gram Equivalent (FGE) limits to prevent inadvertent criticality accidents will have a major effect on the resources required to accomplish this processing and eventual disposal. These limits may even be construed to increase the risk of accidents in the overall waste processing logistical picture.

BACKGROUND

Criticality limits for storage of TRU wastes and residues are derived from criticality safety mathematical models developed at WIPP. These limits are 200 grams FGE for 55 gallon drums and 325 grams FGE for a Standard Waste Box (SWB). This 325 gram limit is significant in that it is the same as the TRUPACT-II[3]

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1. DOE/RW-0006. Rev. 5. "Integrated Data Base for 1989: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics"; plutonium content of buried TRU waste cited in this document is probably underestimated.
2. Mixed Residue A Reduction Report," U. S. Dept. of Energy Rocky, Flats Office, February 26, 1992.
3. Transuranic Waste Packaging Transporter-II: The TPUPACT-II is a Type B transport vessel regulated by the NRC under lOCFR71. The TRUPACT-II has a 14 drum or 2 SWB capacity. Three TRUPACT-II vessels may he carried by a flatbed tractor-trailer.

vessel limit and derived from basically the same conservative modeling assumption. It is the TRUPACT-II limit which constitutes the primary processing constraint, and which -as will be shown -has a very significant impact in terms of resource expenditure and accident risk for the Complex as a whole. What this means is that, although the FGE limit per drum is 200 grams, in reality drums must be packaged at an average fissile content of 325 grams per 14 drums or 23.2 grams/drum, or shipments must be limited to one 200 gram drum per vessel (3 drums per shipment). And when the container assay error is considered, the allowable vessel, and therefore container, fissile mass content is even lower.[4]

WASTE CHARACTERISTICS

Before examining mathematical models of TRU waste physical behavior, some understanding of waste characteristics is prerequisite. TRU Waste and residues are essentially industrial trash with a surface contamination (fixed and loose) of transuranic radionuclides. As TRU Waste is primarily Weapons Grades[5] plutonium, the dominant isotope is ^{239}Pu. The transuranic component of TRU Waste is almost entirely composed of oxides and other stable compounds of plutonium.

Although there are numerous individual waste streams (generation sources, generally known as Item Description Codes or IDC's) that result in transuranic waste, these multiple waste streams can be generally grouped into four categories[6]:

I Solidified aqueous or homogenous inorganic solids (sludges, cementacious monolithic mixtures)

II Solid Inorganics (graphite, glass, metals, masonary, salts, etc.)

III Solid Organics (combustibles, plastics, filters, etc.)

IV Solidified Organics (oils/solvents stabilized in cement)

Ultimately it is likely that the majority of those Category II & III waste types that are compressible or malleable will be supercompacted (compressed into a concentrated contiguous plastic matrix with inherent structural stability and eliminating any appreciable voids in the waste form).

Category I & IV wastes are either processing precipitates or intentionally stabilized (grouted) waste forms.

Only one of all these waste forms is soluble in water -the various salts produced by pyrochemical processes such as Direct Oxide Reduction or Molten Salt Extraction. These salts are packaged in a secondary confinement of metal cans inside drums (or other metal containers). The plutonium component of all categories of waste, in its oxide, hydroxide, or other complexed forms, is very refractory and virtually insoluble in water. The hydrolysis constants[7] for plutonium oxides and hydroxides, for example, are less than 10^{-6}. Also, transportation regulations, repository requirements, and RCRA[8] prohibit corrosives, incompatible chemicals, and free liquids from containers bearing TRU Wastes and/or Mixed (radioactive and hazardous) Wastes, making solubility of waste substrates or radionuclides unlikely in any credible accident scenario.

4. To be exact, the TRUPACT-II criticality limit is: [vessel assay value + (2 X assay error)] $\leq$ 325 grams ^{239}Pu FGE. For example, if assay error is 50%, the actual TRUPACT-II limit becomes 162.5 grams [162.5 + (2 x 162.5/2) = 325].
5. Some sites have significant amounts of heat source plutonium, with the dominant isotope being ^{238}Pu.
6. DOE/WIPP 89-004, Rev. 3, TRUPACT-II Content Codes (TRUCON)
7. The Chemistry of Plutonium, J. M. Cleveland, American Nuclear Society
8. Resource Conservation and Recovery Act

Existing criticality models, however, assume that plutonium in these substrates becomes solubilized (in H_2O) and subsequently concentrated. Documented nuclear industry experience[9] is quite the opposite. It requires prolonged exposure to turbulent alkali solutions at elevated temperatures to remove any significant portions of particulate radionuclides from typical heterogeneous waste forms. Once removed, plutonium oxides are not solubilized and must be collected using forced circulation and filtering. The only other possible method of radionuclide removal would be acid dissolution as in standard plutonium chemical processing technology. Neither scenario represents credible accident phenomenology, given the existing waste controls, and container integrity as described in the following paragraphs.

CONTAINER INTEGRITY

TRU Wastes are stored in DOT Type 17C drums. These are 49CFR173.415 Type A packages which are demonstrated to survive 49CFR173.465 performance tests. Such drums have survived 60 ft. drop tests at Rocky Flats with only a momentary breach of seal integrity upon impact. The TRUPACT-II is designed to carry 14 of these drums in dense pack configuration and fixed within precision metal templates designed to provide secure dunnage, in case of vehicular accident. The TRUPACT-II is also designed to prevent damage to drum payload by absorbing external impacts through its dual metal containment and foam sandwich" configuration. The TRUPACT-II is a Type B container regulated under 10CFR71. It has survived, without any loss of integrity, the 10CFR71.73 performance tests (performed by the Sandia National Laboratory) including:

1) 9 meter free drop test
2) 1 meter puncture drop test
3) 1 hour external fire test
4) the final 1 hour water immersion test.

The demonstrated integrity of the TRUPACT-II and Type A internal packages under hypothetical accident conditions should have a direct bearing on criticality model assumptions, as it has on regulatory requirements for criticality modeling discussed in the following paragraphs.

REGULATORY REQUIREMENTS

The Nuclear Regulatory Commission (NRC) establishes minimum requirements for criticality modeling of transport vessels for fissile materials. The first requirement is found in 10CFR71.55(e):

"A package used for the shipment of fissile material must be so designed and constructed *and its contents so limited that* under the tests specified in § 71.73,[10] the package would be subcritical. For this determination it must be assumed that:

(1) the fissile material is in the most reactive configuration consistent with the *damaged condition of the package and the physical form of its contents;*

9. "Plutonium Recovery from Plutonium Contaminated Combustible Wastes by Washing - Part II," J.O Wilkins & S.J. Wisbey, PWMWP/P209, British Nuclear Fuels plc.
10. Hypothetical accident condition performance tests discussed in the previous paragraph.

(2) water moderation occurs to the most reactive *credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents,* and

(3) there is reflection by water on all sides, as close as is *consistent with the damaged condition of the package.*" [emphasis added]

As the TRUPACT-II survived the §71.73 performance tests without loss of integrity (other than minor structural deformation in the puncture drop test), this is the configuration that the regulations quoted above indicate that should be used in modeling potential criticality accidents. These regulations also indicate that the model should be consistent with the physical form of the waste and internal containers (i.e., "the chemical and physical form of the contents"), also as described in previous paragraphs.

Additionally, 10CFR7 1.57 requires:

a) any number of undamaged packages to be subcritical in any arrangement with optimum interspersed hydrogenous moderation, and

b) 250 packages if each were subjected to the §71.73 hypothetical accident conditions, would be subcritical in any arrangement, closely *reflected on all sides of the stack b- water and with optimum interspersed hydrogenous moderation.* [emphasis added]

Obviously, this means the criticality model should be formulated as depicted below.

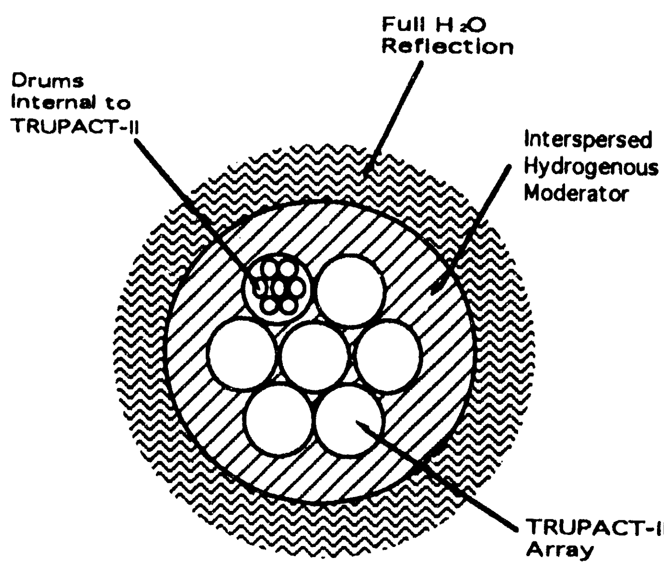

Figure l: Regulatory Model

The NRC regulations were written to c over any size package. Given the size of the TRUPACT-II and the fact that only three are shipped at a time, an infinite array (stack of 250) of TRUPACT-II's is not realistic It would be more conservative, and closer to credible accident scenarios and physical conditions, to model potential for criticality as follows.

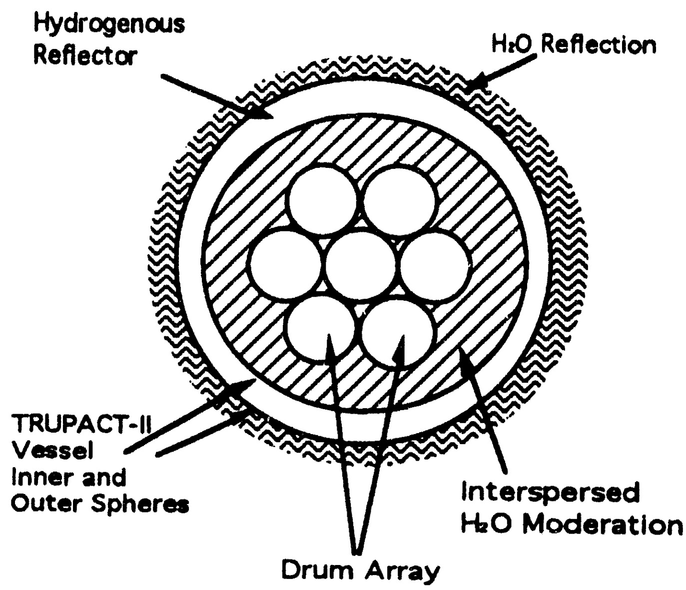

Figure 2: Conservative Realistic Model

The existing TRUPACT-II criticality analysis,[11] however, chooses to model the accident scenario as depicted below.

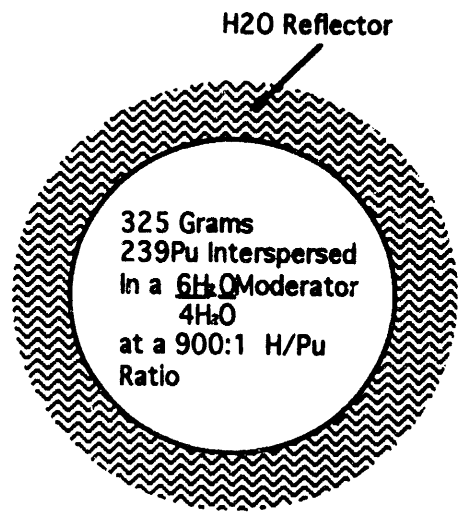

Figure 3: Existing TRUPACT-II Criticality Model

[11]Nupac, TRUPACT-II Safety Analysis Report, para. 6.4.2.

Modeling Conservatisms

Existing criticality models for drum storage assume drums in an infinite array have spheres of pure plutonium metal optimally located for maximum neutron interaction, as shown below.

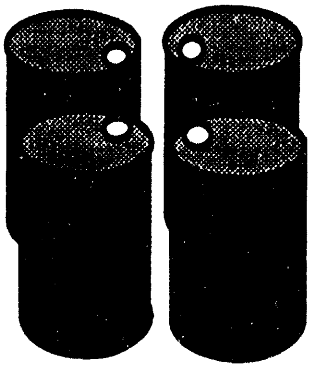

Figure 4: Infinite Array Drum Model

Considering the actual waste characteristics and radionuclide dispersion within the waste matrix previously discussed, such a scenario is incredible, especially over an infinite array. However, the TRUPACT-II criticality model is even more conservative (incredible). It assumes:

1) that all internal container (drum and any containers internal to the drum) integrity is lost
2) that all particulate plutonium oxide is solubilized (in water)
3) that all of the plutonium atoms "gravitate" into a sphere
4) that the sphere also becomes composed of a mixture of polyethylene and water ($6CH_2/4H_2O$ at a 900:1 H/Pu ratio)

This hypothetical accident phenomenology is depicted below (tubing, funnel, and flask courtesy of the illustrator).

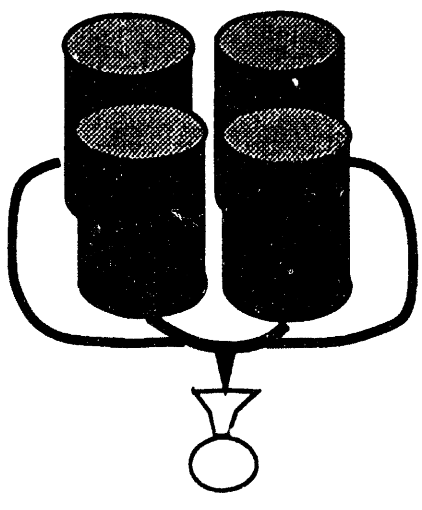

Figure 5: Existing Model Accident Scenario Phenomenology

The criticality limit is consequently set at 325 grams ^{239}Pu FGE as this is the minimum critical fissile mass under such conditions (k_{eff} >0. 95).[12] As the plutonium oxide hydrolysis constant is approximately 10^{-6} (i.e., only 1 in a million molecules actually solubilizes in water), it would be just as logical to establish the criticality limit at 325,000 kg (10^6 x 325 grams). Somewhere between these two extremes the scientific community must find a criticality model that is both conservative and credible. The consequences of failure are discussed in the following paragraphs. It is also worthy of note at this juncture that the accident precursor assumptions of the TRUPACT-II analysis are essentially the same as those of the WIPP Performance Assessment for the repository disposal phase -complete loss of container integrity and complete brine saturation of the repository environment. no one, however, has postulated a massive criticality as a consequence. Perhaps the WIPP should be limited to 325 grams of ^{239}Pu if criticality is possible under these conditions?

COST AND SAFETY IMPACTS

As previously stated, the TRUPACT-II criticality limits are the primary constraints on processing TRU Waste and residues, ,and may be a major contributor to overall risk in the waste disposal process.

Let us examine residues. Residues exist in inventories at radionuclide concentrations far in excess of the 200 gram FGE limit for drums (some sites have reported single drums with up to 6000 grams FGE). Consequently, drums must be repackaged into many more drums (i.e., the contents must be diluted) in order to comply with criticality limits. This entails a significant c ost and a significant worker exposure during the repackaging process.

In considering this repackaging it is important to recognize that: (1) the number of shipments to WIPP is directly proportional to the criticality limit for the TRUPACT-II, and (2) the available space at WIPP is reduced in a manner directly proportional to the criticality limit.

Rocky Flats has approximately 3800 drums of residues currently in inventory. These drums will be processed and repackaged to meet the present 200 gram FGE limit[13] in order to maximize storage capacity and still comply with WIPP criticality limits. If the TRUPACT-II criticality limit is maintained, this will mean shipment of 1 drum per TRUPACT-II, or 5667 shipments of 3 vessels per truckload. Using a ballpark estimate of $150K per shipment[14], transport of Rocky Flats residue inventory to WIPP will cost $850M (5667 x $150K = $850M). If, on the other hand, the TRUPACT-II limit could be raised to 2800 grams to correspond with present drum limits (14 x 200 = 2800), the cost of disposal shipments would be reduced by a factor of 14. 17000 drums divided by 28 drums/vessel = 1214 vessels. 1214 vessels at 3 vessels/shipment = 405 shipments. 405 x $150K = $60.7M. Realistic modeling could save $790M at Rocky Flats alone!

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12. The TRUPACT-II criticality model assumes an infinite array of TRUPACT-II vessels containing the 325 grams of plutonium as depicted in Fig. 4.
13. It is recognized that TRUPACT-II thermal wattage limits could be reason to repackage at less than 200 grams per drum, but that is a separate issue. Also, the number of shipments postulated herein (5667 and 405) were derived assuming an average 10% assay error and therefore actual drum loading of 167 grams FGE.
14. $150K per shipment is probably conservative given the requirements for characterization, aspiration, certification, assay, vessel leak testing, satellite monitoring, emergency preparedness, etc. involved in shipment to WIPP. This estimate was confirmed as not unreasonable by the EG&G Rocky Flats TRU Waste Program Manager. No official estimate is available at this time.

What consequences do criticality limits yield in terms of risk of transportation accidents? The WIPP Final Supplement Environmental Impact Statement cites Department of Transportation accident statistics which indicate a constant risk of vehicular accident of 1.70×10^{-6} per vehicle mile. The route from Rocky Flats to WIPP is 874 miles [15]. If the 325 gram criticality limit is maintained, this means that 8.42 TRUPACT-II accidents are probable in the disposal of Rocky Flats residues [(1.70×10^{-6} accidents/mile)(5667 shipments)(874 miles/shipment) = 8.42 accidents)]. If the limit were raised to 2800 grams FGE the risk would be reduced to 0.6 accidents [(1.70×10^{-6})(405)(874)]. It would be an irony indeed, if a conservative criticality limit were to contribute to accidents we are all striving to prevent.

Finally, let us look at the impact on available space at WIPP. As previously stated, current inventories of TRU Waste are projected at 250,000 m^3. This exceeds the WIPP design capacity of 183,000 m^3. Obviously, disposal space is at a premium. Current estimates for the final construction and test phase costs for WIPP translate to a disposal space value of \$8000/m^3. If Rocky Flats residues are packaged to meet the 200 gram limit (assuming an average assay error of 50%), residue disposal will require 6100 m^3 at a cost of \$49.6 million. If these residues were packaged at 162.5 grams to try to maximize the use of the TRUPACT-II under its current limit of 325 grams (325/2 = 162.5), disposal would require 7630 m^3 at cost of \$61M. Obviously the increased cost of storage would be offset by reduced transportation costs if this option were selected, but the point here is that disposal space at WIPP is very costly and criticality limits have a direct bearing on how efficiently we use it.

CONCLUSIONS

Under the current constraint of the TRUPACT-II, 325 gram FGE limit, the American taxpayer will be forced to pay \$790 million to transport Rocky Flats residues to WIPP, over and above the transportation costs that would be incurred if the TRUPACT-II criticality limit corresponded to the current drum limit of 200 grams ($14 \times 200 = 2800$). This excess will be only a small portion of the cost of transporting residues and 'Hot TRU" throughout the DOE Complex. The drum limit is based on an infinite array configuration and very conservative modeling assumptions as to plutonium location and dispersion (consolidation) within drums. The TRUPACT-II criticality model is far more conservative in its assumptions, far exceeding regulatory requirements, which are generally perceived as conservative themselves. The consequences of these conservatisms are both unreasonable costs for transportation to the disposal site and an unacceptable increase in risk of transportation accidents—eventualities that could be more costly in terms of health effects, resources, and DOE credibility than any other concerns presented herein.

It is time therefore to re-evaluate our transportation and storage criticality models. If criticality testing using surrogate waste forms simulating accident conditions is necessary to support revision of criticality limits, the cost of such testing can certainly be justified. If such testing supported an increase in the drum storage limits as well, the potential benefits speak for themselves.

15. 874 miles one way; 1748 miles round trip

CRITICAL MASS LABORATORY AT ROCKY FLATS

Dr. Robert E. Rothe
EG&G Inc., Rocky Flats Plant

With the onset of "The Cold War," the United States government decided to initiate a vigorous program of nuclear weapons design, development, and production. Thus, Rocky Flats was born in 1952. The plant, northv est of Denver, handled both enriched uranium and plutonium. Safety was always important so the Nuclear Society Group was formed under the able leadership of C. L. Schuske, deceased.

Criticality safety in these early days was estimated by a few sub-critical experimental approaches to criticality and the use of some simple models derived from reactor theory. No computerized calculational methods were available. These early simple *"in situ"* experiments were kept far from critical; and a long extrapolation was necessary to guess at the critical value.

The need to improve accuracy in the knowledge of critical parameters led to the construction, in 1964, of the Rocky Flats Critical Mass Laboratory (CML). It received its first fuel the following year. The first critical experiment involved enriched uranium metal flooded with oil; this happened in September, 1965. The first enriched uranium solution took place in 1967. Since those early formative times, over 1700 critical or critical-approach experiments have been performed spanning at least 25 different programs. These included plutonium, highly enriched uranium, and low-enriched (4.5%) uranium in metal, solution and oxide forms. Programs have featured bare assemblies and ones reflected by a variety of common and bizarre materials, systems which are heavily poisoned with strong neutron absorbers, and both single units and arrays. Two record assemblies deserve special note. One subcritical *in situ* measurement had 185 kg of 93%-enriched uranium metal in one single unit. On another occasion, about half the nation's supply of plutonium metal was stacked into one subcritical vertical array.

These experiments at Rocky Flats ended abruptly in the Fall of 1988 because of certain safety and procedural practices—elsewhere at the plant—being called into question. The CML had paused, in the middle of an experimental program, to clean up and reconfigure the present apparatus when this occurred. Attention quickly focused on problems at the plant site as a whole; and no resources were available to resume the interrupted program. Rocky Flat's perceived procedural shortcomings trickled down to the CML; and that laboratory has not yet recovered from that blow. Its future stands in the balance: decommissioning or resumption.

The thesis of this paper—and that which will be focused upon in the oral presentation—will be a strong argument that safety within, the nuclear industry cannot be assured unless these critical mass laboratories are allowed to resume their important function.

SAND93-1 230C

OPERATIONAL AND SAFETY CHARACTERIZATION OF THE SPR-IIIM FAST BURST REACTOR*

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Sandia National Laboratories

SUMMARY

SPR-IIIM is a modernized, improved version of SPR-III. The new system is expected to improve overall reliability and performance while reducing personnel dose and maintenance frequency. A description of the SPR-IIIM reactor and its features are presented in this paper along with plans for characterizing the reactor's operational and safety characteristics. Enhancements of SPR-IIIM include (a) a larger central irradiation cavity, 7.5 in. ID, (b) a self-aligning safety block, (c) spring-loaded fuel clamping, (d) forced flow cooling across fuel plate gaps, and (e) larger diameter hollow shafts with precision spline bearings to support the reflector control elements.

The critical loading sequence is described as well as the zero and low power calibrations of the reflector control elements, the safety block, and experiment materials. The safety block reactivity worth as a function of separation position and time will be calculated directly from measured data using a finite difference solution of the point kinetics and the measured neutron decay resulting from a safety block scram from delayed critical. A sample was presented using SPR-III and a "constant" prompt neutron generation time. We plan to extend the standard inverse kinetics solution (by D. Minnema) by incorporating a variable quasi-static prompt neutron generation time correction.

For safety characterization of fast burst reactors, we suggest that the total fission yield in a large pulse be treated like risk, i.e., as a "probability function" times the "yield" one would expect if all of the planned reactivity is inserted without preinitiation. G. Hansen has shown that the probability, P, is a "rapidly" decreasing function for prompt critical reactivity insertion values greater than 10¢ [i.e., $\rho(\$) > \$1.10$] and for "neutron-source-divided-by-insertion-rate" values, S/a (n/$), in excess of 5×10^4. Hansen and Wimett further showed that fast burst reactor kinetics behavior can be closely approximated by a "modified" FUCHS model, $Y(t) = Y_F(t) * (1 + \alpha^2\tau^2)$ and a properly selected mechanical relaxation time, τ. But this model is not adequate beyond the elastic range for fast burst systems. Sandia researchers (M. Sherman and D. Coats) are working on a model to encompass the melt and vapor range, but a coupled neutronics-hydro code may be needed to cover the whole range. We hope to first bound the vaporization case with some modifications to the Vented Bottle Model, a code developed to model the design basis accident for a pebble bed space nuclear reactor. With the upper bound vaporization "yield" and "probability" defined, we can establish a practical upper limit yield (a yield with a probability of occurrence in the "credible" range, i.e., $P \geq 10^{-6}$).

* This paper was presented at the Criticality Safety Technology Project Workshop, Monterey, CA, April 26-28, 1993. The work was supported by the United States Department of Energy under Contract DE-AC04-76DP00789.

NEW EXPERIMENTS AT THE LOS ALAMOS CRITICAL EXPERIMENTS FACILITY

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The Los Alamos Critical Experiments Facility (LACEF) is now becoming fully operational following an extended period of shutdown. It is appropriate at this time to review the status of the restart efforts, and to discuss the programmatic efforts which the facility hopes to carry out over the next few years.

The mission statement for the LACEF is presented as a review item. The LACEF programs are aimed primarily at new technologies, prototype design, and service functions. These types of activities present a challenge for writing a Safety Analysis Report (SAR) since experiments at the facility are always expected to involve an element of newness and uncertainty.

The experimental programs proposed for LACEF are arbitrarily divided into four categories: benchmark experiments, applications experiments, basic physics experiments, and prototype device experiments. Several experiments will not be discussed in this talk because they will be covered in more detail in other talks presented at this workshop.

The benchmark experiments all relate to the problem of validation of calculational methods used in making criticality safety decisions. Most of the experiments contain a new or unique feature which has not previously been validated. One set of experiments is aimed at the resolution of anomalous results which strongly disagree with the results of calculations (these experiments all give calculated k-effectives near 0.90). These disagreements between theory and experiment are important, because the theory should correctly predict the results for all physical observables.

The applications experiments are aimed at more specific topics or programs. The Criticality Safety Class provides hands-on training in nuclear criticality safety for operations, supervisory, and management personnel. Approximately two classes per month have been conducted during the past eight months. The Sheba experiments are aimed at determining the properties of excursions in a solution medium. These results should be useful in estimating the consequences of accidents of this type, and would be used in SAR analyses to document accident potential and to address mitigation or emergency response strategies. Equipment qualification and dosimetry are programs in which calibrated radiation fields and bursts are provided from the Godiva or Sheba assemblies for use in the testing of criticality alarms, TLDs, etc. The Godiva and Sheba assemblies are expected to provide primary standards for the radiation fields which are used to qualify equipment and devices.

The basic physics experiments are related to fundamental parameters of nuclear materials, including the delayed neutron fractions for various isotopes.

The prototype device experiments are programs which involve the measurements of properties of specific devices. Among the properties are the power and control rod calibrations, replacement worth measurements, and temperature coefficients of actual physical devices such as the Topaz and Advanced Neutron Source Reactors.

We hope to fund the facility through a combination of a baseline appropriation of \$3M, coverage of the site security costs, and individual programs funds which provide an additional \$1-2M.

MINIMUM CRITICAL MASS ANALYTICAL STUDIES

Rene G. Sanchez
Los Alamos National Laboratory

ABSTRACT

Analytical studies have confirmed that the minimum critical mass for ^{235}U in a critical reactor assembly moderated with high-density polyethylene and surrounded by a thick-beryllium reflector is on the order of 275 g. Similar studies have also shown that the minimum critical masses for ^{233}U and alpha-phase ^{239}Pu in the same type of critical assembly and surrounded by a thick-beryllium reflector are on the order of 185 g and 190 g, respectively.

I. INTRODUCTION

Because of the recent interest in performing experiments to determine the minimum critical mass for plutonium and uranium systems at room temperature, an analytical study has been completed, with the help of the Monte Carlo neutron photon (MCNP) transport computer code, to define minimum critical mass parameters for these systems. The study indicates that for an optimum moderator and reflector, the minimum critical mass for a ^{235}U system is on the order of 275 g and occurs at a $H/^{235}U$ ratio of 331. This result agrees with the experimental value of "...250 to 300 grams..." reported by Jarvis and Mills [1-3] in 1967. In addition, this study also shows that the minimum critical mass for ^{233}U and alpha phase ^{239}Pu in a hydrogen-moderated core with a thick beryllium reflector is on the order of 185 g and 190 g, respectively, and occurs at $H/^{233}U$ ratio of 337 and $H/^{239}Pu$ ratio of 543.

II. URANIUM AND PLUTONIUM SYSTEMS

To minimize the mass needed to maintain an effective multiplication factor, k_{eff}, of 1, it is important to recognize that neutron conservation (namely, the reduction of all losses not associated with the fission process) is essential to the solution of the problem. Because ^{233}U, ^{235}U, and ^{239}Pu have large fission cross sections at thermal energies (see Table I), an optimum hydrogenous quasi-homogeneous core with a thick reflector should reduce the critical mass to its lowest level for these systems.

In this study, several quasi-homogeneous configurations were examined, using the MCNP-4X-C computer code and the continuous energy cross-section data. For each case, a total of one-hundred-thousand source histories was run with the help of MCNP computer transport code. The majority of the configurations studied consisted of stacked high-density 0.635-cm (0.25 in.)-thick polyethylene plates and fissile foils made of 93% ^{235}U, 98% ^{233}U, and 95% alpha phase ^{239}Pu (see Table II). Foils of the same isotopic composition were separated by one, two, or more layers of polyethylene plates. The hydrogenous core was then surrounded by a 33.02-cm (13 in.)-thick beryllium reflector (Figs. 1 and 2). Computer models were then developed using foils of different thicknesses and surface areas. The results indicate that the optimum thickness for the foils is 0.003048 cm (0.0012 in.), with a width of 15.24 cm (6 in.) and length of 15.5575 cm (6.125 in.). Table III shows the results of these studies.

Table I. Thermal (0.0253 eV) cross-section data for fissile nuclide.[a]

Material	σ_a	σ_t	α	η	ν
^{233}U	578.8	531.1	0.0899	2.287	2.492
^{235}U	680.8	582.2	0.169	2.068	2.418
^{239}Pu	1011.3	742.5	0.362	2.108	2.871

[a] Reference 4.

Table II. Isotopic composition of foils.*

Material	^{233}U	^{234}U	^{235}U	^{238}U	^{239}Pu	^{240}Pu
^{235}U Foils	—	1.000%	93.499%	5.501%	—	—
^{233}U Foils	98.45%	0.94%	0.02%	0.590%	—	—
^{239}Pu Foils	—	—	—	—	95.00%	5.00%

* It is probably impossible to conduct an experiment with very thin plutonium foils. However, they were calculated as foils to preserve the comparison with ^{235}U and ^{233}U.

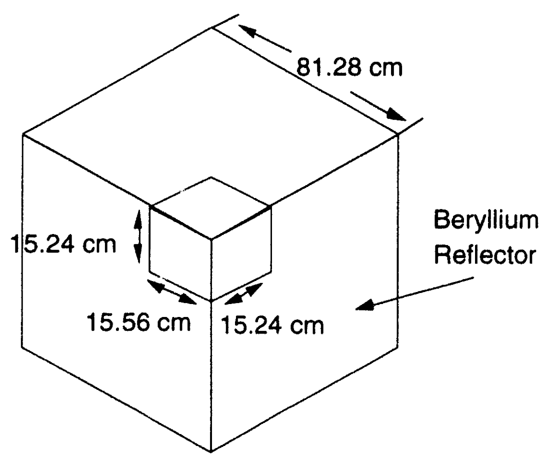

Fig. 1. Beryllium reflector and cubical fuel-cell cavity.

Table III. Data for a hydrogenous core in a thick-beryllium reflector.

^{235}U Foil Dimensions (in.)

Cubical Geometry	Weight of Core Moderator Material (g)	Total ^{235}U Mass (g)	Atomic Ratio $H/^{235}U$	k_{eff}
9 x 9 x 0.003	1439.87	278.9	173	0.878 ± 0.0033
9 x 9 x 0.003	1762.85	278.9	212	0.894 ± 0.0034
9 x 9 x 0.003	2408.81	278.9	289	0.913 ± 0.0038
9 x 9 x 0.003	4183.15	278.9	502	0.856 ± 0.0028
6 x 6 x 0.003	1418.33	278.9	170	0.942 ± 0.0037
6 x 6 x 0.003	1994.22	278.9	240	0.950 ± 0.0039
6 x 6 x 0.003	2495.72	278.9	300	0.964 ± 0.0036
6 x 6 x 0.003	2710.13	278.9	326	0.949 ± 0.0034
6 x 6.125 x 0.0012	2749.00	278.49	331	1.020 ± 0.0026
6 x 6.125 x 0.0012	2893.68	278.49	348	1.020 ± 0.0036
6 x 6.120 x 0.0012	2604.34	277.44	314	1.019 ± 0.0022
Spherical Geometry Radius = 8.82 cm	2746.63	279.30	329	1.017 ± 0.0021

Table III. Data for a hydrogenous core in a thick-beryllium reflector. (Con't.)

^{233}U Foil Dimensions (in.)

Cubical Geometry	Weight of Core Moderator Material (g)	Total ^{233}U Mass (g)	Atomic Ratio $H/^{233}U$	k_{eff}
6 x 6.125 x 0.0012	1735.50	185.54	311	0.995 ± 0.0027
6 x 6.125 x 0.0012	1880.89	185.54	337	0.999 ± 0.0026
6 x 6.125 x 0.0012	1953.24	185.54	350	0.996 ± 0.0027
6 x 6.125 x 0.0012	2025.58	185.54	362	0.996 ± 0.0026
6 x 6.125 x 0.0012	2170.97	185.54	389	0.991 ± 0.0022
6 x 6.125 x 0.0012	2243.32	185.54	402	0.988 ± 0.0030
6 x 6.125 x 0.0012	2388.72	185.54	428	0.979 ± 0.0027

^{239}Pu Foil Dimensions (in.)

Cubical Geometry	Weight of Core Moderator Material (g)	Total ^{239}Pu Mass (g)	Atomic Ratio $H/^{239}Pu$	k_{eff}
6 x 6.125 x 0.00123	1880.89	195.82	441	0.992 ± 0.0032
6 x 6.125 x 0.00123	2896.52	195.05	506	0.982 ± 0.0020
6 x 6.125 x 0.0012	3038.37	190.60	543	0.998 ± 0.0024
6 x 6.125 x 0.0012	3919.00	190.60	701	0.981 ± 0.0020

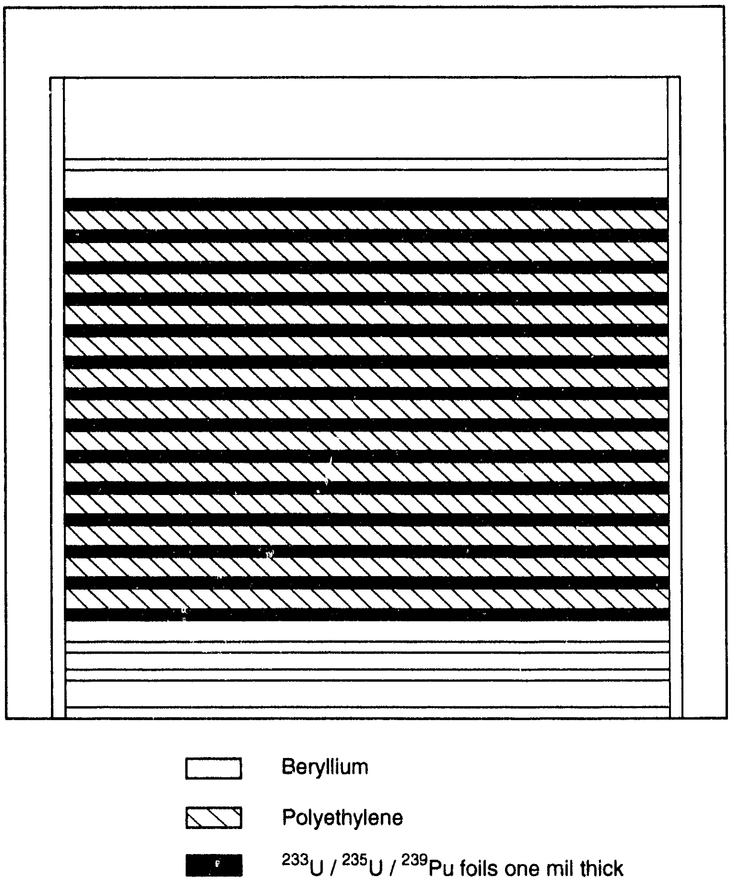

Fig. 2. Arrangement of $^{233}U/^{235}U/^{239}Pu$ foils and polyethylene plates inside a beryllium reflector.

For the case of ^{235}U, the effect of a spherical geometry (Fig. 3) vs a cubical geometry was investigated. The results showed that for the same mass and $H/^{235}U$ ratio, the effective multiplication factor, k_{eff}, is about the same for both geometries. This effect can be explained when we consider that reflected neutrons will have higher probability of interacting with the ^{235}U foils in a cubical geometry (larger area) as opposed to a spherical geometry. Results obtained confirm those published in Ref. 5; Table III shows these results.

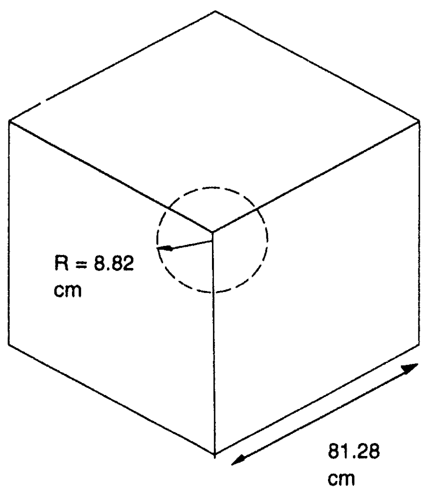

Fig. 3. Beryllium reflector and spherical fuel-cell cavity.

III. CONCLUSIONS

These analytical studies have confirmed that the minimum critical mass for ^{235}U in a hydrogenous core with a thick-beryllium reflector agrees with the experimental value of 250 to 300 g. The studies have also shown that the minimum critical mass for ^{233}U and alpha-phase ^{239}Pu in a hydrogen-moderated core with a thick-beryllium reflector is on the order of 185 g and 190 g, respectively.

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SOURCE JERK MEASUREMENTS ON HIGHLY SUBCRITICAL SYSTEMS

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Fieldable techniques for reactivity measurement have typically used correlation methods or pulsed source methods. The former method is poorly suited to systems which do not have a strong intrinsic neutron source (enriched uranium metal) while the latter requires equipment difficult to field. Work at Pajarito site on the source jerk method of subcritical reactivity determination was initiated by Spriggs [1] several years ago. We have built on this work to develop a fieldable system consisting of a PC clone interfaced to a mini CAMAC crate along with a source transporter or shuffler and a neutron detector.

The ability to measure subcritical reactivity is potentially valuable to criticality safety if it can be done simply and accurately. Concerns about storage density or about the leaching of neutron poisons from material storage vaults, for example, conceivably may be answered with accurate reactivity measurements.

The technique determines reactivity by the analysis of the transient which results from the rapid removal or jerk of a source from a configuration of fissionable material. Fitting measured data with a theoretical expression for the time dependent decay of the neutron population provides the assembly reactivity. We have improved the quality of the fit by fitting for time lag in moving the source as well as the reactivity. Including a lag time in the analysis is reasonable since we expect a finite source travel time. In addition to fitting a lag time, we intend to measure the source travel time to roughly verify the fitting results. Typically the previous work ignored early time data as it could not be fit without destroying the quality of the late time fit. By accounting for the source lag time an improved fit is obtained over both early and late times.

Data analysis has been converted to run entirely on a PC (386). This adds to the easy fieldability of this system as the connection to the laboratory computer center required by the earlier implementation is not necessary.

A prototype source shuffler has been used to take data on a subcritical UH^3 system while some supercritical measurements have been made using the Godiva assembly. A more fully engineered source shuffler is being designed with which we hope to further develop the technique and to make measurements both at Los Alamos and other locations.

1. G. D. Spriggs et. al., "Subcritical Measurements of the WINCO Slab Tank Experiment Using the Source Jerk Technique," Proceedings of the ANS International Meeting on Safety Margins in Criticality Safety, San Francisco, CA, Nov. 26-30, 1989

HEALTH PHYSICS RESEARCH REACTOR

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I have attached a copy of a proposal to acquire and operate the Health Physics Research Reactor at the Los Alamos Critical Experiments Facility (LACEF) at TA-18. The decision of the Department of Energy to provide a base of funding for LACEF makes it cost effective to maintain this primary standard of dosimetry at the Laboratory.

If successful in reactivating the HPRR, I would then propose to incorporate it into a Center for Experimental Radiation Dosimetry and Transport at LACEF (a suggestion made by Rick Brake, HS-4, Health Physics Measurements Group). The Center would feature the HPRR, a bare-metal fast-burst reactor, and Sheba, a low-enriched solution assembly. These two machines would provide neutron and gamma-ray spectra and ratios at the extremes of fissioning assemblies. Singly, and in joint use, they would provide ideal training tools for radiation protection technicians, and a research facility that could accommodate another generation of thesis topics for health physicists. In addition, they would effect a cooperative program with the French Centre de Etudes Valduc, which is already using Silene, a highly enriched solution assembly, for this purpose. With the HPRR and Sheba, we could simulate the prompt and decay radiation that has accompanied many accidents that have occurred throughout the world. In conjunction with Silene and our eight other assemblies at LACEF, they could play a part in solving some of the outstanding problems in basic nuclear data, radiation transport, and accident dosimetry. As indicated in the attached proposal, the application to assessment of the Hiroshima dose is obvious. Lastly, a 14-MeV D-T neutron generator is being transferred to a good-geometry facility that is also located at TA-18.

I plan to visit Oak Ridge in early April to explore the possibility of reactivating the HPRR at Los Alamos. In April 1992, we will begin validation testing of the modified Sheba prior to resumption of operations. We intend to resume steady-state operation of Godiva IV in April 1992; burst operation is anticipated later in the year. A revised Experimental Plan for Skua, our annular-core, bare-metal, fast-burst assembly, is nearly complete. This should permit operations at steady-state and long periods later in the year, leading to resumption of burst operation at a later time. Flattop (with both uranium and plutonium cores) and Big Ten (a metal system with a 10% enriched core and a depleted uranium reflector) were returned to operational status in 1991, and Comet (a simple general-purpose vertical-life machine) has been used for several nuclear criticality safety training classes since October 1991, and ten classes are planned for the remainder of 1992. We plan to resume the safety experiments with highly-enriched uranyl nitrate to support Westinghouse Idaho Nuclear Corporation (WINCO) in late 1992. To improve safety and to minimize errors, we plan to move the experiment from the vertical-lift Plant machine to Honeycomb, a horizontal split table. Experiments with the difficult-to-calculate interacting arrays of moderated elements using Venus (a general-purpose vertical-life machine) are still in the conceptual stage. In addition to training classes, presented jointly with HS-6, several undergraduate and graduate students, and two DOE Fellows, will utilize the assemblies under supervision during the summer of 1992. In addition, we plan to institute an intern program in residence at TA-18 to assist in training criticality safety engineers from throughout the complex.

A PROPOSAL
TO REACTIVATE THE HEALTH PHYSICS RESEARCH REACTOR

The Advanced Nuclear Technology Group of the Los Alamos National Laboratory proposes to reactivate the Health Physics Research Reactor (HPRR) in a DOSAR-type user facility at TA-18 in Los Alamos.

It is proposed to operate the HPRR as a service to continue to provide a well-characterized, mixed (neutron and gamma ray), radiation source for calibration and inter-comparison of radiation dosimetry.

BACKGROUND – Essentially, all radiation exposure standards for humans related to the observations resulting from Hiroshima and Nagasaki. Since those exposures were not instrumented, a 45-year program has been conducted to characterize those exposures. Part of that characterization was conducted in the 1960s when HPRR, a Godiva-like, bare enriched uranium, fast-burst reactor was operated on a tower at the Nevada Test Site to evaluate neutron and gamma-ray exposures on the ground. Although neither the spectra, neutron-to-gamma ratio, nor leakage per fission matched either of the sources in Japan, the data resulting from the study were the best available. Upon retirement, the fast-burst reactor was made the basis of the DOSAR facility at Oak Ridge National Laboratory, where it served as a primary source for calibration and inter-comparison until 1988.

UNIQUENESS – The HPRR is not the only fast-burst reactor in existence. Godiva-like bare-metal assemblies still exist at White Sands Missile Range, Aberdeen Proving Ground, Sandia National Laboratories (Albuquerque), and Los Alamos National Laboratory. However, none of these facilities match the unique features of DOSAR, which include a cylindrical reactor of HEU (with 10% molybdenum) that can be operated outside (to provide better control of capture and scatter radiation) to provide exposure on a nearly flat plane to 1000-m distances. Fast-burst reactors also exist in Russia (about 5 machines, none with HPRR characteristics that we are aware of) and China (about 2 machines, characteristics generally unlike the HPRR).

PHYSICAL SECURITY AND COST – At the time of shutdown, Oak Ridge had promises of about $1.8 M in user fees annually for the HPRR. Nevertheless, DOE determined that this did not constitute "full-cost recovery" and directed the closure. Although not stated, it is likely that DOE determined that even this sum would not adequately provide for both security and operating expenses of the machine. In 1990, additional constraints were placed on the security for large quantities of HEU. Technical Area 18 (TA-18) at Los Alamos, presently houses 10 critical assembly machines, 3 remotely-operated experimental areas, 4 permanent material access areas (MAAs), and several-hundred kilograms of SNM as high-enrichment uranium (HEU), plutonium, and uranium of lower enrichments, in addition to depleted uranium and other materials of interest. This material exists as fabricated burst reactors (Godiva IV - 65 kg of HEU, cylindrical burst reactor, and Skua, 165 kg of HEU annular core burst reactor); benchmarked machines (Flattop, with both uranium and plutonium cores, and Big Ten); radiation sources such as Sheba (the 5% enriched solution critical machine that operates on uranyl nitrate to simulate solution accidents and to validate the response of accidental criticality alarm detectors); and material in storage for several applications. As such, the costs of physical security at TA-18 can be pro-rated over several applications. **There is no additional security cost to add the HPRR!** Even with the extreme security requirements, it has been possible to accommodate uncleared personnel, even foreign nationals, in and around the experimental facility proposed for the HPRR. Of course, advanced planning and detailed arrangements must be made with physical security, but the process to make these arrangements is in place and demonstrated in practice. Exclusive of security and cost associated with DOE-mandated ES&H requirements, the present operating costs at TA-18 are about $4000 per day for each of the experimental facilities.

OPERATION OF HPRR AT TA-18 – In the near term, we propose to install, check out the machine, document, and train operating staff. Installation and checkout will follow normal operating procedures for the site. The original Godiva (Lady Godiva) was designed and built at Los Alamos and operated in the same experimental facility proposed for the HPRR. Personnel from TA-18 assisted in the original checkout of the HPRR. The site safety analysis report (SAR) currently under revision, will be reviewed to ensure that operation of another burst machine can be accommodated. The design basis accidents (DBA) appear to provide an adequate envelope for operation of a fast-burst machine both inside and outside of the proposed experimental area. Operator training should provide no problem for those trained and certified to operate Godiva-IV, because the machines are basically similar.

ULTIMATE CAPABILITY – The HPRR was originally built in an attempt to simulate Little Boy, the weapon employed at Hiroshima. As indicated, it is only marginally successful in that attempt. In 1980, a true replica of Little Boy was built and operated (at steady state) at TA-18 in the experimental area proposed for the HPRR. All components are still available, and it should be possible to resume operations of that assembly as a radiation source. This would allow a side-by-side comparison to validate the 25 years of data from the HPRR. It is also possible to build a mockup of the MK-9, a Little Boy-like weapon that was tested in the atmosphere at the Nevada Test Site and instrumented with radiation instruments. Ultimately, it is possible to return the HPRR to NTS to operate above ground level up to the Hiroshima burst height to more fully evaluate the dosimetry and activation contours on the ground. A National Academy of Science team is currently conducting another reassessment of the Hiroshima dosimetry because of outstanding differences between calculation and observation.

CONCLUSIONS

It is desirable to maintain the primary dosimetric standard within the DOE.

It is cost-effective to maintain HPRR at Los Alamos.

It may be desirable and cost-effective to extend the use of the HPRR to NTS, and to compare the radiation leakage with that from a true replica of Little Boy and the MK-9.

FROM DEUTERONOMY TO DOE RULES

John J. Schinkle
DOE-HQ

The test involving reactor number 4 at the V.I. Lenin Nuclear Power Station was intended to demonstrate safety in the event of a simultaneous loss of utility power and the turbogenerator. The inertia of the rotating equipment would be shown to be sufficient to provide power to feed water pumps and emergency core cooling systems until startup of the standby diesel generators. On April 25, the power was reduced from full power (3200 MWt) to the 700 - 1000 MWt range specified for the experiment. The automatic control system for a group of control rods was turned off. Poor procedures resulted in power reduction below 30 MWt. By 1:00 a.m. on the morning of April 26 the operators struggled to raise the power level to 200 MWt by manual control rod withdrawal; but could raise power no further due to xenon buildup. It appears that the operators were not aware of the extremely precarious condition in which the reactor had been placed. In this configuration, the operating margin had been reduced to one-fourth of the safety limit. They define the operating margin as the rod worth over the first few centimeters of insertion. Even worse was the high flow rate producing coolant conditions very close to saturation, meaning a small temperature increase could cause extensive flashing to steam. This was a terribly important variable because this type of reactor exhibits a positive void coefficient of reactivity. This means that if water flashes to steam, reactivity increases, raising power and temperature, thereby increasing steam production - a highly unstable condition!

From this configuration, the operators bypassed automatic scram circuits at 1:22 a.m. At 1:22:30 a.m., the reactivity printout showed that the reactivity margin had fallen well below the safety limit, the point at which an immediate shutdown should have been initiated, the control rods having been rendered relatively worthless. At 1:23:04 a.m., the turbine stop valve was closed, causing four primary coolant pumps to coast down, producing a sharp rise in reactor power.

In the early-morning hours of April 26, 1986, the reactor near the Ukrainian city of Chernobyl exploded. In the ten days following the accident, millions of curies of fission products were released, on the order of 12 M Ci the first day, falling to 2 M Ci after several days and then rising to 8 M Ci on days eight and nine as the core temperature rose as material was dumped to seal it off. Dose estimates for external and internal exposures are on the order of millions of person-rem.

An operator noticed that the reactor's power had begun to rise. He alerted another who, glancing at the computer printout, shouted that he was going to shut down the reactor and pushed the scram button at 1:23:40 a.m.After a few seconds, there was a thud followed by further thuds from deep inside the building.... an operator looked up at the instruments and saw that the descending control rods had stopped. He immediately disconnected the servo to let them fall under their own weight. They did not move. Then simultaneously there came a terrible tremor together with a sound like a clap of thunder. The walls shook; the lights went out; and a drizzle of plaster-dust rained down from great cracks in the ceiling....

Once again, the world's nuclear community had an opportunity to learn a harsh lesson adding to the previous lessons from SL-1 and Three Mile Island in the U.S.; Leningrad and Mayak in the former Soviet Union; and Windscale in the U.K. Perhaps the three most fundamental lessons from these and numerous other disasters involve:

1) The grave risks that may result when mission takes precedence over safety

2) The grave risks that may result when operations fail to embrace the philosophy of "Conduct of Operations" (INPO - 84~21 and DOE Order S480.19)

3) The grave risks that may result when failure modes go unrecognized or uncorrected.

As tragic as the Chernobyl accident was, it is interesting to note that projected chronic health effects have, thus far, failed to materialize. The leading hematologist in the region stated his views succinctly.

In the U.S., we expend considerable resources in an effort to reduce radiation exposures to what are historically minute levels. These resources are drawn from a national treasury that, were it not for government printing press, would long ago have been declared terminally bankrupt. However, every year around October, or thereabouts, an event occurs that is as predictable as Halloween; specifically Congress passes appropriations for the fiscal year which produces another tidal wave of red ink on top of the ocean of debt currently swamping the country. So the question might be asked, are the expenditures associated with radiation dose reduction cost-effective?

I am not qualified to answer the question and must defer to the experts. They deal with strange criteria such as 8 DAC hours. Like what the hell is a DAC hour anyway and why can't HP's use Pacific Daylight Time like the rest of us? It is worth reflecting on this bar chart (which may be dated, but nonetheless, offers some perspective.) However, I cannot help but observe that there is considerable potential for reduction of annual doses from background radiation for those of us that live in the Rocky Mountain region of the U.S. Annual doses from background radiation could be reduced by as much as 100 mrem per year simply by moving to Ohio.

I have spoken privately with a number of health physicists and it appears there are many important and beneficial requirements in the new Radcon Manual. However, there seems to be a consensus that some of the new health protection requirements are excessive, and wasteful of taxpayers dollars. Surprisingly, there seems to be little open dialogue on the matter. If these perceptions are correct, then I question whether we are fulfilling our moral obligation to our country and the U.S. taxpayers by remaining silent. And never forget, your grandchildren will someday pay for the resources we expend today.

The question of whether the resources expended are justified by the benefits received applies universally, including the commendable goal of protecting and improving our environment.

Laws, regulations, and political activism in support of the goal of protecting and improving the environment provide numerous examples of misguided efforts. A lawyer who was an Earth Day organizer in 1970 and who specializes in environmental law has stated that "there is a real risk of a Chicken Little mentality in the environmental movement" and notes as examples the boycott of colored toilet tissue in 1970, the campaign against disposable diapers which have significant benefits and a trivial impact on landfills and the billions of dollars currently being spent for unnecessary asbestos removal.

Another Earth Day organizer now admits to embarrassment in the strong opposition to the supersonic transport based on the false assertion that nitrogen oxide emissions would destroy the atmosphere. Former EPA chief William Reilly observed the disproportionate reaction to the use of Alar in apple products. It was amusing in a pathetic sort of way to observe Americans ceasing to eat apples but continuing to stuff themselves with junk food.

Rosenberg commented on what he perceives to be wasteful spending in support of the objective of clean water.

I should hasten to state that I became an environmentalist at a very young age. I grew up in a paper mill town at a time when air emissions and liquid effluents were unregulated and untreated. Fortunately, considerable progress has been made in both areas.

In the DOE in recent years, there has also been an increasing emphasis in safety, including "conventional" (i.e., nonnuclear) safety and nuclear safety, with extraordinary initiatives in the latter. The number of nuclear safety Orders is increasing geometrically as can be seen in the following chart, with numerous additional Orders pending. Further, these Orders along with others including the radiation protection Order will be codified as Rules under provisions of the Price Anderson Amendments Act of 1988. The Rules which result from this process will include enforcement provisions involving civil and criminal penalties. For better or for worse, this process will add legal risks to the normal technical risks associated with nuclear operations.

In ancient times, life was much simpler. There were only ten rules. Now with DOE Rule making, there will be tens of thousands of specific rules for nuclear operations.

The question remains as to how environment, safety and health should be dealt with in an era of continuously diminishing federal resources. I believe the answer is clear, and the answer would be the same whether resources were limited or not.

There is an urgent need for those of us with environment, safety and health responsibilities to focus on risk and risk reduction benefits before resources are expended in the name of environment, safety and health.

Having identified the more important risks, the question naturally follows as to what if anything should be done to reduce those Asks. What I propose is that options be identified for reducing risks and that cost estimates for those options be developed. Then ratios for risk reduction to cost can be readily determined. Finally, if one priorities expenditures based upon this cost-benefit ratio, the result will be the maximum risk reduction per resources expended. The conceptual result is shown pictorially on the next chart. At some point, a subjective judgment is made to the effect that further expenditures are not justified due to the minimal benefit in risk reduction. I have titled this process "Optimum Risk Management. "

SESSION III: ACCIDENT ANALYSIS

METHODS FOR PERFORMING CRITICALITY-RELATED ACCIDENT ANALYSES FOR THE SNL HCF SAR*

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SUMMARY

This paper presents methods used to perform criticality-related accident analyses for the SNL Hot Cell Facility (HCF) SAR. This includes the identification and screening of locations and processes within the facility that have potential for criticality. The paper also describes the methods being used to determine the accident frequency of such criticality events as well as the source terms, and dose consequences. The consequences and frequency estimates are combined into risk estimates. Release fractions and building removal fractions are defined for 12 chemical groups. Worker doses are determined using the DOSES computer code and a variety of standard shielding codes. DOSES calculates the dose due to immersion and inhalation from airborne radiocnuclides inside the facility.

These analyses have shown that criticality potential at the HCF is limited only to the steel containment boxes (SCBs) portion of HCF and the fissile material storage area, Rm 108 of building 6580. Workers could receive lethal doses from a criticality incident at either location if they are in the vicinity of the event when it happens. Public and onsite consequences (outside the immediate vicinity) are negligible.

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* This paper was presented at the Criticality Workshop, Monterey, CA, April 27-28 1993.

SESSION IV: EXPERIMENTAL FACILITIES AND MEASUREMENTS

SPENT FUEL SAFETY EXPERIMENT (SFSX): A NEW INITIATIVE

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Calculations to ensure criticality safety in the storage and transport of reactor fuel have traditionally been based on the assumption that all fuel, including spent fuel removed from reactors, is still in its most reactive or "fresh" state, ignoring the reduced reactivity caused by irradiating the fuel in the reactor. This assumption increases costs and imposes unnecessary penalties in the handling of spent fuel. For example, the "fresh fuel" assumption forces utilities to ensure control of criticality by adding neutron absorbers (boron) to spent fuel storage pool and casks. The Electric Power Research Institute has estimated that the nuclear power utilities could save over $500 million by eliminating the boron materials needed just for storage casks to contain the present inventory of spent fuel. The borated material is unnecessary if calculations including the reduced reactivity of the spent fuel are accepted for nuclear criticality safety. Applying these principles to the design of transport casks for spent fuel, the capacity per cask can be increased by factors of three or more, resulting in decreased public and occupational radiation exposure and significant cost savings due to the reduction in the number of spent fuel shipments, handling operations, and casks required. This improvement can be accomplished without compromising acceptable safety margins. To obtain regulatory acceptance, criticality calculations involving the use of burned spent fuel must be validated by comparison with experiments. However, there have been no laboratory spent fuel critical experiments reported in the open literature. There are reports of critical experiments using fresh mixed-oxide fuels that have been fabricated to emulate the U/Pu ratios in spent fuel. Also, commercial nuclear power plants routinely perform reactor physics measurements at the beginning of each fuel cycle that may qualify as spent fuel criticals. The experiment proposed in this summary and presentation will provide a direct measurement of pressurized-water reactor (PWR) spent fuel in a near-critical configuration.

The objective of this experiment is to establish a benchmark for validating criticality calculations using spent fuel from a PWR. An approach-to-critical (or inverse multiplication) technique that has been used extensively to determine the reactivity of fresh fuel will be used to determine the size of the critical array. The experiment will use segments of fuel rods one meter in length arranged in a symmetrical lattice immersed in water. Segments are added to the lattice until the multiplication of neutrons from a small source indicates a close approach to criticality. The number of fuel segments required for criticality can be estimated very accurately by monitoring the neutron count rate without reaching criticality. The innovative aspect of this experiment is the replacement of the center (7-rod) portion of the fresh fuel array with spent fuel segments whose chemical composition has been measured. Fresh fuel rods will again be added to the periphery and the count rate monitored to predict the number of fuel rod segments required to achieve criticality. Two separate spent fuel tests will be performed in addition to the fresh fuel case. The first will use seven rods taken from the center one-meter sections of spent fuel rods. In a PWR, burnup varies as a function of axial position, and all attempts are made to flatten this axial shape as much as possible to achieve the maximum economic benefit from the fuel. The result is a burnup profile that is fairly flat in the center of the fuel but sharply varying at the ends of the fuel due to the lower neutron importance in those regions. These underburned (more reactive) tips are of concern in the criticality safety analysis of burned fuel. Performing a second spent fuel experiment using 1-m long fuel segments taken from the end of a PWR fuel rod will provide a benchmark against which the ability to predict the reactivity worth of the underburned tips may be validated.

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* A United States Department of Energy Facility supported under Contract No. DE-AC04-76DP00789.

The experiments are planned to be performed at Sandia National Laboratories (SNL). A criticality experiment (CX) recently conducted at SNL is similar to the planned SFSX. Personnel and facilities involved in the CX will be utilized in the design of the SFSX. The fresh fuel, lattice plates, and other equipment that were used in a series of fresh-fuel critical experiments performed at Pacific Northwest Laboratories (PNL) will also be used in the SFSX. The spent fuel rod segments will be taken from PWR fuel used in the spent fuel characterization program at the Materials Characterization Center (MCC) at PNL. The fuel rods used in the experiment will be from positions directly adjacent to a rod that has been destructively assayed. The burnup profile for each rod will be measured.

A CRITICAL ASSEMBLY DESIGNED TO MEASURE NEUTRONIC BENCHMARKS IN SUPPORT OF THE SPACE NUCLEAR THERMAL PROPULSION PROGRAM

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SUMMARY

A reactor designed to perform criticality experiments in support of the Space Nuclear Thermal Propulsion (SNTP) program is currently in operation at the Sandia National Laboratories' reactor facility. The reactor is a small, water-moderated system that uses highly enriched uranium particle fuel in a 19-element configuration. Its purpose is to obtain neutronic measurements under a variety of experimental conditions that are subsequently used to benchmark reactor-design computer codes. Brookhaven National Laboratory (BNL), Babcock & Wilcox (B&W), and Sandia National Laboratories (SNL) participated in determining the reactor's performance requirements, design, follow-on experimentation, and in obtaining licensing approvals. BNL is primarily responsible for the analytical support, B&W the hardware design, and SNL the operational safety. All of the team members participate in determining the experimentation requirements, performance, and data reduction. Initial criticality was achieved in October 1989. An overall description of the reactor is presented along with key design features and safety-related aspects.

Criticality assemblies have historically been used to gain knowledge of the neutronic behavior associated with a variety of reactor parameters, to build a data base of information associated with criticality, and to compare the results of specific experimental configurations to predicted values. The power industry and national laboratories have used criticals extensively in the past four decades. With the advent of state-of-the-art neutronic computer codes, criticality and reactivity effects associated with design changes, temperature effects, and other factors can be predicted with greater accuracy and certainty. There are many situations, however, that require neutronic benchmarking by experimentation before a firm reliance can be placed on both a neutronic code and cross-section library. This reliance would allow future design changes to be made to a specific reactor with some knowledge of the uncertainty in the result.

For the SNTP program, a particle-bed reactor that is small, heterogeneous, and maintains a temperature differential of approximately 3000 K from the moderator to the exhaust is being considered. The moderator may be maintained at a temperature of less than 100 K. Exotic or previously untested materials may play important roles in the success of such a reactor. Design margins, controllability, and limiting accident conditions are all important topics that must be addressed using the most accurate, successful, and well-benchmarked neutronic codes. Although a critical assembly cannot address every scenario, or mock up the exact conditions that are expected to be encountered in such a design, it can be used over a range of experimental conditions to model some of the neutronic behavior expected to occur.

It is with this complementary aspect of analytical and experimental analysis that our critical assembly, or CX, was designed and constructed. The CX is currently in operation at the Sandia National Laboratories' reactor facility where both the Sandia Pulsed Reactor (SPR) and Annular Core Research Reactor (ACRR) also operate. It has been licensed, operated, maintained, and staffed in a manner consistent with SPR and ACRR. The CX team is made up of both analysts and experimentalists from each organization. Although our roles overlap significantly, each organization has primary responsibility in an area of expertise. BNL is primarily responsible for the analytical support for the experiments, B&W the hardware design and construction, and SNL the operational safety. All of the team members participate in determining the experimentation requirements, performance, and data reduction. Initial criticality was achieved in October 1989, and we have, to date, safely and successfully conducted over 100 operations.

TOPAZ II FUELING AND CRITICALITY ISSUES

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Phillips Laboratory

The Topaz II Nuclear Space Reactor is being purchased from Russia by the U. S. Strategic Defense Initiative Organization (SDIO) for use in selected high orbit missions. Although the Topaz II reactors have never been flown by the Russians, they are the next generation of Topaz I reactors, which have been flown by the Russians. This "pathfinder" project is part of the US/Russian scientific exchange effort being encouraged to help stabilize the Russian scientific institutions.

The reactor uses 96% highly enriched $U0_2$ fuel with a thermionic nuclear/electric conversion. The moderator is ZrH and the reactor uses axial and radial Be reflectors. The reactor is rated 115 kWt and 6 kWe for a 1-3 year mission. A LiH shadow shield is installed to limit the neutron and gamma fluences to the spacecraft. The coolant for the reactor is NaK. The Russian TFE design permits relatively easy fuel loading.

Nuclear safety is a primary concern to the work being performed by a combined team of Phillips, Sandia, and Los Alamos Laboratories working with the University of New Mexico (UNM). The focus for much of the reactor physics and fuel work is being done at LANL. The fuel may be purchased directly from the Russians or produced at LANL, depending upon cost and political considerations. Thus far, there has been one major neutronics issue of concern. MCNP calculations show the reactor could go critical in an accident mode when the reactor re-enters, after a launch abort, falls into water and is surrounded by sand. Criticality tests performed by the Russians indicate this possibility and a joint US/Russian team of engineers are developing an anti-criticality system to preclude this possibility.

Fuel fabrication uncertainties in Russia require the use of criticality measurements to determine the exact fuel loading configuration prior to launch. Criticality experiments will be conducted in New Mexico and Russia to verify computer codes, ascertain fuel configurations, and determine safety drum and reflector worths. It is anticipated that most of the criticality measurements will done in New Mexico with some minor criticality testing to be done at the launch site.

GODIVA IV REACTIVITY TEMPERATURE COEFFICIENT

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Los Alamos National Laboratory

Operations of the fast burst reactor, GODIVA IV, have been undertaken as part of the restart of the LACEF. We have treated these operations as if we were starting a new reactor. We kept a record of the room temperature, assembly temperature, and initial delayed critical configuration as part of the normal daily procedure with the anticipation that this would serve as a daily verification of the configuration of the material. The static reactivity coefficient measured (0.006 \$/°C) is much larger then we expected from the published coefficients for the GODIVA reactors.

An upper limit for the temperature coefficient can be calculated by assuming a sphere of solid uranium and using the average expansion coefficient ($15*10^{-6}$/°C) for 1.5% molybdenum alloy to get 0.005\$/°C. The static temperature coefficient for Lady Godiva (GODIVA I) was found to be 0.0042 \$/°C. While Lady Godiva was a spherical assembly, but it was made up of several pieces and the gaps between pieces are believed to be responsible for the smaller coefficient. Other published static temperature coefficients include 0.0031 \$/°C for GODIVA II and 0.0031 \$/°C for the Health Physics Research Reactor. Similar numbers are quoted for the dynamic temperature coefficient as determined by the quench of super-prompt-critical operation and accounting for the peak to average power distributions developed during the burst for both machines. The static numbers were obtained by heating the reactors using external heat sources. The number we are quoting was measured over many weeks and as part of the initial operations. Thus the reactor was at a stable temperature for each measurement and had been at that temperature for many hours. GODIVA IV is cylindrical, made up of several pieces, and contains a steel piece in the center.

SESSION V: UNIVERSITY PROGRAMS

COMPUTER SIMULATION OF NUCLEAR POWER PULSES IN SOLUTIONS

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SUMMARY

Several new computer models for predicting the effects of criticality accidents in aqueous fissile solutions have been proposed and tested in recent years. These models have attempted to predict transient pressures and kinetic energy as well as fission rates and energy yields. Both single-region models and multi-region models have been developed, and various methods for representing the production and effects of radiolytic gas have been tried. Computed results have been compared to experimental data from pulsed reactors (KEWB, CRAC, and SILENE).

This paper is a review and comparison of some of the models, including new results using recently revised models for radiolytic gas production and a new two-dimensional hydrodynamic formulation. Fast excursions in water solutions may be regarded as separated into three time regimes: (1) before dissolved gas saturation, short-lived cavities produced by fission fragments cause a small increase in compressibility; (2) after saturation, but before peak pressure, when rapid diffusion of dissolved gas produces rapid bubble growth, a sudden increase in inertial pressure, and greatly increased compressibility; and (3) following pressure relief caused by expansion, when the remaining excess gas is suddenly released from solution in a process similar to violent cavitation. The largest induced experimental pulses in cylinders containing highly enriched uranyl sulphate and nitrate solutions were in the range of about 3 to 6 MJ (about 10^{17} to 2×10^{17} fissions), and the results of our simulations are in fair agreement.

Some new extrapolations to dilute plutonium-water solutions that may have a positive neutron temperature reactivity coefficient are also included. There can be some rather interesting dynamic competition between positive and negative components of feedback reactivity. In shapes with a small negative expansion (e.g., critical assemblies having thin slab geometry), some rather large excursions might be possible. We have simulated hypothetical excursions with yields of several times 10^{19} fissions in extreme situations, but the extrapolation of our models to these plutonium solutions has many uncertainties and is not yet justified by any experimental data.

LADY GODIVA AND REACTIVITY WORTH MEASUREMENTS

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University of New Mexico

The reactivity worth measurements performed in Lady Godiva were duplicated numerically using the Hansen-Roach sixteen-group cross-section library and TWODANT, a deterministic neutron transport code. The purpose of these calculations was to identify bounds in which the Hansen-Roach library is applicable for fast neutron systems. The reactivities were determined from k_{eff} calculations using TWODANT for the case where both a sample and a void were modeled at the center of the assembly. The results from these calculations were mixed: the reactivities from some isotopes (B and Ni) agreed well with experimental values while others (Be, C, and Al) were off by as much as a factor of three. In a few cases (Fe, Co, and Th), the sign for the reactivity change was incorrect. Although uncertainties from sample purity and placement in the asembly may be contributing to the errors in the calculated reactivities, the energy group structure of the Hansen-Roach library appears to be adding to this effect.

Examining the continuous cross-section data for the samples that did not agree with experiment revealed that the absorption cross section varied by a couple orders of magnitude over small energy ranges (~ 1-2 MeV), and these variations were not always well-represented in the Hansen-Roach library. Performing the same calculations using the MENDF cross-section library resulted in much better agreement, as a whole, between the experimentally and theoretically determined reactivities. MENDF is a 30-group cross-section library and has greater energy definition (more groups) at energies above 0.1 MeV (the region in which the Godiva flux is dominant) than the Hansen-Roach library. The additional energy groups in this range may be the likely reason for the improved agreement between experiment and calculation. However, provided the cross sections for the replacement material do not undergo large changes (less than an order of magnitude) in any given group, the Hansen-Roach cross section library seems to predict the reactivity worth of a sample accurately.

The next step in these calculations is to remove the uncertainty of the sample geometry from the flux calculation by applying first order perturbation theory to calculate the reactivities. In these calculations, the unperturbed forward and adjoint fluxes are determined with TWODANT and used to calculate reactivity.

RECONFIGURING A TRIGA CORE FOR UNIQUE APPLICATIONS

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Penn State Radiation Science and Engineering Center

Penn State University has operated its Breazeale Nuclear Reactor since 1955 for the purposes of education, research, and service. A large graduate and undergraduate program in nuclear engineering has evolved which uses the reactor extensively. In addition, there is extensive multi-disciplinary utilization by Penn State, surrounding universities and high schools, and industry. The program is typical of many of the 33 U.S. universities operating research reactors.

The Breazeale Reactor is a one-megawatt TRIGA Mark III pool reactor. The core is supported from a bridge which traverses the length of the pool. Beam tubes penetrate one end of the pool. The following specific examples of projects utilizing the reactor and enhancing its capabilities are discussed. with emphasis on those involving changes to the core configuration:

1. *Frequency Response Measurement*: One of the reactor physics experiments for nuclear engineering undergraduates requires measurement of the TRIGA frequency response. Rotating and stationary semicircular cadmium disks are placed at the core face simultaneously to produce a reactivity perturbation. Compensated ionization chambers show the amplitude and phase shift of reactor power as oscillator frequency is varied.

2. *Approach to Critical*: Pre-college groups are involved in predicting the critical rod position by plotting the inverse count rate.

3. *Boron Depletion*: During the TMI-2 defueling, reactor operators were shown the indications of loss of chemical shim, simulated by a TRIGA core dispersed with borated dummy fuel elements.

4. *Multi-variable Control*: Advanced reactor control theory is demonstrated by controlling a movable experiment of less than one dollar reactivity using experimental algorithms. Core flow and void fraction can be varied to simulate power reactor reactivity mechanisms.

5. *Cold Neutron Sources*: A chamber beside the core allows investigation of typical moderator materials and configurations.

6. *Fuel Loading Density Change:* The fuel cycle was improved dramatically by increasing the TRIGA fuel density from 8.5 to 12 weight percent.

7. *Thermal Column*: A heavy water tank was installed between the core and a beam tube to provide a well-thermalized neutron beam for neutron radiography.

8. *Core Movement*: By modifying the bridge structure the core can be moved with three degrees of freedom (x, y, θ) such that it docks to fixtures throughout the pool. This will also allow studies of beam tube optimization for thermal and epi-thermal beams of radial and tangential configurations.

The flexibility of the Breazeale reactor core and pool configuration along with the ingenuity of many researchers have contributed to active and productive educational and research programs at Penn State.

AN OVERVIEW OF THE NUCLEAR CRIT'CALITY SAFETY PROGRAM AT THE UNIVERSITY OF TENNESSEE

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The University of Tennessee-Knoxville

The Nuclear Criticality Safety (NCS) program at the University of Tennessee-Knoxville is an academic specialization for nuclear engineering graduate students consisting of both special NCS courses and NCS research projects. Two courses are offered sequentially in consecutive semesters. The first course is an introductory course taught by a team of instructors; namely, H. L. Dodds, C. M. Hopper, J. T. Mihalczo, J. T. Thomas, and R. M. Westfall. During the past year, these two courses were presented locally in Oak Ridge, TN, and remotely in Portsmouth, OH, and Paducah, KY, via live teleconference (i.e., live, 3-way, audio and video).

Financial support for the research aspect of the program is currently provided by Martin Marietta Energy Systems, EG&G-Rocky Flats, the U. S. Nuclear Regulatory Commission, and the U. S. Department of Energy. These sponsors currently support seven full-time graduate students and one part-time faculty member. Some part-time graduate students (not supported financially) also participate in the program. The research projects deal with real NCS problems which currently exist and, therefore, are of direct interest to the financial sponsors of the program.

With regard to the productivity of the program during the past year, five papers were presented at national ANS meetings by students describing their NCS research projects (two of these five received outstanding paper awards), two papers have been accepted for publication in Nuclear Technology, two papers have been accepted for presentation at the next NCS Topical Meeting, and most importantly, five students have accepted employment positions in the NCS field.

ACTIVITIES AT NORTH CAROLINA STATE UNIVERSITY RELEVANT TO CRITICALITY SAFETY

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North Carolina State University

There is no formal research program in criticality safety *per se* in the Department of Nuclear Engineering at N. C. State University (NCSU), nor do we offer courses in this specific subject. However, many of our teaching, research, and extension activities are relevant to issues in criticality safety. These activities include coursework in reactor theory, radiation transport, reactor and radiological safety, numerical methods including Monte Carlo, and design projects. Also, our faculty has taught special courses in criticality safety for a reactor fuel vendor, and some of our research in Monte Carlo methods and nodal diffusion methods may be applicable to criticality safety analysis. As part of our strategic planning for both undergraduate and graduate education, we are exploring the need at both levels for curricula in radiological engineering (as a sub-specialty of nuclear engineering). This need is driven by a growing worldwide interest in nuclear waste management, decontamination and decommissioning, spent fuel handling and storage, recycling of weapons grade materials, transmutation of nuclear by-products, improved safety and reliability of nuclear facilities, etc. Many of these areas involve criticality safety issues.

Of perhaps most interest is a recent sponsorship by Los Alamos National Laboratory of a graduate student performing research for a master's degree, with his topic being the analysis and experimental verification of the effects of voids in the "Sheba" device, which is a simple cylindrical homogeneous solution critical assembly. This experience has been an excellent application of the NCSU Traineeship program for the Master of Nuclear Engineering (MNE) degree, which has several industrial and national laboratory sponsors.

A typical MNE degree is completed over a 15-month period, with 6 months spent working on a project and 2 semesters spent at the University taking coursework. The project work is broken up into two time periods, an initial 2-month period prior to any coursework being at the sponsoring organization's facility. Because the student typically has not yet had any graduate courses, this period is spent becoming familiar with the project requirements, the codes and computer systems involved, devising an experimental approach, and getting an initial part of the project done. The student then spends two semesters taking all the classes that are required for the degree, sometimes including special topics related to the project, and in the process develops a much better theoretical understanding of the issues underlying the project. The final four months of the traineeship are then spent completing the project at the sponsor's facility. This latter period is then much more productive than the initial period because the student has a better understanding of the project, and has had two semesters to plan a successful conclusion. We hope to continue this fruitful collaboration with Los Alamos National Laboratory in the criticality safety area, to the benefit of the student, the Laboratory, and our Department of Nuclear Engineering.

One often overlooked benefit of the Traineeship program to the sponsor is the faculty guidance and participation in the project. This can vary from advising the student during his time at the University and periodically reviewing his project progress, to actual participation in the project in the same manner as with a thesis research effort.

Criticality safety appears in the undergraduate curriculum indirectly in some of the student design projects undertaken in the senior capstone design course. For example, this year a group of students are undertaking the preliminary design of a spent fuel dry storage facility to accommodate the total fuel output of a 1,000-MW(e) nuclear plant over a forty-year period. Design objectives are to keep facility costs as low as possible consistent with safety and license acceptability. Students must, of course, assess safety ramifications of many technical issues in addition to criticality safety, such as fuel temperature, radiation shielding and exposure, corrosion, and fission product release. In this process, they learn to address realistic design tradeoffs affecting criticality safety.

Criticality safety has received an enhanced visibility in North Carolina after an incident about a year ago at a nuclear fuel manufacturing plant in which low-enrichment uranium was accidentally discharged to a large waste tank. As a result of its review of criticality safety procedures, the fuel manufacturer decided that its chemical, mechanical, and electrical engineers who design and operate the plant needed to have a better understanding of criticality issues. One of the remedial measures taken was to sponsor a Nuclear Criticality Workshop, presented by one of our faculty to the process engineers. This intensive three-day workshop covered the fundamental concepts needed to understand criticality, nuclear criticality safety principles and techniques of analysis, and sample criticality analyses. It was offered three times last summer and may be repeated in the future as the need arises. It is interesting to note that although fuel plants employ criticality safety specialists, the University can still serve a valuable function in organizing the knowledge base and presenting it in a structured learning environment to other engineers who are not specialists in this field.

These few examples of the emergence of criticality safety issues in the undergraduate and graduate curricula are clearly not unique to the Department of Nuclear Engineering at NCSU. However, we have made a conscious decision in our planning to emphasize more radiological engineering aspects of nuclear engineering in our future coursework and research, in response to the perceived demand for future graduates in this field.

WORKING GROUPS

PARAMETRIC STUDIES WORK GROUP

Hans Toffer and John Pearson
Westinghouse Hanford Company

The work group had an excellent well-attended and productive session on Monday. Due to the large audience including a lot of new faces the progress of the work group activities was discussed. Over the last four years, the effort of the work group has focused on developing a knowledge-oriented data base of all the pertinent nuclear criticality literature. Significant progress has been made with the help of the work group members. Over 1500 data entries are in the data base and over 800 of the entries have been knowledge screened to identify particular parameter studies they contain.

A new concept was proposed for this work group meeting, namely a "value" index on each data base entry. The concept was discussed with the members present, suggestions were received and incorporated in the definitions and groupings of the value index. Subsequent to the discussion, data base literature compiled by major contractor organizations were handed out. Attendees proceeded to go through those listings and "value" index the documents they were familiar with. Several hundred documents were covered. Subsequent to the meeting, additional individuals will be contacted for such input. The response and contributions by the people in attendance exceeded all expectations.

It is very important that the work group activities continue. The one meeting a year may not be sufficient to complete a knowledge band parameter study data base by the end of FY 1995. By that point in time, the data base effort should be completed and available to the criticality experts or other individuals interested in the multitude of parameter studies performed in the past. Goals for the immediate year include screening of additional references for the knowledge base matrix elements and implement the value index in the data entries. The tie into the OSTI criticality data base will be established to take advantage of the resources contained in it. Eventually the parameter study data base will be turned over to OSTI for administration.

EXPERIMENTAL NEEDS WORK GROUP FORECAST OF CRITICALITY EXPERIMENTS NEEDED TO SUPPORT NUCLEAR OPERATIONS IN THE UNITED STATES OF AMERICA: 1994–1999

D. A. Rutherford

ABSTRACT

This report identifies critical experiments forecast for 1994-1999, based on the consensus of the Experimental Needs Identification Workgroup, which is sponsored by the Department of Energy's Nuclear Criticality Technology and Safety Project.

I. INTRODUCTION

From July 27-28, 1993, the Experimental Needs Identification Workgroup (ENIWG) held a meeting to discuss the current and projected need for criticality experiments and facilities. Sponsored by the Department of Energy's (DOE) Nuclear Criticality Technology and Safety Project (NCT&SP), the ENIWG comprises representatives from the following communities: DOE contractors, DOE program offices, special groups working in the area of criticality safety, DOE critical mass laboratories, and the Nuclear Regulatory Commission. At this meeting, the Workgroup identified those nuclear criticality experiments that are necessary to support the DOE's changing programs and diverse production operations. This "Forecast" is generated by the Chair of the Workgroup, with input from the aforementioned groups. This document is considered a "living" document and will be updated periodically.

Current Concerns

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

II. ENIWG OPERATIONS

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

Identifying Experiments

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

Rating Experiments

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

Each experiment listed in the document has a *priority* listing that is one of the following: (1) Maximum practical attention; (2) Required for new or ongoing DOE operation; or (3) Less urgent than (2).

The *status* ranking of each experiment is designated as one of the following: (1) Justification Completed, (2) Justification Being Prepared, (3) Experiment Identified, (4) Anticipated Need, (5) Experiment in Progress, or (6) Experiment Complete.

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

Summary Listing of Experiments and Their Priorities

Table I lists the 59 experiments that have been identified and prioritized. The 21 experiments considered highest priority (maximum practical attention) are listed in Table II.

Table I. Identified and Prioritized Experiments.

	Number of Priority		
Categories	Priority 1	Priority 2	Priority 3
HEU	2	4	0
LEU	2	5	1
Pu	4	2	0
Pu/U	0	1	2
Transport Waste	8	8	0
Baseline	4	3	5
Criticality Physics	1	5	1

Table II. Highest Priority Experiments.

Category/Experiment	Experiment Title
HEU	
104	Advanced Neutron Source
106	TOPAZ-2 Reactor Experiments
LEU	
206	Sheba Reactivity Parameterization
207	Sheba Reactivity Void Co-efficient
Plutonium	
301	Plutonium Solution in Concentration Range from 8–17 g/l
303	Effectiveness of Iron in Plutonium Storage and Transport Arrays
304	Plutonium with Extremely Thick Beryllium Reflection
306	Arrays of 3 kg Pu Metal Cylinders Immersed in Water
Transportion/Waste	
501	Assessment Program for Materials Used to Transport and Store Discrete Items and Weapons Components
502	Waste Processing, Transportation, and Storage Program
502c	Validation of WIPP Hydrogen Generation Calculations
502h	Minimum Critical Mass of Fissile-Polyethylene Mixture
502i	Criticality Studies Which Emphasize/Intermediate Energies
503	Validation of Criticality Alarms and Accident Dosimetry Programs
504	Accident Simulation and Validation of Accident Calculations Program
505	A Program to Evaluate Measurements of Sub-Critical Systems
Baseline Theoretical	
601	Critical Mass Experiments for Actinides
606	Plutonium with Extremely Thick Beryllium Reflection
608	Experiment to Extend Standard ANSI/ANS 8.7 to Moderated Arrays
610	Validation of Calculational Methodology in the Intermediate Energy Range
Criticality Physics	
702	Spent Fuel Safety Experiments (SFSX)

New Category

This subset of new criticality experiments which are needed is intended to cover the area of the applications to storage, transportation, waste, dosimetry alarm systems, training, emergency response, processing, and regulations and standards. The material is divided into two parts – Programs and Specific Experiments. The program areas are further subdivided into specific experiments where appropriate.

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

Training would be included in the continuing capability. The training is divided into three parts. The first is that training provided to those who operate the critical experiments. The second is a continuation, and expansion, of the nuclear criticality safety hands-on 2-, 3-, and 5-day training courses that have been provided for several years. The third type of training is an "intern-in-residence program to allow personnel an opportunity to gain experience in the day-to-day operation of a critical experiment facility. An important adjunct of the training program is the development of a simulator to demonstrate the characteristics of critical systems. It is proposed that this become a "catalog" item that would be developed under the auspices of the DOE and made available to contractors and others at cost.

Programs and experiments included in this category are identified below.

501P Assessment Program for Material Used to Transport and Store Discrete Items and Weapon Components. Priority 1

502P Waste Processing, Transportation, and Storage Program. Priority 1

503P Validation of Criticality Alarms and Accident Dosimetry Program. Priority 1

504P Accident Simulation and Validation of Accident Calculation Program. Priority 3

505P A Program to Evaluate Measurements of Sub-critical Systems. Priority 2

506 Spent Fuel Elements. Priority 2

507 Validation of WIPP Hydrogen Generation Calculations. Priority 3

508 Safe Fissile Mass Thresholds for an Array of Waste Storage Drums. Priority 2

III. RESOURCES AND STATUS OF FACILITIES

The current (1993) status of available critical facilities and their resources are listed below. Although several facilities have been closed, they are listed here for historical reasons. Included in the description of each facility are the

- core technical capabilities (that is, what assemblies or test cells and what materials are available for experiments);
- current documentation (for example, SARs, TSRs, and operating procedures); and
- personnel resources.

A. LACEF

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

Table III. Critical Assemblies at the LACEF.

Assembly	Type	Applications
Big Ten	Large fast-spectrum steady-state benchmark assembly	1, 2, 3, 4
NPR	Prototype zero-power experimental assembly (graphite/uranium compact fuel)	2, 3, 5
Comet	General-purpose vertical assembly machine (portable)	2, 5, 6
Flattop	Fast-spectrum steady-state benchmark assembly	1, 5, 6
Godiva IV	Fast-burst assembly (portable)	1, 2, 4, 6, 7, 8
Honeycomb	Large general-purpose horizontal assembly machine	5, 9, 10
Mars	Large general-purpose vertical assembly machine	3, 5, 6
Planet	General-purpose vertical assembly machine	2, 5, 6
Sheba	Liquid steady-state and burst assembly	1, 2, 4, 7, 8
Skua	Annular-core fast-burst assembly	1, 2, 7, 8
Thor	Spherical plutonium benchmark assembly	6, 9, 10
Venus	Large general-purpose machine (used for solutions)	1, 4, 5, 6, 8

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

Assemblies

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

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

(2) Assembly machines are general-purpose platforms into which fissile, moderating, reflecting, and control components can be loaded for short-range study of the neutronic properties of the materials. The assemblies do not contain fissile material – they only remotely manipulate it.

(3) Solution assemblies are specifically designed to allow critical operations with configurations containing fissile solutions.

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

2. Current Documentation and Personnel Resources. The LACEF staff is trained and certified and documentation is current.

B. Area V, Sandia National Laboratories (SNL)

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

Assemblies

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

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

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

(4) The kiva that houses the SPR reactor has also been used for the CX experiment recently. This critical assembly was used to perform experiments in support of the Space Thermal Propulsion program.

2. Current Documentation and Personnel Resources. The SNL staff is trained and certified and documentation is current.

C. Argonne National Laboratories (West)

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

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

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

D. Hanford Laboratories

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

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

E. Oak Ridge National Laboratory (ORNL)

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

2. Current Documentation and Personnel Resources. The facility has been shut down. There is no trained and certified staff and no current documentation.

F. Rocky Flats

1. Core Technical Capabilities. The Rocky Flats Critical Mass Laboratory (CML) is currently in a "standby" mode. The facility is gradually being defueled, decontaminated, and decommissioned. This has not been completed.

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

Assemblies

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

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

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

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

IV. CONCLUSIONS

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

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

Experiments Categories

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

(1) The majority of Priority 1 experiments are in the Transportation, Waste, Storage and Alarm Systems category (which is a *new* category), with the Criticality Physics and Plutonium categories each having 4 Priority 1 experiments.

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

(2) Criticality safety training is recognized as one of the most important aspects of maintaining our technical capability.

(3) The new priorities in the need for experiments reflect the change in the mission of the DOE and the current thinking in the nuclear community, as well as the continuing for experiments that are recognized as needed to support U.S. processing facilities.

(4) A concerted effort has been made to integrate Physics Criteria for Benchmark Critical Experiments document (see App. C) into this forecast.

(5) An important activity that arose from the meeting was to create an initial draft of criteria for establishing areas of applicability (see App. D).

Resources and Status of Facilities

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

Rocky Flats CML is currently on stand-by status.

Future Directions

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

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