Department of Energy
Washington, DC 20585
October 19, 1995

The Honorable John T. Conway
Chairman
Defense Nuclear Facilities Safety Board
625 Indiana Avenue, N.W.
Suite 700
Washington, D.C. 20004

Dear Mr. Conway:

Enclosed is the Department of Energy's annual report of activities related to the implementation of Recommendation 93-2, Critical Facilities Infrastructure. The Nuclear Criticality Experiments Steering Committee recognizes the importance of resolving the remaining budgetary concerns for Fiscal Year 1996, and plans to provide its recommendations shortly. In the interim, we will keep your staff fully advised of our progress.

Sincerely,

Victor H. Reis
Assistant Secretary
for Defense Programs

Enclosure

cc:
Mark Whitaker, EH-9, w/encl.

DEPARTMENT OF ENERGY
ANNUAL REPORT ON THE STATUS OF
DEFENSE NUCLEAR FACILITIES SAFETY BOARD
RECOMMENDATION 93-2
SEPTEMBER 1995

DEPARTMENT OF ENERGY
ANNUAL REPORT
FOR
DEFENSE NUCLEAR FACILITIES SAFETY BOARD
RECOMMENDATION 93-2

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Defense Nuclear Facilities Safety Board
Recommendation 93-2 Annual Report

Introduction

On March 23, 1993, the Defense Nuclear Facilities Safety Board (DNFSB) issued Recommendation 93-2, Critical Facilities Infrastructure, to the Secretary of Energy. The DNFSB recommended the following:

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.

On May 12, 1993, the Department fully accepted Recommendation 93-2 and submitted an Implementation Plan to the DNFSB on August 10, 1993. The DNFSB accepted this Plan on September 30, 1993. Referring to the Implementation Plan, the acceptance letter stated the following:

> "The DNFSB applauds the Department's setting of department-wide, long-term goals that include well documented critical experiments to confirm the adequacy of criticality computer codes and nuclear data, general critical experiments and training capability, and the improvement of criticality predictability."

The Department is pleased to report that the process which was established back in 1993 to implement this recommendation and manage the critical experiments program has not only succeeded in maintaining this important capability during a time of severe budget pressure, but also has matured significantly. Initially, the Department focused on maintaining the capability to conduct critical experiments and the training of criticality safety professionals. During the past year, however, the Department began viewing criticality predictability from a programmatic perspective. This view fully recognizes the contribution of all five program elements (Experiments, Training, Benchmarking, Analytical Codes, and Nuclear Data) to "the improvement of criticality predictability." Moreover, the Department recognizes that application of improved criticality predictability not only enhances criticality safety, but also could lead to significant cost savings in the handling and storage of fissile material.

The DNFSB also provided a comment concerning its sense of what would be required to successfully carry out the Implementation Plan:

> "The success of the Implementation Plan for Recommendation 93-2 seems highly dependent on the participation of all concerned parties. Vigilance will be needed at high levels to ensure that both the users and suppliers of experiments, computer codes, nuclear data, and training will participate. In the past, because of budget constraints, many concerned parties were unwilling to share responsibility."

The Department agrees with the DNFSB on the issue of shared responsibility, and the Secretary of Energy addressed this issue by tasking the Assistant Secretary for Defense Programs with the responsibility for developing the

Implementation Plan in consultation with all Departmental stakeholders. The Assistant Secretary for Defense Programs is the senior level authority within the Department responsible for implementation of the Nuclear Criticality Predictability Program.

The Implementation Plan established the Nuclear Criticality Experiments Steering Committee (NCESC) whose charter is to provide the Assistant Secretary for Defense Programs with advice on matters affecting the Department's criticality predictive capability. The NCESC consists of representatives of the various stakeholders within the Department who share the responsibility for the Nuclear Criticality Predictability Program. The NCESC and its subcommittees provide the Department with an established forum for the exchange of ideas among the stakeholders with a clear focus on the continuous improvement of criticality predictability and hands-on training for criticality safety professionals. The NCESC is consolidating and prioritizing programmatic needs and making recommendations to the Assistant Secretary for Defense Programs on how to meet those needs. Recommendations from the NCESC to improve the program are being actively supported by senior management.

The NCESC has reviewed and will continue to review the Nuclear Criticality Predictability Program from a systems engineering perspective. One of the program elements requires the Department to maintain its competency in conducting criticality experiments. Because maintenance of competency in conducting criticality experiments requires a long-term commitment from the Department, life-cycle considerations for the facilities that support this program element must be included in the process. Along with planning for the continued operation of existing facilities and potential construction of new facilities, the Department recognizes the need to plan for the eventual decommissioning and decontaminating of these facilities and environmental remediation of the sites where the facilities were located.

The second annual report, contained herein, informs the DNFSB of the overall status of the Department's Nuclear Criticality Predictability Program, including projected funding for Fiscal Year 1996. The report is divided into the following three sections:

> Section 1.0 contains an overall status of the Department's Nuclear Criticality Predictability Program. This section is divided into the five subsections, one for each program element. Each of these program elements is vital to the success of the Department's Nuclear Criticality Predictability Program. The status of each program element is provided with regard to funding, current capability, current requirements, and anticipated future direction.
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> Section 2.0 discusses program coordination between the Department and its criticality predictability customers. Because the Department of Energy maintains the vast majority of capability to conduct critical experiments and hands-on criticality training, as well as the tools necessary for modeling and predicting the state of criticality of fissile systems, the Nation relies heavily on the Department to meet its needs in these areas.
>
> Section 3.0 outlines key issues facing the Department that must be resolved in order to maintain capability and establish a culture that encourages continuous improvement in the Nuclear Criticality Predictability Program. To maintain capability and satisfy anticipated future requirements, the Department is continually canvassing the criticality community to identify requirements as soon as possible so they can be factored into program planning. In addition, budgets are being developed that permit programmatic agility to meet unanticipated requirements as they arise.

1.0 Nuclear Criticality Predictability Program Status

The Nuclear Criticality Predictability Program philosophy is not centered merely on maintenance of capability, but on continuous improvement in this capability. Each of the five program elements (Experiments, Training, Benchmarking, Analytical Codes, and Nuclear Data) is a vital part of the nuclear criticality predictive capability. In the past, baseline funding for each of the program elements was provided by a number of different program offices within the Department. The design and fabrication of nuclear reactors (fast and thermal) and process operations in support of the design, construction, and handling of nuclear weapons required nuclear data acquisition, analytical code development, and in some cases, critical experiments. A coordinated programmatic management strategy for criticality predictability program elements was not undertaken because funding, though not plentiful, was adequate to support each of the program elements. In this environment, the criticality predictability community was able to sustain itself because of clearly identified and funded programmatic needs. When the need for new nuclear weapon and reactor designs waned, so did the funding for the criticality predictability program elements.

During the past year, the NCESC has been studying this situation and has determined that a programmatic perspective is now required to maintain nuclear criticality predictive capability. If this capability is to be maintained and continuously improved, funding for each of the program elements must be carefully managed. In addition, tasks that involve more than one program element must be coordinated. The NCESC is addressing these issues and assessing the impacts on predictive capability vis a vis program element funding levels. This activity will result in a coordinated program in which resource utilization is optimized to maximize the Department's return on its investment in its Nuclear Criticality Predictability Program.

The Department recognizes that maintaining a viable nuclear criticality predictive capability is very important in assuring the continued safe operation of its nuclear facilities. The Department's mission focus has changed dramatically in the past few years from materials production and processing for use in nuclear weapons and nuclear fuels to materials processing to stabilize fissile materials for storage or disposal. Operations that support the disposition of the Department's fissile materials present new criticality concerns because they involve partially moderated systems where intermediate-energy neutrons are major potential contributors to the state of criticality of the system. In the past, the intermediate-energy portion of the neutron spectrum was of little concern to nuclear reactor and nuclear weapon designers. However, to a criticality safety engineer, faced with analyzing processes related to stabilizing fissile material for storage or disposal, the entire neutron energy spectrum, including the intermediate energy region, must be considered. In addition, these systems contain packaging and storage materials, some of which (e.g., silicon) have received scant previous attention with regard to the measurement and qualification of nuclear data. Consequently, a viable nuclear criticality predictive capability is just as important today as it has always been.

Funding for each of the program elements is contained in the following table. Fiscal Year 1995 figures represent funding received during that year, while the Fiscal Year 1996 numbers represent anticipated funding as of September 1995. Because funding is limited, critical experiments will be supported according to priorities that are established by the NCESC. Based on established priorities, the Department will fund one or two critical experiments per year. The critical experiments program is flexible enough to accommodate a priority requirement, should one arise. Projected funding is adequate for maintenance of the Department's capability to conduct critical experiments.

Providing adequate security for the special nuclear materials necessary for conducting critical experiments requires a significant amount of funding. The NCESC has been working closely with Laboratory and Departmental management to assure the availability of special nuclear materials required for critical experiments, while minimizing costs wherever possible without reducing the security posture at experimental facilities.

With regard to training, beginning in Fiscal Year 1996, the Department is instituting a charge-back system for hands-on criticality safety training. Given the anticipated amount of funds collected per student, the projected funding for hands-on criticality safety training is adequate to meet the Department's needs for the foreseeable future.

Resolution of the budget situation for the Nuclear Criticality Predictability Program is a major task facing the NCESC in Fiscal Year 1996. Funding for the experiments and training program elements has been stabilized at an appropriate level for maintaining reasonable capability in those areas. Following recognition of the importance of benchmarking, analytical codes, and nuclear data in maintaining the Department's nuclear criticality predictive capability, the NCESC began reviewing budget information for these program elements in late Spring of 1995. The NCESC is still evaluating whether the scope of the program is adequate to maintain capability, particularly with regard to the new program elements.

	FISCAL YEAR 1995 FUNDING	FISCAL YEAR 1996 FUNDING
LOS ALAMOS CRITICAL EXPERIMENTS FACILITY (LACEF)	$3,200	$3,450
SANDIA NATIONAL LABORATORIES AREA V	590 (1)	300 (1)
TRAINING (AT THE LACEF)	210	120 (2)
BENCHMARKING	2,756	1,100
ANALYTICAL CODES	1,175	450
NUCLEAR DATA	820 --------	555 --------
TOTALS --------------->	$8,751	$5,975

NOTES: ($s in thousands)
(The majority of the funding for analytical codes and nuclear data depicted in this table does not directly support nuclear criticality predictability applications. However, nuclear criticality predictability applications indirectly benefit from these activities.)
(1) funding for Spent Fuel Safety Experiments, Number 702
(2) Defense Programs funding is shown here. Tuition collected will supplement this amount.

The following sections present a current status of each of the Nuclear Criticality Predictability Program elements. Each element is described with regard to its current capability, current requirements, and anticipated future direction. The current capability section provides a status of the current capability in each of the program elements. The current requirements section provides the basis for maintaining or expanding the current capability. The anticipated future direction for each program element is based on current and anticipated programmatic needs within the bounds of fiscal reality.

1.1 Experiments

Maintaining the capability to conduct nuclear criticality experiments cannot be accomplished without adequate facilities, special nuclear materials, and qualified personnel. Facilities and special nuclear materials are absolutely essential, but qualified personnel are really the key elements in the maintenance of this important capability. Retaining quality individuals cannot be accomplished without challenging them and this requires the performance of a variety of meaningful operations involving special nuclear materials. The Department recognizes this and has factored these considerations into its program of critical experiments.

As the demand for new fissile nuclear systems declined, the need for critical experiments associated with the development of these systems declined as well. Nevertheless, critical experiments are still required to support current Departmental missions cited in paragraph 1.0, above. During Fiscal Year 1995, the Methodology and Experiments Subcommittee of the NCESC conducted a survey of the criticality community to determine if any changes to the prioritized list of critical experiments were warranted. Based on this survey, the Methodology and Experiments Subcommittee recommended to the NCESC that no changes be made to the prioritized list of critical experiments at this time. Periodic reviews of this list will be conducted to assure that Departmental needs are being addressed in an appropriate priority. The current prioritized list of critical experiments, contained in Attachment 1 to this report, is being used as the basis for a structured critical experiments program.

1.1.1 Current Capability

The only two Departmental nuclear research facilities that are fully capable of conducting critical experiments are The Los Alamos Critical Experiments Facility (LACEF) and Area V at Sandia National Laboratories (SNL). All other Departmental facilities where critical experiments had previously been conducted are either in operational standby or shut down and awaiting decommissioning. Both the LACEF and Area V are active nuclear research centers. Historically, the nuclear testing done at Area V has not been focused on criticality issues. Rather, it has involved radiation hardness of systems components, nuclear fuel assessments, advanced concept experiments, and recently, a medical radioisotope production demonstration utilizing the Annular Core Research Reactor. Aside from one series of critical experiments, scheduled to be conducted at Area V during the next two years, all other scheduled or proposed critical experiments are being conducted at the LACEF. With its ten critical assemblies, the LACEF currently offers the flexibility required to meet most of the Department's critical experimental needs at one location.

Both the LACEF and Area V currently possess a trained and certified staff for conducting nuclear operations. Because of the decrease in nuclear testing requirements as a result of the end of the Cold War, both facilities have undergone a decrease in staff. The NCESC is aware of this situation and is monitoring it to ensure that staffing levels are maintained commensurate with operational requirements and identified experimental needs.

The Department has determined that the facilities contained within the LACEF are adequate to meet most of the current requirements for conducting critical experiments and training criticality experts. Some of the high priority experiments identified by the NCESC, such as criticality issues associated with plutonium in solution and mixed plutonium and uranium oxides, may require the development of new experimental facilities. The Department recognizes these needs and is including them in future planning according to their priority.

1.1.2 Current Requirements

Eight experiments from the NCESC's priority experiments list are currently either in the planning phase, being conducted, or having results analyzed at the LACEF:

EXPERIMENT #	EXPERIMENT TITLE	STATUS
102	Large Array of Small Units	Planning
206	Solution High-Energy Burst Assembly (SHEBA) Reactivity Parameterization	Ongoing
207	SHEBA Reactivity Void Coefficient	Ongoing
502a	Absorption Properties of Waste Matrices	Planning
503	Validation of Criticality Alarms and Accident Dosimetry Program	Ongoing
504	Accident Simulation and Validation of Accident Calculations Program	Initiated
505	A Program to evaluate Measurements of Sub-critical Systems	Initiated
601	Critical Mass Experiments Program for Actinides	Initiated

One other critical experiment from the NCESC priority experiments list, Number 702, the Spent Fuel Safety Experiments, is funded and will be conducted at Area V, SNL. The necessary driver fuel and spent pressurized water reactor fuel samples are being prepared for delivery to SNL in October 1995. The experiment is expected to be conducted in early Fiscal Year 1996. If funding permits, future plans include performing similar critical experiments at SNL using the same driver fuel and spent boiling water reactor fuel samples.

1.1.3 Anticipated Future Direction

The NCESC and its Methodology and Experiments Subcommittee have established an ongoing process for determining and prioritizing experimental needs. Key members of the criticality safety community actively participate in this process. The primary product of this process is the prioritized list of critical experiments (Attachment 1). This list is the foundation for a structured critical experiments program which is focused on maintenance of this important capability. The Departmental funding for the LACEF is planned to be sufficient to conduct one or two critical experiments per year from the priority list. The Department's critical experiments program is flexible enough to allow unanticipated experimental needs to be met.

Future experimental facility development may be required to support some of the priority experiments identified by the NCESC. For example, if the collaborative effort within the international community does not yield the benchmark data necessary to resolve criticality issues associated with plutonium in solution and mixed uranium and plutonium oxides, new experimental facilities may have to be developed. The most likely location for these new experimental facilities is the LACEF; however, appropriate environmental analysis would have to be conducted in support of a siting decision. In addition, steps are being taken to identify and preserve certain special nuclear materials which are considered to be National assets because of their unique form, composition, or projected cost of regeneration. The NCESC is overseeing required facility development in support of anticipated experimental requirements and special nuclear material National asset preservation.

As for the existing experimental facilities at the LACEF, many of them are now over 40 years old and require an increasing amount of maintenance to assure safe operations. As part of the Department's commitment to maintaining capability in this area, the NCESC will evaluate and recommend support for facility upgrades at the LACEF as appropriate.

1.2 Training

The Department recognizes that hands-on criticality training is absolutely essential in maintaining an effective criticality safety program. The NCESC continued its oversight of the training program during Fiscal Year 1995. Because of limited funding, centralized control over training was deemed necessary to maximize the return on the Department's investment by ensuring that this important training function meets the Department's needs and is provided first to those who need it most. Six training courses were conducted at the LACEF during Fiscal Year 1995.

In addition to overseeing hands-on criticality safety training, the Training Subcommittee of the NCESC continued to identify criticality safety training needs for both Federal and Contractor staffs, determined which facilities and other resources are required to meet identified needs, and began work on determining an equitable funding scheme to support the training. Active participation from the Training Division of the Department's Office of Human Resources played an important role in these activities. This section provides a status of the Department's hands-on criticality training program.

1.2.1 Current Capability

The LACEF is the only Departmental facility that currently conducts hands-on criticality safety training. The NCESC has determined that the LACEF is adequate for this purpose, and developing another facility for hands-on criticality safety training at this time would not be cost effective. The variety of experimental facilities and special nuclear materials stored at the LACEF afford the Department with unique flexibility in structuring hands-on training to meet a broad spectrum of training needs. The current course structure provides attendees with an unparalleled understanding and "feel" for the physics and the art associated with neutron multiplication in the presence of fissile material. Moreover, the Los Alamos staff is capable of tailoring portions of a course to meet specific needs of the attendees.

1.2.2 Current Requirements

A survey of training needs, conducted by the Training Subcommittee, determined that there is a continuing need for this activity. A backlog exists of over 160 requests for admission to the criticality training courses offered at the LACEF. Priorities for admission to these classes have been established as follows: Priority 1: those persons requiring hands-on training in order to perform assigned tasks; Priority 2: those persons requiring hands-on training to meet qualification standards, and; priority 3: those persons desiring this training for career enhancement. Based on identified needs, the NCESC has recommended at least six hands-on criticality safety courses per year as part of the Nuclear Criticality Predictability Program. This does not preclude scheduling of additional courses if unanticipated needs arise and additional funding is made available.

1.2.3 Anticipated Future Direction

Hands-on criticality safety training will continue to be required at the Department for the foreseeable future. The need for nuclear criticality safety hands-on and other types of training media are projected to increase because of requirements for this training in qualification standards that will be implemented for Federal and Contractor employees. Also, in conjunction with the shift in the Department's mission, new materials handling processes and storage configurations may yield a new set of hands-on training requirements. The NCESC, through its Training Subcommittee, is continually assessing requirements and overseeing this important activity. The Department is committed to technical excellence and the continuing development of criticality safety expertise within both its Federal and Contractor staffs.

Aside from the training of nuclear criticality safety professionals, an increasing number of training needs have arisen from the non-proliferation and nuclear emergency response programs within the Departments of Energy, State, Commerce, and Justice, and the Central Intelligence Agency. The LACEF has already hosted some training classes to meet these needs. Because of the unique facilities and staff expertise, and variety of special nuclear materials available at the LACEF, the LACEF Staff will likely continue providing specialized training that supports emerging needs in this area as they are identified.

1.3 Benchmarking

The Department's program of critical experiments is accompanied by a broad assessment of available criticality benchmark data. These measured data represent an important resource for enhancing calculational methods. Until recently, no effort had been made to take full advantage of this resource. In 1992, the Department initiated the Criticality Safety Benchmark Evaluation Project (CSBEP) to identify and evaluate a comprehensive set of critical benchmark data, verify the data to the extent possible, compile it into standardized form, perform calculations of each experiment, and formally document the work. The project was managed through the Idaho National Engineering Laboratory (INEL), but involved nationally known criticality safety experts from a number of Department of Energy Laboratories.

1.3.1 Current Capability

In early 1995, the Department expanded the CSBEP into the International Criticality Safety Benchmark Evaluation Project (ICSBEP) which was accepted as an official activity of the Organization for Economic Cooperation and Development - Nuclear Energy Agency (OECD-NEA). *Also managed through the*

INEL, the ICSBEP members include the United States, the United Kingdom, Russia, Japan, France, and Hungary. This project, led by the United States, established an international forum for the exchange of nuclear criticality benchmark data. The first series of evaluations performed by the ICSBEP was published in May of 1995 as an OECD-NEA handbook entitled, "International Handbook of Evaluated Criticality Safety Benchmark Experiments."

The ICSBEP is focused on the following: to consolidate and preserve the information base that already exists in the United States; to identify areas where more data is needed; to draw upon the resources of the international criticality safety community to help fill identified needs; and to identify discrepancies between calculations and experiments. This program represents a tremendous capability. It provides the United States with the ability to access the global data base of experimental benchmarks to validate calculational methods which simulate the neutronic behavior of the fissile system being analyzed. As an illustration of the benefits of this program, the first evaluation from France includes plutonium in solution data with concentrations ranging from 13.2 to 105.0 grams per liter of solution. There are five experiments reported in this evaluation with plutonium concentrations below 20 grams per liter. These data fill a gap in the United States' data which was considered important enough to warrant one of the top ten priority experiments (Experiment number 301 on the Priority Experiments List, at Attachment 1). There is still a need for data between 7.5 and 13 grams of plutonium per liter, however, given the estimated cost of establishing the capability to perform plutonium in solution experiments in the United States, this single French contribution has saved the Department several million dollars.

1.3.2 Current Requirements

The International Handbook of Evaluated Criticality Safety Benchmark Experiments currently contains 46 evaluations with benchmark specifications for 376 critical or near critical configurations. An additional 101 experimental configurations were found to be unacceptable for use as criticality safety benchmark experiments and are discussed in these evaluations; however, benchmark specifications were not derived for such experiments. Nearly 80 new evaluations are in progress, most of which are from outside the United States. New evaluations will be published and distributed annually. The Handbook is organized in a manner that allows easy inclusion of revisions and additional evaluations as they become available. Continued United States participation in this process is absolutely essential for maintaining capability in producing meaningful benchmark evaluations and deriving further benefit from the international contributions.

1.3.3 Anticipated Future Direction

Large amounts of data exist within the United States that have not been evaluated and documented. In addition, the United States must continue its review of foreign data commensurate with its commitment as part of the ICSBEP process. Continuation of this work at an appropriate level is very important because some of these evaluations would be very useful in supporting the Department's mission needs. The NCESC will continue to monitor this situation and work closely with the benchmarking community in prioritizing requirements and recommending an appropriate level of support.

1.4 Analytical Codes

Analytical codes are central to an efficient criticality predictability program. These codes are indispensable for analyzing accident scenarios required for safety analysis reports. Currently the three general purpose Monte Carlo codes used to model the state of criticality of fissile systems throughout the Department are the MCNP code at Los Alamos National Laboratory (LANL), the KENO code at Oak Ridge National Laboratory (ORNL), and the VIM code at Argonne National Laboratory (ANL). Each of these three-dimensional Monte Carlo codes employs a slightly different calculational methodology. This diversity of methodology provides the Department with significant depth in its criticality modeling capability by allowing for comparison of calculational results from the different analytical codes. Supporting and ancillary codes which are used for scoping calculations or other tasks such as producing volume and flux weighted cross sections for use in the three dimensional Monte Carlo analytical codes are also important analytical tools which must be maintained.

1.4.1 Current Capability

The strength of the United States capability in performing calculational criticality analyses resides in the diversity of the three relatively mature Monte Carlo neutron transport codes cited above. The KENO-Va code is the current production version of the KENO series which has been specialized for criticality applications. Its major features include the energy-multigroup cross sections and neutron-kinematics approach, along with very efficient neutron tracking techniques. The KENO-VI code, which is in the validation and documentation phase, will provide a more general geometry-modeling capability at the cost of some efficiency. The MCNP series of general neutral particle transport codes offers a more rigorous neutron-kinematics treatment based upon energy-pointwise cross sections and a continuous energy mesh. The VIM code system, which also treats energy as a continuous variable, was developed for the fast reactor design program. Consequently, this code system features the most rigorous problem-dependent, resolved and unresolved-resonance shielding techniques. This capability is very important in addressing criticality issues associated with the new Departmental missions which require rigorous analysis of the intermediate-energy range inherent in partially-moderated fissile material storage, transportation, and waste processing systems.

1.4.2 Current Requirements

In addition to ongoing software quality assurance, configuration control and user assistance for the three code systems, top priority enhancements for each code have been identified. For the KENO codes, the associated AMPX Cross Section Processing System should be upgraded to be compatible with the most recent and complete nuclear data source. For the VIM codes, a related improvement was funded in Fiscal Year 1995 and is underway with the installation of the Reich-Moore resonance reconstruction formalism and accessibility to the latest nuclear data file. For the MCNP codes, the installation of a problem-dependent, unresolved resonance shielding capability has been recommended. Finally, all three code communities have proposed the development of ease-of-use features based on graphical user interfaces and additional statistical testing.

1.4.3 Anticipated Future Direction

At the present time, the Department is presented with an outstanding opportunity to incorporate analytical resources developed for other neutronics programs into a world class criticality predictability program. The expertise is in place to enhance existing methods and validate the combined methods and data in support of the new and emerging Departmental missions. This should be done in a way such that independent corroborative capabilities are maintained.

Consequently, the NCESC is pursuing all the recommended code enhancements cited above, as funding permits.

1.5 Nuclear Data

Accurate nuclear data is the foundation of nuclear criticality predictability. Without it, the codes have very limited worth. In order for nuclear data to be utilized, it has to be measured, evaluated, put into standard format, tested, released as part of the Evaluated Nuclear Data File (ENDF), and then processed into the working formats of the three-dimensional analytical and scoping codes. Key activities in this area fall under the Cross Section Evaluation Working Group (CSEWG). Early in Fiscal Year 1995, in response to concerns expressed by the DNFSB Staff about NCESC representation in the CSEWG, the NCESC appointed one of its members as an official representative to the CSEWG. Accordingly, the NCESC is represented at CSEWG meetings and continues to maintain contact with the chairman of CSEWG, as well as other cognizant experts. This liaison is very important because the CSEWG is the established interlaboratory working group that produces the Department's ENDF reference cross section library. The NCESC will continue to work with this group with a view toward continuous improvement of the nuclear data that support the analytical codes.

1.5.1 Current Capability

The Oak Ridge Electron Linear Accelerator (ORELA) is available for measuring nuclear data, but the focus of its program is now nuclear astrophysics. The ORELA is ideally suited for criticality safety applications because it can measure data at high resolution over the energy region important for criticality applications, as well as the other data necessary to provide a complete ENDF/B evaluation. It has supplied data for over 80 percent of the evaluations in the current file, which is referred to as ENDF/B-VI.

The nuclear data programs at the LANL and the ORNL provide the vast majority of evaluations for ENDF/B-VI, which is the most recent and complete data compilation. At the ORNL, in particular, there is significant expertise in the evaluation of the resonance region of the energy spectrum. The author of the SAMMY code, which was developed for that purpose, is at the ORNL. Evaluations are made with full uncertainty files, which are essential for meaningful assessments of the uncertainty in calculated parameters for criticality safety applications because these uncertainties directly impact the calculated margin of subcriticality.

The CSEWG infrastructure exists and can be utilized to upgrade ENDF/B-VI as required by the criticality predictability community. Moreover, the CSEWG can coordinate resources from other National Laboratories and Universities to address unique criticality predictability needs, should they arise.

1.5.2 Current Requirements

Numerous nuclear data deficiencies have been identified which, if corrected, would significantly benefit the criticality predictability community. Most of these require new measurements at the ORELA, followed by a re-evaluation or new evaluation of the ENDF/B-VI file. Current ongoing projects include documenting the ENDF/B-VI Standards, re-evaluating the Uranium-235 cross section data, and reviewing the Uranium-233 nuclear data. These materials are of increasing significance for new Departmental missions involving the handling and storage of nuclear weapons components and processing of waste.

1.5.3 Anticipated Future Direction

To address the deficiencies in nuclear data identified by the criticality predictability community, a multi-faceted program has been proposed. Major program elements would include Measurements, Evaluation, Processing, Data Testing, and International Collaboration. The NCESC, with the help of its Methodology and Experiments Subcommittee, is considering this proposal and will make appropriate recommendations to Defense Programs Management commensurate with available funding.

2.0 Coordination

The NCESC has maintained contact with members of the United States Nuclear Regulatory Commission (USNRC) staff regarding nuclear criticality safety. The USNRC depends on the Department for support in criticality experiments and hands-on training. In one instance during Fiscal Year 1995, the NCESC provided some funding for a joint study with the USNRC concerning "Sensitivity Methods Development and Range of Applicability." This is a very important study because its purpose is to determine the parameter space in which analytical methodology can be reasonably applied to fissile systems.

The NCESC has maintained contact with the various organizations that develop cross section data such as the CSEWG and the AMPX development group at ORNL. This interaction allows the NCESC to remain abreast of new developments and address issues that could impact the Department's commitment to continuous improvement of criticality predictability.

The principal coordinating organization for the Unites States criticality safety community is the Nuclear Criticality Technology and Safety Project (NCTSP). The NCESC will continue to rely on the NCTSP for sound input to the established process for determining and assessing the priority of criticality experiment needs.

Another organization, active in the United States criticality community, is the American National Standards Institute/American Nuclear Society Standards Committee N16. Members of the Methodology and Experiments Subcommittee of the NCESC participate in the standards development process sponsored by this group.

Many members of the NCESC and its subcommittees are active participants in the Nuclear Criticality Safety Division of the American Nuclear Society. Continued interaction with the American Nuclear Society is absolutely essential if the Department is to maintain its commitment to support the needs of the entire criticality safety community.

3.0 Nuclear Criticality Predictability Program Future Direction and Key Issues

Although the NCESC made considerable progress during this past year, much work remains to be done. The following list of issues that the Department considers important will continue to be addressed in the coming year.

* Solidification of the funding support base for all five of the Nuclear Criticality Predictability Program elements.

* Continued implementation of a special nuclear materials management strategy that maximizes materials utility while minimizing security costs without reducing the security posture at experimental facilities.

* Continuously improving the quality of hands-on criticality safety training.

* Periodic review and prioritization of criticality experiments.

* Life cycle planning for facilities required for the Department's program of critical experiments.

Conclusion

The Department recognizes the importance of maintaining an effective criticality safety program to protect the public, workers, Government property, and essential operations from the effects of a criticality accident. An indispensable part of this criticality safety program is the Nuclear Criticality Predictability Program which consists of five program elements: Experiments, Training, Benchmarking, Analytical Codes, and Nuclear Data. Each of these program elements is vital to the success of the Department's Nuclear Criticality Predictability Program. This report presents a status of the program and each of its program elements with regard to funding, current capability, current requirements, and anticipated future needs.

Maintaining a viable nuclear criticality predictive capability is very important in assuring the continued safe operation of nuclear facilities that support Departmental missions. Equally important is the Department's commitment to a systems engineering approach to maintaining facilities that support this program. One of the program elements requires the Department to maintain its competency in conducting critical experiments. Because maintenance of competency in conducting criticality experiments requires a long-term commitment from the Department, life-cycle considerations for the facilities that support this program must be included in the process. Along with planning for the operation of existing facilities and potential construction of new facilities, the Department recognizes the need to plan for the eventual decommissioning and decontaminating of these facilities and environmental remediation of the sites where the facilities were located.

Since accepting DNFSB Recommendation 93-2, the Department has matured significantly in its understanding of what is required to continuously improve its nuclear criticality predictive capability. Maintenance of capability and competence in the five program elements is important, but so is stakeholder involvement in the program management process. Though Defense Programs is responsible for coordinating this process, all stakeholders must share the responsibility for maintenance of this important capability. Much has been accomplished during the past year, and the Department's programmatic view of nuclear criticality predictability has been granted an appropriate priority. With the support of senior management, the NCESC will continue to develop a quality program that meets the Nation's current and future criticality predictability needs.

LIST OF EXPERIMENTS

Number	Title	Priority/Status
102	Large Array of Small Units	P
206	SHEBA Reactivity Parameterization	IP
207	SHEBA Reactivity Void Coefficient	IP
502a	Absorption Properties of Waste Matrices	P
503	Validation of Criticality Alarms and Accident Dosimetry Program	IP
504	Accident Simulation and Validation of Accident Calculations Program	IP
505	A Program to Evaluate Measurements of Sub-Critical Systems	IP
601[a]	Critical Mass Experiments Program for Actinides	IP
702	Spent Fuel Safety Experiments (SFSX)	P
104	Advanced Neutron Source	PDO
105	High-Energy Burst Reactor Experiments	9
301	Plutonium Solution in Concentration Range from 8 to 17 g/l	10
402	Mixed Oxides of Pu and U at Low Moderation	PDO
501	Assessment Program for Materials Used to Transport and Store Discrete Items and Weapons Components	11
502g	Determination of Fissionable Material Concentrations in Waste Materials	7
609[b]	Validation of Calculational Methodology in the Intermediate Energy Range	5

NOTES: PDO: Project-Dependent Only

IP: In Progress

P: In the Planning Phase

[a] Includes 605a (Neptunium Delayed Neutron Fraction Measurement)

[b] Includes 502i (Intermediate Energy Neutron Criticality Studies)

(Priority numbers indicate the original experiment priority assigned by the Methodology and Experiments Subcommittee without adjustment due to initiation of higher priority experiments.)

ATTACHMENT 2

LOS ALAMOS CRITICAL EXPERIMENTS FACILITY SEMI-ANNUAL PROGRESS REPORT, OCTOBER 1, 1994 - MARCH 31, 1995

LA-UR-95-2151

LOS ALAMOS CRITICAL EXPERIMENTS FACILITY

SEMI-ANNUAL PROGRESS REPORT

October 1, 1994 - March 31, 1995

First Half FY95

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1.0 PROGRAM HIGHLIGHTS

1.1. PROGRAM HIGHLIGHTS

R. Paternoster and R. Anderson

The first half of FY95 was a demonstration of the multifaceted character of Pajarito Site and LACEF. Numerous programs involving Category 1 special nuclear materials were completed or in progress during this period. These activities supported emergency response capabilities and programs relating to the understanding of the static and dynamic properties of chain-reacting systems.

Programs were related to basic neutron physics of subcritical multiplication and properties of critical systems as well as to excursions in solution media. Other programs using the LACEF and LACEF personnel fulfilled national security needs involving Category 1 configurations of fissile materials. Currently no other U.S. national laboratory facility could complete this diverse set of programs with Category 1 nuclear materials.

The SHEBA (Solution High-Energy Burst Assembly) machine is the only operating solution critical assembly in the U.S. (at least five solution burst assemblies are known to be operating in the Former Soviet Union). SHEBA recently completed the benchmark void coefficient experiments (Section 4) and research on solution excursions (Section 5). SHEBA performed irradiations to test criticality alarm system (CAS) for both Y-12 and TA-55, while completing irradiations for Rocky Flats accident dosimetry. The Flattop benchmark assembly was used for a series of replacement measurements to determine the critical mass of Np^{237} in a fission spectrum (Section 2).

Subcritical measurements are related to our fundamental understanding of the physics of static criticality. They measure the safety of a specific system and validate calculational methodology. There are several methods for making measurements of this type, two of which are being implemented and compared at LACEF. We anticipate that a test comparison will be made for all known methodologies for making subcritical measurements in the near future.

Safety training activities (Section 7) were completed during the first half of FY95. Two Nuclear Criticality Safety Classes were completed. The NCSC classes include three "hands-on" lab exercises involving construction of multiplying assemblies and operation of a critical assembly.

The construction of two new SNM (Section 8) vaults dominated the scene in Kiva 3. The large "massive" vault became operational and was loaded on March 1. This high-tech vault increases the storage capacity at TA-18 while reducing high-security concerns in the remaining TA-18 vaults. The Godiva vault was completed in mid-November and installation of the Godiva transport system began. Acceptance testing of the Godiva transport system and automated door was completed on March 30. Godiva-IV nuclear operations are set to resume in early April.

Documentation, compliance, and personnel training activities continued during this period. In accordance with DOE Order 5480.22, a new set of TSRs (Technical Safety Requirements) were produced, reviewed, approved within the Laboratory, and submitted to the DOE on February 24 (Section 10). Almost coincident with this we received notice that the new LACEF Safety Analysis Report was approved by the DOE.

2.0 CRITICAL EXPERIMENTS CORE CAPABILITY

2.1. CRITICAL MASS OF Np_{93}^{237}

R. Sanchez, J. Bounds, P. Jaegers

Large quantities of actinide elements are produced in operating power reactors. Some of these actinides have been separated from irradiated fuel elements and are being stored in liquid form. This is adequate for short-term storage. However, for long-term storage, these liquids will be converted into oxides and metals. At the present, there is great uncertainty about the critical mass for some of these actinide elements in their oxide and metal form. Knowing precisely the critical mass of these elements not only will validate storage mass limits reported in the standard ANSI/ANS-8.15-1981, "Nuclear Criticality Control of Special Actinide Elements," but will optimize the geometry needed for safe disposition of these materials. This progress report describes the experiments that have been performed to determine the critical mass of Np_{93}^{237}.

It is well known that Np_{93}^{237} is produced primarily in uranium reactors through the following nuclear reactions:

$$U_{92}^{238}(n, 2n)U_{92}^{237} \rightarrow \beta \rightarrow Np_{93}^{237}$$

$$U_{92}^{235}(n, \gamma)U_{92}^{236};\ U_{92}^{236}(n, \gamma)U_{92}^{237} \rightarrow \beta \rightarrow Np_{93}^{237}$$

Its half-life is approximately $2x10^6$ years and decays into Pa_{91}^{233} by alpha emission.

Experiments performed in the past six months consist of placing small samples of neptunium, uranium, or empty aluminum cans in the center of the FLATTOP assembly. The FLATTOP critical assembly is operated above delayed-critical by inserting the control rods all the way in. The worth of each sample is then estimated through the measured asymptotic reactor period and the Inhour equation. These measurements were repeated tens of times to obtain better statistics and reduce the error of the measurements. Table 2.1 shows the dimensions, weights and isotopic composition of the samples used for these replacement measurements. It is important to point out that the neptunium sample is clad with nickel.

The experimental results showed that the uranium sample is 3.41 ± 0.39 ¢ more reactive than the neptunium sample. In addition, the uranium sample was 22.44 ± 0.22 ¢ more reactive than the empty aluminum can and the neptunium sample was worth 18.95 ± 0.32 ¢ more than the empty aluminum can. These experimental results were compared against computational results obtained from TWODANT and MCNP and shown in Table 2.2. As seen from Table 2.2, the TWODANT computational and experimental results are in good agreement, which benchmark the cross section data for Np_{93}^{237}. Note that the MCNP kcode computational results cannot distinguish these small increases in reactivity between the samples even after 1000 generations. The TWODANT and MCNP computational models are shown in Fig. 2.1.

Once the cross section data for Np_{93}^{237} was benchmarked, we were able to use the TWODANT code and estimate its critical mass for a bare neptunium sphere. The calculation yielded a mass of 56 ± 2 kg at a density of 20.45 g/cm^3.

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Table 2.1. Properties of the replacement samples.

	Uranium Sample	Np-237 sample	Empty Al can
Weight of metal	29.909 g	28.393 g	-------
Weight of can	------	0.773 g	0.476 g
Dimensions			
Length (in.)	0.5015	0.4890	0.4975
Outside diameter (in.)	0.4990	0.4865	0.4865
			Al wall thickness
		Nickel clad	
Thickness			0.010
Ends (in.)	-----	0.00350.010	
Sides (in.)	-----	0.0057	
Isotopic composition, wt%			
Uranium		**Neptunium**	
U-234	1.1	Np-237	99.87
U-235	93.2	Other elements	0.13
U-236	0.2		
U-238	5.5		

Table 2.2. Experimental and computational results.

Experiment	
	$\Delta\rho$ (U-235-Np-237) = 3.41 ± 0.39 ¢ $\Delta\rho$ (U-235-Al) = 22.44 ± 0.22 ¢ $\Delta\rho$ (Np-237-Al) = 18.95 ± 0.32 ¢
Computational TWODANT	
	$\Delta\rho$ (U-235-Np-237) = 2.05 ¢ $\Delta\rho$ (U-235-Al) = 22.72 ¢ $\Delta\rho$ (Np-237-Al) = 20.66 ¢
MCNP	
	k_{eff}(Np-237) = 1.00081 ± 0.0.0004 k_{eff}(U-235) = 1.00035 ± 0.0.0004 k_{eff}(Al) = 1.00029 ± 0.0.0004

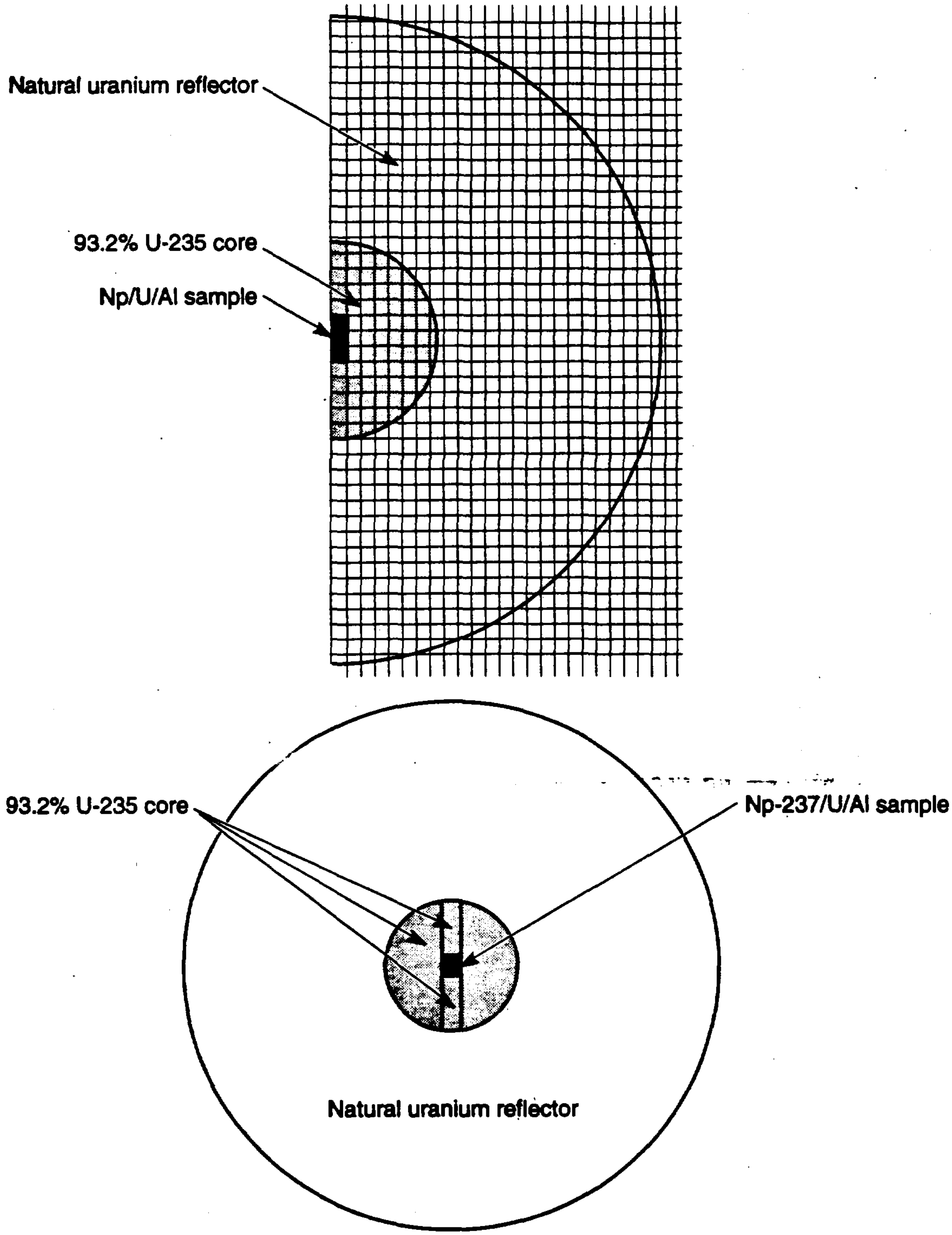

Figure 2.1. TWODANT and MCNP computer models.

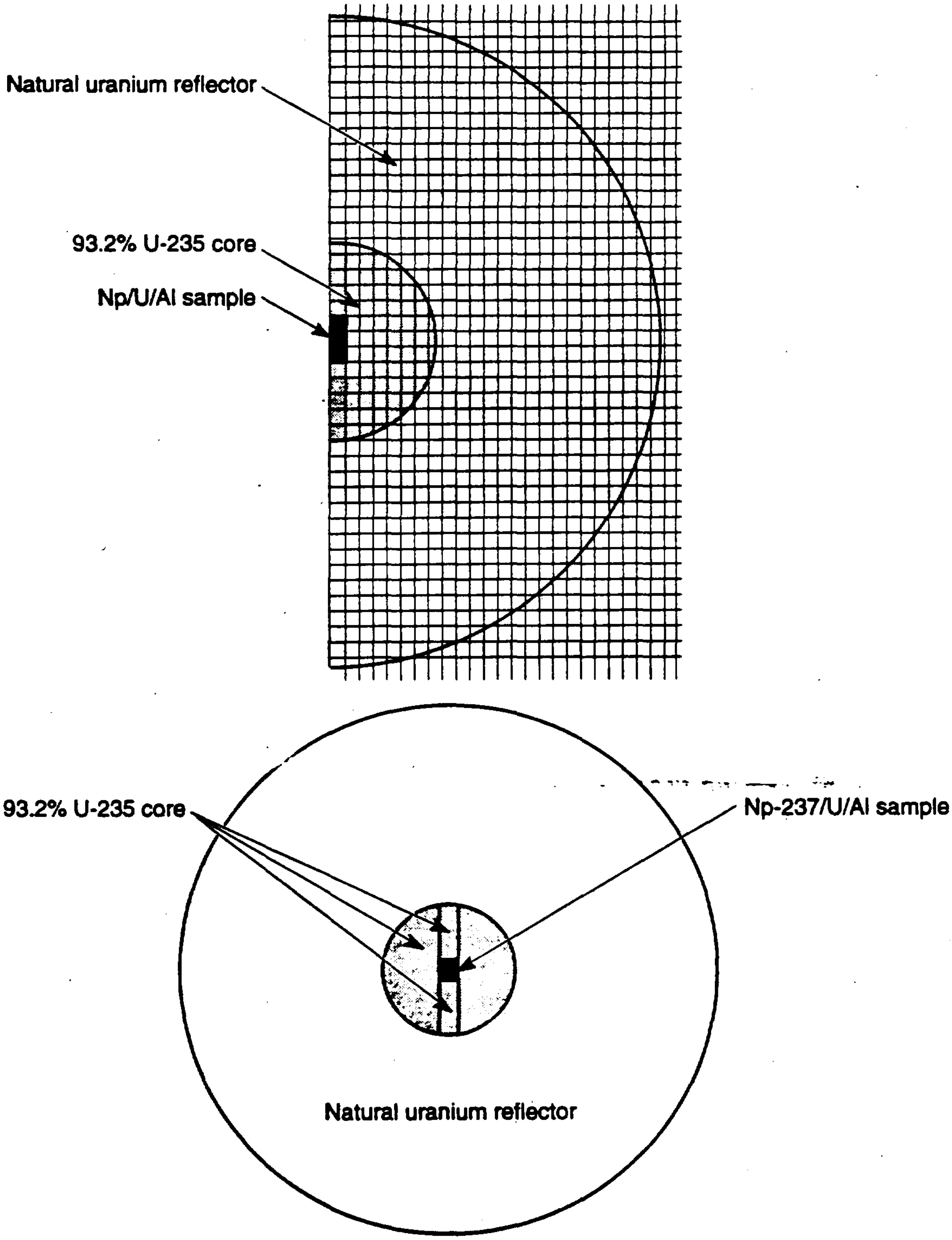

Figure 2.1. TWODANT and MCNP computer models.

2.2. DESIGN OF A CRITICAL ASSEMBLY FOR TESTING INTEGRAL PROPERTIES OF MATERIALS

D. Hayes

Introduction

Implementation of the Defense Nuclear Facilities Safety Board (DNFSB) Recommendation 93-2 (criticality experiment capability) includes expanding and updating the current nuclear criticality database. To that end, the Nuclear Criticality Experiments Steering Committee (NCESC) has compiled and prioritized a list of experiments solicited from the criticality community. In response to the NCESC list, a critical assembly is being designed at the Los Alamos Critical Experiments Facility (LACEF) to incorporate elements of several experiments. Specifically, elements of

1. Experiment 102 - Large Array of Small Units,
2. Experiment 501 - Assessment Program for Materials Used to Transport and Store Discrete Items and Weapons Components,
3. Experiment 502a - Absorption Properties of Waste Matrices, and
4. Experiment 609 - Validation of Calculational Methodology in the Intermediate Energy Range

are being designed into the assembly.

Description

Design of the assembly centers around the fuel elements, which are 5-L right circular cylinders filled with $U(93.1)O_2(NO_3)_2 * 6H_2O + HNO_3 + H_2O$ (U(93.1)NH). The cylinders are 0.25-cm-thick 304SS with a height to diameter ratio of one. Initially, four elements, in a square lattice reflected by 10 cm of polyethylene, will be used. Future modifications include fuel changes (to UO_2 and PuO_2, for example) and increases in the array size (to 2x2x2, 3x3x3, etc.).[1]

Operation

Honeycomb (Fig. 2.2), a horizontal split table, is chosen as the platform for the assembly. Two fuel elements will be placed on the "movable" side and two will be placed on a jackscrew table mounted on the "stationary" side. Closure is achieved by hydraulic ram ("movable" side)and the jackscrew table, using a 1/M approach.

Materials will be introduced interstitially and externally to the fuel elements to determine their integral properties as moderators, reflectors, and absorbers.[2] Some materials of interest are Al_2O_3, CaCl, cellulose, celotex, concrete, depleted uranium, expanded borated polyfoam, Fe_2O_3, firedike, foamglas, kerosene, lead, plexiglas, polyethylene, PVC, SiO_2, and TBP. Concrete (building material) will be used as a reflector at varying distances from the fuel elements to investigate room-return effects.

Varying the spacing of fuel elements and the size and position of nonfissile materials results in a varying neutron energy spectrum. Thus, the spectrum may be adjusted to a specific energy range for evaluating material properties.[3]

---

[1] Experiment 102

[2] Experiments 501 and 502a.

[3] Experiment 609.

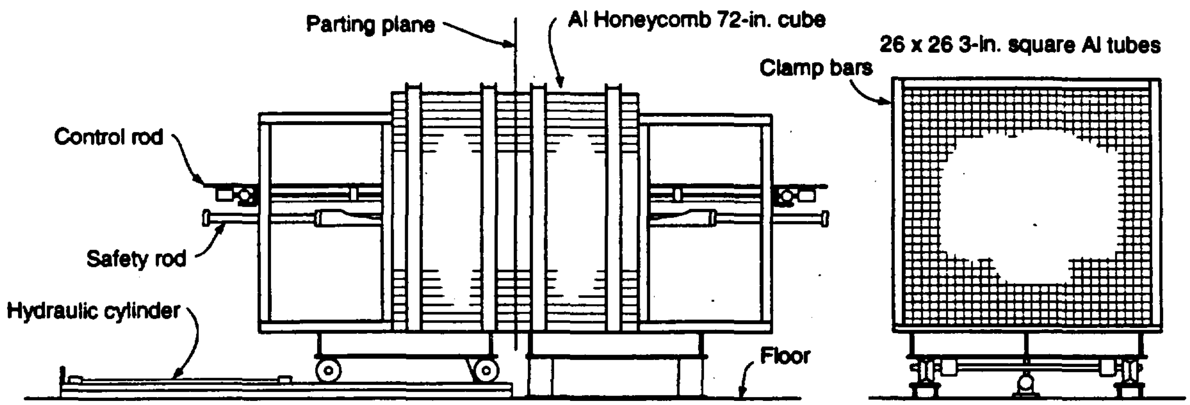

Figure 2.2. Current Honeycomb configuration.

Expected Results

Critical dimensions of the various systems will be measured, yielding information on the physics of interacting fuel elements and materials. The data will be evaluated to determine the effects of fuel element size, geometry and fuel type. Room return effects will be investigated. Integral measurements of non-fissile material properties will be used to evaluate computer models (MCNP and KENO, for example), associated cross-sections, and thermal treatments ($S(\alpha,\beta)$).

Design Progress

Budget concerns and time constraints control the design of this assembly. As a result, the design requires use of existing equipment and on-hand fissile materials. Thus, Honeycomb and U(93.1)NH from the WINCO Slab Tank Experiment were chosen.

For simplicity, four fuel elements compose the initial assembly. This necessitates use of a reflector to achieve delayed critical. Ten centimeters of polyethylene serve as the reflector, with the added benefit of isolating the assembly from room return.

MCNP k_{eff}-calculations have been performed to determine appropriate geometries. Preliminary calculations use 1000 neutrons per cycle, 15 inactive cycles and 100 active cycles (Fig. 2.3). Based on k_{eff}-calculations and mechanical simplicity, polyethylene is placed adjacent to the top and bottom of the fuel elements, and the distance between the fuel elements (face to face separation) and the polyethylene walls is varied (Figs. 2.4 and 2.5). Figure 2.6(a-f) shows k_{eff} as a function of fixed fuel-element position and varying distance to the polyethylene walls. Figure 2.7(a-e) shows k_{eff} as a function of fixed distance to the polyethylene walls and varying fuel element separation.

Fuel element spacing of 10-cm was investigated as a geometry for studying interstitial material properties. Figure 2.8 depicts k_{eff}, with fuel elements fixed at 10-cm separation in "x," as a function of separation in "y" (the reflector is adjacent to the fuel elements). Figure 2.9 shows k_{eff}, with 10-cm separation in "x" and 0 cm separation in "y," as a function of distance between the polyethylene walls and the fuel elements.

Additional k_{eff} calculations were performed for a single fuel element using TWODANT, a Two Dimensional Diffusion Accelerated Neutron Transport Code, and compared with MCNP. A significant difference exists between the calculations. The results are tabulated in Table 2.3.

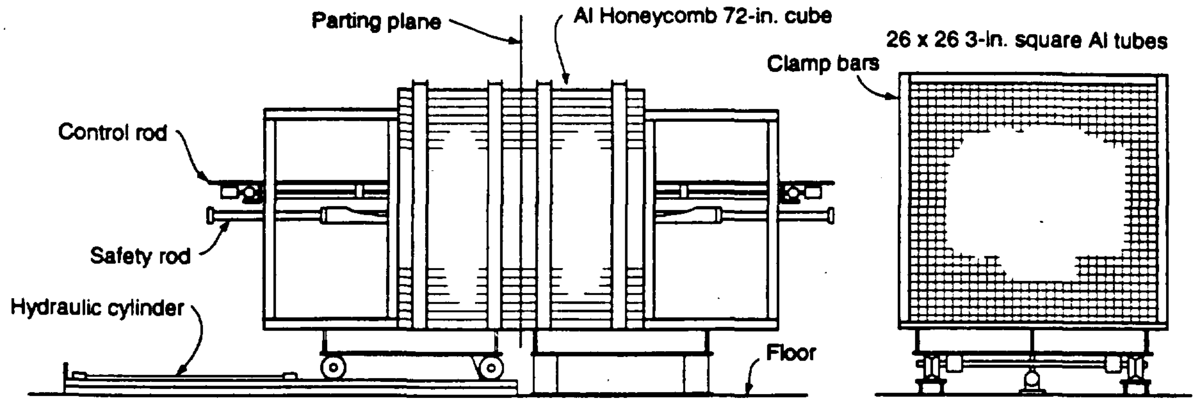

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Table 2.3. Single-unit k_{eff} calculations.

Case	k_{eff} from MCNP	k_{eff} from TWODANT ENDF	k_{eff} from TWODANT HR16
Bare 5L Cylinder	0.62262 ± 0.00095	0.62485	0.60581
TWODANT-HR16 Dimension Search	1.02033 ± 0.00115	1.01796	1.0000
TWODANT-ENDF Dimension Search	1.00085 ± 0.00116	1.0000	0.98181
Bare 5L Sphere	0.61981 ± 0.00096	0.62603	0.61057

Results from the dimension searches were evaluated using different cross-sections/codes. ENDF cross-sections for TWODANT were for a fast (metal) system. The Hansen-Roach (HR16) anomaly is being investigated.

Future Effort

MCNP calculations of increased detail, including interstitial materials, will be performed to support control system, mechanical design, and administrative operational requirements. A Design Requirements Document and Experiment Plan will be developed.

Honeycomb modifications include removal of the box tubes, installation of new fixtures, and installation of a digital control system.

```
2x2x1 405gU/l U(93.1)O2(NO3)2 304SS Cyl .25cm thick 2cm sep poly box
C Cell Cards
1  1 -1.5579 -1 5' -6  u=1  imp:n=1
2  0       -1  6  -7  u=1  imp:n=1
3  2 -7.92  #1  #2     u=1  imp:n=1
4  0       -2  3  -4  fill=1  imp:n=1
5  like  4  but  trcl  (21.54 0 0)  imp:n=1
6  like  4  but  trcl  (0 21.54 0)  imp:n=1
7  like  4  but  trcl  (21.54 21.54 0)  imp:n=1
8  0       #4 #5 #6 #7 3 -4 11 -12 15 -16 imp:n=1
9  3 -0.92  -8  4  11  -12  15  -16  imp:n=1
10 3 -0.92  -3  9  11  -12  15  -16  imp:n=1
11 3 -0.92  10  -11  -8  9  14  -17  imp:n=1
12 3 -0.92  12  -13  -8  9  14  -17  imp:n=1
13 3 -0.92  14  -15  -8  9  11  -12  imp:n=1
14 3 -0.92  16  -17  -8  9  11  -12  imp:n=1
15 0       8:-9:-10:13:-14:17  imp:n=0

C Surface Cards 3cm to poly
1  cz  9.52  $ir
2  cz  9.77  $or
3  pz  -0.25  $Bottom of Can
4  pz  18.02  $Top of Can
5  pz   0.00  $Bottom Solution
6  pz  17.561 $Top Solution
7  pz  17.77  $Top Gap
8  pz  28.02
9  pz  -10.25
10 px  -22.77
11 px  -12.77
12 px  34.310
13 px  44.310
14 py  -22.77
15 py  -12.77
16 py  34.310
17 py  44.310

C Control Cards
kcode  1000 1.0  15  115
ksrc   0  0  8.885    21.54  0  8.885
       0  21.54  8.885    21.54  21.54  8.885
C Material Cards
m1     1001.50c  5.9150-2    8016.50c  3.7873-2    7014.50c  2.0746-3
       92235.50c  9.6656-4    92238.50c  7.0730-5
mt1    lwtr
m2     26000.55c  5.9360-2    24000.50c  1.7430-2    28000.50c  7.7200-3
       25055.50c  1.7400-3
m3     1001.50c  7.8990-2    6000.50c  3.950-2
mt3    poly
```

Figure 2.3. Sample MCNP input (2 cm unit separation, 3 cm to polyethylene walls).

```
2x2x1 405gU/l U(93.1)O2(NO3)2 304SS Cyl .25cm thick 2cm sep poly box
C Cell Cards
1  1 -1.5579 -1 5 -6 u=1 imp:n=1
2  0        -1 6 -7  u=1 imp:n=1
3  2 -7.92  #1 #2    u=1 imp:n=1
4  0        -2 3 -4  fill=1 imp:n=1
5  like 4 but trcl (21.54 0 0) imp:n=1
6  like 4 but trcl (0 21.54 0) imp:n=1
7  like 4 but trcl (21.54 21.54 0) imp:n=1
8  0        #4 #5 #6 #7 3 -4 11 -12 15 -16 imp:n=1
9  3 -0.92  -8 4 11 -12 15 -16 imp:n=1
10 3 -0.92  -3 9 11 -12 15 -16 imp:n=1
11 3 -0.92  10 -11 -8 9 14 -17 imp:n=1
12 3 -0.92  12 -13 -8 9 14 -17 imp:n=1
13 3 -0.92  14 -15 -8 9 11 -12 imp:n=1
14 3 -0.92  16 -17 -8 9 11 -12 imp:n=1
15 0        8:-9:-10:13:-14:17 imp:n=0

C Surface Cards 3cm to poly
1  cz 9.52  $ir
2  cz 9.77  $or
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6  pz 17.561 $Top Solution
7  pz 17.77  $Top Gap
8  pz 28.02
9  pz -10.25
10 px -22.77
11 px -12.77
12 px 34.310
13 px 44.310
14 py -22.77
15 py -12.77
16 py 34.310
17 py 44.310

C Control Cards
kcode  1000 1.0 15 115
ksrc   0 0 8.885    21.54 0 8.885
       0 21.54 8.885    21.54 21.54 8.885
C Material Cards
m1     1001.50c 5.9150-2    8016.50c 3.7873-2    7014.50c 2.0746-3
       92235.50c 9.6656-4   92238.50c 7.0730-5
mt1    lwtr
m2     26000.55c 5.9360-2   24000.50c 1.7430-2   28000.50c 7.7200-3
       25055.50c 1.7400-3
m3     1001.50c 7.8990-2    6000.50c 3.950-2
mt3    poly
```

Figure 2.3. Sample MCNP input (2 cm unit separation, 3 cm to polyethylene walls).

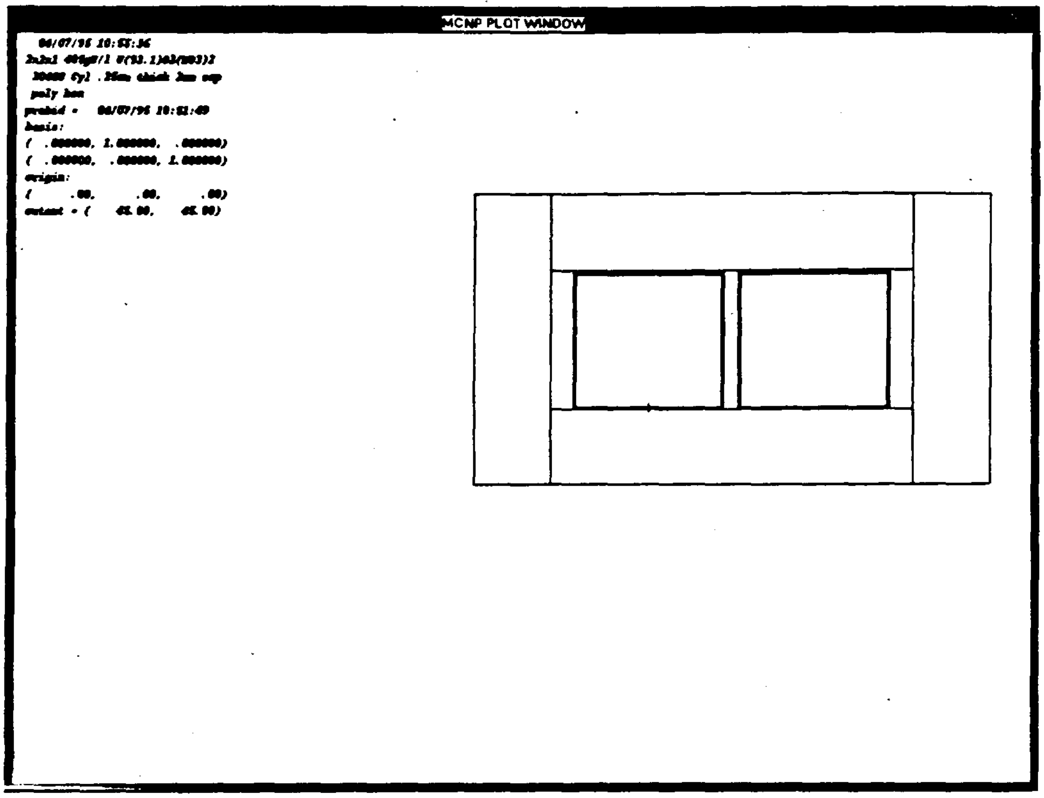


Figure 2.4. Side view of fuel elements and polyethylene.

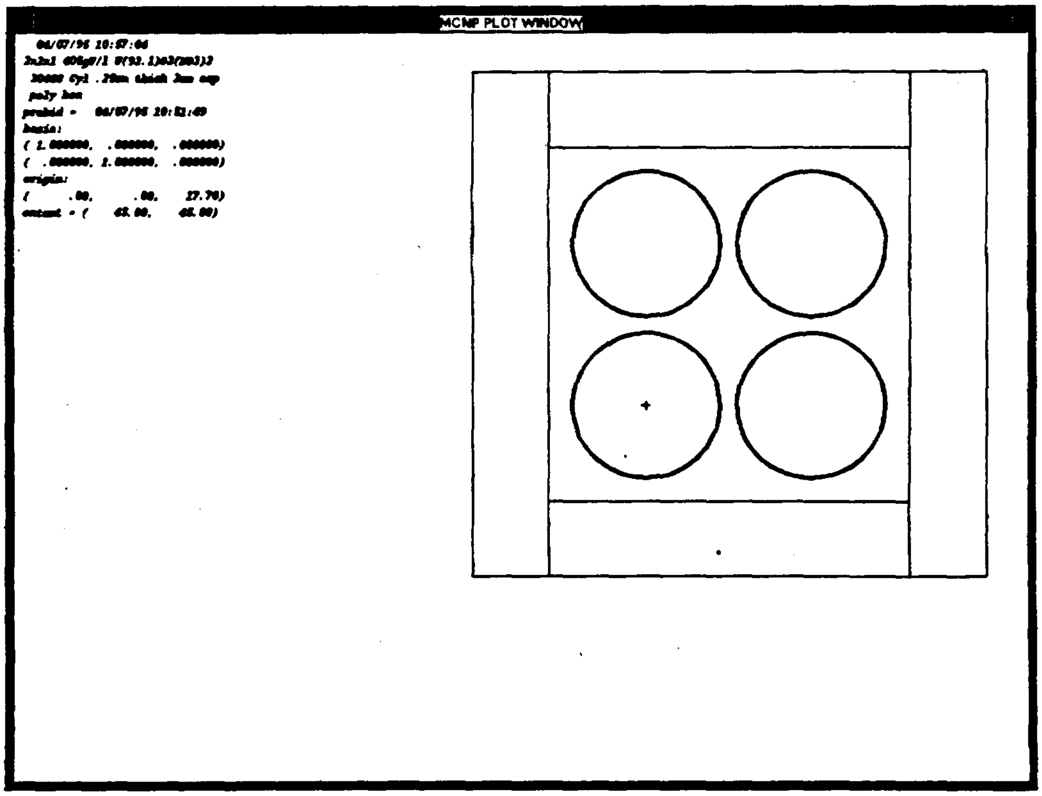


Figure 2.5. Top view of fuel elements and polyethylene (central plane).

Figure 2.6(a-f) shows k_{eff} as a function of fixed fuel element position and varying distance to the polyethylene walls.

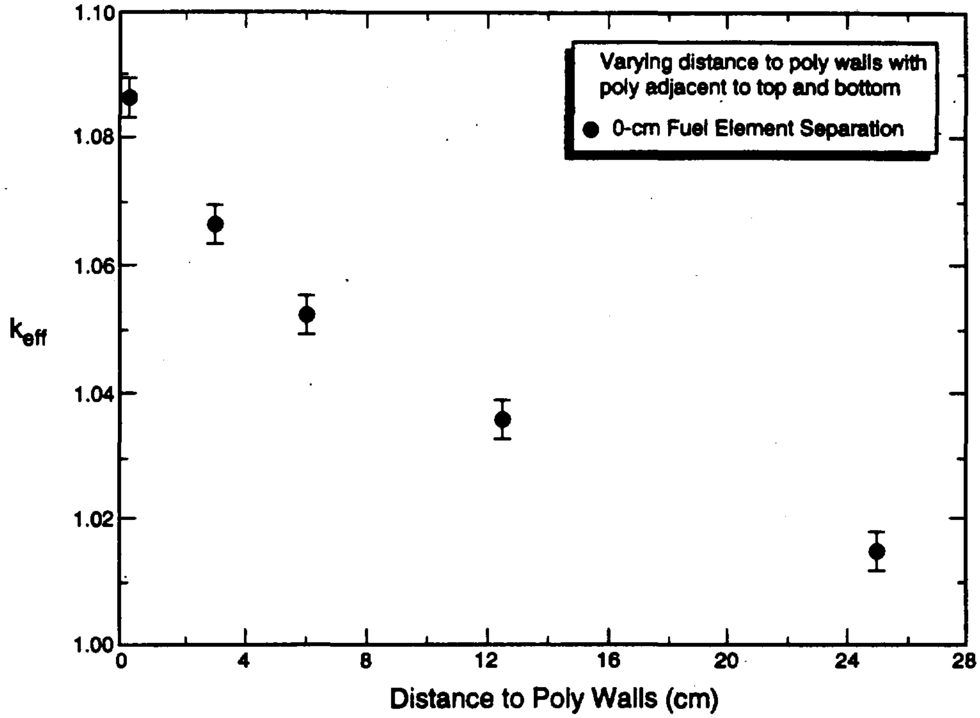


Figure 2.6(a).

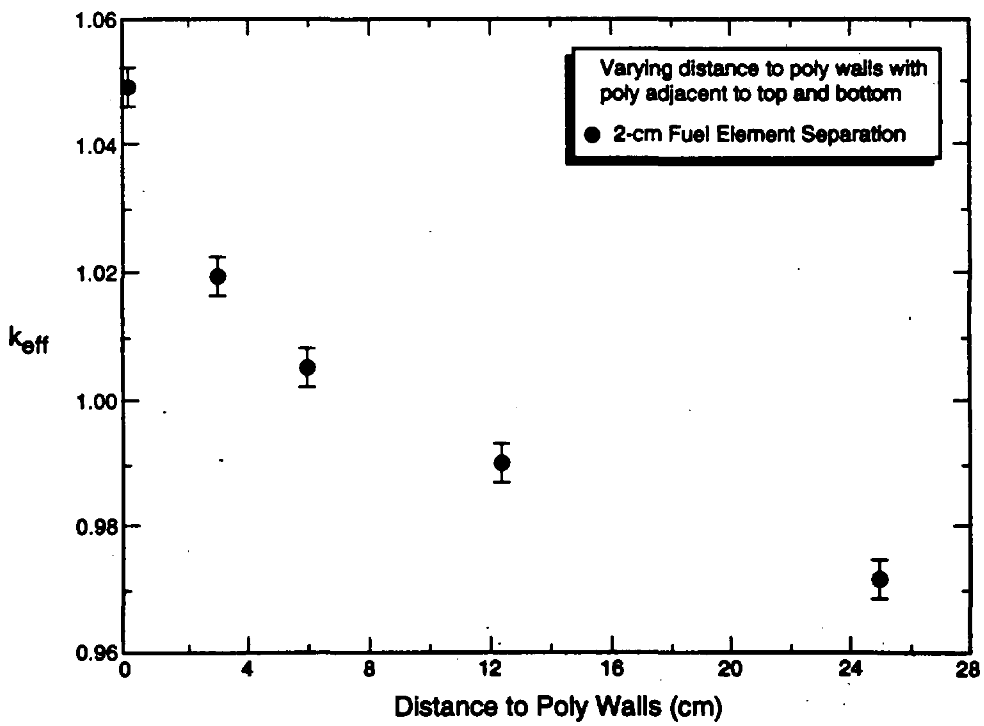


Figure 2.6(b).

Figure 2.6(a-f) shows k_{eff} as a function of fixed fuel element position and varying distance to the polyethylene walls.

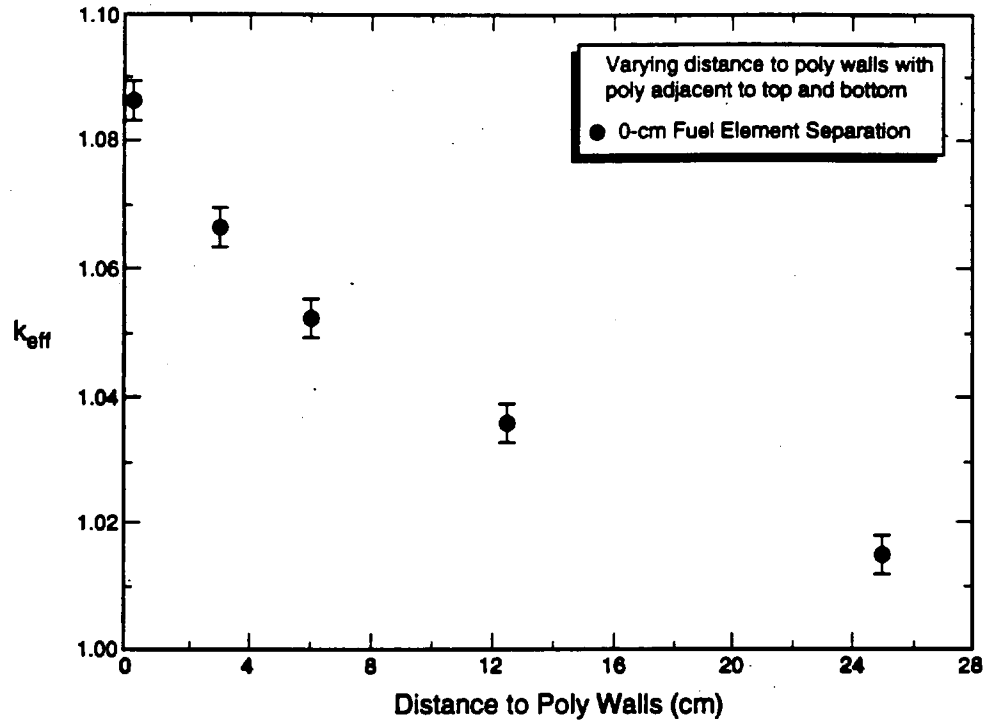


Figure 2.6(a).

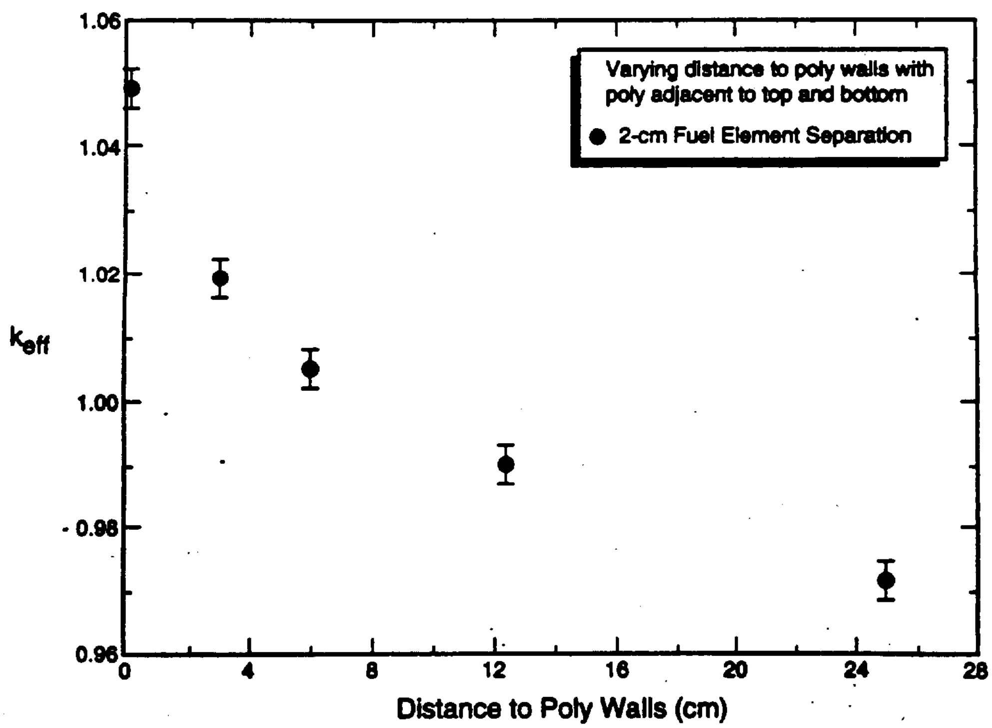


Figure 2.6(b).

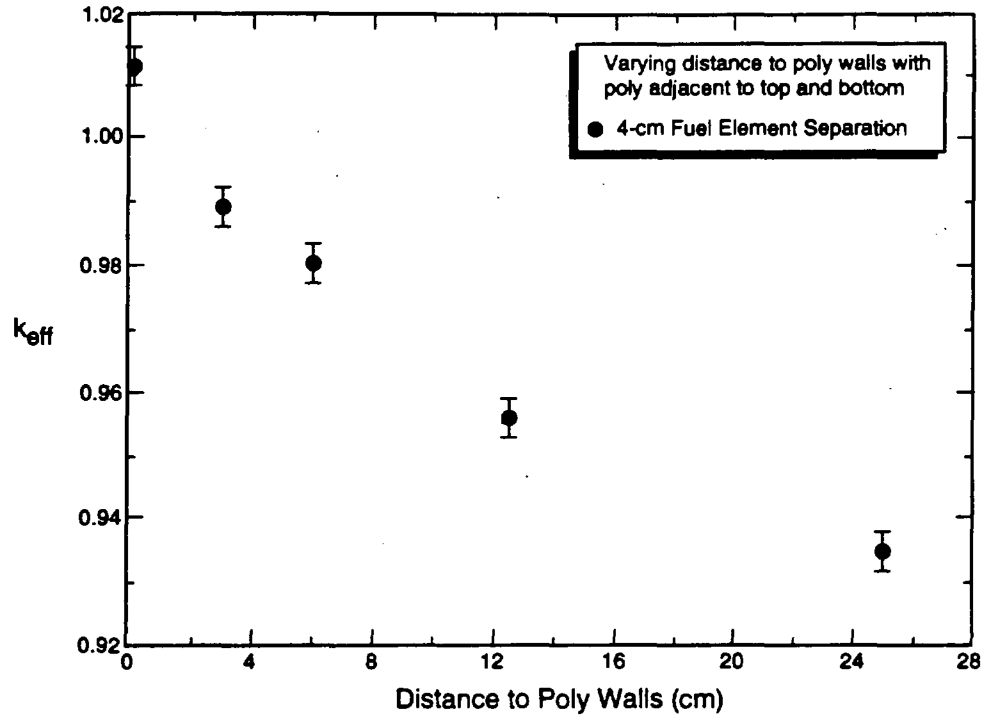


Figure 2.6(c).

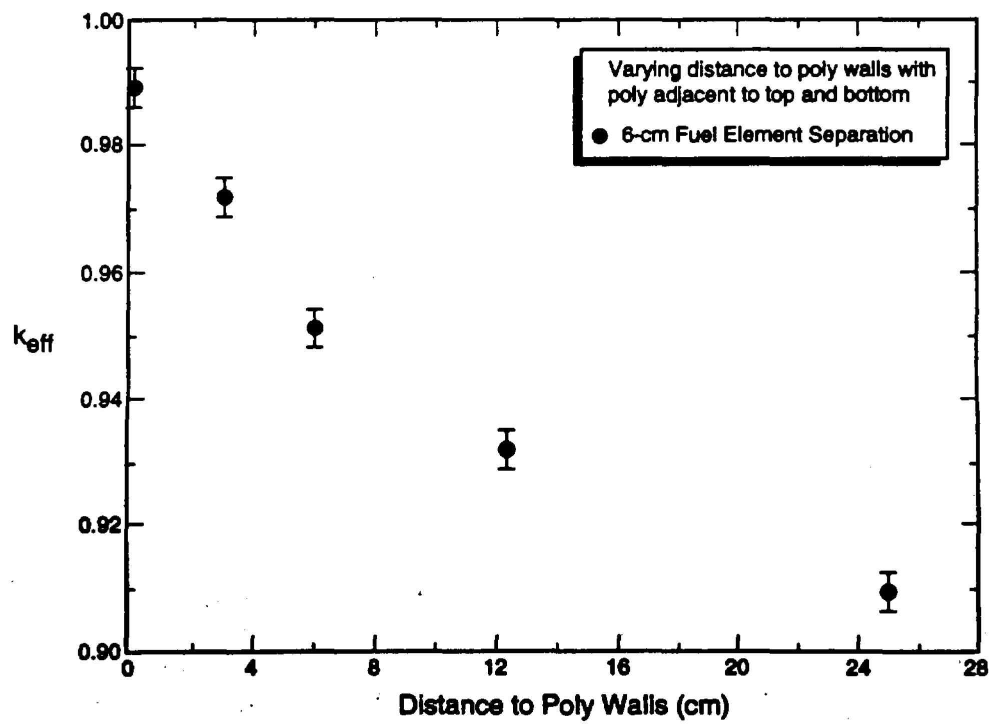


Figure 2.6(d).

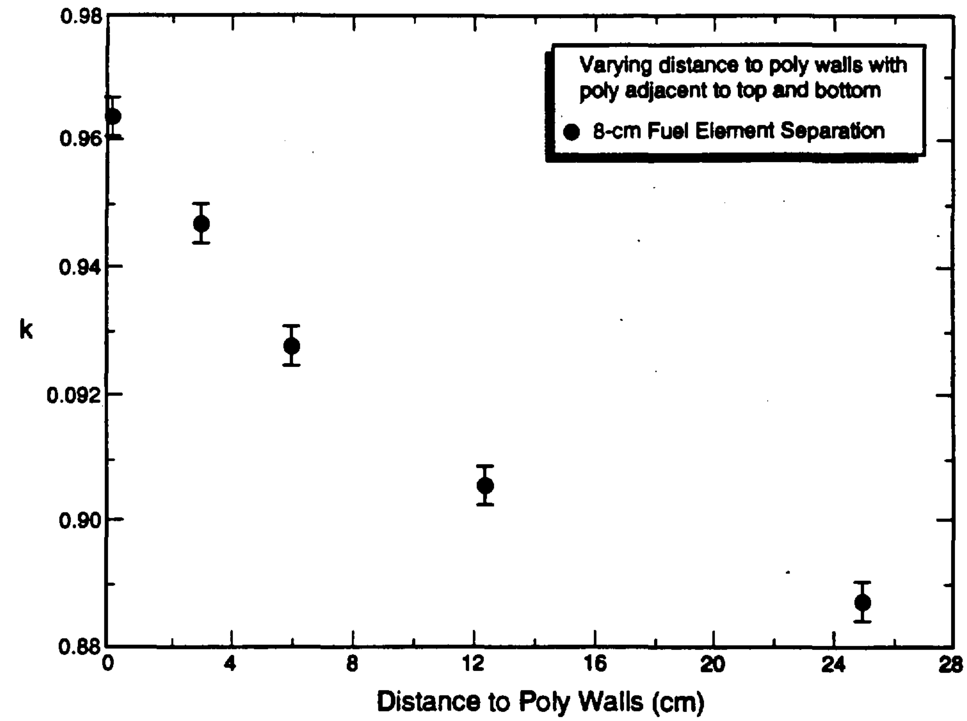


Figure 2.6(e).

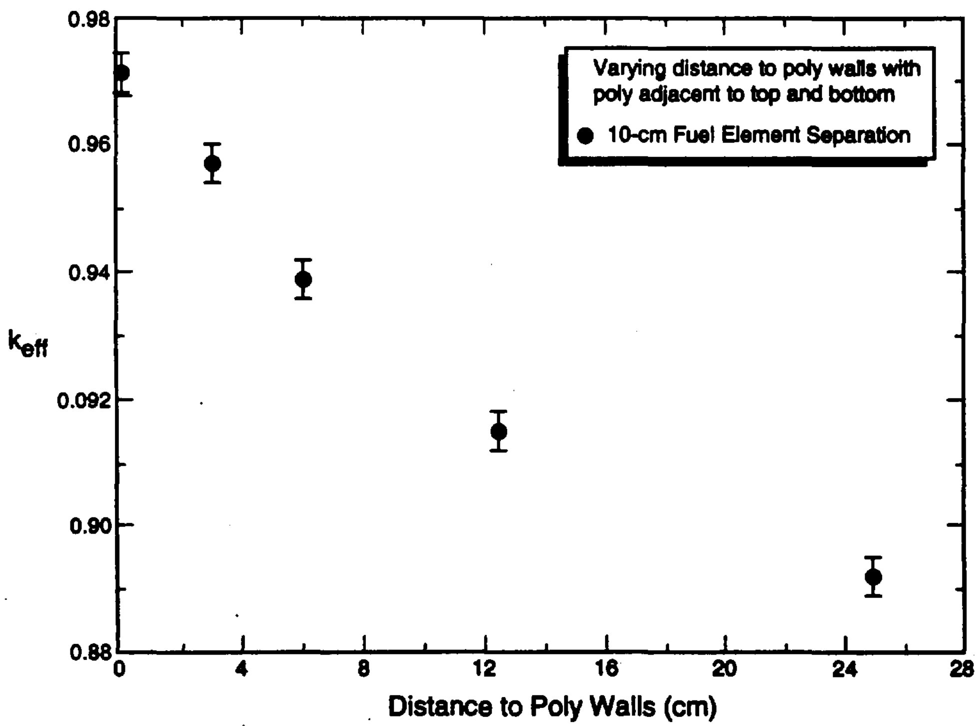


Figure 2.6(f).

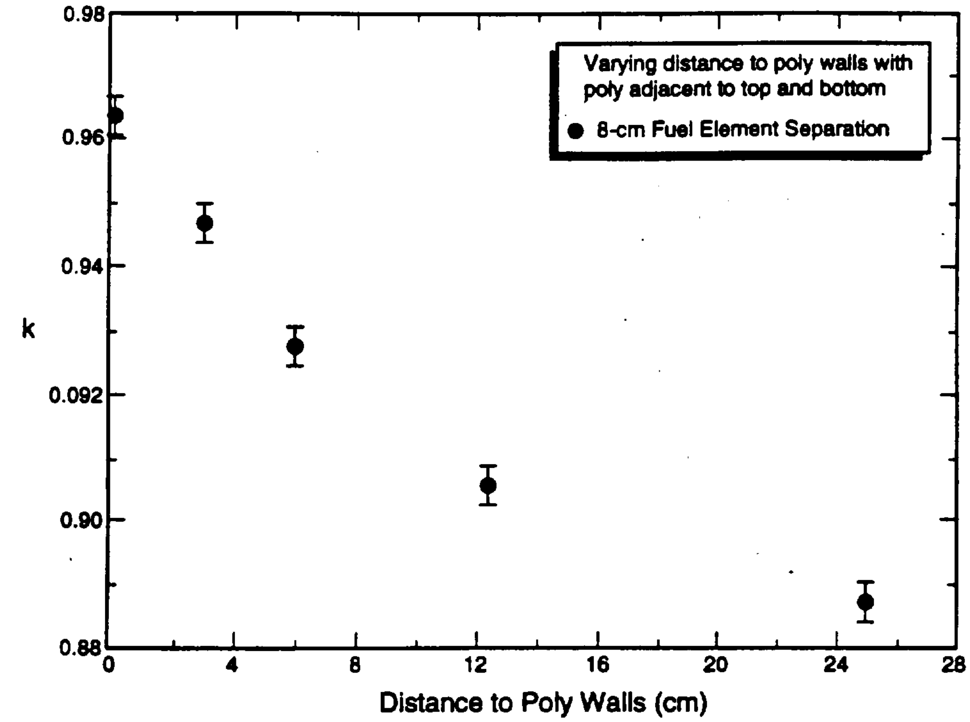


Figure 2.6(e).

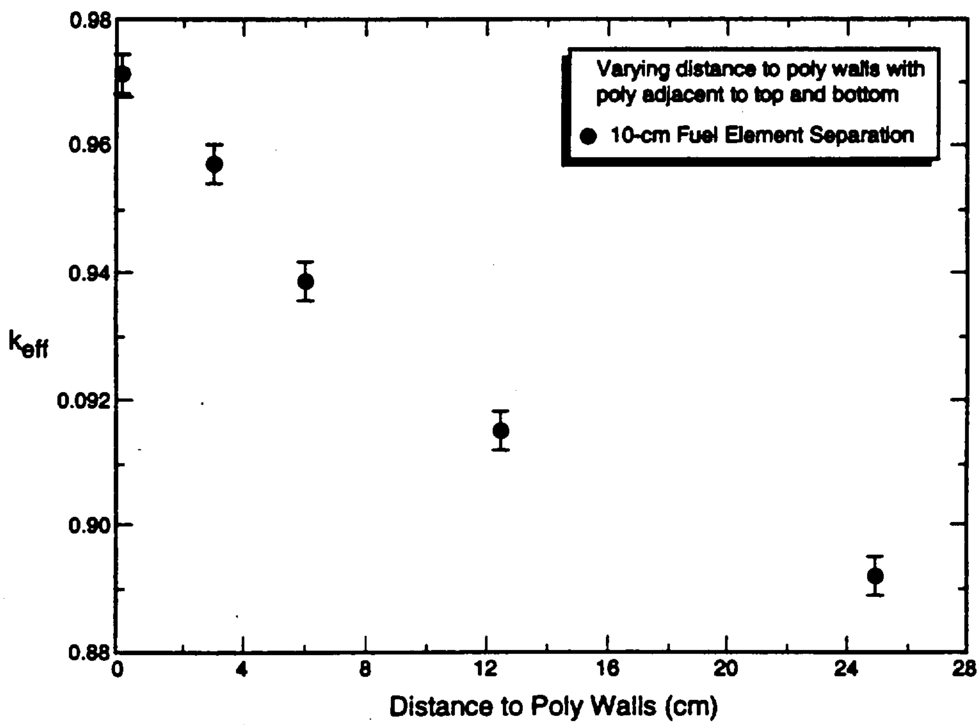


Figure 2.6(f).

Figure 2.7(a-e) shows k_{eff} as a function of fixed distance to the polyethylene walls and varying fuel element separation.

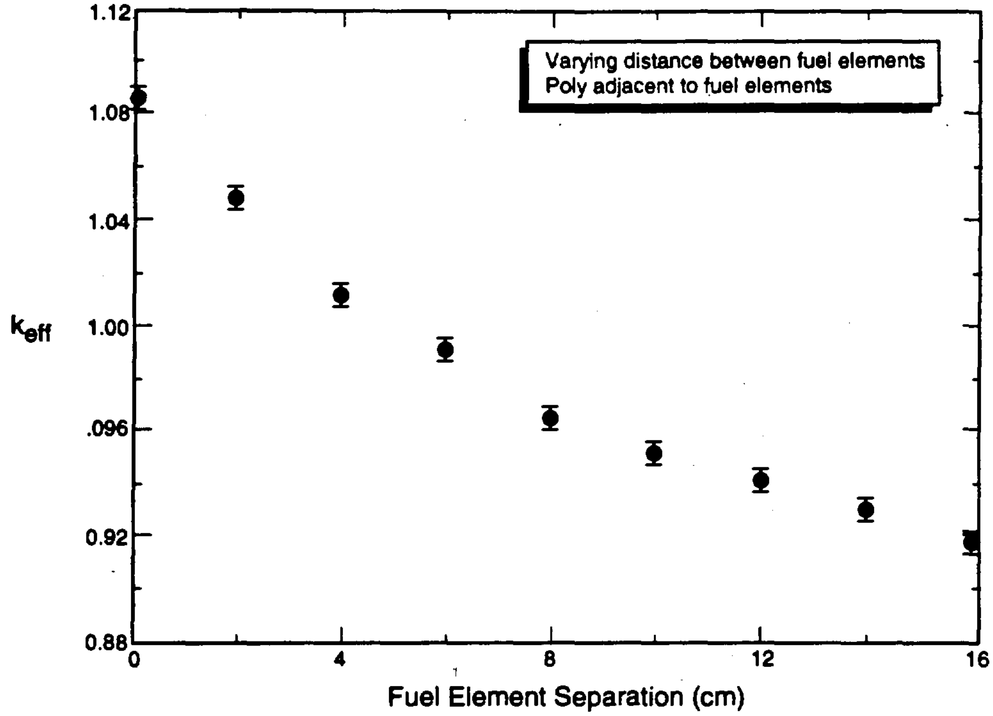


Figure 2.7(a)

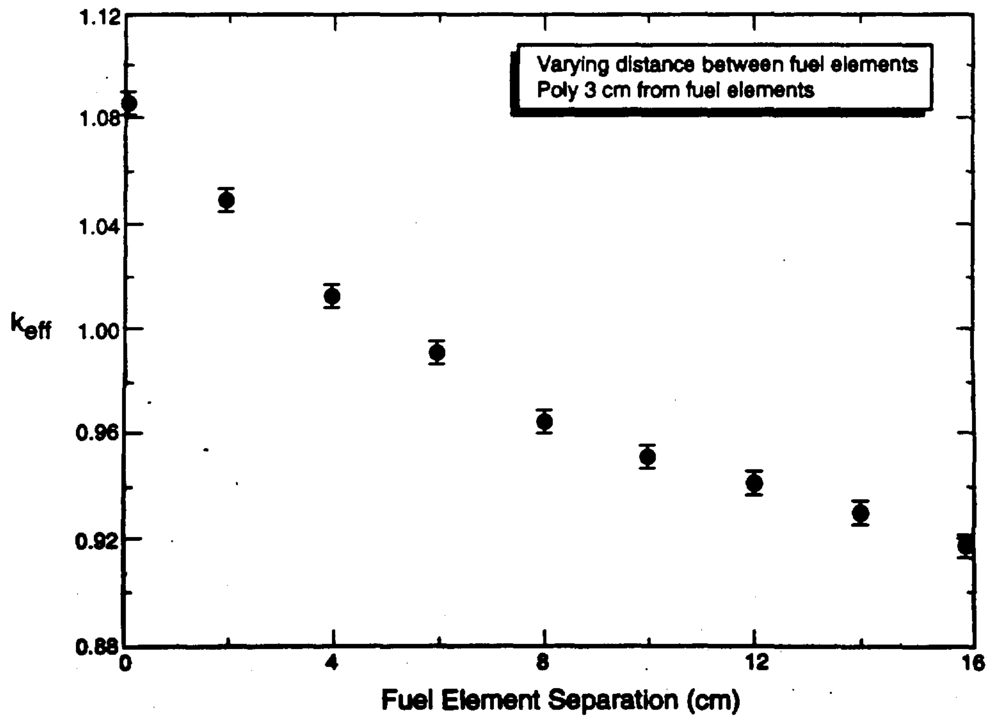
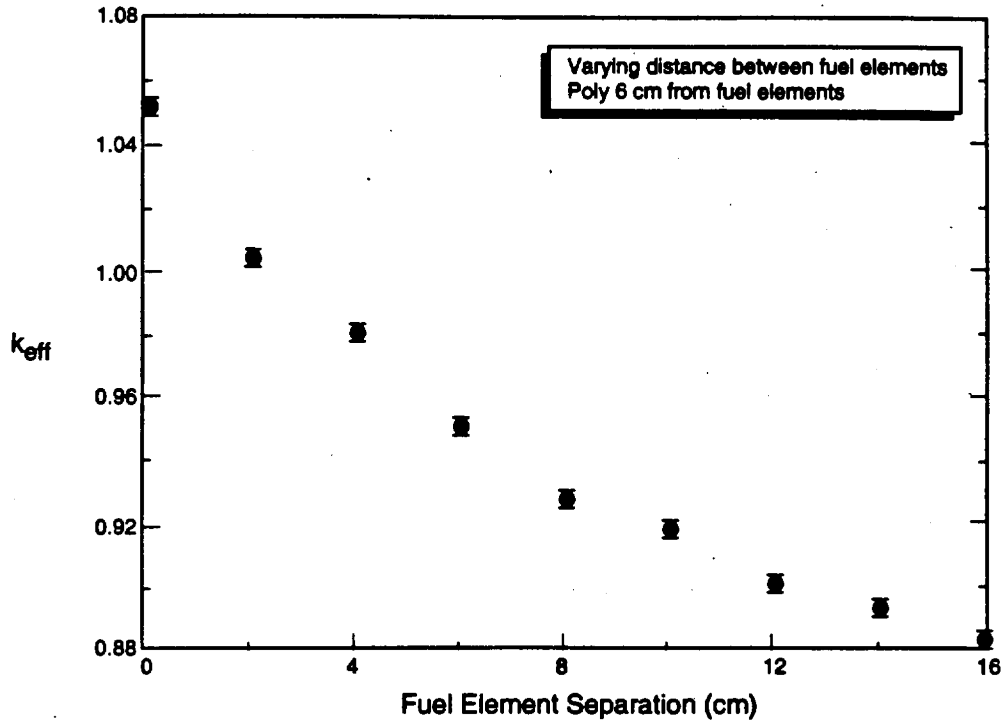


Figure 2.7(b)



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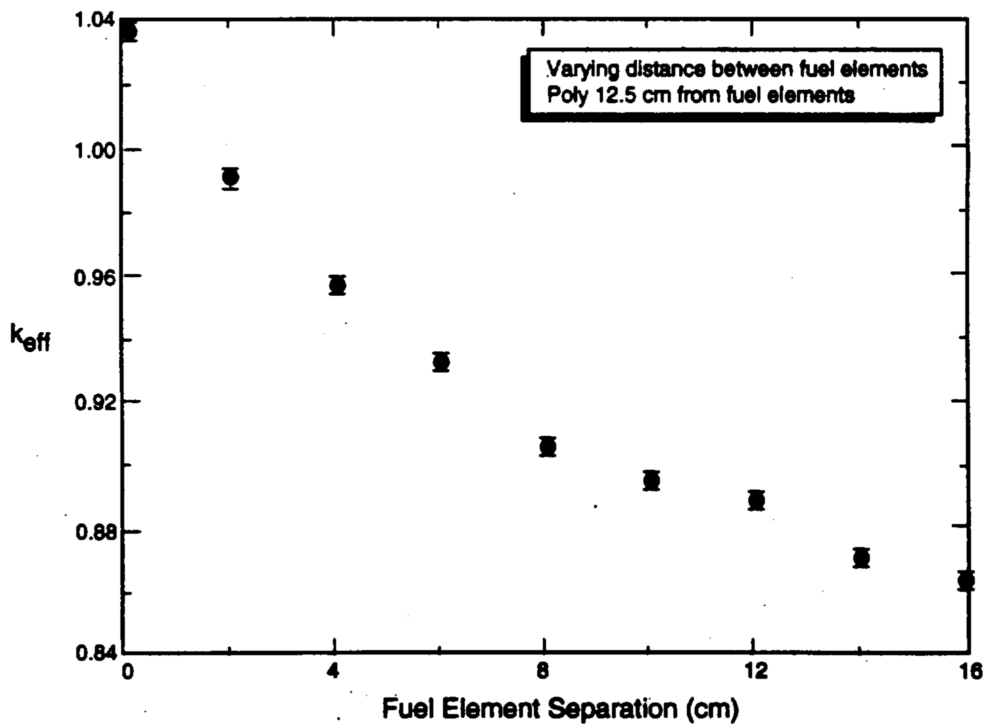

Figure 2.7(d)

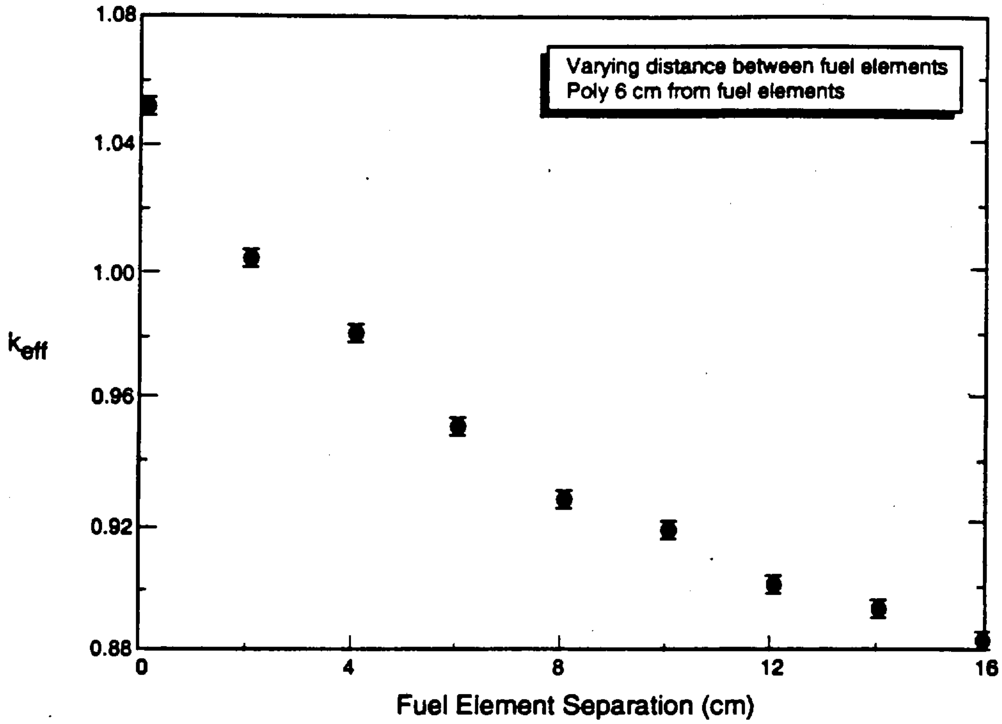

Figure 2.7(c)

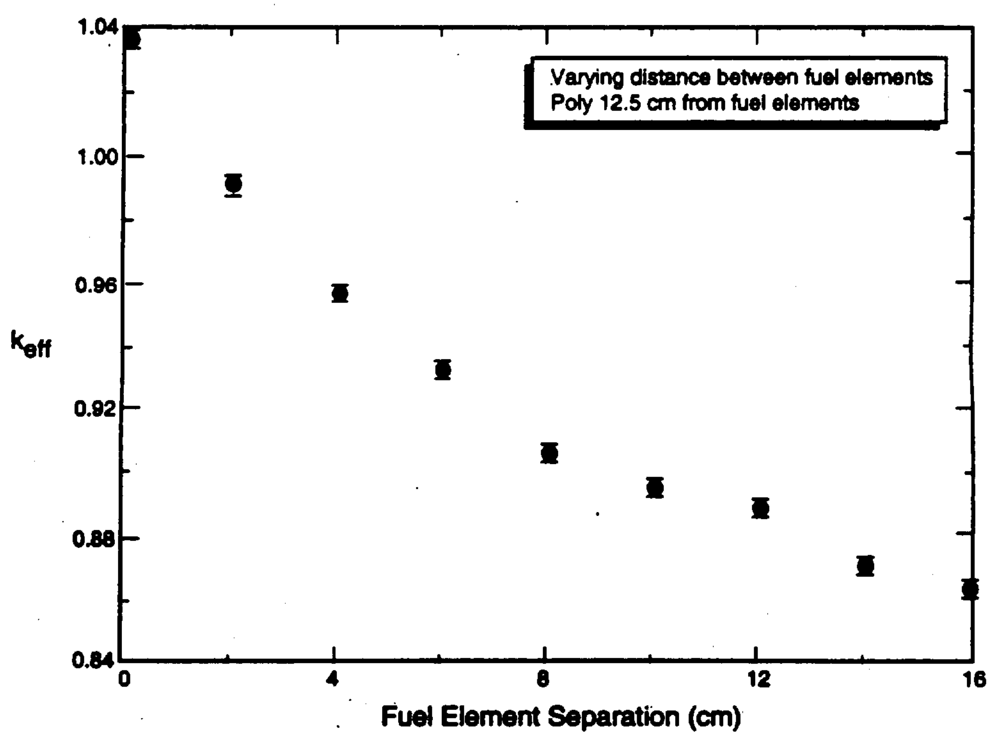

Figure 2.7(d)

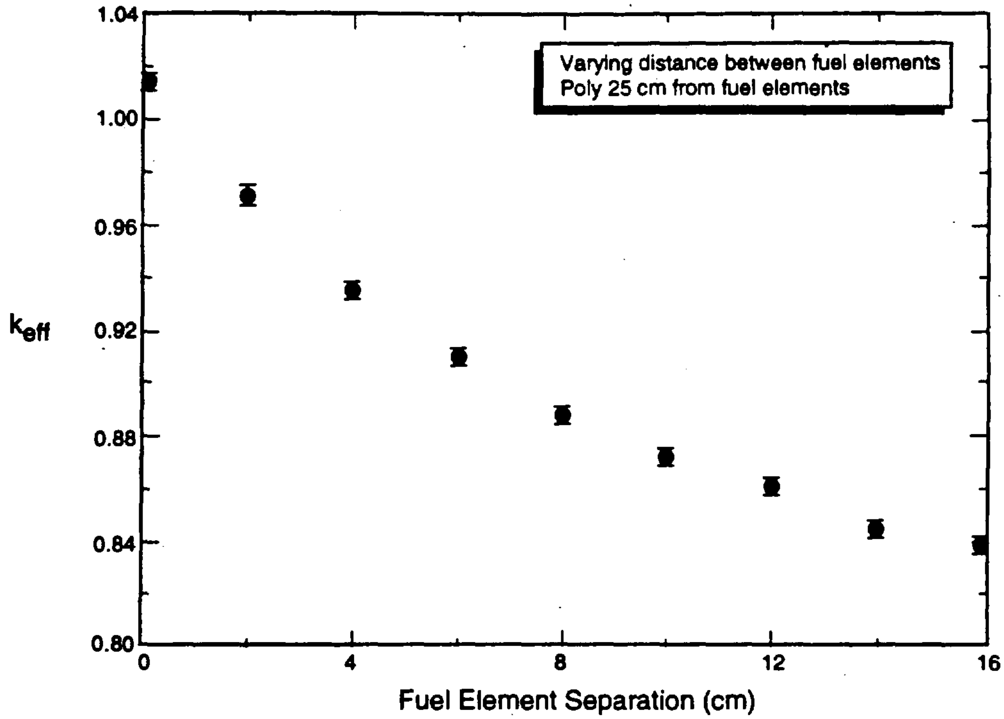

Figure 2.7(e)

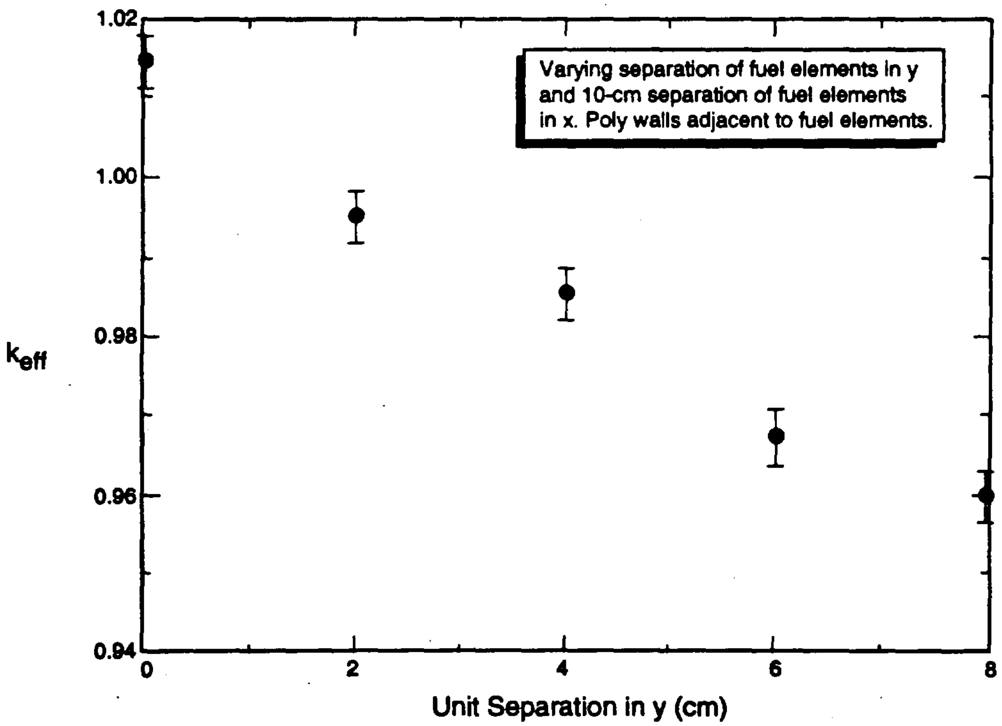

Figure 2.8. k_{eff}, with fuel elements fixed at 10cm separation in "x," as a function of separation in "y."

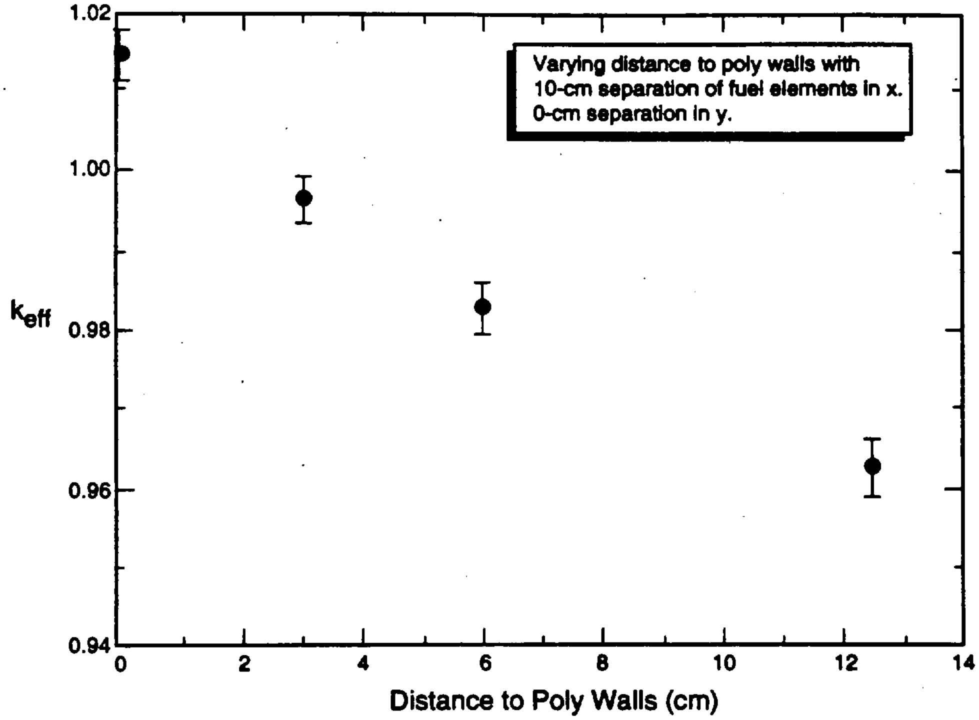

Figure 2.9. k_{eff}, with 10 cm separation in "x" and 0 cm separation in "y."

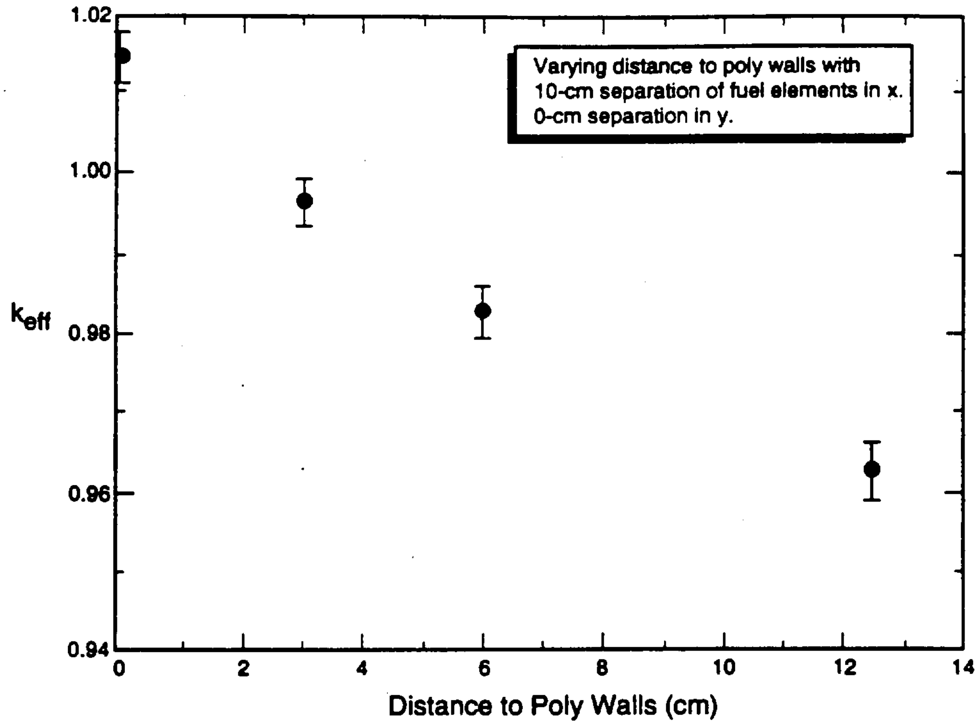

Figure 2.9. k_{eff}, with 10 cm separation in "x" and 0 cm separation in "y."

2.3. FISSION CHAMBER MEASUREMENTS

J. Bounds, P. Jaegers, D. Barton, and D. Rutherford

A standard technique for measuring the absolute fission cross section in a critical assembly has been the use of fission chambers. Fission chambers typically contain microgram quantities of fissionable material plated inside proportional counters. Taking the ratio of fission counts for different isotopes gives what is known as a spectral index for the assembly. The actual number of fissions is the product of the distribution of neutron energies and the respective cross-section of the isotopes at those energies. These spectral indices thus give experimental validation points to computer codes that calculate them.

An effort is currently underway to continue and expand the spectral indices measurements for the LACEF assemblies. The fission chambers being used are the four-barrel design, shown in Fig. 2.10. Each of the four barrels is a proportional counter containing a fissionable isotope. Two chambers are being used now; one has a U-238/U-235/Np-237/Pu-239 combination, and the other a U-233/U-235/U-238/blank combination. A typical data set is shown in Fig. 2.11. The spectra show a characteristic double-hump due to fission fragments.

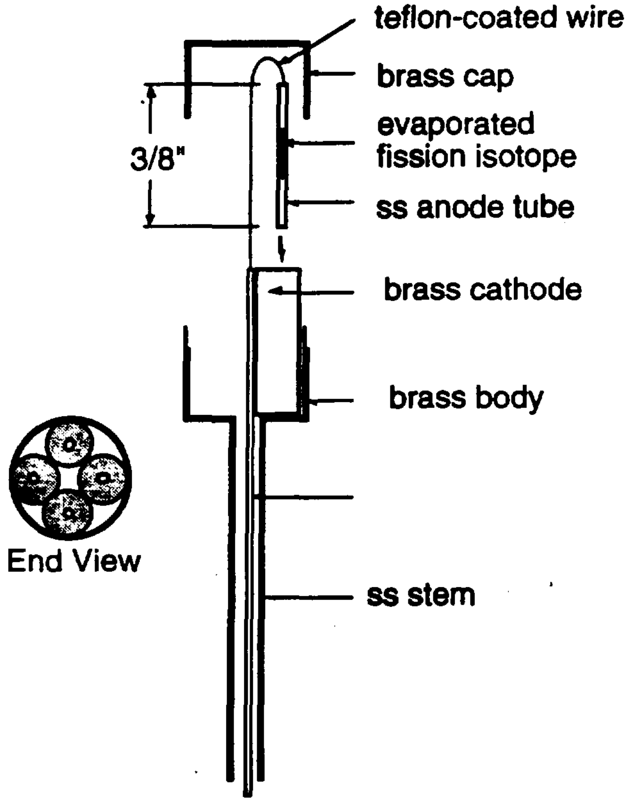

Figure 2.10. Design of the four-barrel fission chamber.

Figure 2.11. Data from a fission chamber in Big Ten. Four spectra are acquired and displayed simultaneously. The second spectrum above is from the blank barrel.

A quick run was made with a chamber centered outside of Big Ten rather than on an axis inside of Big Ten. A comparison of spectra inside and outside the assembly appears in Fig. 2.12. Note the dramatic change in relative fissions, attributable to room return of neutrons.

Data has now been acquired with each chamber in Big Ten. The chamber will next be used in Flattop and then either SHEBA or SKUA. This data will be tied to the benchmark measurements of D. M. Gilliam and J. A. Grundl.

The previous fission chamber data had been acquired by people who have since retired or moved to other jobs. The present effort serves both to obtain valuable data and to maintain competency in LACEF for doing such measurements.

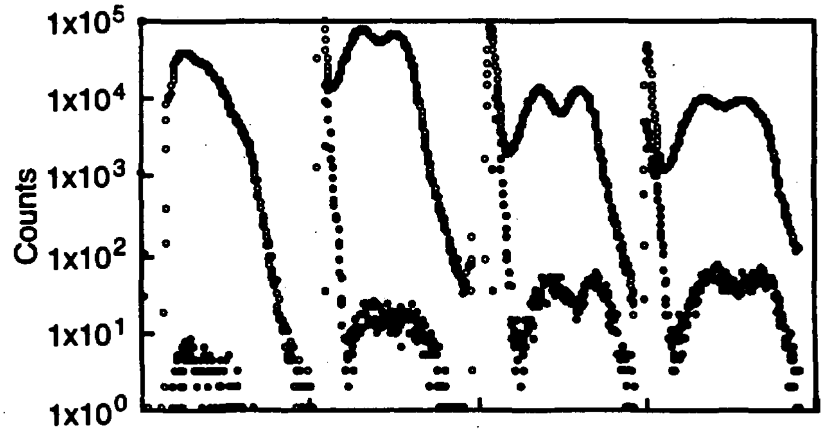

Figure 2.12. Comparison of spectra for a fission chamber inside of Big Ten (open circles) and outside of Big Ten. Note the change in relative intensities.

Assessment Form 2

Date: 11/6/95

Assessment Form 2 No.: C-COO-1-4 page 1
Review Area: The Conduct of Operations Program for the Y-12 Plant
Responsible Individual: W. A. Condon

Finding -- A statement of fact documenting a deviation from an applicable Federal law, DOE Order, standard, safety requirement, performance standard, or approved procedure.
Concern -- Any situation while not in violation of any written procedure, in the judgment of the assessment team member indicates less than optimal performance and could be the indicator of more serious problems.
Observation -- Any situation while not in violation of any written procedure or requirement, in the judgment of the assessment team member is worthy of raising to the attention of site management in order to enhance overall performance.
Noteworthy Practices -- Practices that are notable and will have general application to other DOE facilities for the improvement of overall safety or performance.

I. Identification Section

A. Statement

(Provide exact wording of the potential or final Finding, Concern, Observation or Noteworthy Practice):

Finding:

Occurrence reporting criteria of Y60-161 does not meet DOE Order 5000.3B requirements. For example, violations of procedures and items of management interest are not required to be reported.

Background:

Y60-161, Occurrence Reporting, was developed to implement the requirements of DOE Order 5000.3B. However, Y60-161 does not adequately implement the requirements in 5000.3B and many applicable criteria have been omitted from the site reporting matrix. As a result, occurrences which meet 5000.3B reporting criteria are not being identified and reported. Additionally, management has not been sensitized to the importance of the reporting process and the threshold of reporting occurrences is too high. Examples of reporting inadequacies include: (1) failure to comply with procedure requirements and DOE commitments for performance of material control and accountability inventories, and (2) numerous CSA violations which were identified as below the reporting criteria but were not evaluated as a programmatic deficiency.

B. Information Requested

(List any information needed to further evaluate this item):

N/A

2.4. VOID MEASUREMENT IN BIG TEN

J. Bounds, P. Jaegers, D. Hayes

To do Rossi alpha measurements in Big Ten, a detector must be placed inside the assembly. The four-barrel fission chambers have too little material to efficiently capture neutrons. To determine what size detector could be placed in Big Ten, total excess reactivity was measured as a function of the volume of a void. Up to a 10"-in.-long by 1 1/2-in.-diameter cylindrical void can be supported. The data obtained is shown in Fig. 2.13.

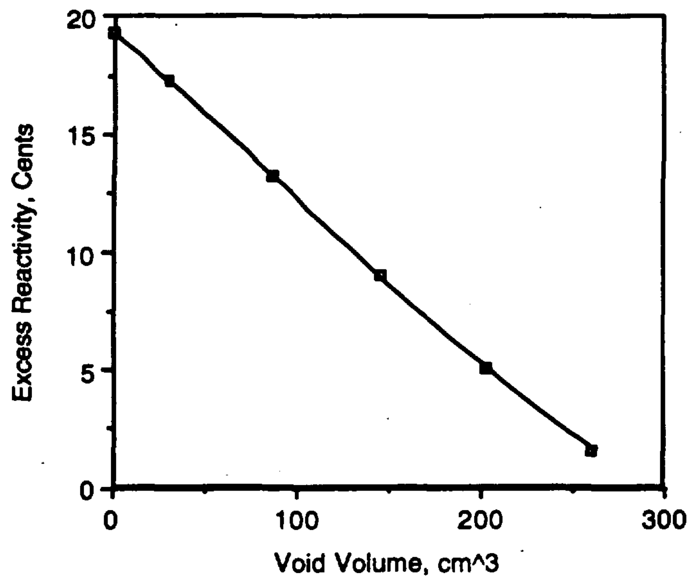

Figure 2.13. Total excess reactivity as a function of void size in Big Ten.

2.5. A STOCHASTIC, FOUR-ENERGY-GROUP MODEL OF A NUCLEAR ASSEMBLY

(The following is a summary of a dissertation by William Leon Myers, Ph.D. He received a Ph.D. from the University of Illinois at Urbana-Champaign, Department of Nuclear Engineering (Dr. Roy A. Axford, Advisor) while supported as a Graduate Research Associate at the LACEF. It illustrates LACEF's continued support of higher education in the area of crtiticality safety and understanding of the basic physics of neutron chain reacting systems.)

The time evolution of the neutron population of a nuclear reactor system is essentially a stochastic process. Each step in the life of a neutron, which is strongly influenced by the amount, nuclear cross section, and geometric arrangement of the materials present, can be dealt with in a probabilistic manner. The chaotic behavior of the fundamental processes contribute to the fluctuations of the neutron population away from expected or average values. These chaotic behaviors are as follows: the probabilities of the various neutron scattering, radiative capture, and fission interactions; variations in the number of neutrons born during a fission; variations in the kinds of delayed neutron precursors born during a fission; and variations in the number of neutrons introduced into the system by extraneous sources. In a reactor, with a large number of individual chain reactions taking place, each initiated by independently emitted neutrons from an extraneous source and subject to the chaotic nature of the fundamental processes mentioned above, the neutron population is going to fluctuate in a stochastic manner.

Two extreme cases are intuitively clear as examples of stochastic behavior of a nuclear reactor system. Consider, as a first example, a supercritical system that contains only a weak neutron source

(Refs. 4, 6, 7, 8). For such a system, the mean neutron population will be proportional to the source and it will increase exponentially in time. The actual neutron population, however, will be strictly zero until the first source-neutron appears. Thereafter, the population may increase very rapidly or it may die out to zero again. In neither case is the actual neutron population likely to resemble the mean population. For example, for a given burst of the fast critical assembly Godiva, the subsequent number of fissions and the time width of the burst were predictable and were verified by experiment, but the corresponding time of occurrence [wait time] of the peak fission rate varied (Ref. 4) The fluctuations in the wait time could not be predicted from a deterministic formulation. As the other extreme case, a power reactor may contain on the order of 10^{15} neutrons. For such a large number of neutrons, one would expect that the fluctuations in the population would be small compared to the mean value. Although relatively small, the fluctuations may nevertheless be quite interesting and their study might reveal significant information about the dynamic behavior of the system.

Model and Theory

For this research, the formulation of a stochastic, four-energy-group, space-independent model of a nuclear assembly was studied. The model included one group of delayed neutron precursors and an extraneous source. The model treated the evolutionary birth and death processes of a neutron chain in a nuclear assembly as a multi-dimensional Markovian process. The probability balance equation was derived by accounting for all the birth and death processes occurring in the assembly and then was changed into its differential-difference form. Using the probability-generating-function technique, the differential-difference equation was transformed into a first order partial differential equation in the probability generating function. Equations for the following functions were derived from the characteristic equations for the various moments of the first order P.D.E. for the probability generating function: neutron and delayed neutron precursor population probability distribution functions, the deterministic neutron and delayed neutron precursor population functions (means, variances, and covariances), and the extinction probabilities of the neutron chains.

The probability distribution function for the population of neutrons and delayed neutron precursors is defined as follows:

$P(n_1, n_2, n_3, n_4, m, t) =$ Probability distribution function that defines n_1 neutrons to be in energy group 1, n_2 neutrons to be in energy group 2, n_3 neutrons to be in energy group 3, n_4 neutrons to be in energy group 4, and m delayed neutron precursors to be present in the system at time t.

where

$$\sum_{n_1=0}^{\infty} \sum_{n_2=0}^{\infty} \sum_{n_3=0}^{\infty} \sum_{n_4=0}^{\infty} \sum_{m=0}^{\infty} P(n_1, n_2, n_3, n_4, m, t) = 1.0$$

Numerical Results

Numerical calculations were made to find the probability distribution functions for the population of the four-energy-group neutrons and delayed neutron precursors (up to two neutrons per energy group or combined total in the system), the deterministic functions (means, variances, and covariances of the energy-dependent population for neutrons and delayed neutron precursors), and extinction probabilities of neutron chains for a 93% enriched U-235 fissile assembly. Results were calculated for three different cases:

1. An infinite fissile assembly with delayed neutrons ignored and no extraneous sources present,
2. A finite fissile assembly with delayed neutrons ignored and no extraneous sources present, and

(Refs. 4, 6, 7, 8). For such a system, the mean neutron population will be proportional to the source and it will increase exponentially in time. The actual neutron population, however, will be strictly zero until the first source-neutron appears. Thereafter, the population may increase very rapidly or it may die out to zero again. In neither case is the actual neutron population likely to resemble the mean population. For example, for a given burst of the fast critical assembly Godiva, the subsequent number of fissions and the time width of the burst were predictable and were verified by experiment, but the corresponding time of occurrence [wait time] of the peak fission rate varied (Ref. 4) The fluctuations in the wait time could not be predicted from a deterministic formulation. As the other extreme case, a power reactor may contain on the order of 10^{15} neutrons. For such a large number of neutrons, one would expect that the fluctuations in the population would be small compared to the mean value. Although relatively small, the fluctuations may nevertheless be quite interesting and their study might reveal significant information about the dynamic behavior of the system.

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where

$$\sum_{n_1=0}^{\infty} \sum_{n_2=0}^{\infty} \sum_{n_3=0}^{\infty} \sum_{n_4=0}^{\infty} \sum_{m=0}^{\infty} P(n_1,n_2,n_3,n_4,m,t) = 1.0$$

Numerical Results

Numerical calculations were made to find the probability distribution functions for the population of the four-energy-group neutrons and delayed neutron precursors (up to two neutrons per energy group or combined total in the system), the deterministic functions (means, variances, and covariances of the energy-dependent population for neutrons and delayed neutron precursors), and extinction probabilities of neutron chains for a 93% enriched U-235 fissile assembly. Results were calculated for three different cases:

1. An infinite fissile assembly with delayed neutrons ignored and no extraneous sources present,
2. A finite fissile assembly with delayed neutrons ignored and no extraneous sources present, and

3. A finite fissile assembly with delayed neutrons considered and no extraneous sources present.

Each case had four different parts:

1. The neutron chain was initiated by a neutron of energy group 1.
2. The neutron chain was initiated by a neutron of energy group 2.
3. The neutron chain was initiated by a neutron of energy group 3.
4. The neutron chain was initiated by a neutron of energy group 4.

All of the numerical integrations done in the calculations used a fourth-order Runge-Kutta technique. The time step for the numerical integration was chosen to be a fraction of the shortest energy-dependent neutron lifetime. The root finding routine used to find the extinction probabilities came from the LANL subroutine library called CLAMS (Ref. 3).

The four-energy-group cross sections used in the calculations were a collapsed version of the Hansen-Roach sixteen-energy-group cross sections (Ref. 5). A FORTRAN program was written that collapsed the cross section data from the Hansen-Roach sixteen-energy-group set (Ref. 5) into a four-energy-group set using collapsing weights obtained from the program TWODANT (Ref.1). The energy groups were collapsed as follows:

Group 1: Hansen-Roach group 1
Group 2: Hansen-Roach group 2
Group 3: Hansen-Roach groups 3-4
Group 4: Hansen-Roach groups 5-16

For the finite fissile assembly calculations, neutron leakage probabilities were estimated from TWODANT output and used as input to calculate equivalent neutron leakage cross sections. Figure 2.14 shows the TWODANT model used.

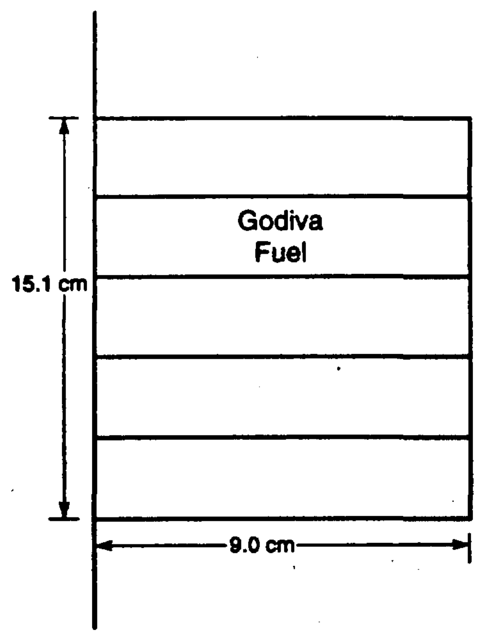

Figure 2.14. A Godiva-like model used to represent a 93% enriched U-235 finite fissile assembly for the TWODANT calculations.

Other nuclear data, such as the probability mass distribution for prompt fission neutrons, $\bar{\nu}$, the average number of neutrons born per fission, and the energy-dependent neutron lifetimes, were calculated using the methods described in Ref. 10.

The probability distribution functions are denoted in the figures as follows:

$$P_{0000} = P(0,0,0,0,t)$$
$$P_{1000} = P(1,0,0,0,t)$$
$$P_{2000} = P(2,0,0,0,t)$$
$$P_{1100} = P(1,1,0,0,t)$$
etc.

For the cases where one group of delayed neutron precursors is included in the model, the probability distribution functions are as follows:

$$P_{00000} = P(0,0,0,0,0,t)$$
$$P_{10000} = P(1,0,0,0,0,t)$$
$$P_{20000} = P(2,0,0,0,0,t)$$
$$P_{11000} = P(1,1,0,0,0,t)$$
etc.

Figures 2.15 through 2.18 and Tables 2.4 through 2.6 are some examples of the results found from the research.

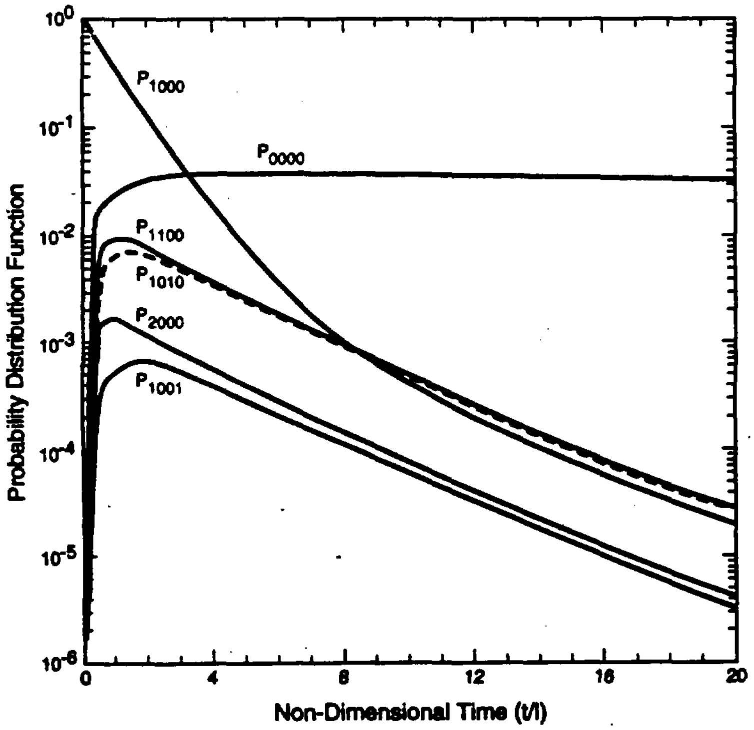

Figure 2.15. Probability distribution functions for the case of a neutron of energy group 1 initiating the neutron chain in an infinite 93% enriched U-235 system with no sources present and no delayed neutrons in the model.

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$$P_{2000} = P(2,0,0,0,t)$$
$$P_{1100} = P(1,1,0,0,t)$$
etc.

For the cases where one group of delayed neutron precursors is included in the model, the probability distribution functions are as follows:

$$P_{00000} = P(0,0,0,0,0,t)$$
$$P_{10000} = P(1,0,0,0,0,t)$$
$$P_{20000} = P(2,0,0,0,0,t)$$
$$P_{11000} = P(1,1,0,0,0,t)$$
etc.

Figures 2.15 through 2.18 and Tables 2.4 through 2.6 are some examples of the results found from the research.

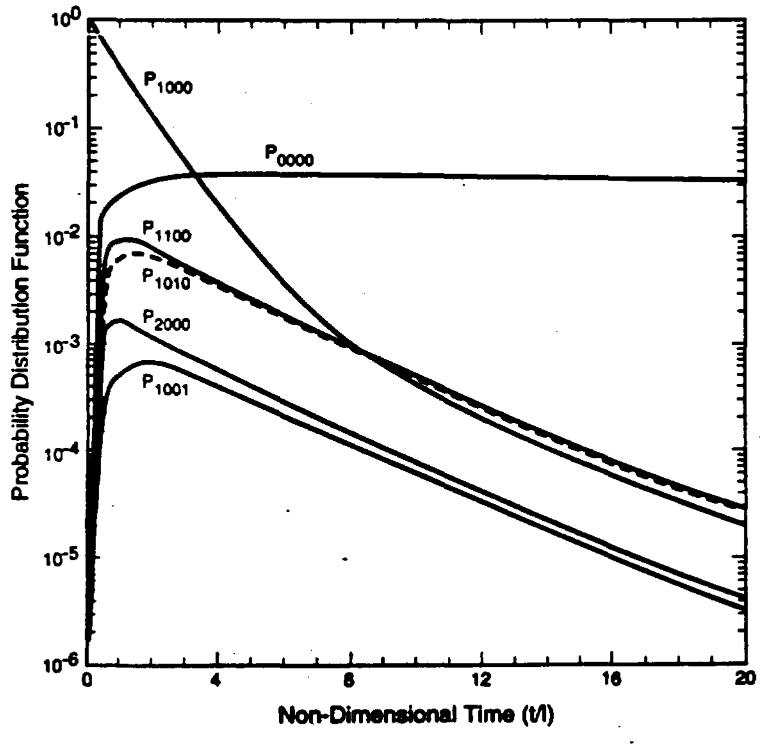

Figure 2.15. Probability distribution functions for the case of a neutron of energy group 1 initiating the neutron chain in an infinite 93% enriched U-235 system with no sources present and no delayed neutrons in the model.

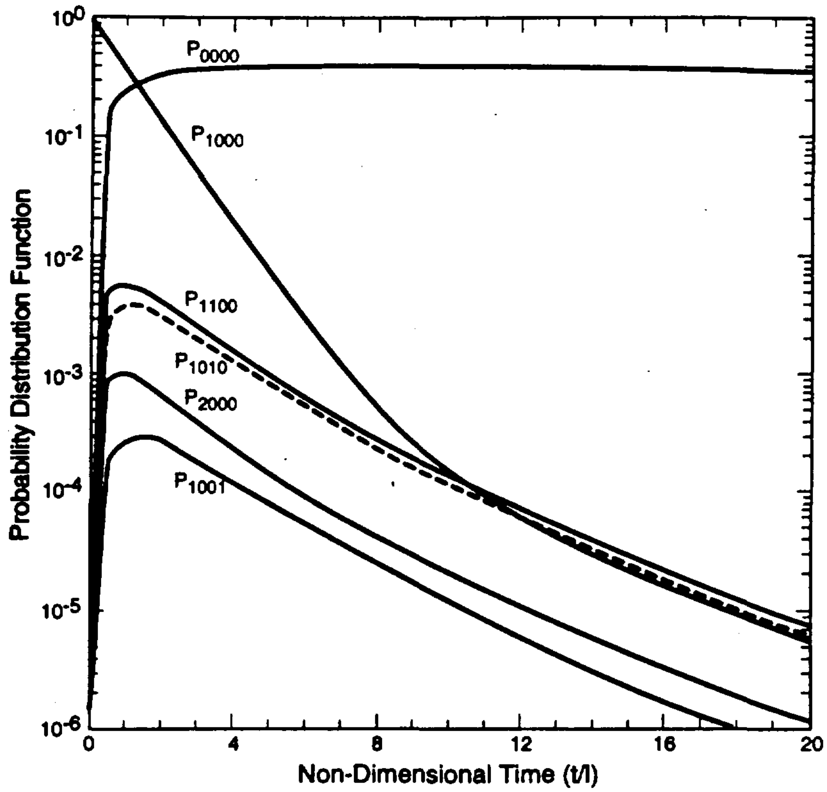

Figure 2.16. Probability distribution functions for the case of a neutron of energy group 1 initiating the neutron chain in a finite 93% enriched U-235 system with no sources present and no delayed neutrons in the model.

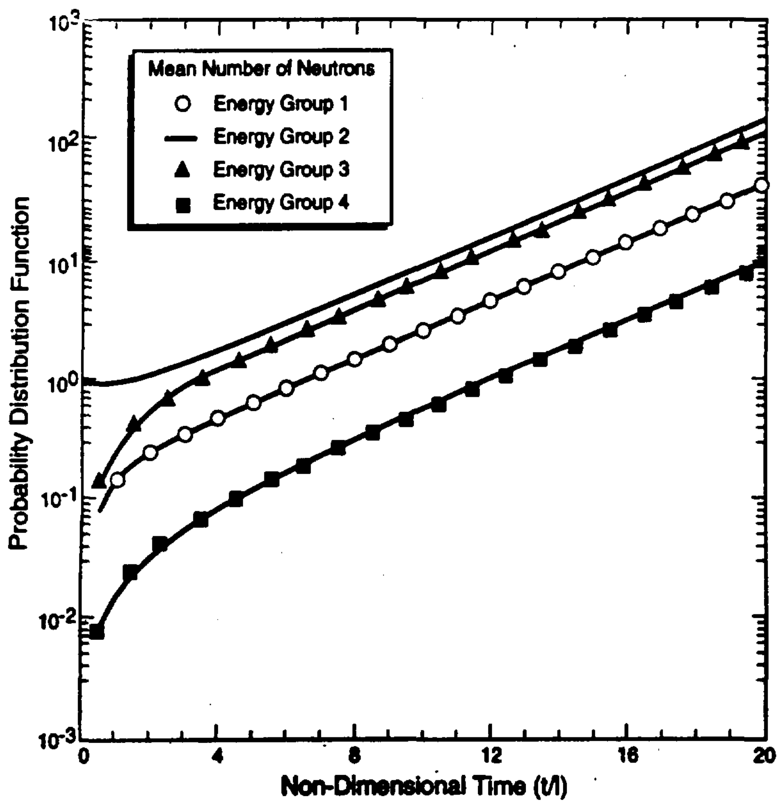

Figure 2.17. Deterministic functions for the case of a neutron of energy group 2 initiating the neutron chain in a finite 93% enriched U-235 system with no sources present and no delayed neutrons in the model.

Figure 2.18. Deterministic functions for the case of a neutron of energy group 2 initiating the neutron chain in a finite 93% enriched U-235 system with no sources present and with delayed neutrons included in the model.

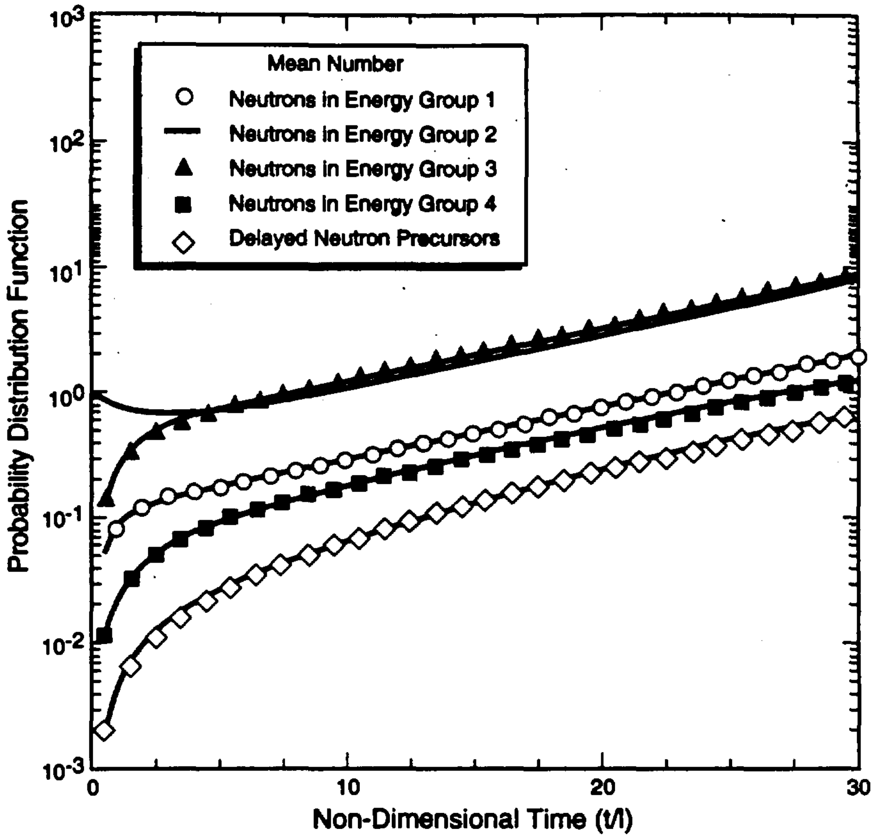

Table 2.4. Extinction probabilities of neutron chains caused by single neutrons introduced into an infinite 93% enriched U-235 system.

Energy Group	Extinction Probability
1	7.39E-02
2	1.43E-01
3	2.06E-01
4	3.66E-02

Table 2.5. Extinction probabilities of neutron chains caused by single neutrons introduced into a finite 93% enriched U-235 system.

Energy Group	Extinction Probability
1	5.88E-01
2	6.53E-01
3	6.84E-01
4	5.52E-01

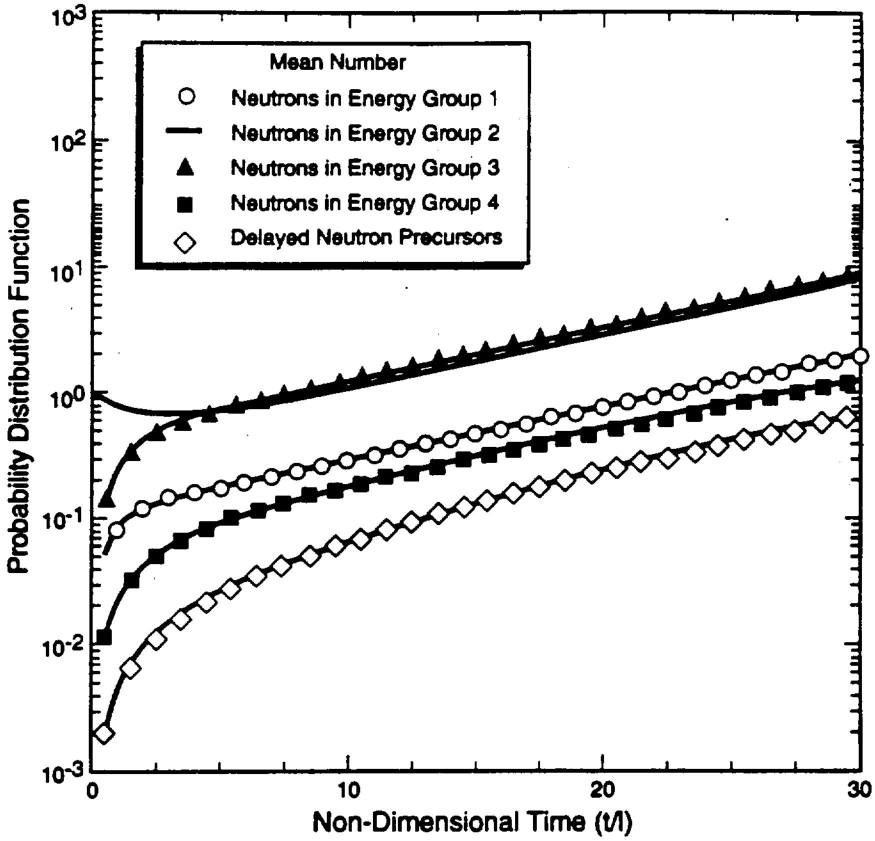

Figure 2.18. Deterministic functions for the case of a neutron of energy group 2 initiating the neutron chain in a finite 93% enriched U-235 system with no sources present and with delayed neutrons included in the model.

Table 2.4. Extinction probabilities of neutron chains caused by single neutrons introduced into an infinite 93% enriched U-235 system.

Energy Group	Extinction Probability
1	7.39E-02
2	1.43E-01
3	2.06E-01
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Energy Group	Extinction Probability
1	5.88E-01
2	6.53E-01
3	6.84E-01
4	5.52E-01

Table 2.6. Extinction probabilities of neutron chains caused by single neutrons introduced into a finite 93% enriched U-235 system (delayed neutrons included).

Energy Group	Extinction Probability
1	5.94E-01
2	6.36E-01
3	6.82E-01
4	5.50E-01

Discussion About Numerical Results

The results were compared to show the effects of neutron leakage, delayed neutrons, and different-energy neutrons starting the chain reactions in the fissile assembly. Data in the literature, to use for comparisons, was scarce. The results from the four-energy-group model for the absolute extinction probability of a neutron chain in an infinite U-235 fissile assembly compared favorably with the one-energy-group model found in the literature (Ref. 2).

Comparison of results between infinite and finite assemblies gives a perspective on the effects of neutron leakage to a fissile assembly. Neutron leakage is a very large contributor to the loss term for the finite assembly. This is especially noticeable when comparing the extinction probabilities of the neutron chains initiated by neutrons of the same energy for both cases. Because the material properties are the same for both models, the relative chance of a neutron being lost is less for the infinite case. Therefore, the extinction probability of the neutron chain is smaller for the infinite system. Also, comparison of the mean neutron populations from the deterministic results shows that the infinite system populations build up much faster. Differences between the neutron population probability distribution functions are not as obvious for the two cases. The difference that is most obvious is the asymptotic value that P_{0000} seems to be heading toward. The value of P_{0000} should approach the value of the extinction probability.

Comparison of the models with and without delayed neutron precursors is similar to comparing two different fissile assemblies, one with its prompt multiplication constant K_p equal to the total multiplication constant K_{eff} of the other assembly. The results of the model without the delayed neutrons do not have the time-lag of population buildup associated with the decay time of the delayed neutron precursors. This can be seen by comparing the mean neutron populations from the deterministic results for the two models. There isn't much difference in the extinction probabilities of the neutron chains calculated for the two models. But note that if a delayed neutron precursor is in the system, the extinction probability is zero until it decays and a neutron is released. For the model of the fissile assembly chosen for this study, the data used for the calculations has all the delayed neutrons being born into energy group 3. One could state that the extinction probability of a neutron chain started by a delayed neutron is equal to the extinction probability of a neutron chain started by a source neutron of energy group 3.

In comparing the asymptotic value that $P(0,0,0,0,0,t)$ reaches and the calculated value of the absolute extinction probability of a neutron chain for all the cases considered, the asymptotic value of $P(0,0,0,0,0,t)$ undershoots the absolute extinction probability value by about 30-50%. Further numerical investigations were inconclusive as to why this occurred.

Summary, Conclusions, and Future Work

This study demonstrated from start to finish how to define, derive, and calculate the probability distribution functions for the population of neutron and delayed-neutron precursors, the deterministic functions, and extinction probabilities of neutron chains started by different-energy neutrons by treating the evolution process of neutron chains in a fissile material as a Markov process. The study demonstrated how to approximate a finite system in a space-independent model by treating neutron leakage as a non-fission absorption process. It compared the results from an infinite-system model with a finite-system model and analyzed the effects of including delayed neutrons in the model. The results were compared to show the effects of neutron leakage, delayed neutrons, and different-energy neutrons starting the chain reactions in the fissile assembly.

For an initial analysis of a problem for which one is investigating effects that depend of the neutron energy on the static and dynamic response of a fissile assembly, the methods and analysis used in this research have an advantage over other techniques (Monte Carlo methods and other deterministic analysis). Transient Monte Carlo analysis is very computer-time intensive and the results from such an analysis are basically the probability of a certain value and a range around that value. Such a study may not justify the computer time when a simpler, faster method of finding the same results is available. Many deterministic codes find the mean values, but don't find the variances (first moment around the mean). The theoretical variances are important to know when extracting reactor physics parameters from experimental measurements (for example, static analysis of a fissile assembly using the Feynman Variance to Mean method [Ref. 9,11]). The methods used in this research, in a fairly simple and expedient manner, are able to find the probability distribution functions or neutron and delayed-neutron populations, the deterministic functions (mean and variances), and the extinction probability of a neutron chain for a space-independent, four-energy-group model of a fissile assembly (no feedback).

Some practical applications of this research are to help model zero-power critical assemblies or critical experiments and to analyze experimental data from them. Probability distribution functions for the neutron population can be used to estimate "wait times" for burst mode operations of some critical assemblies. Also, future work could involve modeling experiments that use detectors that measure both fast and thermal neutrons. The statistics from these experiments could be compared with the stochastic formulation of the model to deduce the degree of subcriticality of the system or find other parameters of interest in reactor physics. A stochastic formulation of a newly conceived criticality experiment could help to evaluate safety limitations or predict results important for the design.

An obvious expansion of the work is to make the model spatially dependent. More detail can be included with respect to geometry and material distributions. Inclusion of feedback effects in the model can be studied by using time-dependent reaction probabilities to incorporate the feedback effects. Also, expansion of the model to include more groups of delayed neutron precursors (six groups) could be done to investigate how the differences in delayed neutron fractions from fissions in each energy group affect the system. An analysis with a multi-energy group model with six groups of delayed neutron precursors for each energy group might yield more information than a mono-energetic model with six groups of delayed neutron precursors.

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3.0 SUBCRITICAL MEASUREMENTS

3.1 SOURCE JERK

C. A. Goulding, A. A. Robba, and J. J. Malanify

We have been reactivating the source jerk series of measurements. The apparatus was reassembled in the basement of the high bay and made operational this fall. Aside from reassembling the old apparatus, we also made some improvements on the source-transfer instrumentation. With these improvements, the source transfer time was reduced to 200 ms. There still seems to be some time-dependent background associated with the source not entering the shield immediately.

We made some subcritical measurements on a U^{235} assembly. The measurements reproduced the calculated reactivity at k_{eff} of 0.9 but deviated from the calculated values as the reactivity was lowered. We believe that this effect was the result of several causes:

1. Time dependent background,
2. Non-normal mode components in the driven system, and
3. Neutron detectors viewing the source neutrons.

We have redesigned the system to remedy these problems. First, the source jerk system has been completely redesigned and is under construction. Second, we will analyze the data not using the prompt drop but rather just the tail to see :f the agreement improves. Third, we will use smaller neutron detectors that will be completely shielded from the source by the assembly.

3.2. SUBCRITICAL MEASUREMENTS AND COMPUTATIONS FOR THE HAND STACK EXPERIMENT

P. Jaegers and C. Goulding

Many definitions are used for the multiplication of a subcritical assembly. The multiplication of a subcritical assembly is highly dependent upon the position of the source in the assembly. This paper will discuss the progress in understanding and the application of these many definitions. Both computational models and experimental data will be used to determine the multiplication of a subcritical assembly, namely the hand stack experiment. Additionally, it will be shown that many of the definitions of multiplication in use have fundamental assumptions that are not valid under a wide variety of situations.

There are in general two types of neutron multiplication as defined by Serber (Ref. 1, the first being the net multiplication, also called the leakage multiplication, and the second is the total multiplication. In practice there are no simple analytical formulations that relate the neutron reproduction factor k_{eff} to either the net or total multiplication of a subcritical system; although the few expressions that have been formulated have severe limitations.

The net or leakage multiplication M_L of a subcritical assembly is defined as the net number of neutrons that are produced per source neutron and is written as

$$M_L = \frac{S + F - A}{S} \quad , \tag{1}$$

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where S is the number of source neutrons, F is the number of neutrons produced from induced fissions, and A is the total number of neutrons absorbed. This includes both absorbed source and fission neutrons. If we write a neutron balance for the system as

$$S + F - A - L = 0 \quad , \tag{2}$$

where L is the number of neutrons that leak from the system, we can rewrite the leakage multiplication as

$$M_L = \frac{S + F - A}{S} = \frac{L}{S} \quad , \tag{3}$$

so that the net multiplication is equal to the total number of neutrons that leak from the system per source neutron, and hence the name leakage multiplication.

Ideally, one could measure the net multiplication of a system by first taking a neutron count of a bare source in some fixed geometry with respect to a detector system. One would then take a second measurement with the same source-detector geometry with the source embedded in the assembly. The net multiplication would simply be the second measurement divided by the first. In general, it is extremely difficult to perform such a measurement because all neutron detectors have a response that depends upon neutron energy; therefore, one would prefer to perform the measurements such that the detector sees approximately the same neutron energy spectra for both measurements. This can be accomplished by performing the first measurement with all of the non-fissile components of the assembly present, i.e. reflectors and moderators, then performing the second measurement with the fissile material present. We can define the net multiplication of a source in an assembly with no fissile material present as

$$M_{nf} = \frac{S - \widetilde{A}}{S} = \frac{\widetilde{L}}{S} \quad . \tag{4}$$

Since some source neutrons are always absorbed, M_{nf} will always be less than one. We will now define the apparent net multiplication, M_{AL}, as the ratio of the net multiplication of the subcritical system to that of the net multiplication of the system with no fissile material as

$$M_{AL} = \frac{\dfrac{S + F - A}{S}}{\dfrac{S - \widetilde{A}}{S}} = \frac{S + F - A}{S - \widetilde{A}} = \frac{L}{\widetilde{L}} \quad . \tag{5}$$

We can then use the procedure as outlined above to determine the apparent net multiplication of a subcritical assembly, because the leakages with the fissile material and without the fissile material are both readily measurable. Additionally one can easily model such an experiment using a fixed-source type of calculation because neutron transport codes will usually determine the integral neutron leakage of the modeled system, and thus the apparent net neutron multiplication can be determined.

The total multiplication M_T of a subcritical assembly is defined as the total number of neutrons produced per source neutron and is written as

$$M_T = \frac{S + F}{S} \quad , \tag{6}$$

where S is again the number of source neutrons and F is the number of neutrons produce from induced fissions. Experimentally, it is difficult to measure the total multiplication of a subcritical system. Some methods (Refs. 2,3) based upon statistical models have been developed to measure the total multiplication, however we will not go into them here. Needless to say there is no easy method, such as that employed to determine the net multiplication, by which the total multiplication can be determined. It is however possible to infer the total multiplication of an assembly from the net multiplication. If we write the total and the net multiplications as

$$M_T - 1 = \frac{F}{S} \tag{7}$$

and

$$M_L - 1 = \frac{F - A}{S} \ , \tag{8}$$

the ratio of equations (7) and (8) becomes

$$\frac{M_L - 1}{M_T - 1} = \frac{F - A}{F} \ . \tag{9}$$

If we additionally assume that

$$F = \int \nu\Sigma_f(\vec{r},E)\phi(\vec{r},E)\ d\vec{r}dE = \overline{\nu\Sigma_f}\int \phi(\vec{r},E)\ d\vec{r}dE \tag{10}$$

and

$$A = \int \Sigma_a(\vec{r},E)\phi(\vec{r},E)\ d\vec{r}dE = \overline{\Sigma_a}\int \phi(\vec{r},E)\ d\vec{r}dE \ , \tag{11}$$

Eq. (9) reduces to

$$\frac{M_L - 1}{M_T - 1} = \frac{\overline{\nu\Sigma_f} - \overline{\Sigma_a}}{\overline{\nu\Sigma_f}} \ . \tag{12}$$

where $\overline{\nu\Sigma_f}$ and $\overline{\Sigma_a}$ are the spatially and energy averaged fission and absorption cross-sections. Computationally one can easily determine the total multiplication of a subcritical assembly by performing a fixed source calculation, and then utilizing information from the integral tables and Eq. (6) to determine the total multiplication.

Many formulas have been employed to relate the k_{eff} of a system to the total neutron multiplication. Probably the most famous and misused is

$$M_T = \frac{1}{1 - k_{eff}} \ . \tag{13}$$

Equation (13) is based upon the assumption that the source is a normal-mode source distribution. That is, the source distribution has the same energy and spatial distribution as the source, which satisfies the homogeneous transport equation. This assumption is not bad as long as k_{eff} is close to 1.0. However, as k_{eff} becomes smaller, this assumption is no longer valid, because a neutron source placed in the system becomes the overriding source of neutrons, and modes other than the normal mode will exist.

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The subcritical system examined was the hand stack experiment, which has a low multiplication. The hand stack experiment consists of a stack of interleaved 93%-enriched uranium foils and Lucite plates. The uranium foils are approximately 23 cm x 23 cm x 0.0076 cm, and the Lucite plates are 35.6 cm x 35.6 cm x 1.27 cm. Additionally there is a 7.62-cm-thick reflector of Lucite on both the top and the bottom of the stack, and a californium neutron source was placed in the fourth Lucite plate. The experiment was performed by starting with an initial configuration of five Lucite plates, no uranium, and the Lucite reflector plates. Neutron count data was collected for 300 s, then two Lucite plates were added and data was collected again. This process was repeated until 11 Lucite plates had been added. Once this was complete, the initial configuration was reassembled with the addition of 6 uranium foils, and the stacking was performed once again until 12 uranium foils were stacked. The apparent net multiplication was then determined. Additionally, the computer code Threedant (Ref. 4) was used to simulate the experiment via a fixed-source calculation and the k_{eff} of the system was calculated. These results are presented in Tables 3.1 and 3.2.

Table 3.1. The apparent net multiplication as determined experimentally and computationally.

Number of Uranium Foils	Experimental Apparent Net Multiplication	Calculated Apparent Net Multiplication
6	4.284	3.834
8	7.081	6.211
10	10.801	9.688
12	15.963	15.022

Table 3.2. Total multiplication as computed from fixed source and k_{eff} calculations.

Number of Uranium Foils	Fixed Source Total Multiplication	Calculated k_{eff}	Total Multiplication Based upon $M_T = \frac{1}{1 - k_{eff}}$
6	3.742	0.667	3.002
8	5.573	0.758	4.139
10	8.188	0.829	5.842
12	12.280	0.884	8.614

As one can see, the experimental and calculated apparent net multiplications are within 11% and agree fairly well. The discrepancies between the experimental and calculated net multiplications can be easily assigned to both experimental and computational errors due to the approximations used in the computer code. On the other hand, the two calculated total multiplications differ by 20 to 30%. The two calculated total multiplications differ so greatly because the assumptions for the k_{eff} based multiplication are not valid for a system, such as the hand stack, which is not being driven at normal mode.

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4.0 SOLUTION ASSEMBLY PHYSICS

4.1. SHEBA VOID EXPERIMENTS

C. Cappiello, K. Butterfield, and A. Criscuolo

The Solution High Energy Burst Assembly (SHEBA) is a solution-fueled reactor which uses 5% enriched uranyl fluoride solution for fuel. One of the parameters of interest in solution reactors is the formation and the effect of the voids caused by radiolytic gas production. The investigation of this phenomenon in SHEBA was separated into two parts. The first part used large aluminum "voids" to benchmark the MCNP and 3DANT calculations of the effects of voids. The second part of the investigation included a series of "high-power" free-run experiments in SHEBA to produce radiolytic gas and observe the effects. This section describes the results of the aluminum void experiments. The results, to date, of the bubble formation experiments are presented elsewhere in this progress report.

The void experiments will be used to benchmark the calculations done in Refs. 1 and 2. The experiments were performed by positioning relatively large aluminum void simulants at various locations in the SHEBA tank and finding the critical height of the system for each void position.

The experimental apparatus is shown in Fig. 4.1. Three void simulants and a void manipulation apparatus were fabricated. Two of the voids fit around the central thimble of the critical assembly vessel and one fit the inside contour of the outer wall of the vessel. The two large voids were fabricated to match the calculational voids in Ref. 1 and the third void is 1/3 the size of the calculational voids. All of the voids are cylindrical shell segments. The two inner voids are 574 cm^3 and 190 cm^3. The outer void is 558 cm^3. Figures 4.2, 4.3, and 4.4 show the dimensions of the voids. The voids are made of aluminum to reduce the contribution of the material to the neutronics of the system. Aluminum corrodes slowly in the uranyl fluoride, but for short (on the order of days) periods of time, there was no noticeable effect.

The manipulation apparatus consisted of an aluminum stalk that passed through a gas seal in the CAV cover and attached to a ball-screw mechanism. The ball screw was turned with a stepping motor that was controlled from the Kiva 1/SHEBA control room. Two limit switches are installed on the ball screw: one to stop the void simulant just above the bottom of the tank and one to stop the simulant when it is completely retracted from the fuel.

The critical height of the SHEBA system was measured with the three void simulants (one at a time) in the middle position, the bottom position, and in several intermediate positions. The following sequence of operations was followed for each measurement to ensure that negative or very small positive reactivity insertion occurred between steps.

1. Position the simulant at the upper limit switch (above the expected critical height of the fuel).
2. Fill the CAV and find delayed critical (DC).
3. If the void simulant is to be lowered to the center-level position from above the fuel, lower the fuel level at least below the centerline position of the void simulant (i.e., if the no-void delayed critical height is 45.3 cm, lower the level to a least 22.6 cm). If the void simulant is to be lowered from the center-level position to the bottom-level position, lower the fuel level at least 4 cm below the unperturbed delayed critical height.
4. Move the simulant into the desired position.

5. Adjust the fuel height to reestablish DC.
6. Adjust the fuel height as described in step 4 above.
7. . Go to step 4, and repeat as necessary.

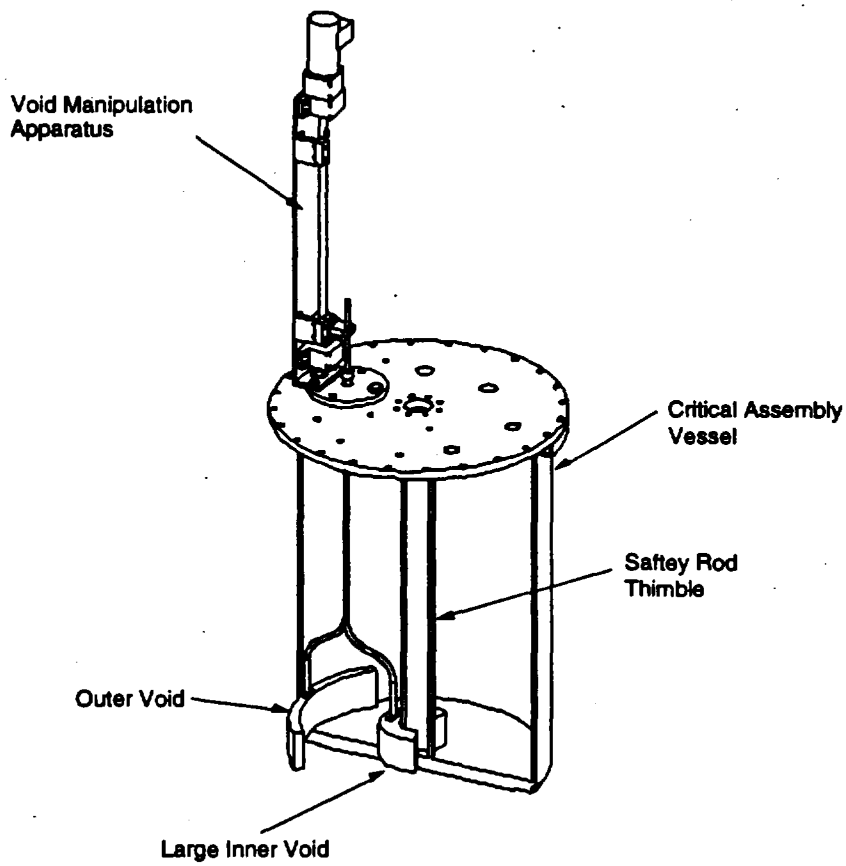

Figure 4.1. Experimental apparatus.

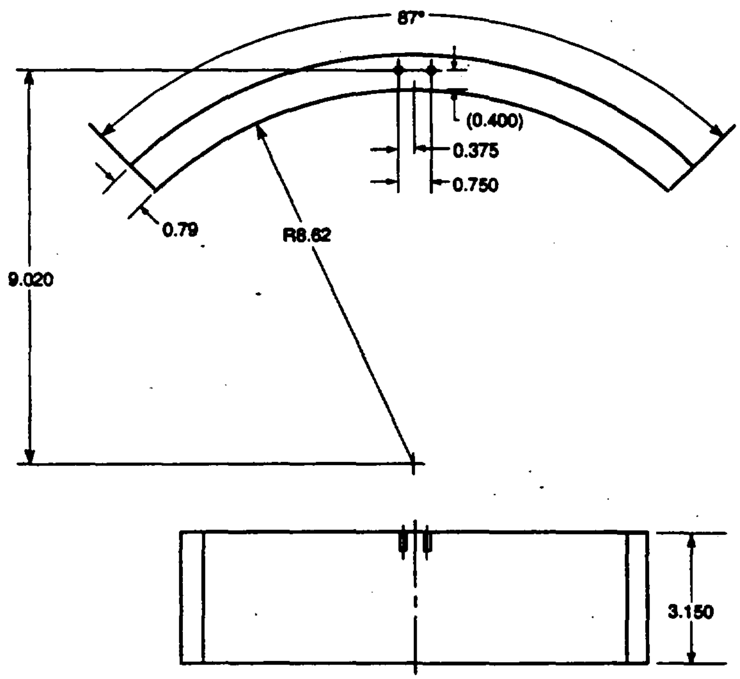

Figure 4.2. Outer void.

5. Adjust the fuel height to reestablish DC.
6. Adjust the fuel height as described in step 4 above.
7. . Go to step 4, and repeat as necessary.

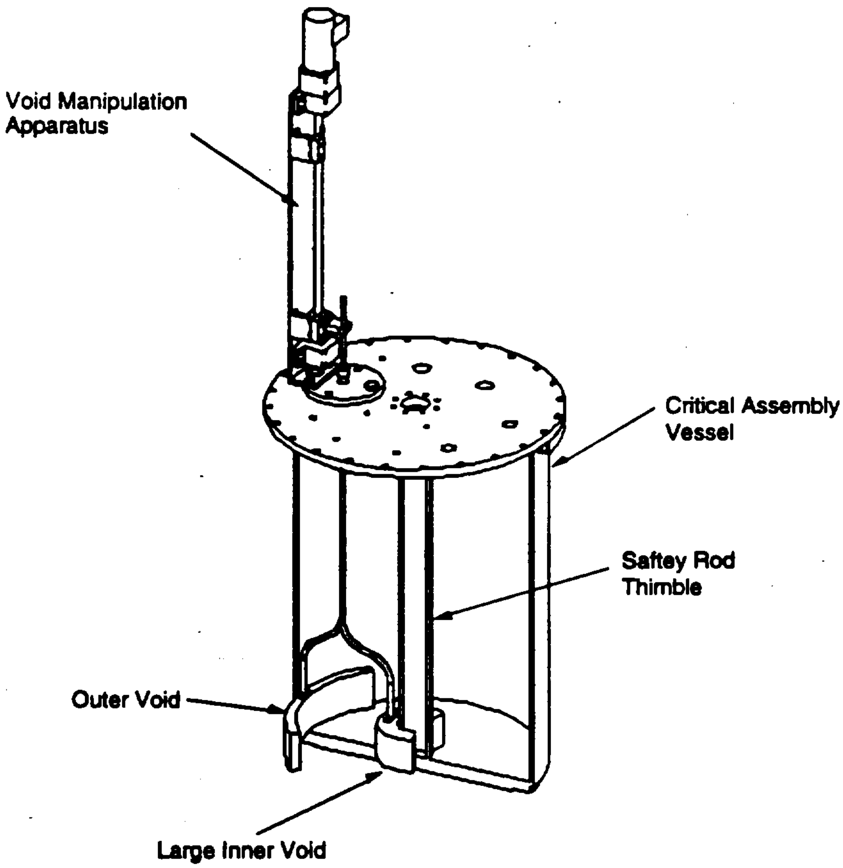

Figure 4.1. Experimental apparatus.

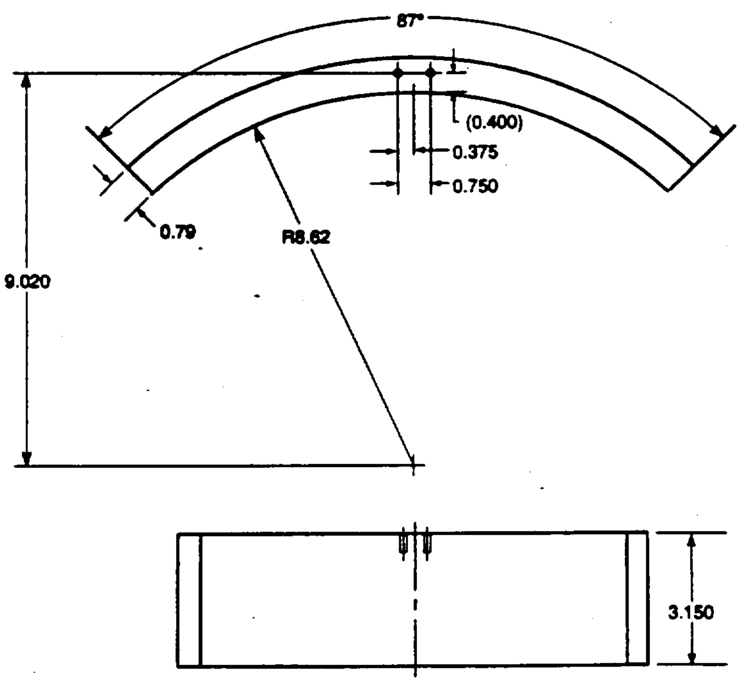

Figure 4.2. Outer void.

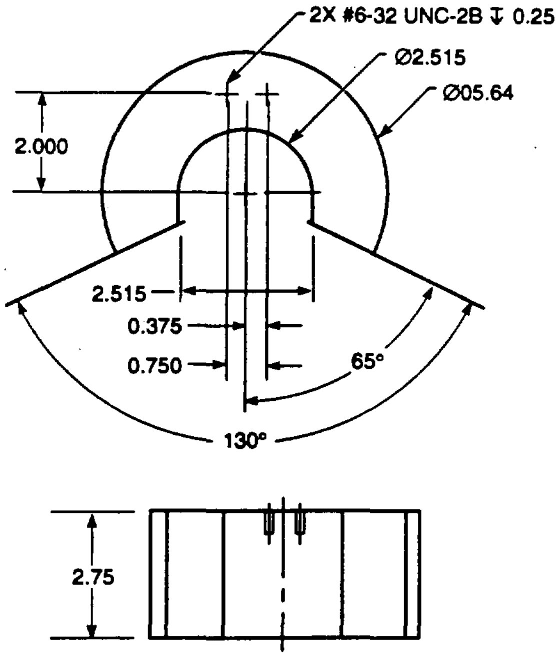

Figure 4.3. Large inner void.

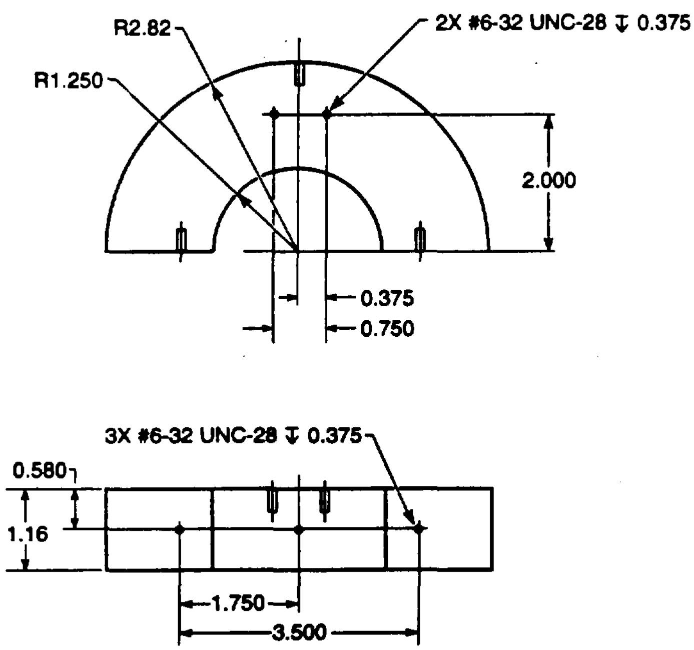

Figure 4.4. Small inner void.

Tables 4.1, 4.2, and 4.3 show the data for the void at the outside edge, the large void in the center and the small void in the center, respectively. A schematic diagram depicting the layout of the system (Fig. 4-5) and the meaning of DC and "Void Position" for these experiments is included with Table 4.1.

The first experiment was the outer void experiment. The data taken was within the calculational tolerance of the expected data. However, when the large center void was used, the worth of the void at the center had been grossly underestimated by the calculations and the experiment was terminated. At that time, the third void was constructed to be ~1/3 the size of the large inner void and the experiment was then repeated.

Table 4.1. Void at the outside edge.

Void Volume = 558.2 cm^3, Δheight due to void = .31 cm
Date: 8/24/94

Void Position	DC (cm)	Temperature (°C)	DC at 29.3° C	ΔDC due to void (cm)
No void	45.60	28.1	45.72	
Void at top, 10 cm	45.91	29.1	45.93	-0.10
Void at top, 10 cm	45.90	29.3	45.90	-0.13
Void at 27 cm	46.33	29.3	46.33	0.30
Void at bottom	46.02	29.3	46.02	-0.01
Void at 12.7 cm	46.01	29.3	46.01	-0.02

Table 4.2. Large void at the center.

Void Volume = 574.04 cm^3, Δheight due to void = .31 cm
Date: 8/30/94

Void Position	DC (cm)	Temperature (°C)	DC at 25.7 °C	ΔDC due to void (cm)
No void	45.19	24.4	45.32	
Void at top, 9 cm	46.25	25.3	46.29	.93
Void at center, 25 cm	>48.43 Not Critical	25.7	>48.43	>3.07

Table 4.3. Small void at the center.

Void Volume = 190.09 cm^3, Δheight due to void = .11 cm
Date: 9/20/94

Void Position	DC (cm)	Temperature (°C)	DC at 25.7 °C	ΔDC due to void (cm)
No void	45.05	21.8	45.07	
Void at top, 6 cm	45.34	21.8	45.36	0.18
Void at center, 25 cm	46.13	21.8	45.15	0.97
Void at 26 cm	46.13	21.8	45.15	0.97
Void at 21 cm	46.03	21.9	46.04	0.86
Void at 16 cm	45.81	22.0	45.81	0.63
Void at 11 cm	45.56	22.1	45.55	0.37
Void at 6 cm (top)	45.35	22.1	45.34	0.16
Void at 0 cm (no void)	45.08	22.1	45.07	

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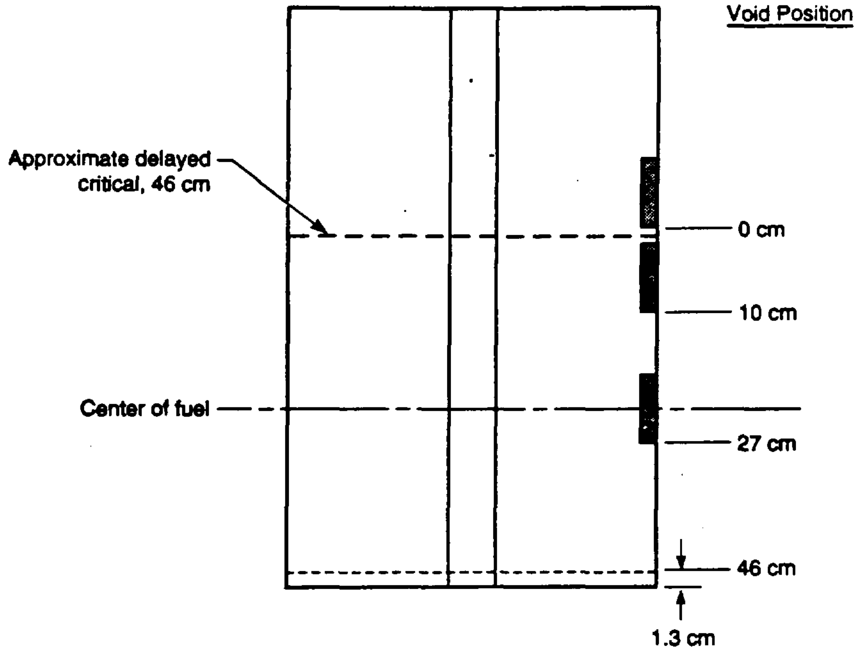


Figure 4.5. SHEBA assembly vessel with outer void positions shown.

Table 4.4 is a comparison of the calculations with the experimental data. For the void on the outside edge, the results agree fairly well with 3DANT, including the prediction that the void at the outer corner has a negative worth. This is because the fuel is being displaced from an area of low importance toward the center where it is worth more. MCNP did not calculate a negative worth in this area and overestimated the worth of the void at both locations.

Table 4.4. Calculations with experimental data.

Void/Location	Ref. 1 MCNP (cm)	Ref. 1 3-DANT (cm)	Ref. 2 3-DANT (cm)	Experiment Large Voids, (cm)	Experiment Small Inner Void (cm) Small/Corrected to Large Void
Delayed Critical	41.31 ± 0.3	44.13		45.0	45.0
Outside/Top	0.50 ± 0.3	-0.19	-0.17	-0.13	
Outside/Middle	0.98 ± 0.3	0.17	0.32	0.30	
Outside -- Difference	0.48 ± 0.6	0.43	0.49	0.43	
	Large Void	**Large Void**	**Small Void**		
Inner /Top	1.01 ± 0.3	0.33	0.19	0.93	0.18/0.544
Inner / Middle	1.08 ± 0.3	0.27	0.75	>3.07	0.97/2.92
Inner -- Difference	0.07 ± 0.6	-0.06	0.56		0.79/2.37

Figure 4.6 is a plot of the small inner void data and the results of Refs. 1 (corrected for void size) and 2.

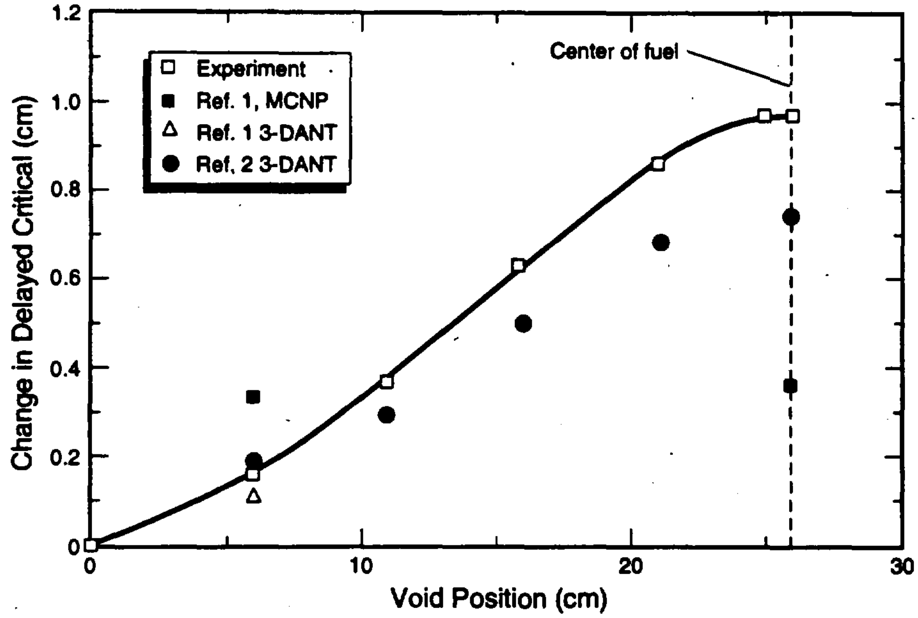


Figure 4.6. Plot of the small inner void data.

For the void at the inner thimble, the Ref. 1 calculations overestimated the value of the void at the top of the fuel and all of the calculations greatly underestimated both the worth of the void at the center and the large change in moving the void from the top to the center.

References

1. Steven G. Walters, "Reactivity Effects of Void Formations in a Solution Critical Assembly," Los Alamos National Laboratory report LA-12716-T.

2. R. J. LaBauve and Joseph L. Sapir, "Sheba-II as a Criticality Safety Benchmark Experiment," Draft document LA-UR 95-160.

4.2. COEFFICIENT OF THERMAL EXPANSION FOR URANYL FLUORIDE

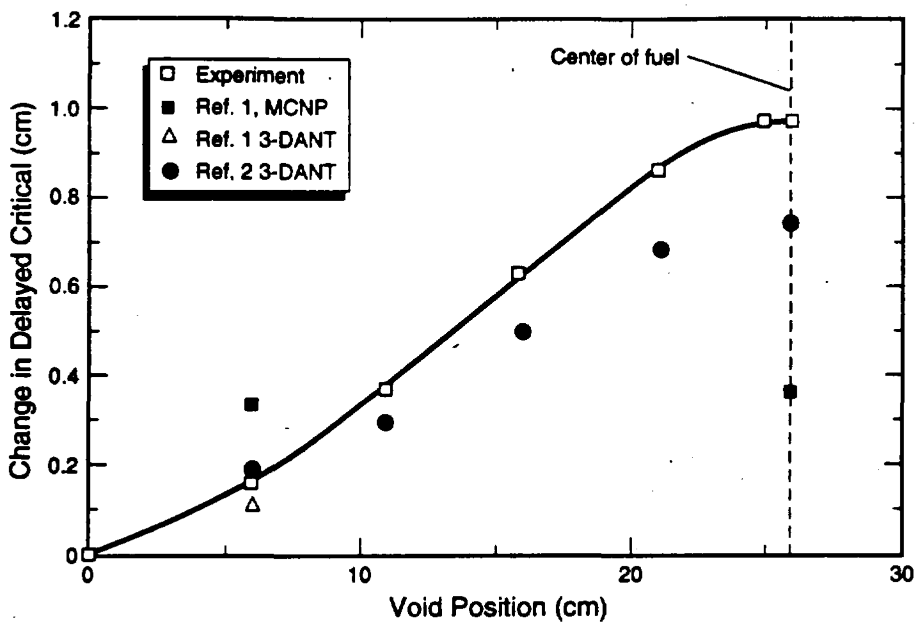
S. Rojas

In response to a desire to calculate the heat transfer associated with SHEBA runs, the coefficients of thermal expansion for the uranyl fluoride solution used in the assembly were calculated. Outlined below is the approach used for calculating the coefficients.

The coefficient of linear thermal expansion for uranyl nitrate may be calculated by considering the following relations:

$$\Delta V = V_0 \beta \Delta T$$

Figure 4.6 is a plot of the small inner void data and the results of Refs. 1 (corrected for void size) and 2.



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The coefficient of linear thermal expansion for uranyl nitrate may be calculated by considering the following relations:

$$\Delta V = V_0 \beta \Delta T$$

$\beta = 3\alpha$

where:

ΔV = change in volume

V_o = original volume

β = coefficient of volume thermal expansion

ΔT = change in temperature

α = coefficient of linear thermal expansion

Plugging in the volume for a cylinder, these expressions may be reduced to

$$\alpha = \frac{h_2 - h_0}{3h_0\Delta T} \quad ,$$

where

h_2 = ultrasonic level value and

h_0= integrated level value.

To get a very rough idea of the coefficients, the following approximate data was used:

h_2 = 42.53 cm,

h_0 = 42.30 cm, and

ΔT = 10 °C.

Using this data in the expression for α yields

$\alpha = 1.8 \times 10^{-4}$ and

$\beta = 5.4 \times 10^{-4}$.

From the next SHEBA run, we will try to collect more accurate values for h_2, h_0, and ΔT to get better approximations for the expansion coefficients for use in computer simulations.

5.0 EXCURSION PHYSICS

5.1. COMPUTER SIMULATION OF SHEBA EXCURSIONS

R. Kimpland, K. Butterfield, and C. Cappiello

A computer model, which simulates the dynamic behavior of the SHEBA assembly during excursions, has been developed at LACEF. This model is a simple lumped parameter model, which combines the neutron point kinetics equations with simple thermodynamic expressions for temperature and density. In addition, a radiolytic gas model has been developed to simulate the production and migration of radiolytic gas bubbles in an aqueous fissile solution. The results produced by this model have been compared with experimental data from the SHEBA assembly.

The goal of this work is a better understanding of the basic physics of aqueous fissile solutions, in particular, the reactivity feedback mechanisms present during an excursion and the phenomena of radiolytic gas formation and migration. It is anticipated that information gained from this work will be of use in other areas such as criticality accident analysis and in the design of the medical isotope production reactor (MIPR).

The effect of radiolytic gas on an aqueous fissile solution can be quite significant. During slow excursions, the radiolytic gas can provide a large negative reactivity insertion. During fast excursions above prompt critical, the radiolytic gas can produce a transient compression of the fissile solution. Radiolytic gas is formed by the process of radiation nucleation during high-power operation. Fission fragments slowing down in the aqueous medium dissociate water into dissolved hydrogen and oxygen gas. At some threshold point, enough dissolved gas exists so that the radiation nucleation process can occur along fission tracks. On the macroscopic level, this process is not well understood. An equation for the amount of radiolytic gas in an assembly after some threshold point is given by

$$\frac{dV_g}{dt} = C\frac{dE}{dt} - \frac{V_g}{1}$$

where the rate of gas production is proportional to the rate of energy being dumped into the system multiplied by an adjustable parameter, and the migration of gas bubbles out of the assembly is modeled by giving the bubbles a mean bubble lifetime.

The neutron point kinetics equations are coupled through reactivity feedback with an energy equation and an equation of state given by

$$\frac{dE}{dt} = MC_p\frac{dT}{dt}$$

and

$$\frac{d\rho_1}{dt} = -\rho_1\alpha\frac{dT}{dt}$$

where ρ_1 is the solution density, α is the isobaric compressibility, and T is the solution temperature.

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where ρ_1 is the solution density, α is the isobaric compressibility, and T is the solution temperature.

Three reactivity feedback mechanisms have been identified for the SHEBA assembly. The first is a neutron temperature feedback, which accounts for hardening of the thermal neutron spectrum with increasing solution temperature. The second is volumetric expansion, which accounts for the decrease in density of the solution with increasing temperature. The third is radiolytic gas feedback, which accounts for the radiolytic gas displacing fuel from regions of high importance to regions of low importance. An expression for the reactivity of the SHEBA assembly is given by

$$\rho = \rho_O - \alpha\Delta T - \phi\Delta V - \psi V_g$$

where ρ_o is a step insertion of reactivity and α, ϕ, and ψ are the neutron temperature, volumetric, and radiolytic gas feedback coefficients. A series of transport calculations, using the discrete transport code TWODANT, was performed to determine these feedback coefficients.

Figure 5.1 shows the model's prediction for a \$0.29 step insertion in the SHEBA assembly. The model has the ability to track the core-height change due to liquid expansion as well as expansion due to the radiolytic gas. The effect of these expansions can be seen clearly from the reactivity and power curves. A comparison between the model and experimental data from a \$0.29 free run in SHEBA is shown in Fig. 5.2(a and b).

The model demonstrates all the main features of the SHEBA free runs. The results of the comparison between the model and the experimental data are encouraging. Future work will concentrate on the radiolytic gas formation and migration mechanisms. Also, new methods for calculating reactivity feedback effects will be considered.

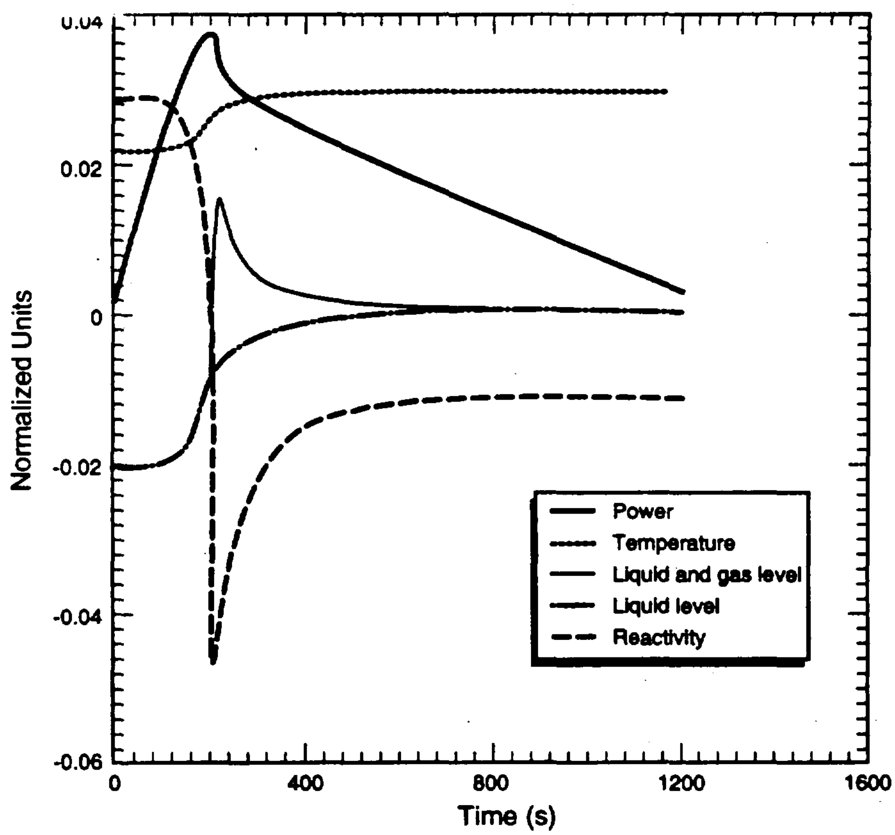
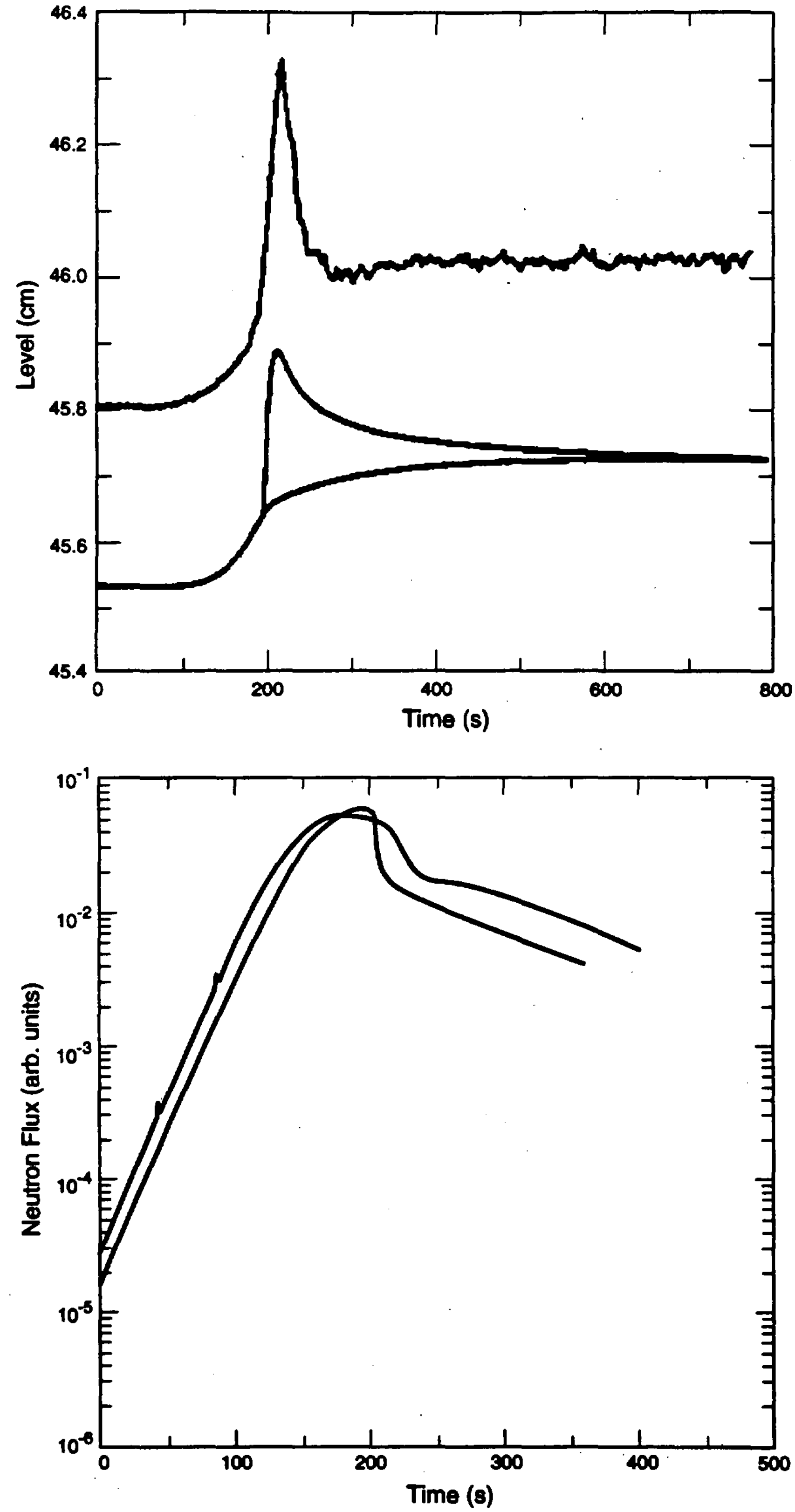


Figure 5.1. Model's prediction of a \$0.29 excursion in SHEBA.



(a)

(b)

Figure 5.2(a and b). Comparison between the model and experimental data from a $0.29 free-run in SHEBA.

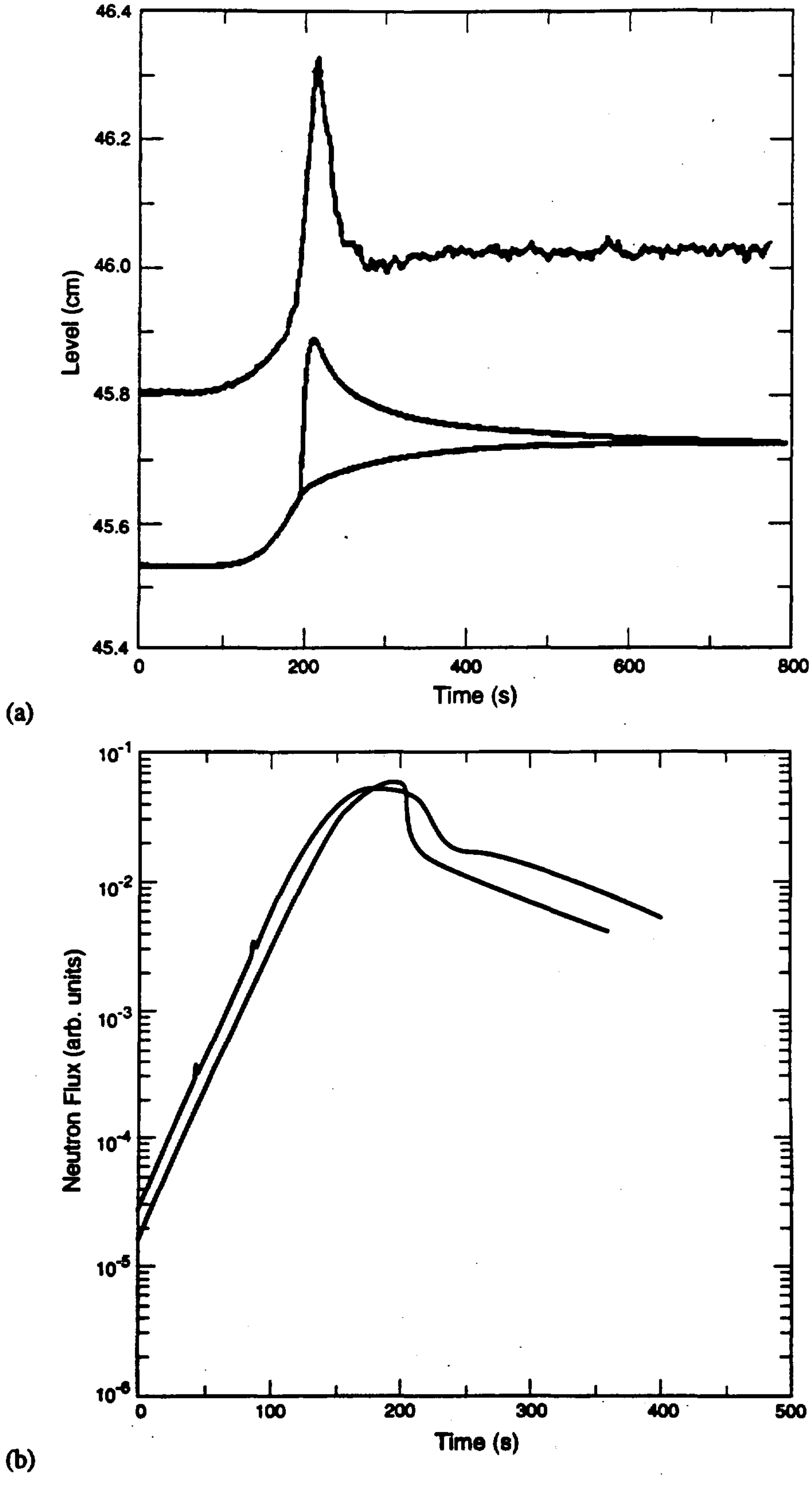


Figure 5.2(a and b). Comparison between the model and experimental data from a \$0.29 free-run in SHEBA.

5.2. CRITICALITY SAFETY STUDIES PLUTONIUM GEOLOGIC STORAGE

D. Hayes, R. Kimpland, W. Myers, R. Sanchez, W. Stratton

During the past three months, criticality calculations have been underway for mixtures of plutonium, silicon dioxide (SiO_2) and water (R. Sanchez) and for mixtures of U^{235}, graphite, and water (D. Hayes). These calculations make use of the ONEDANT (neutron transport) code with Hansen-Roach 16-group cross sections. The extent of the studies includes the full range of moderation from a metal system, bare and reflected, to the limiting concentration (the asymptote) of the fissile metal in the diluent, i.e. SiO_2, graphite, or water. In addition, the neutron lifetimes of these systems have been or are being determined over the full range of moderation (W. Myers, D. Hayes, R. Sanchez) to provide input to the Pajarito Dynamics Code (PAD), a computer program designed to calculate the fission and explosive energy release from a postulated supercritical system (R. Kimpland).

The plutonium studies are underway to provide information about the possible behavior (in geologic times) of plutonium mixed with cylinders of SiO_2 and buried deep underground in soil or rock of the same composition. The basic questions asked and to be answered are as follows. 1) Given the vagaries of climate, geology, and local chemistry over geologic eras, can circumstances arise that will allow the plutonium-SiO_2 mixture to disperse, mix, or otherwise be distorted and diluted with more SiO_2 and water (up to a few weight percent) so that the mixture (as modified) could become a critical system? The only extant example from geologic history is the Oklo reactor phenomenon of two billion years ago. Most of the criticality calculations of the Pu-SiO_2-water systems are illustrated in Fig. 5.3 (R. Sanchez). 2) Given, however improbable, that a critical system could be created, could this critical system (now a postulated reactor) be found in circumstances that would cause an explosion to occur? An explosion is defined as a fission energy release in a sufficiently short time that pressures developed would lead to motion outward (kinetic energy) sufficient to cause physical damage to its surroundings of SiO_2 soil or rock. The studies of the Oklo phenomenon have not uncovered any evidence of an explosion. Given completion of the criticality data, these will be examined to determine if any circumstance could lead to more than a very low fission power without the potential of developing an explosion. When possible, the appropriate conditions will be placed in the PAD code to examine the characteristics of a fission energy transient should one be postulated.

The graphite-U^{235}-water studies are underway to provide data for a criticality study of materials comparable to the Pu-SiO_2 water studies. Results of critical experiments exist for low, medium, and very large values of the carbon/uranium atom ratio thus providing immediate tests of the computational scheme. The computed critical data are illustrated in Figs. 5.4-5.7 (D. Hayes). Additionally, transient experiments have been completed for at least two graphite-uranium systems of widely differing atom ratios. These are the Kiwi-TNT experiment of 1965 and the controlled transients of the TREAT reactor that is owned and operated by the Argonne National Laboratory. The former was a destructive experiment (graphite vapor destroyed the reactor) while the latter was limited to a temperature rise of about 600 °C centigrade. Given successful operation of the PAD computer program, data for these graphite reactors will be used as initial conditions for calculated transients of the two reactors. Successful calculation of these two systems will provide confidence in the calculation of Pu-SiO_2-water systems should this be necessary.

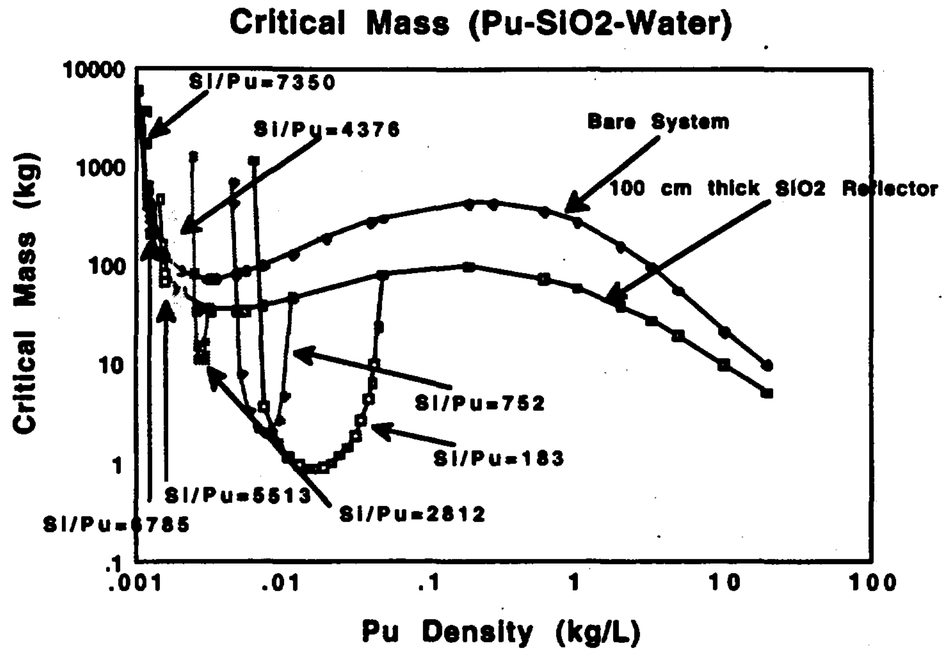


Figure 5.3. Critical masses of SiO_2-water-moderated Pu spheres.

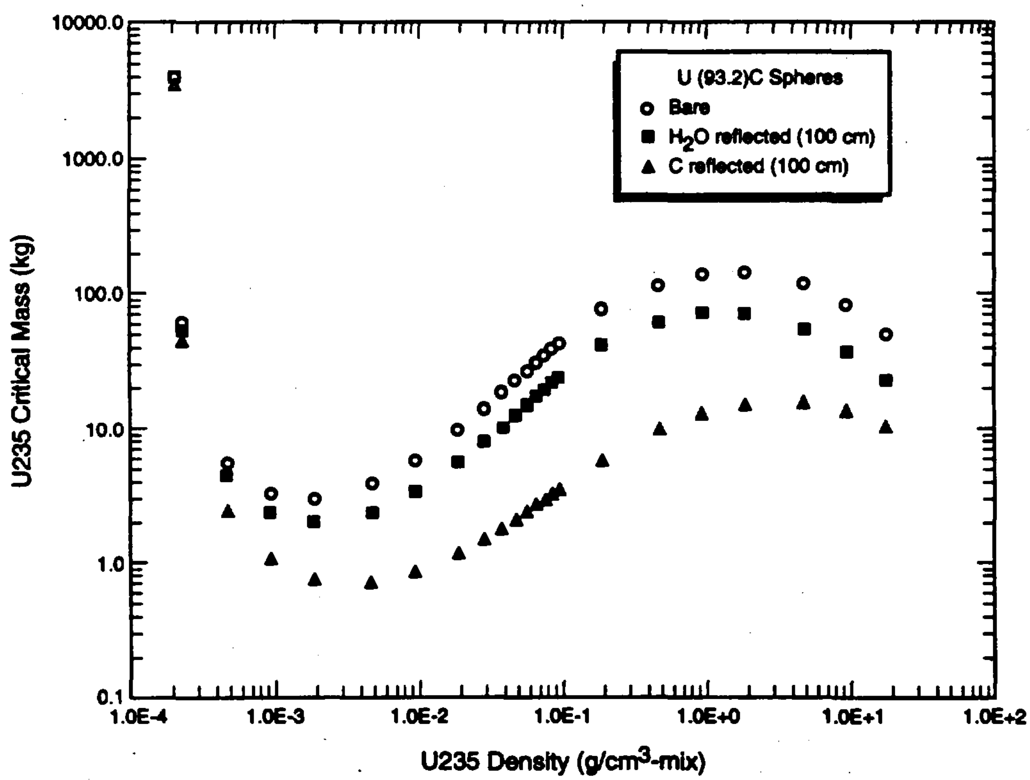
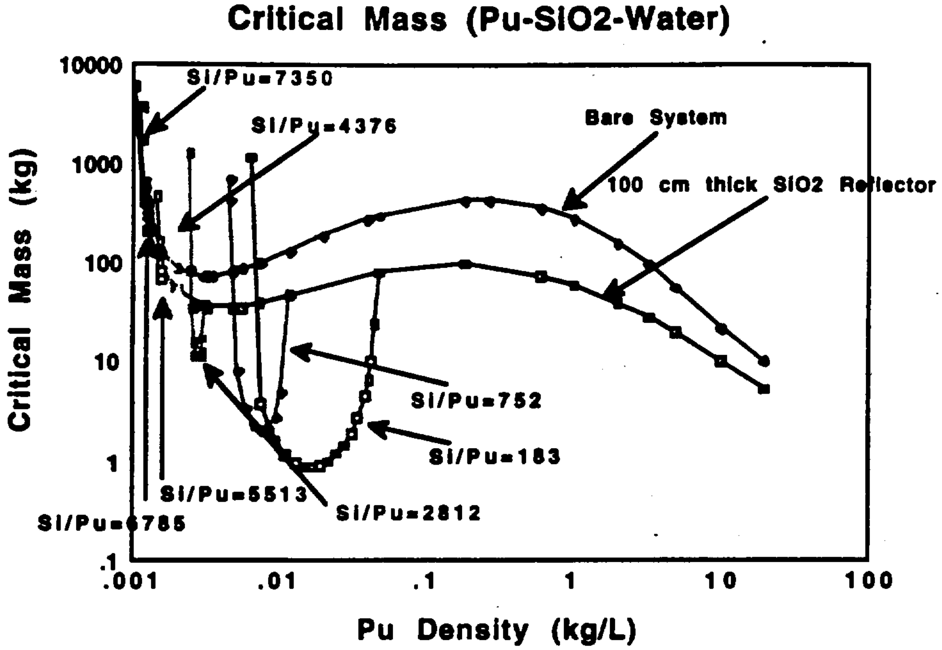


Figure 5.4. Critical mass of U(93.2)C spheres.



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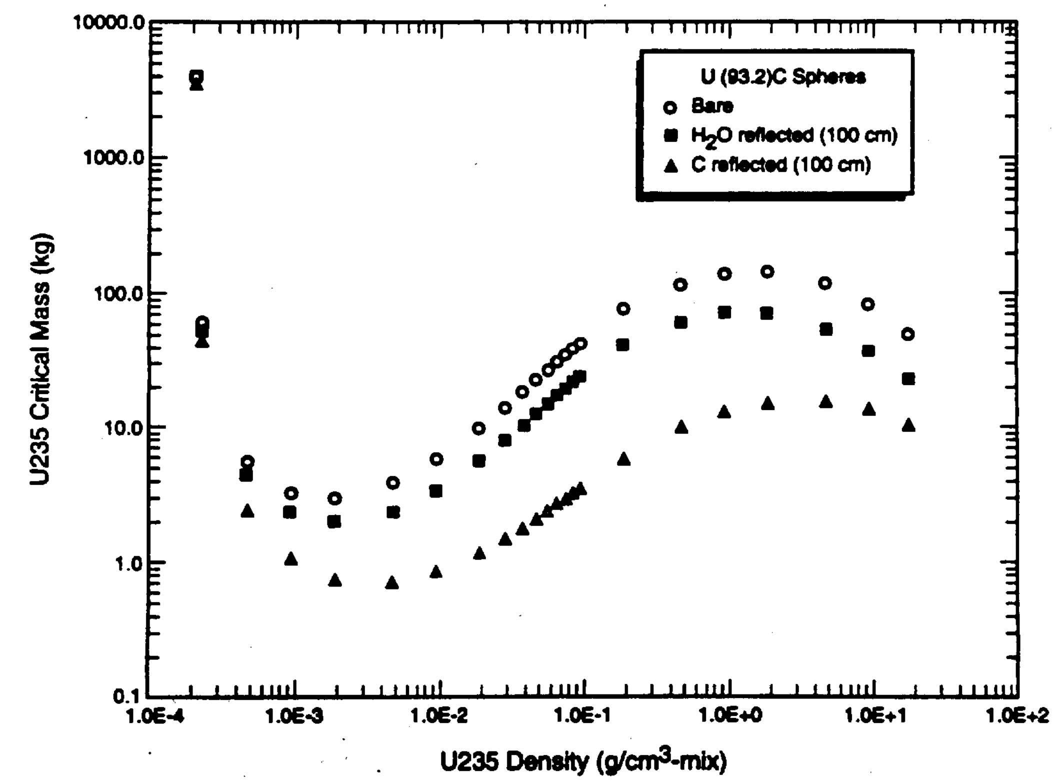

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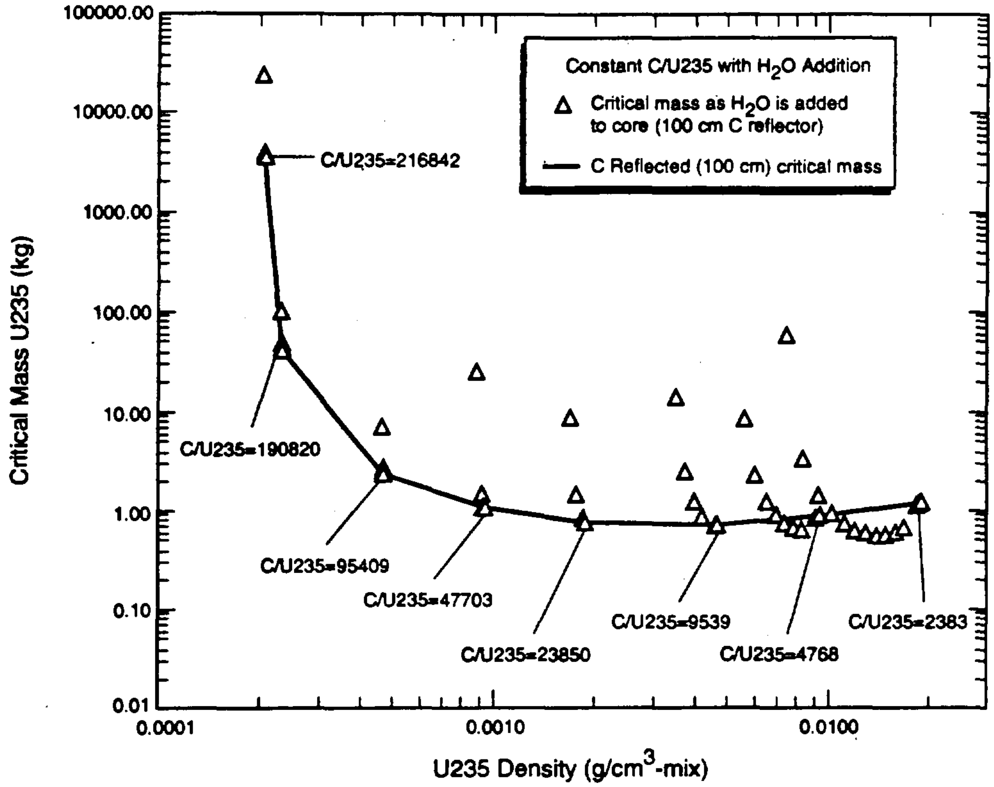


Figure 5.5. Critical mass of graphite-reflected U(93.2)C spheres as a function of H_2O addition to the core.

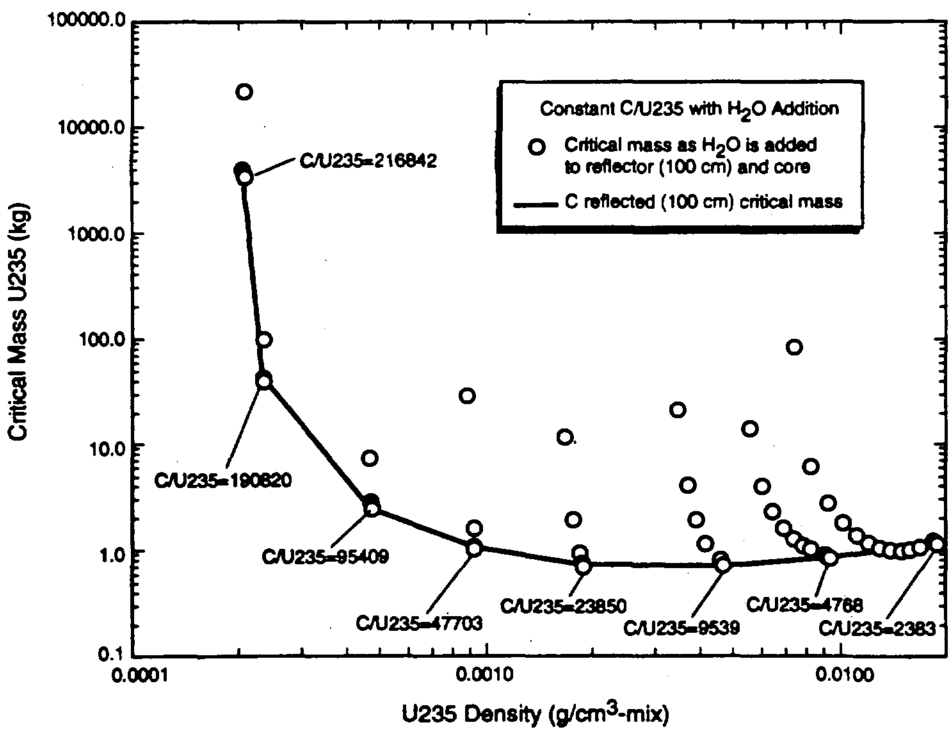


Figure 5.6. Critical mass of graphite-water-reflected U(93.2)C spheres as a function of H_2O addition to the core.

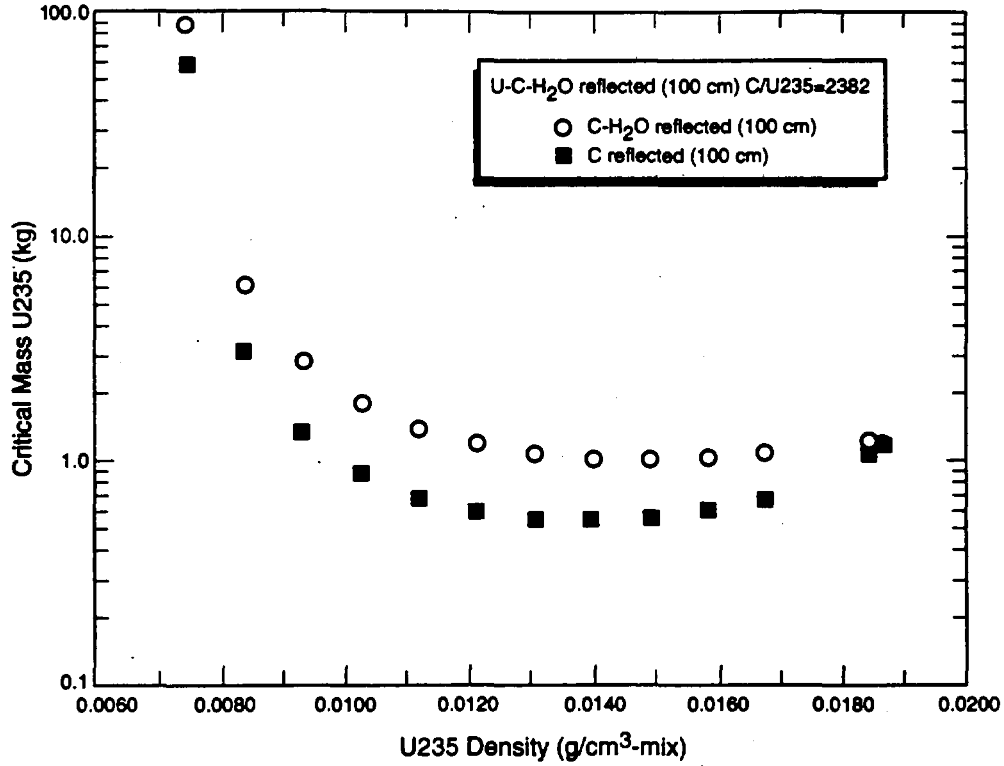
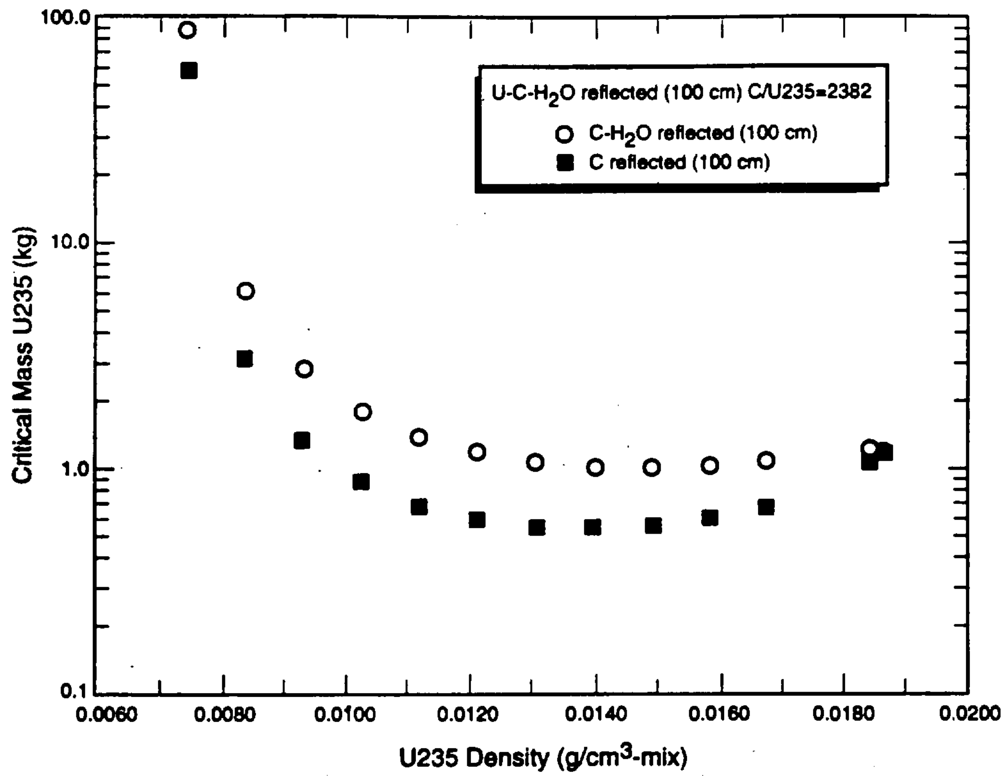


Figure 5.7. Critical mass of graphite-reflected and graphite-water-reflected U(93.2)C spheres as a function of H_2O addition to the core.



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6.0 DOSIMETRY

6.1. DOSIMETRY AND CRITICALITY ALARM TESTING ON SHEBA

K. Butterfield and W. Casson

Dosimetry measurements are an ongoing effort in collaboration with William Casson of ESH-4. We have determined a calibration factor between the integrated current from the SHEBA ^{3}He ion chamber and the neutron dose rate at 3 m. We have also calibrated a set of various-sized Bonner ball neutron counters that give a measure of the SHEBA neutron energy leakage spectrum. These calibrations are now being used to measure the total dose delivered to the standard Los Alamos and Rocky Flats personnel accident dosimeters. These measurements are made using various sized "free run" experiments that reproduce the transient power curve that would be seen in a solution accident. The free runs are also producing data that is used to model the kinetics of solution accidents as described in this report by Robert Kimpland. SHEBA will be used in the upcoming inter-laboratory dosimetry comparison workshop involving a number of the DOE nuclear laboratories.

SHEBA is also being used to test the criticality alarms used in the Los Alamos Plutonium Facility and by the Y-12 plant in Oak Ridge. These laboratories use different alarm systems and have different requirements that need to be tested. The Los Alamos Plutonium Facility is currently concerned that their alarm system will not latch up at high dose rates. The Oak Ridge facility is more concerned with verifying that their alarms detect the minimum accident of concern and with determining the radius of coverage for this detector because they have many separate buildings in a relatively small area. To further these measurements Oak Ridge has supplied building block material that is typically found in their buildings to build enclosures around their detectors. In November, SHEBA was used to determine the radius of coverage using three separate alarm systems and establish the integrated ion chamber current that corresponds to the ANSI standard for the minimum accident of concern. We plan to verify the proper operation of all 40 Oak Ridge criticality alarms in the future.

6.2. RADIATION ACCIDENT DOSIMETRY (RADS) WORKSHOP

Criticality accident dosimetry is an important part of the radiation protection program at DOE facilities that have potential for personnel exposures in the event of a criticality accident. The typical dosimetry system is not able to respond, nor to provide sufficient information for the high neutron doses associated with such events. The specialized systems used for accident dosimetry usually depend upon activation foils and high-range TLD chips. The activation foils, used for deriving the neutron component of the dose, have response curves that poorly relate to the desired neutron to personnel dose effect. As such, the derivation of personnel doses, often referred to as unfolding, is a complex process which requires both a good understanding of the physics involved and experience in performing the calculations from real exposure data. It is also necessary to be able to perform real hands-on training to be able to maintain the technical expertise for proper use of criticality dosimeters. This training has been unavailable since the shutdown of the HPRR facility at Oak Ridge in 1987.

To maintain the necessary capability to analyze dosimeters after an accident at Los Alamos National Laboratory, the Environmental, Safety, and Health (ESH) Division requested that the Godiva and SHEBA assemblies be made available to perform exposures of accident dosimeters in well-characterized fields under simulated accident conditions. Concurrently, the Rocky Flats Plant requested that the LACEF facility and ESH-4 test the accident dosimeter in use there with as many simulated accident fields as possible. The neutron fields at distances of 3 and 6 m from SHEBA were measured using a Bonner sphere system, TLD-based dosimeters, and activation foils. The results have been used to analyze the data from several tests of criticality dosimeters, both from Rocky Flats and Los Alamos, and for testing the reproducibility of the reference measurements systems in normalizing the neutron fields during individual SHEBA runs.

The same type of neutron field characterization is to be carried out in the near future on Godiva IV. Those results plus the SHEBA results will be used in an upcoming DOE-wide nuclear accident dosimeter intercomparison study, planned for June 1995 and funded by EH.

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7.0 TRAINING

7.1. TRAINING ACTIVITIES

Nuclear Criticality Safety Classes (R. E. Anderson, J. A. Bounds, K. B. Butterfield, C. C. Cappiello, T. P. McLaughlin (ESH-6), R. R. Paternoster, R. G. Sanchez, and S. Vessard (ESH-6))

The Nuclear Criticality Safety Class has been offered slightly more often than once a month since the facility restart in June 1991. Listings of the personnel who have attended the training during the period from August 1994 through March 1995 are presented in Table 7.1. This training is intended primarily for nuclear materials handlers and supervisory personnel in the DOE complex, with occasional participation by persons from outside the DOE. To maintain a high level of instructional quality, attendance is limited to an enrollment of approximately 15 persons per class.

During the class, the students engage in hands-on manipulation of nuclear material and build a stack of uranium foils and Lucite plates to achieve a multiplication of approximately 4. The students then continue to add to the stack, which is assembled by remote control, until a multiplication of approximately 125 is achieved. Finally, the students observe a critical assembly operation (currently this is done with the Flattop assembly) and are allowed to operate the assembly under direct supervision of LACEF personnel.

This training is a highly effective demonstration of the principles used to determine the safe handling procedures for nuclear materials in real world situations.

Table 7.1. Attendance at Nuclear Criticality Safety Courses August 1994 — March 1995.

	Dates	Laboratory	Non-Lab
1.	Aug. 1-3, 1994	—	24 Com Rad
2.	Aug. 9-11, 1994	1	8
3.	Sept. 13-15, 1994	2	15
4.	Nov. 1-3, 1994	—	15
5.	Feb. 7-9, 1995	2	12
	TOTALS	5	74

8.0 ENGINEERING AND CONSTRUCTION

8.1. HIGH-SECURITY VAULTS

Two new security vaults were completed during the first half of FY95. The first of these vaults is designed for high-security, long-term storage of material contained in shipping containers and storage containers. The vault, designed and fabricated by Sandia National Laboratories, Albuquerque, was constructed of heavily reinforced, massive concrete "blocks." The door of the vault is also constructed of interlocked, highly reinforced, massive concrete blocks. Each block and its associated attachment hardware had to be custom fit on-site following delivery, requiring much more time to construct than originally anticipated.

A second vault was constructed to house the Godiva critical assembly machine and to allow Godiva to be put into the vault, remotely following operations. This required development of mechanisms that would disconnect all of the control and power cables, pull Godiva into the vault, close the door, lower the locking plate, and spin the lock, all accomplished from the Control Room located approximately 1/4 mile away.

8.2. REACTIVITY INSERTION OF AN ACCIDENTAL GODIVA-IV ASSEMBLY FALL

Robert Kimpland

The Godiva IV critical assembly has been modified so that it can be transported with ease from its storage vault to a desired operating position out on the Kiva 3 floor. The assembly, which is mounted atop a cart, runs along a track set into the Kiva 3 floor. This work examines the reactivity insertion caused by the assembly accidently falling over onto the concrete floor during transport.

A set of MCNP calculations, which attempt to model the godiva assembly both in its normal upright position and in a fallen over position on the floor were made. The MCNP model includes the godiva core, top of the godiva stand, and 8 sq. ft of concrete floor (8 in. thick). The core portion of the model includes the glory hole, safety block, control rod cavities in the fuel rings, and the 3 steel clamps.

The 99% keff confidence interval for godiva in the normal upright position was 0.98559 to 0.98776 (approximately \$2.11 subcritical). In the fallen down position the central axis of the core is approximately 10 inches from the concrete floor, this distance being the closest approach from the center to the edge of the godiva stand. The 99% keff confidence interval for this position was 0.98987 to 0.99210 (approximately \$1.42 subcritical). The additional reactivity due to reflection from the concrete floor is \$0.69~0.35. However, these calculations assume that the safety block is fully inserted. Without the safety block in place the core is far subcritical (approximately \$22.00 subcritical) and the effect of the concrete floor is negligible.

8.3. SIGNAL CABLE PLANT UPGRADE, KIVA 1

The data-acquisition-cable plant to Kiva 1 and the SHEBA building was upgraded to provide better signal quality for experimenters.

Assessment Form 2

Date:

Assessment Form 2 No.: C-COO-2/DUO-3 page 2
Review Area: Material Conditions/Housekeeping Inspection in 9204-2E
Responsible Individual: J. Angelo

III. Approval Section (Signatures)

Originator J. W. Angelo DBBoyd For Date 11/9/95

Approved DBBoyd Date 11/9/95

Suggested Corrective Action:

- Improve effectiveness of training and management oversight of activities, panels are properly secured after use, and material deficiencies are reported and corrected.
- Fix the broken fasteners.

IV. Contractor/DOE Response
(Provide results of Contractor/DOE review with technical basis and references.)

N/A

Accepted By: ______________________ Date ____________

8.4. ADDITION OF A HEPA FILTER SYSTEM TO SKUA

A HEPA filter system for the Skua critical assembly machine was designed and fabricated. This system will capture fission fragments generated during burst operation of the machine and result in a lower contamination level of the kiva.

8.5. SHEBA MODIFICATIONS

Several modifications were made to SHEBA during the report period to enhance the operation and to correct deficiencies noted during operation. These modifications included replacing a needle valve in the cover gas outlet line with a fixed orifice and removing a failed pressure gauge from the pump bypass line. In addition, an experiment thimble was designed and fabricated to allow insertion of samples into the Sheba tank for irradiation without physical contact with the SHEBA fuel.

8.6. SUBCRITICAL MEASUREMENTS USING THE SOURCE JERK TECHNIQUE

Hardware was completed for use in investigating the source jerk technique for use in subcritical measurements in fissile material arrays. A second generation of hardware is now in the design phase that will enhance the measurements.

8.7. SITEWIDE DOSIMETRY SYSTEM

We have been in frequent contact with the vendor of our sitewide dosimetry system. The system is due to ship in April and will consist of 12 weatherproof neutron and gamma dose-rate stations. A station will be located in each kiva and in or near the SHEBA building as well as at each railroad gate leading to the kivas. Other stations will be in the main office building on site, at the office building offsite and closest to Kiva 2, and in the parking lot at the point of closest approach of the main road. The dose rate ranges of the system will exceed what is presently in place in the kivas and on the roof of the main office building. Data will be transmitted from the stations to the control rooms. A computer program is being written that will read, archive, and display all of the dose rates on a single screen.

8.8. TRAINING AIDS

Two types of training aids for use in the DOE Office of Professional and Technical Training and Development were completed during the report period.

The first type of training aid that was completed included two displays of cutaways for motors, fuses, and batteries, for example, for the Electrical Principles course. The second type of training aid was two complete closed-loop pumping systems, each consisting of two pumps, a variety of valves, a water tank, and instrumentation and controls.

A proposal for development of a Criticality Simulator training aid has been submitted for funding.

9.0 PROGRAM DEVELOPMENT

9.1. MEDICAL ISOTOPE PRODUCTION REACTOR (MIPR)

LACEF has been working with Babcock and Wilcox on the Medical Isotope Production Reactor (MIPR) concept. The MIPR concept uses a homogeneous-solution fuel with a recirculation loop to continuously extract Mo-99 and other fission product medical isotopes. We have partnered with CST-12 (Moses Attrep and Bob Rundberg) to assist with the extraction process chemistry.

1. Draining of the WINCO Slab Tanks

The WINCO Slab Tanks were used for criticality experiments using a fuel of 93% enriched uranyl nitrate. Each tank (2 ea.) held ~ 36 L of fuel. The WINCO experiments were completed in 1991. Plans now call for using this fuel to investigate the possibility of producing medical isotopes with a solution reactor.

In preparation for these experiments, the Slab Tanks were drained and the uranyl nitrate was returned to the 10-L storage bottles. The draining process involved fixturing to hold the slab tanks during draining and the entire draining process required the use of chemical suits and respirators.

2. Criticality Analysis of the SHEBA Fuel Conversion
(R. Kimpland)

The present SHEBA fuel will be replaced with a 20% enriched uranyl nitrate fuel in an effort to study the MIPR concept. A series of transport calculations using TWODANT and MCNP have been performed to determine the composition of the new fuel and its characteristics. In addition, a set of calculations was performed to determine the criticality safety of the new fuel while being stored in the SHEBA fuel tanks.

Using TWODANT, a concentration search was performed to determine the composition of uranyl nitrate fuel. The fuel was assumed to be $UO_2(NO_3)_2+H_2O$, with an H_2O density of 1 g/cm^3. Figure 9.1 shows the results of this search. With a fuel loading of 0.15 gU/cm^3, the core would have a critical height of approximately 47 cm, which produces the desired geometry of a right circular cylinder. MCNP predicts a k_{eff} of 0.99924±0.0007 for this fuel loading at 47 cm. Therefore, the 0.15 gU/cm^3 fuel loading has been chosen for the new SHEBA fuel.

A series of TWODANT calculations were performed to determine the reactivity characteristics of the core. Figure 9.2 shows the reactivity of the core in dollars vs the height of the core. Around delayed critical the core has a reactivity worth of $0.42/cm of fuel. TWODANT was also used to determine the reactivity worth of the borated epoxy safety rod. Fully inserted, the rod is worth $-3.85. The β_{eff} for the above calculations was taken to be 0.007.

A series of MCNP calculations were performed to determine the criticality safety of the new fuel in the SHEBA fuel tanks. The MCNP model included the four SHEBA fuel tanks centered inside a hollow sphere of concrete. The inner diameter of the concrete shell is 6 ft and the outer diameter is 10 ft. This model represents an overly conservative simulation of the fuel tanks in the SHEBA pit. The model also assumes that each tank is completely filled with fuel. The predicted k_{eff} of this system was 0.77100±0.00042. A second calculation was also performed in which the void surrounding the fuel tanks was replaced with water, simulating the pit being completely flooded. The predicted k_{eff} of this model

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It is hoped that the actual experiments with the uranyl nitrate fuel will validate these transport calculations. It will be of interest to determine whether the methods used to calculate the reactivity characteristics of the core are adequate for such fissile solutions.

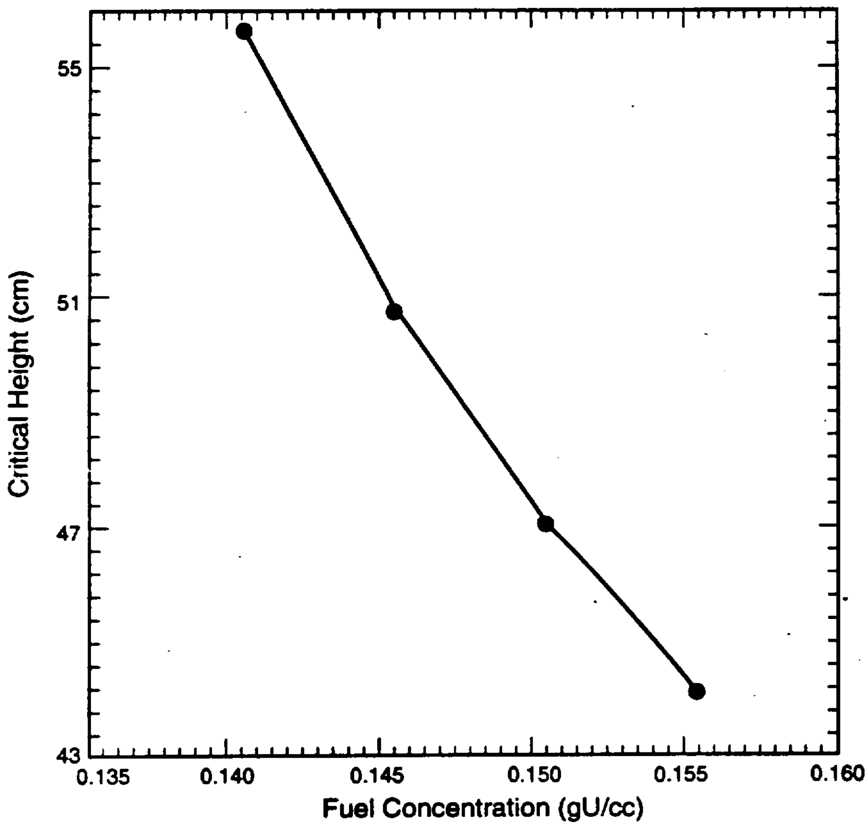


Figure 9.1. Critical height vs fuel concentration for 20% uranyl nitrate in SHEBA.

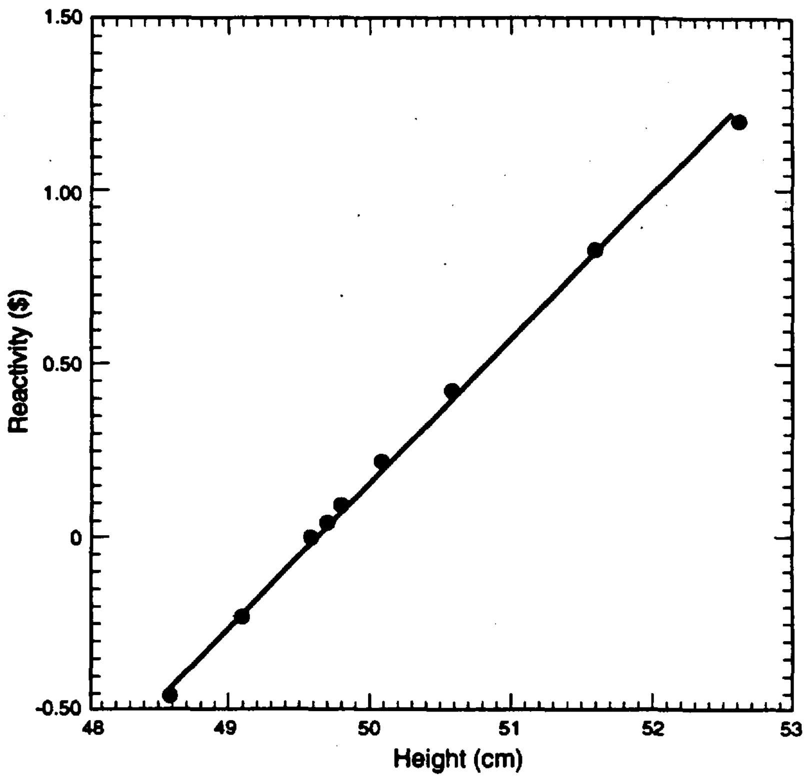


Figure 9.2. Reactivity vs height for 20% uranyl nitrate fuel (0.15 gU/cm³) in SHEBA.

9.2. CRITICALITY TRAINING SIMULATOR FUNDING PROPOSAL

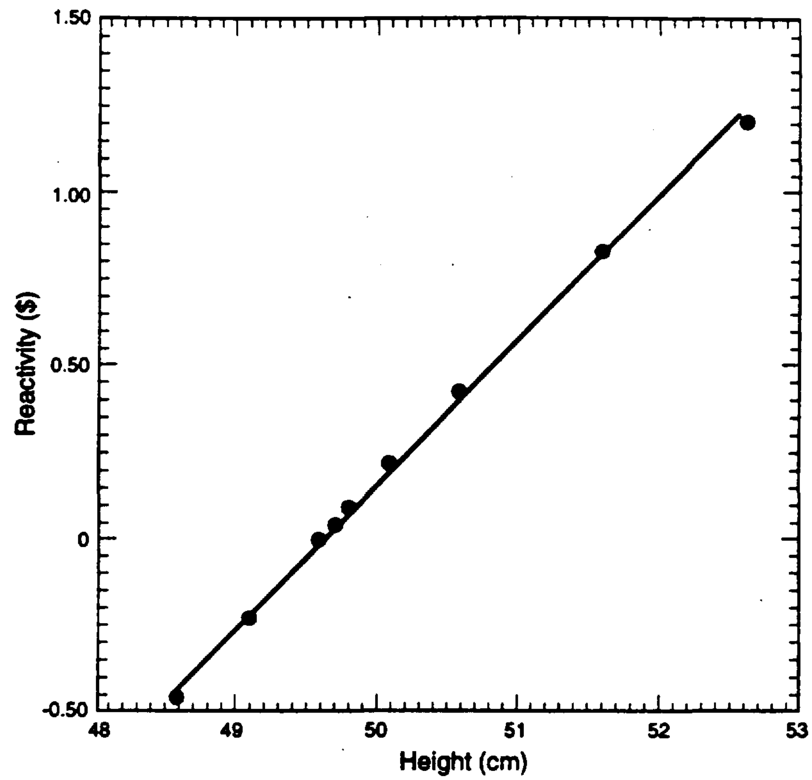
A proposal was written with Al Criscuolo to solicit funding for a training simulator that would provide realistic, hands-on experience for fissile material handlers and associated personnel (supervisors, radiological control technicians, and craft support personnel, for example). The proposed portable simulator would provide an excellent addition to current training methods by presenting several different criticality scenarios that could be quickly selected by an instructor in a class room setting.

9.3. HONEYCOMB DESIGN REQUIREMENTS DOCUMENT

A draft Design Requirements document was written for a proposed array experiment using the Honeycomb assembly. The proposed experiment involves forming a supercritical array of poly-moderated barrels using a horizontal split table. This document outlines the preliminary safety, operating, control system, and mechanical/electrical requirements for performing the experiment.

9.4. KIVA I DIGITAL CONTROL SYSTEM UPGRADE

The ladder logic coding and system design for upgrading Kiva I to digital control (via CR I hardware) was begun during this reporting period. Shown on the following page is a schematic of the anticipated control system upgrade for Kiva I. As seen in the Fig. 9.3, the Kiva I upgrade will involve expansion of the SHEBA control system rather than completion of an entirely new Kiva I control system. The ladder logic code under development is for control of the Honeycomb horizontal split table within the software structure established by SHEBA.



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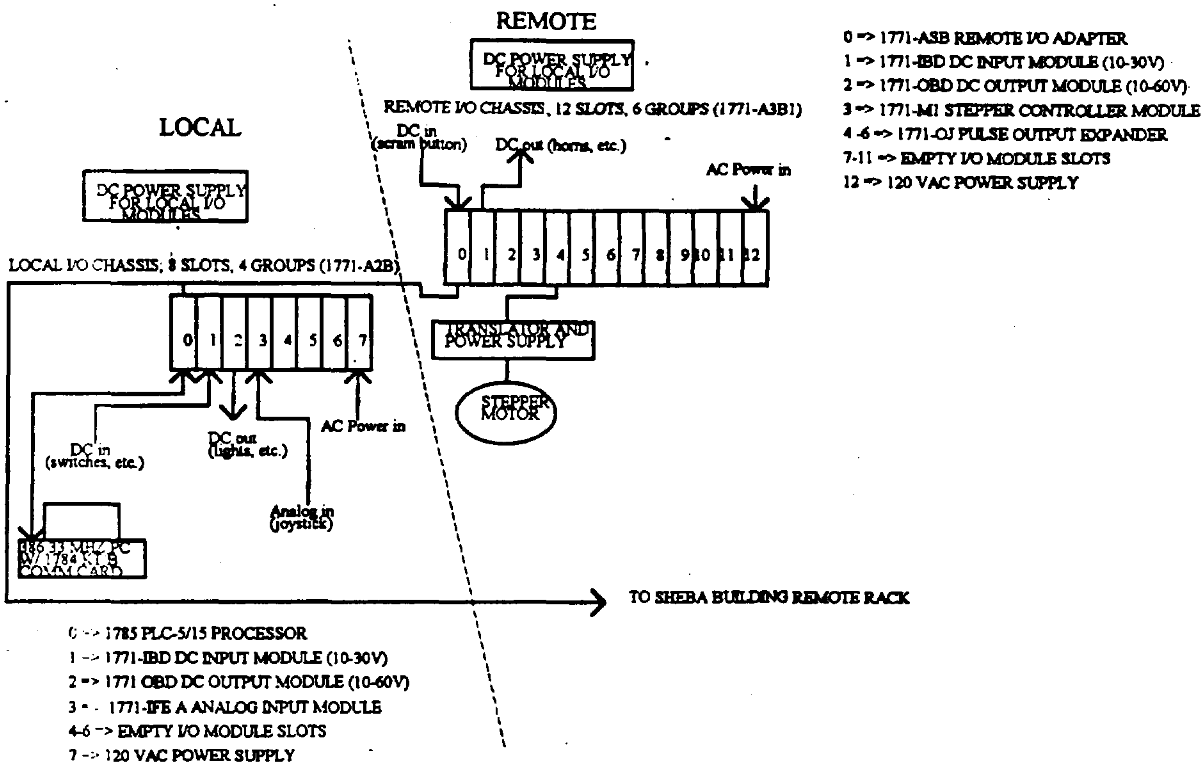


Figure 9.3 Anticipated Kiva I control system.

9.5 ULTRASONIC TAMPER-INDICATING DEVICE (TID) MEASUREMENTS

W. Myers, S. Rojas

Introduction

A technique proposed by NIS-7 personnel involves analyzing the frequency response of a barrel to determine whether it has been opened or moved (an ultrasonic TID). Because of its unique experimental facilities, Pajarito site was chosen as the location for field experiments to investigate the feasibility of such a concept. The tests were carried out at TA-18 in Kiva III using 55 and 65 gallon drums containing SNM. The measurement technique involves placing two magnetic transducers on the barrel at right angles to one another. The transducer on the top emits a 20-30 KHz sinusoidal frequency while the transducer on the side measures the response and phase shift of the barrel. In this way, a "template" is made which may be used as a baseline to compare all subsequent measurements against. An IBM computer with an A/D card and custom software was used for data acquisition and the temperature inside the room was logged throughout the duration of the tests. Figure 9.4 shows the basic setup for the experiments performed.

Results

A total of 29 data sets were taken using 22 barrels, some of which were opened, moved, or left fixed. When the barrels were opened or moved, a marked change in the baseline frequency response occurred. Figure 9.5 shows strong agreement between two response plots of a barrel that was not moved or opened. Figure 9.6, on the other hand, indicates a definite change in the baseline frequency response of one barrel that was moved from its original location and opened. As evidenced by Figs. 9.5 and 9.6, the goal of using ultrasonic interrogation as a TID appears feasible.

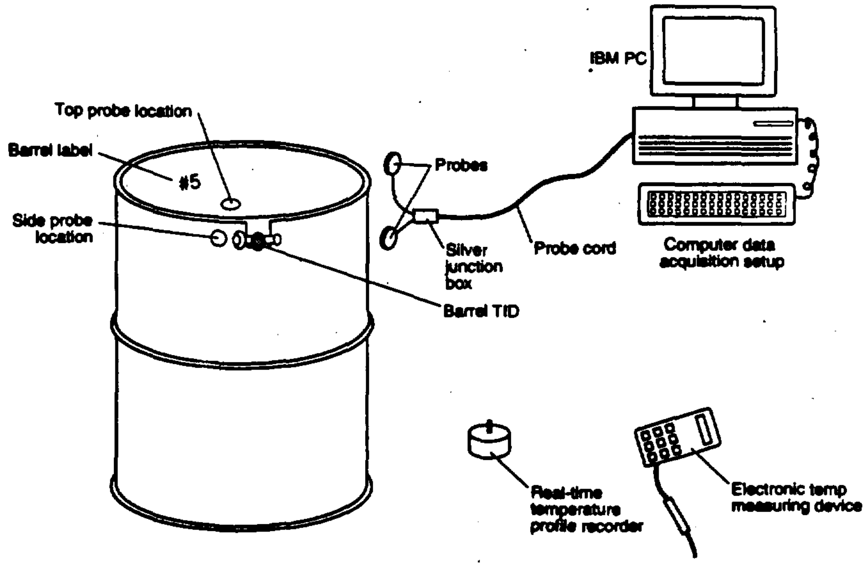


Figure 9.4. Basic set-up for ultrasonic TID measurements.

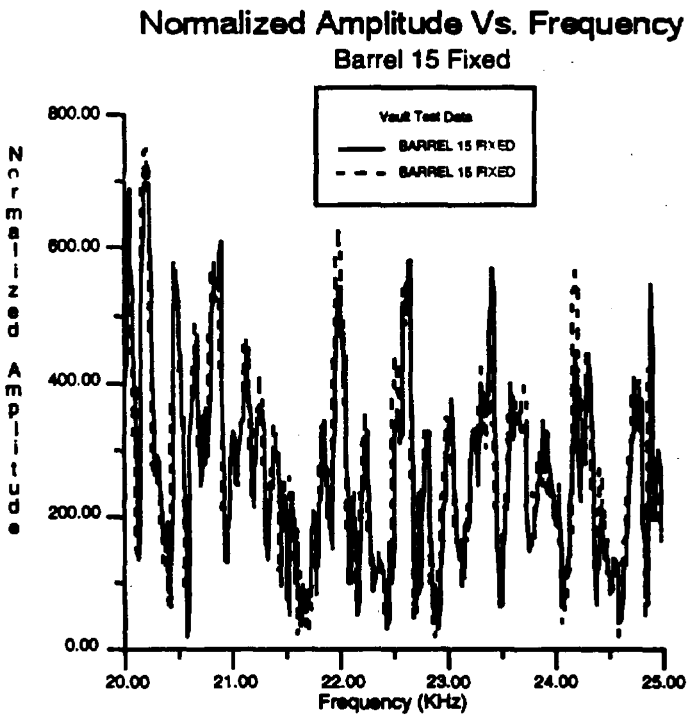
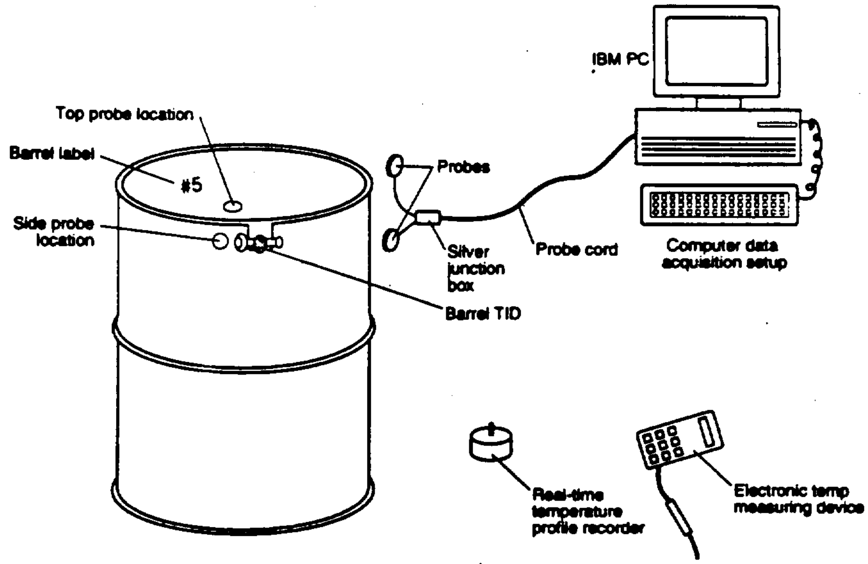


Figure 9.5. Response comparison for Barrel 15 fixed.



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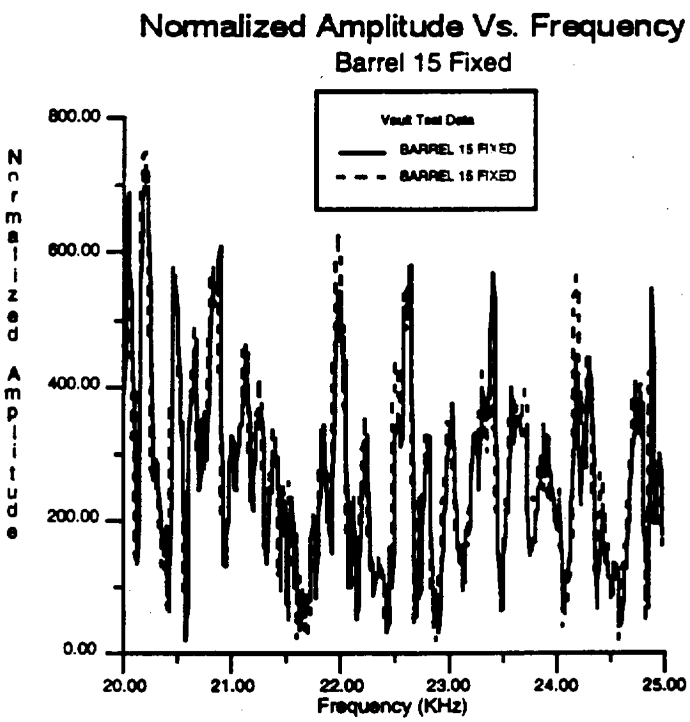

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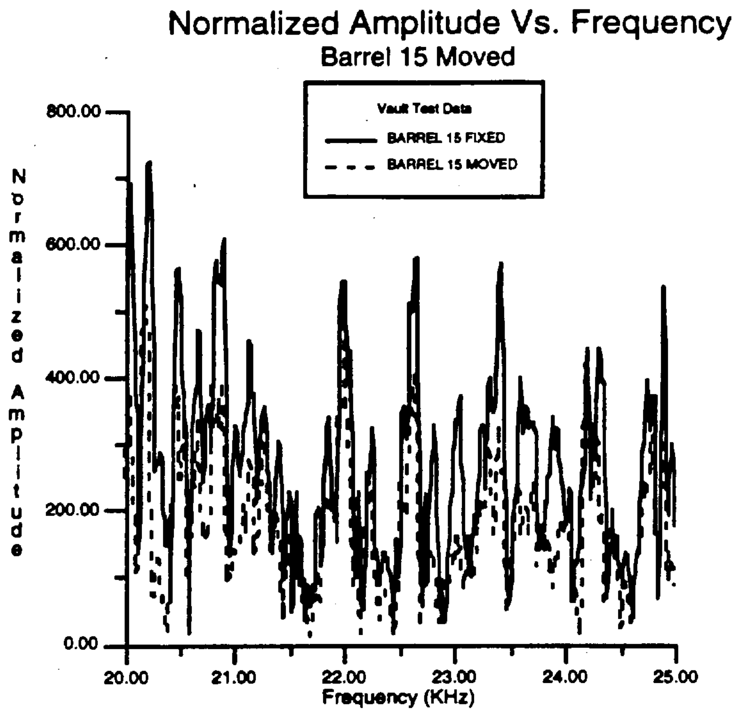

Figure 9.6. Response comparison for Barrel 15 moved.

Future Research

If a loudspeaker could be used to remotely excite the barrels and the frequency response information could be obtained from remotely mounted lasers, this method could provide a valuable non-intrusive, cost-effective material control and accountability tool. Future research might focus on the feasibility of loudspeaker-excited systems and the availability of laser-based accelerometers. Such a system might prove invaluable as a portable verification tool in IAEA-type inspections.

10.0 DOCUMENTATION

10.1. DOCUMENTATION

1. Completion of Draft TSRs

In accordance with DOE Order 5480.22, a set of draft Technical Safety Requirements was submitted on February 25, 1995. The document was issued as a Los Alamos controlled publication: LA-CP-95-11. When reviewed and approved by the DOE, these will replace the current Technical Specifications contained in LA-6016-SOP, Rev. 2 (October 1987). Prior to submittal, the document was iteratively reviewed by NIS-6 members of the LACEF Nuclear Safety Committee (primarily R. Anderson, E. Mullen, K. Butterfield, and C. Cappiello), by members of the Los Alamos Reactor Safety Committee (primarily J. Graf, J. Schlapper, S. Bowdenstein, and T. Schmitt) , and by several members of ESH-3 (primarily M. Bowidowich, C. Nelson, and J. Bueck). The new TSRs provide a broad envelope for operation of the LACEF and the Hillside Vault consistent with the new SAR. As with most things the new TSRs bring new capabilities and new requirements. Some of these are as follows:

- Specifications for entering radiological control areas surrounding the kivas during critical operations
- Specifications to operate assemblies outside
- Reduced requirements for assembly operations
- Additional requirements for surveillance procedures

2. Maintenance Implementation Plan (MIP) Review and Revision

DOE ALOO reviewed the LACEF MIP and identified vulnerabilities in the maintenance program. The MIP was updated to cover these vulnerabilities and to describe current practices and procedures that have developed since the MIP was submitted to DOE for review. A comprehensive maintenance categorization of systems at TA-18 was also completed and work is now underway on a Master Equipment List.

3. Completed Revision of all Maintenance Procedures for Assembly Machines

All of the preventive maintenance procedures for the LACEF Critical Assembly Machines have been updated to reflect Los Alamos National Laboratory standards for procedures.

4. Design and Drafting Control Procedure and V&V Procedure Updates to Include Unreviewed Safety Question Determination (USQD) Procedures

The "LACEF Procedure for the Validation and Approval of New and Modified Critical Assembly Machines" and the "Advanced Nuclear Technology Design and Drafting Control Procedure" were revised to include the LANL Unreviewed Safety Question Determination procedures. In addition, the graded approaches in both documents were revised to present one graded approach common to both procedures.

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11.0 MEETINGS AND CONFERENCES

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Burst Reactor Embedded Topical Meeting
R. Paternoster

An international embedded topical meeting on Physics, Safety and Applications of Pulse Reactors was organized in conjunction with the Winter ANS Meeting from 13-17 November 1994, in Washington, D.C. Sixty papers were presented with 24 from the US, 21 from Russia, 4 from PRC, 3 from Japan, 2 from France, 2 from Kazakhstan, and 1 from Texas A&M.

1995 Nuclear Criticality Technology & Safety Project (NCTSP) Workshop
R. Sanchez

During the past six months, we have continued the planning for the 1995 Nuclear Criticality Technology & Safety Project (NCTSP) workshop. We have selected the city of San Diego, CA, as the hosting city for this workshop and the Catamaran hotel as the hotel where the workshop will be held. More information about this meeting and the final meeting agenda will be issued soon.

ICNC'95 International Conference on Nuclear Criticality Safety
R. Paternoster

The fifth ICNC is scheduled for 17-22 September 1995 in Albuquerque, NM. The conference chairperson is Norm Pruvost, and Rick Paternoster has been asked to act as deputy chairperson. There are 140 technical papers accepted for the meeting with 80 oral presentations and 2 poster sessions with 30 posters in each. On Friday a tour of either Sandia or Los Alamos is being offered. As part of the Los Alamos tour we will try to show Control Room 1, SHEBA, and the assemblies in Kiva 2.

12.0 PUBLICATIONS, REPORTS, MEMOS

12.1. PUBLICATIONS, REPORTS, MEMOS

Paternoster, Anderson, and Mullen, "Technical Safety Requirements for the Los Alamos Critical Experiments Facility and the Hillside Vault (PL-26)," Draft submitted to the DOE, LA-CP-95-11,

Paternoster, Bounds, Sanchez, and Miko, "Physical Characterization of the Skua Fast Burst Assembly."

Paternoster, Flanders, and Kazi, "Accident Analysis for U.S. Fast Burst Reactors."

Paternoster, Kimpland, Jaegers, and McGhee, "Coupled Hydro-Neutronic Calculations for Fast Burst Reactor Accidents."

Paternoster, Hetrick, and Cappiello, "Maximum Consequence Accident Analysis for the Los Alamos Critical Experiments Facility (LACEF) and TA-18 (Pajarito Laboratory)," Draft Los Alamos report, LA-12717-MS.

Presented status of Np_{93}^{237} critical mass experiments and their experimental results to the Nuclear Criticality Experiments Safety Committee (NCESC) on February 23, 1995. Participated in two criticality safety classes.

A thesis paper entitled "Conceptual Design of a Digital Control System for Nuclear Criticality Experiments" was presented at the summer ANS meeting in New Orleans. The paper presented the digital control system, Monte-Carlo calculations, and preliminary mechanical design concepts for a proposed critical experiment involving two "slab" tanks filled with uranyl nitrate.

11.0 MEETINGS AND CONFERENCES

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Burst Reactor Embedded Topical Meeting
R. Paternoster

An international embedded topical meeting on Physics, Safety and Applications of Pulse Reactors was organized in conjunction with the Winter ANS Meeting from 13-17 November 1994, in Washington, D.C. Sixty papers were presented with 24 from the US, 21 from Russia, 4 from PRC, 3 from Japan, 2 from France, 2 from Kazakhstan, and 1 from Texas A&M.

1995 Nuclear Criticality Technology & Safety Project (NCTSP) Workshop
R. Sanchez

During the past six months, we have continued the planning for the 1995 Nuclear Criticality Technology & Safety Project (NCTSP) workshop. We have selected the city of San Diego, CA, as the hosting city for this workshop and the Catamaran hotel as the hotel where the workshop will be held. More information about this meeting and the final meeting agenda will be issued soon.

ICNC'95 International Conference on Nuclear Criticality Safety
R. Paternoster

The fifth ICNC is scheduled for 17-22 September 1995 in Albuquerque, NM. The conference chairperson is Norm Pruvost, and Rick Paternoster has been asked to act as deputy chairperson. There are 140 technical papers accepted for the meeting with 80 oral presentations and 2 poster sessions with 30 posters in each. On Friday a tour of either Sandia or Los Alamos is being offered. As part of the Los Alamos tour we will try to show Control Room 1, SHEBA, and the assemblies in Kiva 2.

COMMITTEE MEMBERSHIP

NUCLEAR CRITICALITY STEERING COMMITTEE (ALL DOE FEDS)

	PHONE NO.	FAX NO.	ORGAN.
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Max Clausen	6-8217	7705	FM
Al Evans	3-4896	9513	ER-13
Ivon Fergus	3-6364	4672	EH-34
George Kachadorian	202-275-6445	7710	HR-33
Burton Rothleder	3-3726	1182	EH-31
Bob Walston (nonvoting)	505-845-5694	6431	DOE/AL

METHODOLOGY AND EXPERIMENTS SUBCOMMITTEE (FEDS AND CONTRACTORS)

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TRAINING SUBCOMMITTEE (FEDS AND CONTRACTORS)

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Tom Mclaughlin	505-667-7628	5-4970	LANL

ACRONYMS

ANL	Argonne National Laboratory
CSBEP	Criticality Safety Benchmark Evaluation Project
CSEWG	Cross Section Evaluation Working Group
DNFSB	Defense Nuclear Facilities Safety Board
ENDF	Evaluated Nuclear Data File
ICSBEP	International Criticality Safety Benchmark Evaluation Project
INEL	Idaho National Engineering Laboratory
LACEF	Los Alamos Critical Experiments Facility
LANL	Los Alamos National Laboratory
NCESC	Nuclear Criticality Experiments Steering Committee
NCTSP	Nuclear Criticality Technology and Safety Project
OECD-NEA	Organization for Economic Cooperation and Development - Nuclear Energy Agency
ORELA	Oak Ridge Electron Linear Accelerator
ORNL	Oak Ridge National Laboratory
SHEBA	Solution High-Energy Burst Assembly
SNL	Sandia National Laboratories
USNRC	United States Nuclear Regulatory Commission