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SAFEGUARDS PAPERS

from

ANS/AIF WINTER MEETING

NOVEMBER 1968

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Office of Safeguards and Materials Management

UNITED STATES ATOMIC ENERGY COMMISSION

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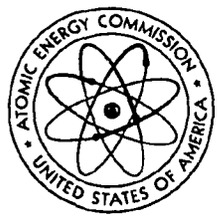

April 15,1970

Office of Safeguards and Materials Management
UNITED STATES ATOMIC ENERGY COMMISSION

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ABSTRACT

This document assembles under one cover all papers devoted to nuclear safeguards which were presented at the 1968 Winter Meeting of the American Nuclear Society. There are nine papers covering International Safeguards which were presented at the joint ANS/AIF plenary session. Eight technical papers were presented at the ANS session on Fissionable Materials Safeguards Technology.

INTRODUCTION

In order to organize and desseminate information regarding nuclear safeguards, it is the policy of the Office of Safeguards and Materials Management to publish pertinent papers in the Safeguards field. Many such papers were presented at the 1968 Winter Meeting of the American Nuclear Society which was held in Washington, D.C., November 10-15,1968. The papers were presented in two separate sessions:

International Safeguards - Joint ANS/AIF Plenary Session
November 12,1968

Fissionable Materials - ANS Technical Session
Safeguards Technology November 13,1968

As indicated by the following Table of Contents, the papers from the plenary session, International Safeguards, were published by the American Nuclear Society in the "Proceedings of the Plenary Sessions." Summaries of the papers presented at the technical session, Fissionable Materials Safeguards Technology, were published in the ANS "Transactions," Volume II, Number 2, November 1968. The purpose of this document is to assemble under one cover all papers presented at the two sessions. The complete papers from the technical session, in place of the summaries, have been obtained from the authors.

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*Papers published by the American Nuclear Society "The International Conference on the Constructive Uses of Atomic Energy" Proceedings of the Plenary Sessions. Reprinted by the kind permission of The American Nuclear Society.

FISSIONABLE MATERIALS SAFEGUARDS TECHNOLOGY*

*Abbreviated papers were published in ANS Transactions, vol. 11, no. 2, November 1968.

PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON
THE CONSTRUCTIVE USES OF ATOMIC ENERGY

INTERNATIONAL SAFEGUARDS

Joint ANS/AIF Session

Chairman
Sigvard A. Eklund (IAEA)

Technical Secretary
G. R. Keepin (LASL)

Tuesday PM
November 12, 1968

INTERNATIONAL SAFEGUARDS
ANS/AIF SPECIAL SESSION

Introductory Remarks

G. R. KEEPIN *(USA)*

I'd like to welcome all of you to this Joint Session of the American Nuclear Society and the Atomic Industrial Forum on the subject of International Safeguards. As it turns out, our Chairman for this Session also happens to be our first speaker, so I have been asked to introduce him. It is my great pleasure to do so, although such a prominent world figure in the area of international safeguards scarcely needs any introduction to this audience. I am referring, of course, to Dr. Sigvard Eklund, the Director General of the International Atomic Energy Agency in Vienna. Dr. Eklund.

S. EKLUND *(IAEA)*

Thank you, Dr. Keepin. I first have some introductory remarks to this Special Session on International Safeguards. Following these, I shall proceed directly to my presentation of the first paper in this Session, entitled "The IAEA Safeguards System."

Two significant international treaties have been opened for signature during the past one and one-half years; viz., the treaty for the prohibition of nuclear weapons in Latin America and the treaty for the non-proliferation of nuclear weapons. When the latter was commended by the United Nations General Assembly in June this year (1968), it marked the consummation of several years of intensive negotiations in the UN Disarmament Committee.

On several earlier occasions, as for instance in the Atomic Industrial Forum Panel meeting in November last year, I have dwelt upon the need for a non-proliferation treaty. I will now only recall the figures I quoted then: Around 1980, there will, according to the best estimates available, be a worldwide production of at least 70 tons of fissile plutonium per year, 25 tons of which will be produced in present non-nuclear weapon countries. This corresponds to a production of fissile plutonium that would be adequate to manufacture at least some 100 bombs of minimum size per week in those countries. These figures show the urgency for the conclusion of a non-proliferation treaty, and, with deep satisfaction, I greet the progress that has been made so far. It is gratifying that the NPT has already been signed by 82 nations.

Both the Non-Proliferation Treaty and the treaty for the creation of a Latin American nuclear-free zone recognize the necessity of international safeguards and controls to ensure and demonstrate that nuclear materials, designated for peaceful purposes, are not diverted to military uses.

The concept of international safeguards is a fairly new one in international relations. The notion that a State's activities in any field should be subject to supervision by an international organization may understandably be unpalatable, as it does have implications on perhaps the most cherished concept of all; viz., that of national sovereignty. In the future, it is likely there will be a more general need for international regulation and control of technical activities; for instance, through such organizations as the International Telecommunication Union and the International Civil Aviation Organization. I would also point to the obvious international interest to control the traffic in narcotics. The principle of international verification rights has recently been incorporated in the Antarctica Treaty and the Outer Space Treaty. However, in no other area have control provisions of such a complexity been designed and firmly institutionalized as in the nuclear field, and it is natural that some objections of a general nature have been and are still being raised against safeguards. There have been legitimate fears among the operators of nuclear facilities for possible interference from safeguarding authorities. There have been apprehensions about possible industrial espionage and also the fear that safeguards might hamper the general development of nuclear science and technology in a country. I believe these fears are based, to a large extent, on a lack of understanding of what safeguards are and how they are implemented.

It must also be realized that safeguards have existed and have been implemented for more than ten years on an international basis; first, under bilateral arrangements consequential to the assistance agreements between, for instance, the USA

and other countries. Both EURATOM and ENEA have established regional safeguards systems, and the IAEA international safeguards system has been in operation since 1961. It is obvious, however, that from a political point of view we have to make a distinction between regional systems on the one side and truly international systems on the other.

In spite of the experience that has been gained from quite extensive safeguards operations, we still have to regard safeguards as in the very beginning of development, both from a conceptual and methodological point of view. It is very fortunate that it has been possible to devote one session of an important international meeting to international safeguards. This session should help both to inform about safeguards and to define the problems that remain to be solved.

THE IAEA SAFEGUARDS SYSTEM

S. EKLUND, *Director General*
International Atomic Energy Agency
Vienna, Kaerntnerring, Austria

THE IAEA SAFEGUARDS SYSTEM

The International Atomic Energy Agency is enjoined by its Statute of 1957 to establish and administer safeguards to ensure that nuclear materials, facilities, or equipment intended for peaceful use are not diverted to further any military purposes. Restated in more practical terms, the statutory objective of Agency safeguards is to detect at any early time any illegitimate uses, or diversion for unknown uses, of safeguarded nuclear materials. Thereby, Agency safeguards would deter, and indirectly help to prevent the use of nuclear materials for military purposes. This, of course, is a political, qualitative objective to which technical considerations have to be applied in order to arrive at a specification for a safeguards operation. I will not dwell upon the political connotations and problems of IAEA safeguards, nor will I try to compare them with other safeguards systems.

The Agency's Statute also lists in some detail the rights and responsibilities of the IAEA in applying safeguards. On the basis of these fundamental but somewhat sketchy statutory requirements, the Agency's Board of Governors has set up the so-called "Safeguards System" of the Agency. This system, which is established by means of a document containing a set of agreed conditions, rules, and procedures, has been developed through a series of revisions and additions since 1961, so that it now applies to all facilities in a nuclear fuel cycle except uranium enrichment plants. On the basis of the document, the Agency's Department of Safeguards and Inspection is elaborating detailed safeguards procedures for the various types of facilities.

Under the IAEA Statute, the application of Agency safeguards is mandatory if a State obtains assistance for a project from or through the Agency. Otherwise, safeguards are only applied *upon request.* Such a request may be for the Agency to undertake the safeguards responsibility under a bilateral international agreement, or a State may ask the Agency to apply safeguards to any or all of its nuclear activities, either unilaterally or under a multilateral treaty. In all cases, the prerequisite for the application of Agency safeguards is the conclusion of an international agreement between the Government of a country and the Agency.

The first version of the Agency's safeguards system was adopted by the Board in April 1961. It was then applicable only to small research reactors. A revised and expanded system, applicable to all power and research reactors, was adopted in 1965, and in 1966 and 1968 extensions were added for fuel reprocessing and fabrication facilities, respectively. The revisions and extensions have approximately been in step with the requirements arising from the increasing safeguards responsibilities of the Agency, starting first with low-power research reactors, but now including three industrial power plants, one reprocessing plant, and several pilot fuel fabrication plants.

So far, the Agency's Board has approved 40 safeguards agreements with 30 countries, and under these, Agency safeguards are applied to 69 reactors with a total thermal power of 3200 MW(th), and almost as many other locations with nuclear material. The Agency is now safeguarding a great number of individual facilities in different countries, and the safeguards system and detailed procedures have naturally been developed for this type of operation. In most of the countries concerned, the only parts of the fuel cycle under safeguards are transport of cold fuel to a reactor, use in the reactor, storage, and transport of irradiated fuel. Procedures may then have to differ

from those that can be used in a country where a complete or nearly complete fuel cycle is to be safeguarded. Nevertheless, the Agency's system, as conceived for individual facilities rather than a fuel cycle, contains the essential elements of any safeguards system, and its fundamental provisions will undoubtedly retain their validity for some time to come.

The Agency's safeguards system contains the following four basic technical provisions for the implementation of safeguards:

1. The Agency has the right to review the design of "nuclear facilities for the *sole purpose* (and I repeat, for the sole purpose) of satisfying itself that a facility will permit the effective application of safeguards."

2. The Agency requires that each facility under safeguards should maintain a record of its operation, of its inventory of nuclear materials; i.e., the amount and detailed specifications of the materials it handles, the extent of losses if any, etc. I would like to emphasize that an exact and up-to-date record of the quantity and location of nuclear materials together with their correct specifications is the *sine qua non* of any safeguarding system and, of course, also very much in the interest of any facility operator.

3. Routine reports based on the records kept by a country should be rendered to the Agency at periodic intervals. If there is any kind of untoward or unusual incident at the plant, involving actual or potential loss of safeguarded nuclear material, such an incident must be reported immediately to the Agency.

4. The Agency must have the right to send its inspectors into a Member State for the purpose of verifying the reports the State has made. In the present state of the technology, supporting safeguards, any kind of safeguards without the right of on-the-spot verification, i.e., inspection, would be meaningless.

The safeguards system also carefully lays down general principles, specifying among other things, that the Agency shall implement safeguards in a manner designed "to avoid hampering a State's economic or technological development" and "to be consistent with prudent management practices required for the economic and safe conduct of nuclear activities." I believe it is true that we have proved to safeguarded nuclear industry that safeguards, and especially inspections, do not hamper their operations or increase their costs. The operators of two major nuclear power plants under Agency safeguards in the UK and USA have, on several occasions, testified to the fact that inspections have not proved unduly burdensome.

In connection with the design review and inspections, the specter of industrial espionage has often been raised. I would like to state that neither during an examination of the design features nor in the course of the inspection of a plant are the inspectors likely to require access to design details or materials selections for important components that can be said to be of a commercially secret nature. The general design features, which are of importance for the purpose of inspections, cannot be regarded as industrial secrets as many of them are available in the open literature. The safeguards document also places a strict obligation on the Agency to protect any secret information it obtains in the course of its safeguards work. Finally, a Member State under Agency safeguards has the right to refuse acceptance of a nominated inspector.

The safeguards document finally describes the circumstances under which nuclear material may be exempted from safeguards, or safeguards be suspended or terminated. It also defines the conditions under which material can be transferred outside the jurisdiction of a State in which it is being safeguarded.

The question of sanctions in the case of a detected diversion of nuclear materials is one that has received rather little attention in the past; therefore it may be interesting to dwell briefly upon it here.

The Agency's Statute foresees a two-step procedure to determine whether or not there is a noncompliance by the State with the basic obligations of the safeguards agreement. The first step consists of a determination by the staff of inspectors of noncompliance (which obviously does not include minor deviations from agreed safeguards procedures such as a slight delay in the submission of routine reports). After this the matter is submitted by the Director General to the Board of Governors who determine definitely and ultimately whether or not there is noncompliance.

There would then follow a series of quasi-automatic measures that are set forth in the Statute:

> "The Board *shall* call upon the recipient State or States to remedy forthwith any noncompliance which it finds to have occurred. "The Board *shall* report the non-compliance to all members and to the Security Council and General Assembly of the United Nations."

The proper sanctions in the case of a noncompliance are thus decided upon by the United Nations Security Council and General Assembly.

If and when the Non-Proliferation Treaty and the Treaty for a Latin American Nuclear-Free Zone come into force, the requirements on the Agency's safeguards system will be changed somewhat, at least in the case of some countries. These treaties will call for open-ended safeguards agreements, involving all present and future activities in a country. The first such agreement was concluded in September this year between the Government of Mexico and the Agency. In several countries that type of agreement would give the Agency the responsibility to safeguard complete or nearly complete fuel cycles. In such cases it will become possible to streamline the operation to some extent (and when a major part of a fuel cycle is under safeguards, chances also increase for an early detection of any diversion, which would make the safeguarding task easier in one sense). It will probably become necessary to adapt the safeguards system to such changed circumstances but I am certain that any adaptation can be performed on the basis of our present system and that the main provision will remain conceptionally unchanged

There are some other developments that will also influence the IAEA safeguards operations in the future: the efforts now being devoted to research and development in safeguards, and the establishment of materials accounting and safeguards systems on the national level. Each government in a country with a fairly large nuclear program must have a great interest toward establishing controls, i.e., national safeguards, to avoid losses of expensive materials and the risk of these materials getting into the hands of subversive elements. The existence of a national materials accounting and safeguards system, if harmonized with the international safeguards, will clearly decrease the workload the international control involves.

It is only during the last year or so that significant sums have become available for research and development work in safeguards. The work that has been started in laboratories in several countries will, in the long term, no doubt become of great significance for international safeguards, and has already begun to contribute to a better understanding of the subject.

There is a huge need for research and development of both fundamental and applied character. To dwell briefly on some of the important areas, I would consider it essential that, first of all, more work be devoted to clear and *quantitative* definitions of the aims and objectives of safeguards procedures. This applies to both the fundamental procedures of the present safeguards document and to the detailed procedures applied in specific safeguards operations. It will require a systematic analysis of the process involved in more-or-less complete fuel cycles at different throughput levels. Such systems analysis work should also give results that will lead to conceptual developments in the safeguards system.

Second, there is much to be done toward developing ways and means of achieving the objectives that have been defined. This relates to both general and specific safeguards procedures and the tools and instruments that can be employed. These lines of research are being pursued in several of the Agency's Member States and will be described in other papers of this session. The Agency is keeping in contact with the work on the national level to ensure that the Agency's safeguards operation will benefit as soon as possible from results.

When, as I hope, international safeguards become a normal aspect of the peaceful uses of nuclear energy, it will also be realized that safeguards can be greatly facilitated in many cases if some simple precautions are taken in the design of a facility. Here, both the safeguarding authority and the facility operator stand to gain very much indeed for a very small cost.

It will probably take several years before results of the research and development work, which to a great extent has only begun, will yield concrete results in the form of simplified safeguards procedures. I believe, however, that some time will also elapse before international safeguards are to be generally applied under a non-proliferation treaty. The IAEA in its safeguards operation has to strike a very careful balance between the requirements that have to be met; viz., acceptability of the system to a State under safeguards, maximum efficiency of the operation, and credibility to the international community. This can be achieved only by regarding the system and operation as flexible and adapting it to changing circumstances and technical development. It is the Agency's purpose to do this, and specific provision is indeed made for this in the safeguards document.

THE EURATOM SAFEGUARDS SYSTEM

FERNAND SPAAK, *Director General of Energy*
Commission of the European Communities, Brussels, Belgium

It is a great privilege for me to address this highly qualified audience, which includes leading representatives in the fields of industry, management, and science from countries all over the world. And I am very grateful to the sponsors of the Atomic Industrial Forum and the American Nuclear Society for affording me this opportunity to give you a general outline of the system of safeguards applied by the European Community for Atomic Energy. I am pleased, in particular, to present to you the political principles involved in this system, its originality and its value as an existing reality, as well as its use as a testing bench for a further, perhaps wider, approach.

More than eleven years ago in Rome, on March 25th, 1957, six European States—the names of which are certainly familiar to this distinguished audience—signed a Treaty establishing the European Community for Atomic Energy, and setting up, in a general framework of economic and scientific interdependence, a system of safeguards covering in practice all the nuclear peaceful activities within the Community. Pursuant to the Treaty, the exclusive responsibility for applying such a safeguards system is given to the Commission of the European Communities—the European executive—which has had its headquarters in Brussels since the merger in July 1967 of the three former bodies: the Commission of the European Atomic Energy Community, EURATOM; the Commission of the European Economic Community; and the High Authority of the Coal and Steel Community.

The Euratom system was the first multinational safeguards system put into operation. It was conceived by the Six to provide each other with guarantees ensuring that the buildup of an efficient nuclear industry in Europe would be pursued on a basis of mutual confidence, by preventing the diversions of materials from their declared uses.

Under the Euratom Treaty, the Commission must ensure that ores, source materials, and special fissionable materials in the territories of the Member States are not diverted from their intended uses as stated by the users, and that the provisions concerning supplies and any special undertaking concerning safeguards entered into by the Community in an agreement concluded with a third country or any international organization are observed.

To enable the Commission to fulfill all its obligations, the Treaty requires that anyone setting up or exploiting facilities for the production, separation, or use of source materials or special fissionable materials, or for the processing of irradiated nuclear fuels, shall make a declaration to the Commission setting out the basic technical characteristics of such facilities. An additional requirement is that operating records must account for ores, source materials, and special fissionable materials used or produced, and that these records have to be made available to the safeguards authority of the Community.

Furthermore, the Commission has the right to send inspectors into the Member State Territories; these inspectors shall have access at all times to all places and data, and to any person dealing with materials, equipment, or facilities subject to safeguards.

To implement its safeguards system, Euratom has issued various regulations that form the structure and embody the actual means for applying safeguards. The facilities are required to maintain a material accounting system and to make periodic declarations to the Commission regarding the materials subject to safeguards. The Commission checks the accuracy of these

declarations by accounting and physical methods; i.e., the parties making the declarations are responsible for them and the Commission verifies them.

These measures provide the Commission, at any time, with an accurate picture of the Community situation regarding nuclear materials and their use, as well as discourage any misuse or undeclared use.

Because of the interconnection of Euratom activities, the Commission has further means of verifying the accuracy of the declarations received and of assuring the detection of any diversion of material. The rights and competences granted by the Treaty to the Community, and in particular to its supply Agency in the field of supply transactions, give the accounting branch of the Euratom safeguards system the possibility of further checking the accuracy of the data and the calculations based upon them.

The validity of a safeguards system cannot rely solely on remote controls applied through documents and records, however, and this is where the other important instrument available to the Commission comes in—the on-site inspection.

Euratom inspectors are appointed by the Commission after security clearance, and entrusted with their specific tasks after consultation with the Member States. They have full status as Community officials, which assures them of continuity and stability of their assignment and makes them immune from pressure or influence by national governments or industry.

The governments and the utilizers have never raised objections to the Commission's choice nor has the work of the inspectors ever been hampered by the governments or private enterprises. At the request of the State concerned, the inspectors may be accompanied on their assignments by representatives of that State acting as observers, provided these observers do not interfere with the activities of inspection.

Inspectors enjoy the widest powers of investigation; they have access to all places and data at all times, and to all persons who, by reason of their occupation, deal with material, equipment, or facilities subject to safeguards.

The volume of inspections carried out by Euratom officials is considerable—there are approximately 280 installations within the Community holding nuclear materials now, with the number constantly increasing: At the end of 1960, first year of application of inspections, there were 12; at the end of 1964, there were 110; and in 1967, there were 411.

The nuclear enterprises being inspected regularly cover the whole fuel cycle: mines, mills or other plants processing natural uranium, power reactors (of which, at present, 17 are in operation with a total installed capacity of approximately 2500 MW), approximately 70 research reactors, several research centers and laboratories, 8 fuel fabrication and 7 reprocessing plants. All these nuclear plants have different potentialities of diverting nuclear materials from their declared uses; consequently, the intensity of inspections required varies from one facility to the next.

The Euratom inspection system has been adapted to this wide range in accordance with the basic quantitative and qualitative criteria established by the Commission after study of the problems and their optimal technical solutions.

These criteria were examined in discussions with other national and international organizations involved in the same fields. Permitting a case-by-case assessment of the situation at each individual installation, they provide for different intensities of inspection, the highest being the resident or continuous inspection adopted for those plants holding large quantities of material of a very sensitive nature.

One of the most important aspects of the safeguards system created by the Treaty is to guarantee the fulfillment of any special international engagement undertaken by the Community in addition to ensuring a regular and equitable use of nuclear material throughout the Community. In those cases, the scope of the system can be widened to include, for instance, equipment and even nonfissionable material.

Thus, in the agreements for cooperation in the peaceful uses of nuclear energy concluded with the United States, the United Kingdom, Canada, Brazil, and the Argentine, the Community has undertaken the exclusive responsibility for establishing and implementing a safeguards system giving maximum assurance that materials and equipment supplied by these countries are being utilized solely for peaceful purposes.

All these agreements provide for consultations and exchanges of visits between the parties to give assurance that equipment and materials are not diverted from their peaceful destination. In particular, the consultations agreed by the Community and some of the above-mentioned third countries give to the latter the possibility to verify, by appropriate scientific methods, the correct application of the safeguards covering the nuclear material supplied by them to the Community.

Among other procedures, joint technical working groups have been established to permit experts from both sides to meet regularly, exchange experiences, compare data, carry out joint exercises of analysis and sampling in certain facilities, and study jointly the most appropriate

safeguards methods applicable to specific types of nuclear plants.

To conclude this particular subject. I think I can say that the implementation of the safeguards clauses contained in the international agreements concluded so far by the Community has been satisfactory for all parties concerned.

One major problem has appeared in the course of consultations with our international partners: the adoption of new techniques assuring greater effectiveness of safeguards while maintaining or improving the protection of industrial or commercial secrets. Actually, the right of nuclear operators to be protected against the unwanted spread of information concerning their activities must be given the highest consideration.

Our Community is aware of the need for international collaboration to coordinate the development and assure the compatibility of new techniques on a worldwide basis. Active studies have been undertaken within the Community, with the participation of the Commission, to explore and to perfect the technical possibilities for easing the burden of safeguards. The first objective is to detect certain strategic points in each plant where the flow of fissionable materials can be checked with accuracy; and second, once such strategic points are spotted, to replace—either entirely or partially—the work of the inspectors by instruments capable of measuring the flow of the materials. This second objective is extremely important if we look at the progressive expansion of nuclear energy. The number of nuclear facilities is constantly increasing, and the number of inspectors will have to increase accordingly. In other words, in a few years it will be hard to keep pace with such dynamic expansion if we satisfy ourselves with present-day techniques. The manpower requirements for an efficient safeguards system will justify a reexamination of the allocation of a very large number of highly qualified personnel to this task, with the consequent training and financial implications.

Instrumentation and automatic data processing may be the solution to the problem by complementing, if not replacing, the work performed by men.

This description of the system, and the way it is applied and inserted in an international context, lead to some general considerations. First, the great advantage of our safeguards system is that it was written into one of the Treaties of Rome, one of the cornerstones of a united Europe's structure.

The Euratom safeguards, as the Treaties themselves, have been established for an unlimited period and are not subject to a termination clause. No Member State may at any time escape all or any part of the safeguards rules without withdrawing entirely from the Treaties—a possibility none of them provides for.

Thus, the legal framework of the Euratom safeguards system distinguishes it from other systems that are founded only on the consent (which may be withdrawn at any time) of States whose acceptance of the application of international safeguards is confined to a contractual basis.

Furthermore, the safeguards carried out by the Commission on behalf of the Member States must be considered in the framework of the new nuclear structure being created in Europe, together with other elements as the joint research, the dissemination of information, the nuclear common market, and others; the closed interrelationship between such elements is the condition for a complete economic integration.

There is no need to further underline the value this structure embodying the Euratom safeguards represents for everyone of us in Europe. Whatever its aim, a safeguards system, by its very nature, imposes sacrifices. In the case of the Euratom safeguards system, the common sacrifices have been accepted by the Six in view of common advantages. Refusing to make sacrifices today would automatically imply renunciation of the substantial advantages already acquired and of those to still be acquired in the process toward European economic integration.

I would now like to say a few words about the attitude taken by the Commission and the Member States regarding the treaty of non-proliferation of nuclear weapons. The fact that some of our Member States, among many other nations, have signed it, proves there is no incompatibility between the basic goals of the treaty and those of the Treaties of Rome, which were subscribed with a view to a peaceful and orderly process toward a multinational integration. However, for the very reasons that I have just given, none of the Member States intends to accept the implementation of a treaty that might affect the balance of rights and obligations, or of sacrifices and advantages established by the Treaty of Rome.

This is why the Member States who have already signed have declared that the ratification of the Treaty will take place only after a mutually satisfactory agreement in the field of safeguards has been concluded between the International Atomic Energy Agency in Vienna and our Community.

In view of the Member States, and of the Commission, such an agreement must maintain the existing Euratom Safeguards System intact. At this stage, we can only say that the text of the non-proliferation treaty actually provides a legal

basis for an acceptable agreement and that, on its part, the Commission is willing to tackle this problem in a constructive spirit.

Another consideration concerns the extent of the application of Euratom safeguards. They are carried out by the Commission on a mandatory basis throughout the whole territory of the Community to all nuclear materials produced or imported in the Community. The fact that each person or facility everywhere within the Community is compelled to conform with the safeguards applied by the Commission, means that the Member States, in setting up such a mandatory and general system, have yielded the essential part of their power in this field to the Commission. The Community is thereby able to implement its rules, decisions, and sanctions without national ratifications.

It must be pointed out that safeguards do not apply to materials that are in the course of being specially prepared or stored in a military establishment for purposes of a Member State's defense. Nevertheless, this does not prevent the Commission from having a complete view upon all movements of nuclear material in the territory of the Community, including to and from military programs.

Non-discrimination is the bedrock of an effective safeguards system; the Euratom rules apply to all Community enterprises or persons in the same way, whatever their nationality.

There have been some references to the existence of practical difficulties and obstacles in the application of safeguards to the industry of a nuclear power, in the military sense of the term.

From our own experience, we are able to say these obstacles can be overcome provided there is the political will to do so; as a matter of fact, all the nuclear peaceful activities of France, the only nuclear power within the Six, are placed under the safeguards of the Commission, exactly like those of the other five Member States.

It must be stressed that the extension of safeguards to all nuclear materials in the Community considerably reinforces the effectiveness of the safeguards themselves. This enables the declarations submitted by facilities to be checked at any time or place, and it allows the Community to verify, *a posteriori*, the validity of the declaration; i.e., even when the materials that have been involved in a particular program have been transferred to another program.

Finally (but my list of considerations is not exhaustive), there is a direct relationship between the Commission and the facilities and persons holding nuclear material; the Commission acts as the only competent authority for the application of safeguards on behalf of the Member States, who are not involved in the process, having renounced their prerogatives in this field in its favor. Thus, the possibility of a nationally coordinated conspiracy to withdraw any material from safeguards and move it from one national facility to another becomes nearly impossible, as the Commission exercises a direct control over each facility.

Ten years of this first multinational experience have proved its effectiveness which, we believe, remains unequalled. We are determined to improve this effectiveness by proceeding with an active, and hopefully wider, international collaboration by harmonizing manpower and new techniques in accordance with reality and with the necessities of a changing world.

SAFEGUARDS AND NUCLEAR MATERIALS MANAGEMENT IN THE USA

DELMAR L. CROWSON, Director
Office of Safeguards and Materials Management
U.S. Atomic Energy Commission, Washington, D.C. 20545

INTRODUCTION

In carrying out its responsibilities under the Atomic Energy Act of 1954 as amended, the U.S. Atomic Energy Commission has fostered the development of nuclear power, has regulated the industry with regard to radiation hazards, and has taken steps to ensure that special nuclear material is used as intended, to advance the welfare and security of the country.

For a number of years, all significant quantities of special nuclear material were handled in government-owned facilities and were subject to direct AEC control measures. The safeguards measures which were applied to licensees (private industry) were based on the assumption that financial responsibility, together with penalties imposed by the Atomic Energy Act for misuse, would ensure adequate control measures for the limited quantities in the hands of licensees. However, the nuclear industry is now rapidly developing, with the consequence that special nuclear material is becoming a significant article of commerce throughout the nuclear reactor fuel cycle, depicted in Fig. 1.

The quantity of uranium and plutonium expected to be introduced into the fuel cycle is staggering. Figure 2 depicts an AEC estimate that six million kilograms of uranium will be required annually for fuel fabrication in the domestic cycle by 1980.[1] The total fuel cycle cost, as depicted in Fig. 3, is expected to reach two and one-half billion dollars at the same time.[1] As a result of such predictions, added impetus was given to the AEC review of its safeguards responsibilities and practices, and in May 1966 a program was initiated to strengthen domestic safeguards to meet the anticipated increase in the quantities of materials in the hands of licensees, where practically all the material will be privately owned within a few years.

In connection with our examination of the nature of the growing private nuclear industry, we have identified three general elements of safeguards.

1. *Physical security*, which includes measures such as vaults, locks, seals, guards, fences, and alarms intended to prevent diversion by providing a high probability of immediate detection. Measures to ensure reliability of employees may be included in this category.

2. *Accountability*, which includes measurement, accounting, auditing, and inventory procedures designed to provide an accurate knowledge of material quantities and locations.

3. *Surveillance* or monitoring, which is the use by the safeguards agency of its own technical personnel and monitoring instruments to obtain independent information on a random or continuous basis about material flow or inventory, performance of a plant, process or security, or the accountability system.

The extent to which each of these elements is employed depends considerably on the environment under which safeguards are applied.

The current AEC domestic safeguards system has been effective in operating in the environment for which it was designed; namely, in plants owned or operated exclusively for the government, and other plants producing materials and products that are classified for security reasons, and as a consequence, receive optimized security protection. Heretofore, relatively insignificant amounts were leased to the private sector for commercial purposes and the domestic safeguards system has been successful in providing the necessary degree of safeguards with respect to these amounts. However, the amounts of material in the private sector and the amounts of material that are involved in operations no longer requiring protec-

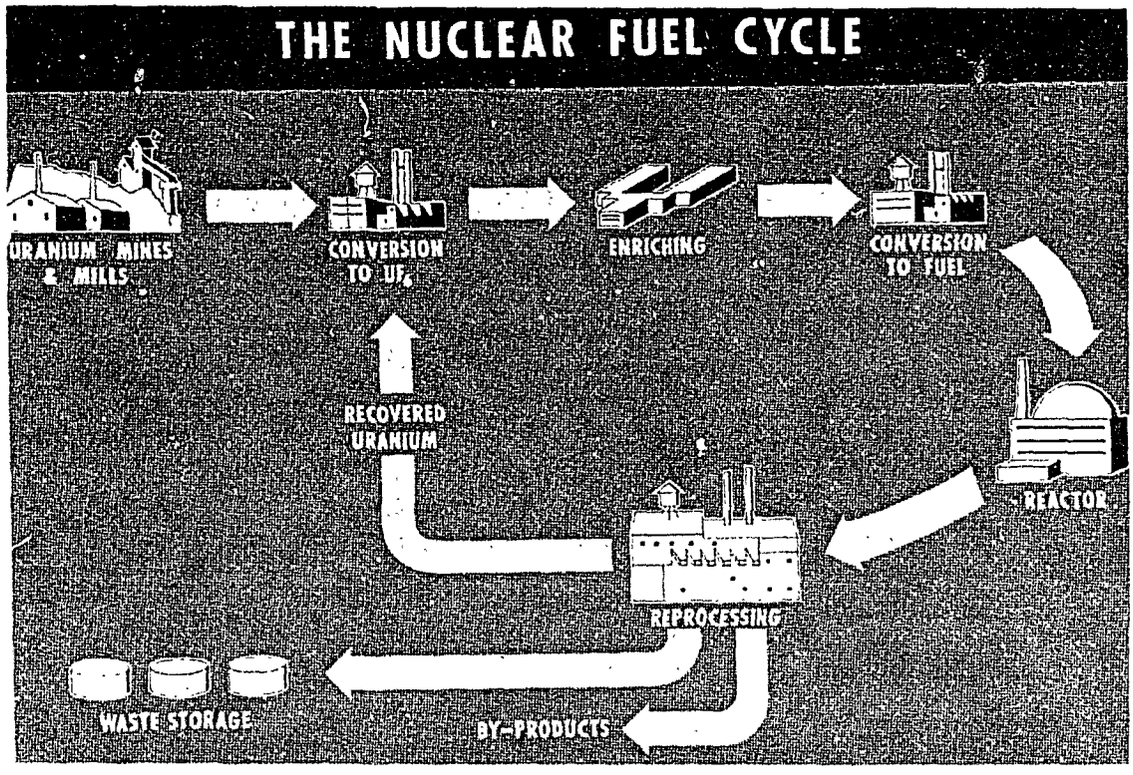

Fig. 1. The nuclear fuel cycle.

tion as classified security information has already begun to expand rapidly. Consequently, the AEC safeguards program must be adapted to meet the changing situation.

We recognize that it is necessary to conduct studies to clarify the issues, to develop improved techniques, and to start implementation of an appropriately adapted safeguards system or systems immediately. Preliminary decisions are being based on past experience and evaluation of information currently available. Studies, analyses, and field tests now underway and contemplated will provide a more solid base for making decisions within the next two years.

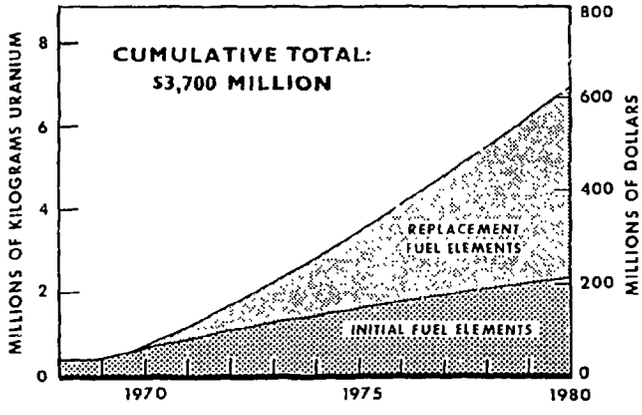

Fig. 2. Fuel fabrication.

The AEC has an extensive safeguards research and development program underway at this time which is being rapidly expanded to accommodate the needs of the safeguards program as such needs are identified. The AEC safeguards R&D efforts are being fully coordinated with other concerned US Government agencies and with appropriate international organizations.

The subject of this paper, "Safeguards and Nuclear Materials Management in the USA," is quite broad and cannot be fully covered without some discussion of safeguard policies and the entire scope of the research and development program. However, these subjects have already been discussed at length with many in this audience, and will be further discussed by others during this meeting. Therefore, I shall confine my remarks to a brief description of the domestic AEC safeguards program and a detailed discussion of one of our major R&D efforts, which we call systems studies, that is designed to help us identify those areas in the fuel cycle where specific research and development efforts can be applied to solve the most pressing problems at the earliest possible date.

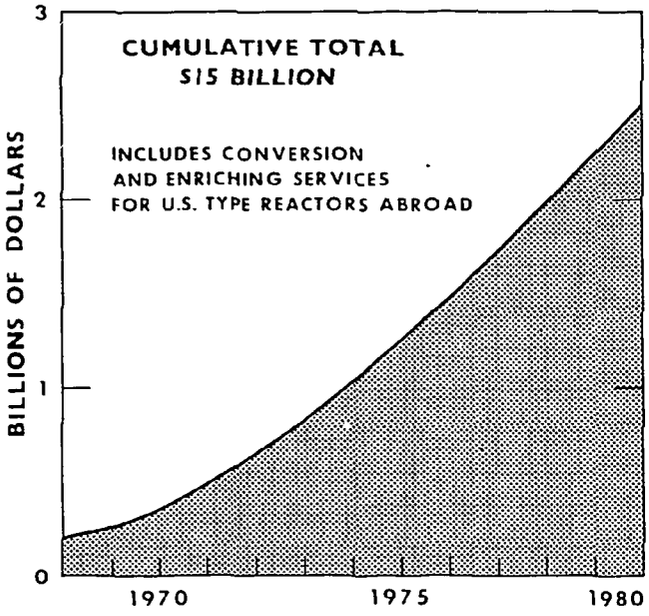

Fig. 3. Total fuel cycle costs.

US DOMESTIC SAFEGUARDS SYSTEM

Our current domestic safeguards and nuclear materials management systems consist, in addition to the security requirements applicable to government operations (not licensees), of two basic features; namely, a system of records and reports designed to establish how much material *should be* on hand, and a system of inspections designed to establish how much material *is* on hand. The procedures designed to accomplish these primarily accountability and surveillance actions include:

1. The AEC requires that transfers of materials be documented and reported to the AEC where central transfer journal records are maintained.
2. The AEC requires submission of material balance reports semiannually. These reports summarize all transactions for the preceding six months and include all consumption and losses and the results of the most recent physical inventory.
3. The AEC maintains detailed records of all special nuclear material (SNM) possessed by contractors and licensees and of the SNM shipped to foreign nations under bilateral and multilateral agreements.
4. AEC contractors and certain licensees are required to take physical inventories at least annually and to report the results on the next required material balance report. Also, AEC contractors and certain licensees are required to develop formal material control procedures and submit them to the AEC for review and approval.
5. Annually, AEC employees visit the plants of certain contractors and licensees to perform a safeguards inspection during which statistical checks of inventories are made, including sampling and AEC laboratory analysis of materials. During these annual inspections, plant safeguards and materials management practices are thoroughly inspected and any inadequacies are made the subject of senior management discussions with a view toward corrections as required.

With the exception of certain small quantity holders, these procedures apply to all contractors and licensees in the United States. We are now developing security procedures that will be applicable to licensees.

To assist us in program evaluation and research and development planning, we have considered what modifications might be required to our current domestic safeguards accountability and surveillance system; these modifications should enable us to maintain an effective accountability and surveillance capability, as the environment in which safeguards will be applied continues to change. The initial modification we anticipate will be to provide for the collection of safeguards related data in plants with less dependence upon plant personnel than provided for by the current system. We expect to be able to accomplish this by the use of available instruments to record data and the possible use of AEC resident inspectors (to the extent that our present resident-inspector evaluation program identifies the need for such resident inspectors in domestic plants). Successively, we would expect to continue to modify our system by optimizing, from an economic and functional standpoint, the use of instruments and resident inspectors (if resident inspectors are found to be feasible). We would expect, even in the case where resident inspectors are used, that as new instrument capability is developed we would increase our dependence on instruments and have an increasing reduction in our dependence and need for resident inspectors. We have oriented our research and development effort toward achieving a capability for optimizing our use of automatic instruments, and minimizing, if not eliminating entirely, our dependence upon resident inspectors. Beyond this we have recognized a hypothetical further modification to our safeguards capability which would provide for the connection of automatic instrumentation through

tamperproof links to a central computer to facilitate near-instantaneous material balance capability.

US Material Control System

Our present US material control system contains three features that make it particularly applicable to safeguards purposes.

Administration and Regulations. AEC license-exempt contractors and licensees have direct custodial responsibility for the material, and must adhere to certain contractual and regulatory requirements involving operating practices and reporting functions.

A Double Ring of Material Balances and Performance Evaluation. The plant, depicted by the inner ring of Fig. 4, performs periodic physical inventories, material balance summations, and operational status reports which portray plant performance and material control conditions.

The government functions, depicted by the outer ring, consist of periodic audits of records, measurements, and inventories. This includes independent verification of the material and evaluation of plant procedures and practices.

A Central Information System. This accounting process contains the locations and quantities of all materials in the system and acts as a historical data bank in which the accumulated history of any site is maintained. Much of that system is now computer-based, providing an excellent framework for future expansion.

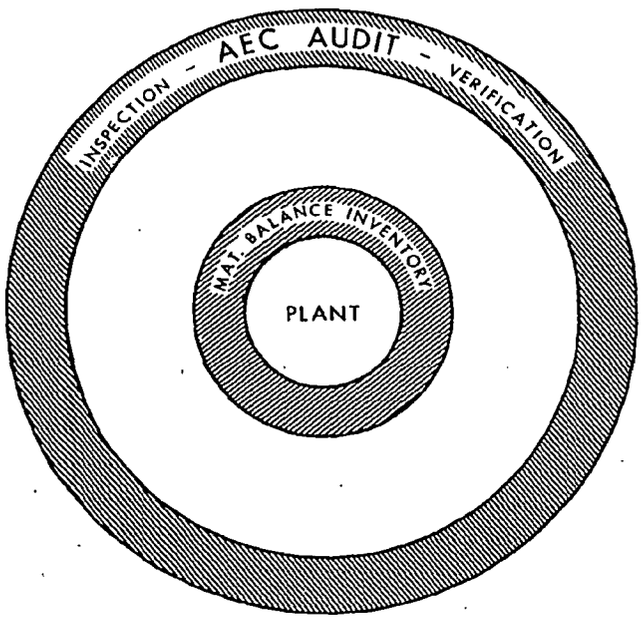

Fig. 4. US material control system.

These three features of our material control system provide us with the foundation for the material control portion of a safeguards system. I would like to add that our material control system is the result of some 20 years of experience and labor, and one of which we are justifiably proud.

BACKGROUND FOR SYSTEMS STUDIES

We have chosen the "systems" approach to analyze and identify necessary modifications to our current system to meet the new challenge. We recognize, and will avoid, the continuing hazard of such an approach that results from a natural desire to want to re-engineer the entire world of things and people.

Quite a few of those who concern themselves with the history of problem solving by the "systems" approach place the birthday of the method back during World War II, some 25 or more years ago. My systems analysts claim that Plato laid the foundations of systems analysis when he wrote *The Republic* over 2000 years ago! Plato admits to imperfections of any existing system of things and people and asks if it can be established that improvement "is possible, and if so, how?" Plato then goes on to ask "What is the smallest change . . . : one thing, if that is enough, or else two, or as few in number as possible and least far reaching" that will approach his perfect system.[2] We are following Plato's advice and seeking out the minimum change that will improve our system, and if we work things right, the price tag will also show a minimum.

There is a three-element common denominator in all systems analyses which is closely analogous to the "scientific method." First, we define the underlying logic of the current system. Generally, we consider that the US safeguards accountability system is based on the material balance concept, which is simply jargon for a principle of "conservation of material"; i.e., what went into the area is either still there or came out. Second, we develop a mathematical model of the operation which, in its simplest form for safeguards, merely states that we demand to know what happened to the material left over when we subtract the output of the material balance area and the material remaining in it from the amount of material put in. This is a simple idea, but when we introduce Plato's condition, "the imperfect nature of man and his measurements," and try to follow the inherent measurement uncertainties of any such system, we have troubles. The third and last element in our common denominator is the

determination of an "optimum" array of parameters in the model; that is, what do we measure, where, and how well? This is the point of maturity in our analysis and is, by no means, straightforward.

CURRENT SYSTEMS STUDIES

We are currently engaged in three efforts which come under the broad heading of systems studies. These are the Nuclear Process Analysis Studies, Evaluation of the Resident Inspection Experiment, and Inventory Verification Procedures.

Nuclear Process Analysis Studies

Nuclear Process Analyses were initiated early in 1968. These studies are aimed at determining in a more quantitative way the role and effectiveness of material control systems as a safeguards tool. The studies bring together into a single undertaking the fields of statistics, operations analysis, accounting, analytical chemistry, nuclear engineering, physics, data processing, and nuclear process technology.

The immediate goals of these efforts are:

1. Assess our present safeguard system's ability to detect diversion and find the loopholes for hiding diversion.
2. Isolate the individual sources of uncertainty and determine their contribution to the overall uncertainty.
3. Determine the costs associated with reaching given levels of effectiveness.
4. Determine the information needs of the AEC for the material-control phase of an effective safeguards system.
5. Establish guidelines and statistical bases for evaluating and using indexes that measure the performance of the system and indicate when investigative action should be taken.

To reach these goals, we are carrying out in-depth studies of contemporary processes, measurements, and practices. It is important that we learn both the ultimate capabilities and historical performance of the material control system if we are to optimize its effectiveness. Concurrent with the systematic analysis of the system, we are developing a process model to simulate material flows and propagate the associated measurement errors to determine limits of uncertainty at various points in the fuel cycle.

The models that have been developed over the past few months in the uranium hexafluoride to oxide or metal conversion processes are undergoing continuous experimental verification as new data are collected from the plants. On-site nuclear process studies have been started within the past month at an enriched-uranium fuel fabrication plant, and modeling of plutonium fuel fabrication will begin next month. We expect to extend our models to enriched-uranium and plutonium-fueled reactors in early 1969. The plants have been carefully chosen to represent a continuous flow of material through the cycle. We should be able to have a closed fuel-cycle model for enriched uranium and plutonium by the Summer of 1969.

Evaluation of the Resident Inspection Experiment

In July of 1967 we initiated a one-year Resident Inspection Test Program at four of our commercial material conversion and fabrication sites and at the US reactor fuel reprocessing plant. That experiment was just recently completed and is being evaluated at this time.

Inventory Verification Manual

An Inventory Verification Manual is being prepared by one of our prime contractors. This work is an extension of some excellent basic theoretical work and current operational experience of the AEC. The initial work on the manual was done by personnel from AEC's Field Offices and Headquarters groups. The ultimate objective of inventory verification is to be able to certify, as a result of the AEC surveys, the amounts of special nuclear materials in plant inventories.

SOME BASIC CONSIDERATIONS

I think it is evident from my brief descriptions of our system studies effort that our immediate efforts are at the operating level. One of our first goals in evaluating our safeguards system is to strengthen our statistical basis for exercising material control. This is a twofold approach. The first is the improvement of the statistical procedures themselves and the second is in providing the mechanisms for obtaining and applying realistic parameters for use in statistical evaluations.

At the first line of control, that of the plant itself, a shortage or loss of material would normally be indicated when the magnitude of the MUF (material unaccounted for) exceeds the limits of error arising from measurement uncertainties. Thus, we emphasize that it is the variance of the MUF which really reflects the sensitivity with which we can detect diversion by material balance accounting. Once the variance is established, a statistical upper limit can be constructed such

that if the observed MUF exceeds that limit it can be concluded with a reasonable and calculable certainty that a real loss or diversion has taken place.

The most significant feature we are illustrating by the discussion of MUF is the kind of information needed to be able to place the detection of diversion on an objective basis. It is evident that the systems analysts must have fairly detailed and exact knowledge of the total error structure in a material balance. Secondly, it is evident that reliable information must be used on a sound statistical basis. As a consequence, we consider mechanisms for determining, reducing, and evaluating measurement uncertainties as a key part of our safeguards studies effort. We are applying the knowledge gained from our in-depth studies and from use of our process simulator to all phases of our control system. We expect to see practices and procedures used in operating components which keep the effects of measurement uncertainty at a minimum consistent with cost and effectiveness. We also intend to increase our use of such concepts in evaluating new control systems and in monitoring the performance of the system.

I would now like to recall your attention to the second material balance ring formed by the central control agency, the AEC. In actual practice, the second balance ring is only partially closed by independent action of the AEC. The balance is formed in part by shipper-receiver measurements, by estimation of waste discards, and by verification of inventory quantities by the AEC.

INITIAL RESULTS

The known loopholes in the second balance ring are illustrated in Fig. 5. They are:

1. Non-verifiable inventory items
2. Lack of suitable technology for the reactor operator to accurately measure his input, output, and inventory
3. The need to verify waste discards or similarly to be assured they are not overstated.

Non-Verifiable Items

Non-verifiable items are defined as those items for which the inspection team does not actually verify the total nuclear material content because of excessive uncertainty inherent in the measurements of those materials. These items include irradiated and unirradiated fuel elements, certain scrap and waste materials, and material contained in certain types of processing equipment. Normally such items can only be identified as being present, and the national inspection group will normally accept the stated value for material content. These items may also constitute part of product output of the same plant, thus widening the gap in the second balance ring. As a consequence, we are sponsoring development work in the area of nondestructive assay, particularly portable and transportable devices, to increase our measurement capability in this area. I will discuss some of these developments later.

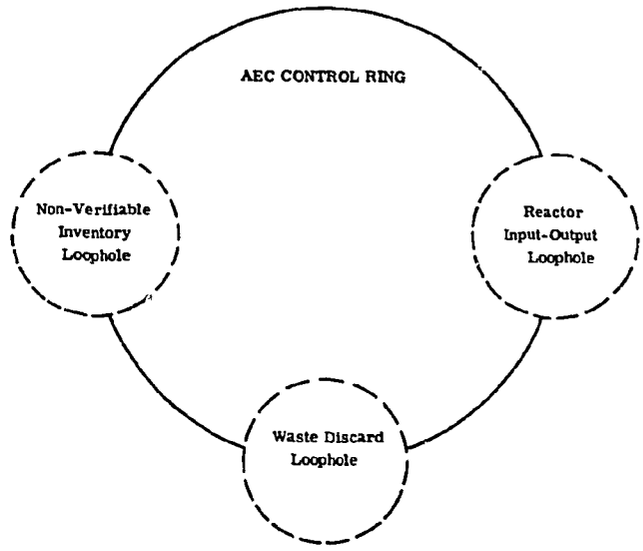

Fig. 5. Main loopholes in second control ring.

Fuel-Fabrication Reactor Chemical Plant Interfaces

The measurement interfaces between fuel fabrication and the reactor, and the reactor and chemical processing present difficult problems from the standpoint of independent verification. For the unirradiated fuel elements, nondestructive devices offer the best hope of a practical solution by instrumental means. The problem with irradiated fuel elements presents a much more difficult case, both in theory and in its practical aspects. We are also heavily committed to research and development related to nondestructive measurement devices in this area. However, the most fruitful approach for the immediate future is to center attention on verifying the input to chemical processing plants by a combination of surveillance and independent measurement. From our work on our Resident Inspection Experiment, we may also conclude that such a method can provide effective safeguards in this critical area.

Verification of Waste Discards

When the control agency does not independently measure waste discards, there always exists the

possibility that waste discards could be overestimated and thus mask the diversion of product materials. The approach taken by the USAEC in controlling waste discards in its government cost-type operations is to set discard limits based on plant historical performance. The aim here is to verify waste discards and compare them to what is considered within the range of process capability. Independent methods of physical verification by means of instruments are dependent upon the development of the devices mentioned earlier.

Resident Inspection

Two basic approaches were considered in our work on Resident Inspection. The first was to determine the manpower and equipment requirements for the inspection team to form and maintain material balance data independent of plant personnel. Early indications are that this approach would require some 14 to 18 people for a chemical processing plant and about an equal number for a major material conversion and fuel fabrication plant. The second approach was to consider the value of resident inspection in our domestic fuel cycle where a ring of independent measurements already exists to some extent for feed and product materials. At this point, we also considered the value of resident inspection to the AEC's performance evaluation function. It appears that in a closed measurement ring one or two inspectors could make a valuable contribution to safeguards at material conversion and fabrication sites. They could do much in closing the second balance ring and in placing the AEC evaluation effort on a continuous basis.

The problem was somewhat different for a chemical processing plant because of the current uncertainties in reactor predicted values for plutonium. As a result, we found it was necessary to test the ability of the inspection team to verify chemical plant inputs by a combination of surveillance and independent measurement. Based on what we then considered the current state-of-the-art in measurements, we concluded that the task would require some 12 to 14 people in total, with some of those people working at an independent safeguards laboratory.

Historical Performance

One finding of our current studies of historical data is that significant discrepancies in material balance do occur which have no relation to diversion. The major discrepancies that are observed in monitoring a material control system are measurement uncertainties, such as inadequate random sampling and poor technical estimates. We find blunders (as opposed to uncertainties), e.g., errors in transcription, reporting of data and item identification as indicated by a limited check weighing study, to be an important factor in endeavoring to estimate the precision and bias of inventory holdings from samples of the inventory. We find that blunders must be virtually eliminated before reasonable estimates of measurement uncertainties can be made.

Examination of historical performance also shows that, in general, the long-term MUF is not zero but on the order of 0.5% of throughput. The common conclusion is that such apparent shortages are due to unmeasured losses or measurement bias. However, from a safeguards standpoint, continuation of the same MUF could also indicate or mask a consistent diversion—a matter of equally serious concern from a national security and a business profits standpoint. It is then a common thing for the safeguards agency to be faced with the questions—loss? bias? or diversion? One approach to handling problems of this type is for the safeguards agency to have a detailed knowledge of the processes, mechanisms of losses, measurements, and practices. This tradition of intimate knowledge of the plant process and control system is another key feature of the US material control system.

FUTURE CONSIDERATIONS

From the previous discussions, one would envision that for the immediate years the US material control system would continue to operate as a series of control rings which close as plant physical inventories and AEC independent inventory verifications are performed according to optimum safeguards schedules. These types of controls might be supplemented with continuous inspection and independent measurement in particularly sensitive parts of the fuel cycle such as in chemical processing, and in plutonium and enriched-uranium fabrication processes.

Our research in nondestructive assay is expected to: provide methods for reducing the number of non-verifiable items in an inventory, thus closing loopholes in our accountability system; provide tools for independent measurement by the inspectors; provide analyses of isotopic abundance on a near-instantaneous time scale; and provide a technology spin-off which has already been advertised as a quality control item in the nuclear industry.

Nondestructive assay methods are expected to reduce some of the difficulties of sampling of complicated or heterogeneous systems. The high penetrability of nuclear radiation and the radiation from subsequent induced reactions is expected to provide a means for circumventing many expen-

sive and time-consuming chemical techniques. The detection of both neutrons and gamma rays emanating from the interrogated material can provide a quantitative assay of the amount and isotopic content of the safeguarded item. We are expecting field demonstrations of these methods applied to the assay of scrap and product materials during the coming year.

Nondestructive assay methods are being covered in more detail in other sessions of this meeting.

The lag between the time when a diversion might take place and the time when the current safeguards system would detect such a diversion is one of the factors requiring modification when we recognize the unclassified and expanded environment in which the system would be operating in the foreseeable future. Ideally, an instantaneous response, highly automated system is desired. However, our efforts are aimed at a system with a rapid response time but not necessarily instantaneous. There are many encouraging trends in this direction. Historically, the incentive for computerized inventory control has been critical mass considerations. Such systems, fortunately, provide an excellent foundation for future safeguards systems. Another encouraging trend toward rapid response is the increasing development and use of methods of nondestructive measurement in some parts of fuel fabrication.

For the US system to improve its response time, both new measurement devices and information handling systems are being developed. Such systems must have practical plant application and be justifiable on an overall economic basis. More importantly, they must be sufficiently reliable for safeguards purposes. It should be noted that computerized data-handling systems using automated sensing devices are also subject to both human and instrumental error. As a result, their reliability as a replacement for conventional methods must be carefully tested and thoroughly demonstrated. Thus, as a general rule, we would hope to proceed to highly automated material control systems in a stepwise manner by developing, proving, and placing in practice the equipment needed to achieve that end.

Finally, I must reiterate our basic position on our entire safeguards research and development program. It remains oriented around four basic objectives:

1. The system must be technically effective
2. The system must be inexpensive relative to the cost of the material involved
3. The system must be depersonalized to the maximum extent possible
4. The system must create minimum interference with normal plant operation.

I must also assure you that I can see no need for any abrupt changes in our current system. When research and development progress dictates, cost benefit studies will begin. After these results are evaluated, the proposed modifications will be openly discussed with the industrial community. Changes finally decided upon will be implemented in a progressive and gradual manner to ensure maximum effectiveness with minimum perturbation.

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TECHNICAL REQUIREMENTS FOR INTERNATIONAL SAFEGUARDS

F. ARSENAULT and V. SHMELEV *Division of Operations*
Department of Safeguards and Inspection
International Atomic Energy Agency, Karntner Ring 11
Vienna, Austria

I want to start my presentation of this paper with the statement that before one can discuss the technical requirements for international safeguards it is necessary to define the objective of safeguards.

For the purpose of discussion we can state the safeguards objective in general and relative terms, which should be compatible with any of the objectives now envisioned for an international safeguards system. The objective of international safeguards is to provide timely detection of any illegitimate use, or diversion for unknown uses, of safeguarded nuclear materials, and thereby to deter, and to the extent possible to prevent, illegitimate uses. It is seen from this statement that a basic safeguards requirement is a knowledge of the location of nuclear material and of the uses to which it is put. This requirement can be satisfied by a system of nuclear materials accounting combined with a system of verification, designed first to assure the validity of the accounting statements, and second, to assure legitimacy of use. It is apparent that a primary technical requirement for safeguards is the establishment of valid criteria and standards for nuclear materials control.

It is possible to have two or more equally valid systems of nuclear materials control that are completely different in their methods and even in their basic philosophy. In fact, systems of nuclear materials control differ significantly within the international community. While it is quite possible to apply safeguards successfully in such a situation, it is clear that the cost of safeguards application is very much a function of the degree of uniformity encountered, first in the control philosophy and secondly in the methods employed by the various control systems.

Governments generally recognize the need for national controls over nuclear material for reasons of public welfare. They must avoid physical destruction of property and lives, accidental exposure of the general population, and overexposure of workers in the field. Some governments apply special controls directed against deliberate diversion of nuclear materials by persons or parties under their jurisdiction.

In a country with State ownership and control (e.g., UK and USSR), the material control function is integrated with operations—however, it is desirable even in such cases to have some separation of responsibility to provide assurance of adequate implementation. In other countries where private or individual operations are carried out (e.g., USA and Japan), appropriate safeguards can only be assured by external, preferably centralized, control.

An international safeguards agency has a role similar to that of the centralized national control organization. It has been suggested on several occasions that the key to future international safeguards lies in the automated instrumentation of industrial facilities, and the extended use of automatic data processing. These statements are, of course, welcome to those of us charged with the routine implementation of safeguards, but any attempt to implement these suggestions requires uniform recognition of the basic requirements of safeguards and the relationship of these requirements to national and industrial interests.

Largely, it can be said that data collected by an external safeguards agency can never be better than that which is collected by an interested, prudent, and honest operator of any given facility. On the other hand, it is possible that the standards of precision and accuracy required for effective safeguards may be greater than that required for routine management of a facility. When this situation arises, the availability of improved data will no doubt be welcomed by the plant operator, provided they are acquired by techniques that are

(and now we quote from the Agency's Safeguards Document) "...consistent with prudent management practices required for the economic and safe conduct of nuclear activities."

For purposes of discussion, let us now assume that reasonable standards of nuclear materials control are being applied by the operators of safeguarded nuclear facilities, and that a system of communication exists which provides adequate accounting information to the safeguarding agency. The remaining essential element of an effective safeguards system is some method of verifying the accounting data. The techniques to be employed for this purpose will vary considerably. They must be tailored to the existing conditions, taking into account the type of facility, the level and kind of technology, and the compatibility of the local nuclear materials control system with those general principles and methods that have been adopted for safeguards purposes.

During past years, development of quite specific and sophisticated verification techniques has been carried out on the assumption that they would be essential, or at least useful, for safeguards purposes. It was, in fact, the lack of practical applicability of some of these techniques that has stimulated the current widespread interest in the subject of systems analysis.

It must be noted, however, that until recently systems analysis might well have been inappropriate because international safeguards have dealt largely with isolated facilities. The application of systems analysis to an isolated facility would probably result in a highly effective set of methods for that facility which might not necessarily be the best for the case when this facility is safeguarded as an integral part of the fuel cycle. The point here is that a high degree of control at one point in the fuel cycle can be relatively useless if the same control cannot be achieved at other important points in the cycle. The term "fuel cycle" itself can imply various things in the context of international safeguards. A country with a well-developed nuclear industry may indeed possess all the facilities usually associated with a fuel cycle; i.e., fuel fabrication plants, reactors, and irradiated fuel reprocessing plants. In a country with less nuclear development, the fuel cycle might well begin with the receipt of fresh reactor fuel and end with the shipment of irradiated fuel. It is useful to keep this in mind when dealing with the subject of systems analysis for international safeguards.

We can identify the three tasks essential to the development of effective safeguards, assuming that the basic objective has been adequately stated. We require:

1. An adequately detailed technical description of the fuel cycle to which safeguards are to be applied.

2. A specification of the technical results to be achieved by the application of safeguards to the cycle so the safeguards objective can be attained.

3. A selection of techniques, existing or possible, that could be utilized or developed to achieve the required technical results.

That is, of course, a technical approach. One must also keep in mind the boundary conditions imposed by the political and economic facts of life, some of which we have already mentioned.

The first of these tasks is what we mean when we use the term "systems analysis." Several Member States of the Agency are carrying out such studies at present and the results will be made available to us. The parties performing this work have maintained excellent, and from our point of view, most useful contacts with the Agency in this area.

The second task mentioned above—a specification of the technical results to be achieved by the application of safeguards—is also being studied by the Member States in connection with the systems analysis studies. No one has yet produced these specifications for all parts of the fuel cycle. However, the requirements for certain parts of the cycle have been defined sufficiently well to permit an assessment of the relevant techniques. In other words, the third task identified above is currently being pursued, either on the basis of partial results of requirements studies or in anticipation of these results.

To understand and evaluate the work being done by the various groups who are performing these three tasks, the Agency must, at the very least, have a well-developed conceptual picture of the various elements involved. We would like to illustrate the manner of our approach to this problem by referring to Fig. 1. The figure shows one section of the fuel cycle, which is, in many ways, the simplest section and certainly the one with which the Agency has had the greatest experience.

We arbitrarily begin with the shipment of fuel elements from the fuel fabrication plant and end with the dissolution of irradiated fuel at a reprocessing plant. We have subdivided this section into segments on the basis of the characteristics of the fuel material and, separately, on the basis of the environment in which the fuel is found. The picture is simplified for the purposes of discussion; for example, it would be possible to show a

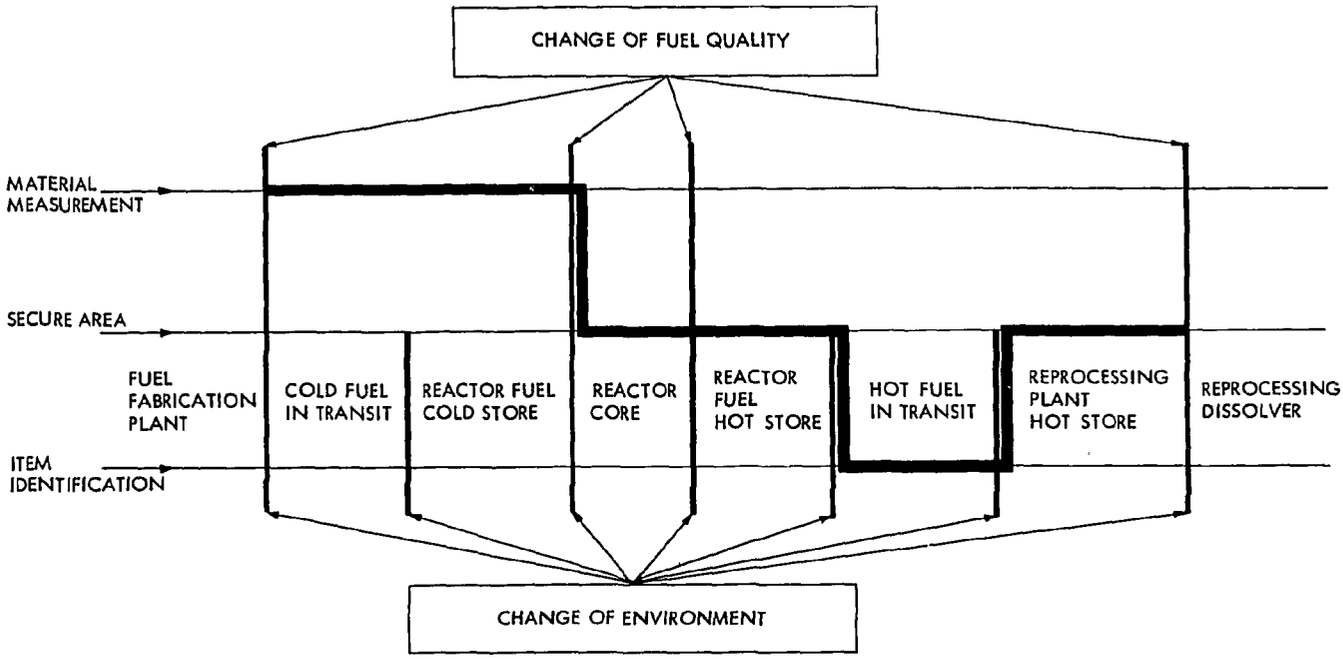


Fig. 1.

segment for the handling operation between cold storage and reactor core.

The accounting record describing the situation in this section will be based on the operator's statement concerning the quantity of nuclear material in each segment (e.g., a Material Balance Statement). The verification of this statement by the Agency may be accomplished by applying one or more of the following categories of safeguards methods:

1. Material measurement
2. Secure area
3. Item identification.

We want to define and discuss each of these terms, since many of the apparent differences of opinion in the field of safeguards methods arise simply from differences in terminology.

1. *Material measurement* involves the determination of element and isotope quantities by direct or indirect measurements and the application of fairly well established principles of materials accountability to determine that all material is accounted for. It should be noted that this method is dependent on the existence of an adequate method for determining material quantities.
2. *Secure area* (physical security) involves a well-known principle which has been widely applied in other industries. We wish to define it somewhat narrowly, however, to avoid confusion with the next category of safeguards methods. The term, secure area (physical security), means that material cannot be removed from the area without detection of the act of removal; i.e., the time between diversion and detection is very small.
3. *Item identification* involves the unique identification of a closed container of material, the integrity of which cannot be violated without detection. When these conditions are established it is possible to ignore the quantity of material contained and to assure against diversion by a periodic check on the location of the container. Note that we *ignore the quantity*, and *periodically check the location*. If the quantity of material is given significance, this method is merely a useful technique for determining an inventory for material balance purposes; if the location check is continuous, it amounts to physical security or a secure area.

Returning to the diagram, we see there are a number of possible combinations of these methods. The optimum combination can be chosen by a cost-effective comparison of the applicable methods after we have identified the available techniques appropriate to each method.

Let us assume that techniques are available for all methods, for example:

1. That there exists an adequate nondestructive assay method for both irradiated and unirradiated fuel, and that a method exists for adequately determining burnup and production through measurements at the reactors

2. That devices permitting unique identification can be attached to the fuel elements upon shipment from the fuel fabrication facility and upon discharge from the reactor, and that the reactor can be sealed with such a device—this, of course, presumes that the safeguarding agency also removes the devices when necessary

3. That physical security can be achieved.

I hardly need point out that few of these assumptions are valid at the present time. The work on development of such methods and techniques is in progress and we will hear about some of it in other papers read at this session.

A cost-effectiveness comparison of the possible methods for each segment permits the selection of the optimum combination of methods across this section of the fuel cycle. The diagram permits this to be expressed graphically. If one moves across this diagram, only along the vertical and horizontal lines, choosing the most cost-effective route, one traces the most appropriate combination of safeguards methods for this section of the fuel cycle. We are confident that such a diagram can be drawn for the entire fuel cycle.

Of course, something is missing from the picture we have drawn. It is the specification of the technical results to be achieved by the application of safeguards to each of the segments of the fuel cycle. This, the second in our list of tasks, is in several ways the most interesting and the most difficult of the three tasks to be accomplished. We will not dwell on the subject, but only mention in passing that it involves not only a determination of the significance of material quantities, and of the uncertainty in those quantities, but also an additional parameter—time.

We have tried to outline the circumstances relating to technical requirements for international safeguards. It is possible now to list certain conclusions that may be drawn concerning these requirements.

1. An attempt should be made to establish internationally acceptable standards and criteria for nuclear materials control.

2. The present efforts to define and describe the fuel cycle from a safeguards viewpoint should be continued until an adequate picture is obtained of the relevant segments.

3. The results to be achieved by the application of safeguards should be quantitatively specified. To say that we wish to prevent diversion of nuclear material is not enough; we should specify how much material, over what period of time, and under what conditions.

4. From the results of systems analysis the relevant segments of the fuel cycle may be defined; for each segment the techniques appropriate to each of the three safeguards methods should be identified and an attempt made to estimate the cost of applying these techniques and quantitatively assessing their effectiveness in achieving the desired results.

The International Atomic Energy Agency has the three major prerequisites necessary to cope successfully with these technical requirements:

1. It has established the "safeguards system" which lays down the basis for safeguards technical development.

2. It has a nucleus of qualified and competent staff to direct the technical development and to utilize its results.

3. Most important, it enjoys the ever-growing scientific and technical support of the Member States in the field of technical development for international safeguards.

NEW METHODS AND TECHNIQUES IN NUCLEAR SAFEGUARDS RESEARCH AND DEVELOPMENT

G. ROBERT KEEPIN *University of California,*
Los Alamos Scientific Laboratory,
P.O. Box 1663, Los Alamos, New Mexico 87544

INTRODUCTION

The implementation of an effective nuclear safeguards and materials management system requires direct physical methods of detecting, identifying, and quantitatively analyzing fissionable materials in various practical configurations containing both fissionable and nonfissionable materials. To be most effective and useful, such assay methods should be nondestructive, rapid, accurate, and capable of being carried out under a wide range of both laboratory and field conditions; e.g., in situ or on-line assay in materials processing plants, or in mobile isotopic assay stations for in-the-field use. The difficulties of representative sampling in chemical (destructive) assay of heterogeneous and complex systems are largely obviated in new nondestructive assay methods, primarily because of their characteristic high penetrability through bulk materials.

PASSIVE AND ACTIVE INTERROGATION METHODS

Nondestructive assay methods can be divided into two main categories: passive assay, and active interrogation. Passive assay methods involve observation of both neutrons [from spontaneous fission or (α, n) reactions—as in ^{240}Pu or ^{239}Pu + light elements] and gamma rays (emitted following α decay—as in ^{239}Pu and ^{235}U) which are uniquely characteristic of individual fission species. The naturally occurring gamma lines having sufficient intensity for passive assay applications are typically a few hundred kilovolts or less in energy, and hence, have limited penetrability through dense materials. For many practical assay problems, this lack of penetrability, together with the absence of suitable passive signatures for many fissionable isotopes, often severely limits the usefulness of passive methods, and one must employ active interrogation techniques that provide the higher penetrability required. From an inspection and surveillance standpoint, it may also be noted that active interrogation techniques are inherently more difficult to subvert or circumvent than are the simpler passive techniques.

Active interrogation employs an external source of highly penetrating neutrons or photons to induce fissions in the material under investigation. Neutron sources are currently used for active interrogation studies at Los Alamos, Karlsruhe, and Trombay, while a linear-accelerator-produced bremsstrahlung beam is used for the same purpose at Gulf General Atomic. Neutron sources were chosen for active interrogation in the Los Alamos safeguards program because of:

1. The high effective penetrability of fast neutrons in nuclear materials generally
2. Sharp, well-defined neutron fission thresholds which provide incisive isotopic discrimination
3. Readily available, inexpensive, compact neutron sources (e.g., D-D, D-T neutron generators, ^{252}Cf, etc.) of the required intensity, simplicity, and reliability for practical assay applications.

In active interrogation methods using either neutron or photon interrogation, quantitative assay is based on detailed observations of the delayed and prompt neutrons and gamma rays from fission. In general, the characteristics of these emanations are quite similar for both neutron- and photon-induced fission, so the major difference between active assay methods using neutron or photon interrogation lies largely in the magnitude and energy variation of the basic fission cross sections involved. In addition, due to the

inherent differences in the nature of the interrogating beams, the techniques used for observing the fission-produced response signals (neutrons or gamma rays) will be correspondingly different.

The delayed regime has the advantages of complete time-separation from the interrogating pulse, and permits the use of simple, inexpensive counting circuitry and a minimum of data reduction equipment. The characteristic differences in yields and kinetic (time-dependent) response of the delayed neutrons from the various fission species provide a unique method for analysis of individual fission isotopes in unknown mixtures of fissionable and nonfissionable materials.[1] The experimental techniques involved in delayed-neutron assay are rapid, nondestructive, and relatively simple and inexpensive. Representative experimental accuracies obtained using delayed-neutron assay methods are: ~2% for relative isotopic abundance determinations in plate-type (e.g., MTR) reactor fuel elements, and ~0.5 to 1% accuracy in determining absolute amounts of fissile materials in small or thin samples (thin fuel plates, pins, rods, pellets, etc.) using appropriate standards.[2] Of course the accuracy of assay (using either delayed- and/or prompt-neutron response) in composite bulk systems is dependent on the relative abundance of the constituent fission isotopes.[3] Based on the experimental accuracies obtained thus far, delayed-neutron assay techniques should soon become competitive with destructive chemical analysis for many practical assay problems in the nuclear industry—especially where representative sampling is difficult or essentially impossible.[2]

The basic data required for the practical application of delayed-neutron assay techniques include absolute delayed-neutron yields (per fission) as a function of incident-neutron energy. Absolute delayed-neutron yield measurements or delayed-to-prompt-neutron yield ratios provide a basis for effective discrimination between the thermally fissioning species (notably ^{239}Pu and ^{235}U). Recent measurements[4] of the absolute yields of delayed neutrons from 3.1 and 14.9 MeV neutron-induced fission have been carried out for all of the major fission species (^{233}U, ^{235}U, ^{238}U, ^{239}Pu, and ^{232}Th). These measurements show clearly that the delayed-neutron yield per fission of every isotope studied decreases significantly in going from 3.1 to 14.9 MeV fission—a result which is expected from the known behavior of fission mass and charge distribution, but which is in direct contrast to previous measurements at other laboratories in both the USA and the USSR.

In addition to delayed neutrons, delayed-fission gamma rays and prompt-fission neutrons and gamma rays, as well as capture gamma rays, may be used to provide very useful characteristic signatures of individual fission species. To this end, a number of measurements are being carried out at different laboratories on fundamental neutron and gamma-ray yield and energy characteristics; also, practical response signatures are being investigated and catalogued for various types of fissionable materials using both neutron and photon interrogation techniques.

As already noted, capture gamma rays in the different fissionable species should provide unique signatures for isotopic assay. The differences in neutron binding energies of different isotopes produce corresponding differences in the energy spectra of the capture gamma rays [e.g., as seen by a Ge(Li) spectrometer]. To suppress the large background of prompt fission gammas, one may count the desired capture gamma rays in anticoincidence with fission events (i.e., prompt-fission neutrons observed with high-efficiency detectors). The feasibility of this anticoincidence scheme for studying capture gamma signatures has been investigated using a reactor neutron source at Karlsruhe.[5]

To acquire basic data for photon interrogation techniques, the yields of prompt and delayed neutrons from photofission have been measured at Gulf General Atomic[6] with an electron linear accelerator over the energy range from near the various photofission thresholds (in the neighborhood of 5.5 MeV) up to ~8 MeV; i.e., below the photoneutron thresholds of most other materials (notable exceptions being ^{2}H, ^{6}Li, ^{9}Be, ^{13}C, ^{118}Sn, and ^{207}Pb). These fundamental data have shown the basic isotopic yield differences among the major fission species as a function of bremsstrahlung end-point energy.

Delayed gamma rays from fission may either provide unique isotopic signatures or complement the delayed-neutron assay methods already developed, depending on the specific gamma-ray characteristics measured. The gross delayed gamma-ray intensity (total photons/fission-sec) over the early time range from ~0.1 to 100 sec after fission depends on the isotope undergoing fission in much the same way as the delayed-neutron emission. Of particular interest for active interrogation would be expected large differences in the yields of high-energy (penetrating) gamma rays from ^{235}U(n, F) and ^{239}Pu(n, F), which can be resolved with high resolution Ge(Li) detectors. In this case, interrogation with subthreshold neutrons could be used to assay ^{239}Pu and ^{235}U separately, in the presence of a large quantity of ^{238}U, as occurs for example in power reactor fuels.

The combined measurement of both delayed-neutron and gamma-ray response to active interrogation should prove particularly useful for assay

applications in which the environment of the nuclear material is unknown; viz., because of large differences in the attenuation characteristics of neutrons and gamma rays, such combined data can give important information about the materials surrounding, or interspersed with, the fissionable material under investigation.

The characteristic resonance structure in the neutron fission cross sections of the different fissile isotopes offers additional methods for nondestructive assay. This structure is most pronounced in the neutron energy range from 0.3 eV to roughly 10 keV, where the resonance peaks are typically orders of magnitude larger than the valleys. These characteristics have been applied in two rather different techniques for nondestructive assay. The first technique employs a slowing-down-time spectrometer[7,8] which is based on fission cross-section variations as a function of neutron slowing-down time (i.e., neutron energy) in a large lead block. This method provides useful discrimination between ^{235}U and ^{239}Pu, particularly in the lower-energy resonance region ($\lesssim$100 eV), provided self-shielding effects are not too large. The low-energy region is useful in assay of low-enrichment fuels (i.e., few percent enrichment, as in light-water reactor fuels), while the upper end of the resonance region and the unresolved resonance region must be used for highly enriched fuels, as in fast breeder reactors.

The second technique[9] employs a steady-state beam of epithermal neutrons which is passed through a sample of fissile material, and the transmitted beam is then monitored with thin-foil fission detectors containing the same fissile isotope(s) as the sample being assayed. This "resonance self-indication" technique is especially sensitive for ^{239}Pu assay because of the large (3000 b) resonance in ^{239}Pu at 0.3 eV. Another direct result of these resonance self-indication studies has been an experimental delineation of the effectiveness of ^{239}Pu, ^{235}U, and ^{233}U neutron beam filters for enhancing fissile isotope discrimination. The use of thick (~ 150 mil) beam filters can clearly be combined with other characteristic signatures such as neutron-induced hard gamma lines from fission and neutron-capture gamma rays (e.g., to provide increased discrimination between ^{235}U and ^{238}Pu). In this connection it is noteworthy that oftentimes a suitable combination of assay techniques and signatures may be used to provide an extremely effective "hybrid" method, especially tailored for a particular assay application.

Other techniques for nuclear material assay include calorimetry (for Pu analysis based on alpha heating), x-ray fluorescence analysis, both of which are being developed at KFZ, Karlsruhe, Germany, and the MIST (Minor Isotopes for Safeguards Technology) Program being undertaken by the U. S. Arms Control and Disarmament Agency.

PARALLEL COMPUTATIONAL PROGRAMS

For maximum efficiency and rate-of-progress in safeguards research and development work, it is essential to parallel experimental programs with theoretical analyses and computer simulation of delayed-neutron and gamma-ray response of various composite, practical systems to active interrogation. Monte Carlo and SN transport codes are being used at Los Alamos, for example, both for analysis and to provide guidance for experiment, and help determine the precision of basic data required to achieve a specified accuracy in isotope assay applications. Such calculations have proved extremely valuable in pointing the way toward the most fruitful lines of approach in safeguards research and development.

An excellent case in point is the neutron transport calculation of delayed-neutron response from small amounts of fissile material interspersed in large amounts of various scrap materials.[10] These parametric computations, summarized in Figs. 1 and 2, point up the very useful general re-

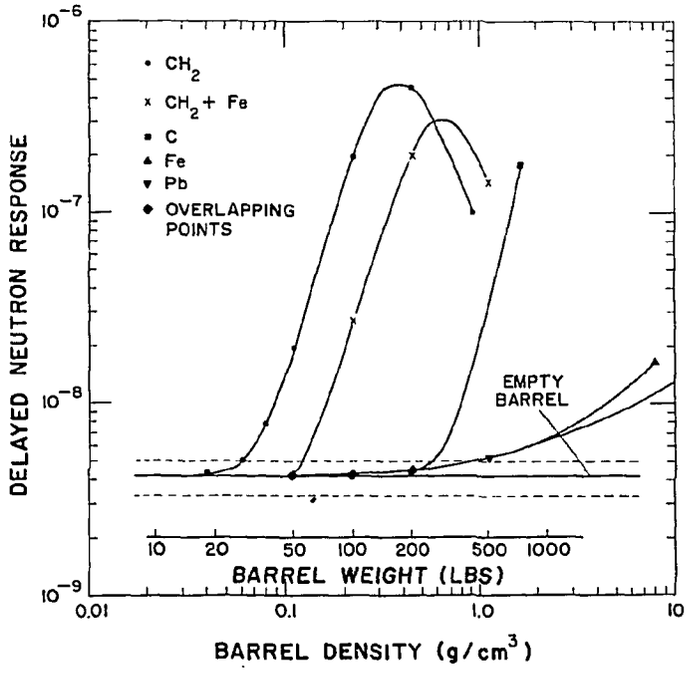


Fig. 1. Delayed-neutron response of a simulated scrap barrel containing 1 g of ^{235}U at 15 cm in representative moderator or matrix materials—14-MeV source. The dashed lines indicate empty-barrel (no moderator) response ±20%. (The ±20% error span represents reasonable accuracy limits for practical scrap assay.)

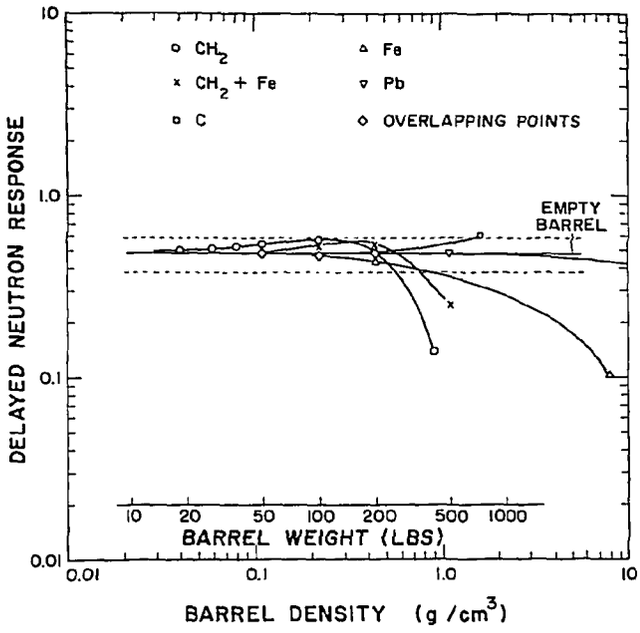


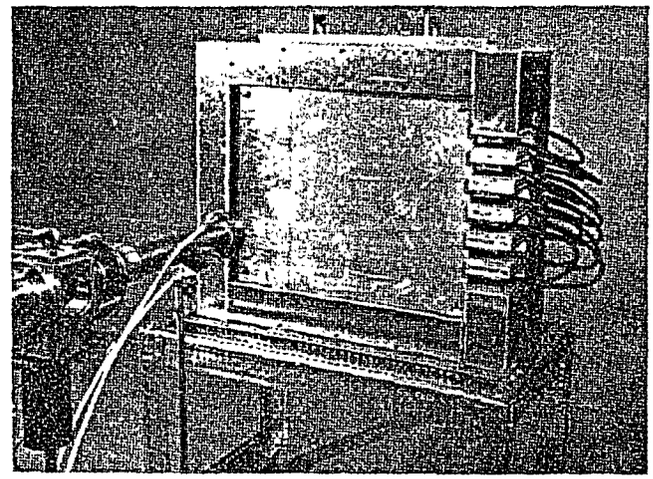
Fig. 2. Delayed-neutron response of a simulated 55-gal scrap barrel surrounded by a 2.5-cm polyethylene reflector. The response has been normalized according to the new "add-a-gram" differential response method.[10] The dashed lines indicate empty-barrel (no matrix material) response ± 20% (the ± 20% error span being typical of acceptable accuracy limits for practical scrap assay).

sult that barrels (55-gal size, commonly used for scrap storage) containing almost any common scrap material other than hydrogenous compounds, are effectively transparent to fast-neutron interrogation and delayed-neutron assay (Fig. 1). Assay for fissile material is thus essentially independent of composition of the matrix in which the fissile material is interspersed. The case of a hydrogenous matrix material is the most difficult for neutron assay applications, but even here calculations have shown that very effective fissile-material assay can be carried out by the use of a moderator-reflector together with appropriate ("add-a-gram") normalization of delayed-neutron response.[10] The resulting dramatic improvement in delayed-neutron "flat" response for bulk hydrogenous systems is clearly illustrated by comparison of Figs. 1 and 2.

INSTRUMENTATION: DETECTORS AND SOURCES

In general, detection systems for nuclear safeguards are adapted from state-of-the-art detector technology to meet the special requirements of safeguards assay problems. In many cases one seeks high-efficiency detectors (e.g., for fast-neutron counting), and in other instances, high-resolution detectors [e.g., Ge(Li) for gamma spectrometry] are required.

For active interrogation applications it is desirable to have the highest possible detection efficiency for both neutrons and gamma rays to minimize the strength of the interrogating neutron (or photon) source and thus reduce the associated



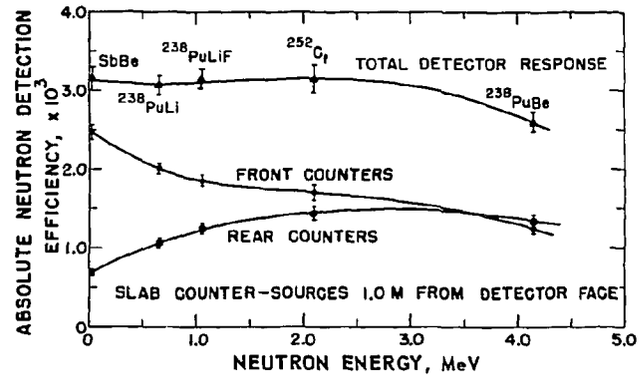


Fig. 3. High-Efficiency Neutron Detectors. Specialized detectors have been developed for nuclear safeguards inspection and surveillance applications; one such detector, the "N-6 slab detector," is pictured here near the target of the "Accelerator I" mobile neutron source. The graph shows the high efficiency of the detector as a function of neutron energy. The indicated "flat" efficiency-vs-energy characteristic for the sum of the front and back counter banks (i.e., total detector efficiency) is necessary for most applications involving assay of large complex arrays of nuclear material such as fuel elements, nuclear devices, scrap, etc.

shielding requirements, personnel hazards, etc. An example of a recently developed high-efficiency neutron detector having a "flat" (energy-independent) response is shown in Fig. 3.

The sources available for neutron interrogation studies are small, inexpensive, high-yield neutron generators, accelerators, and radioactive sources (e.g., ^{252}Cf), as well as reactor neutron beams. A typical compact, mobile neutron generator is shown in Fig. 4. For all such sources, considerable variability of neutron bombarding energy may be obtained by:

1. Using either the D-D reaction (2.5-MeV neutrons) or the D-T reaction (14-MeV neutrons)
2. Degrading or "spectrum tailoring" of source neutrons by surrounding the target

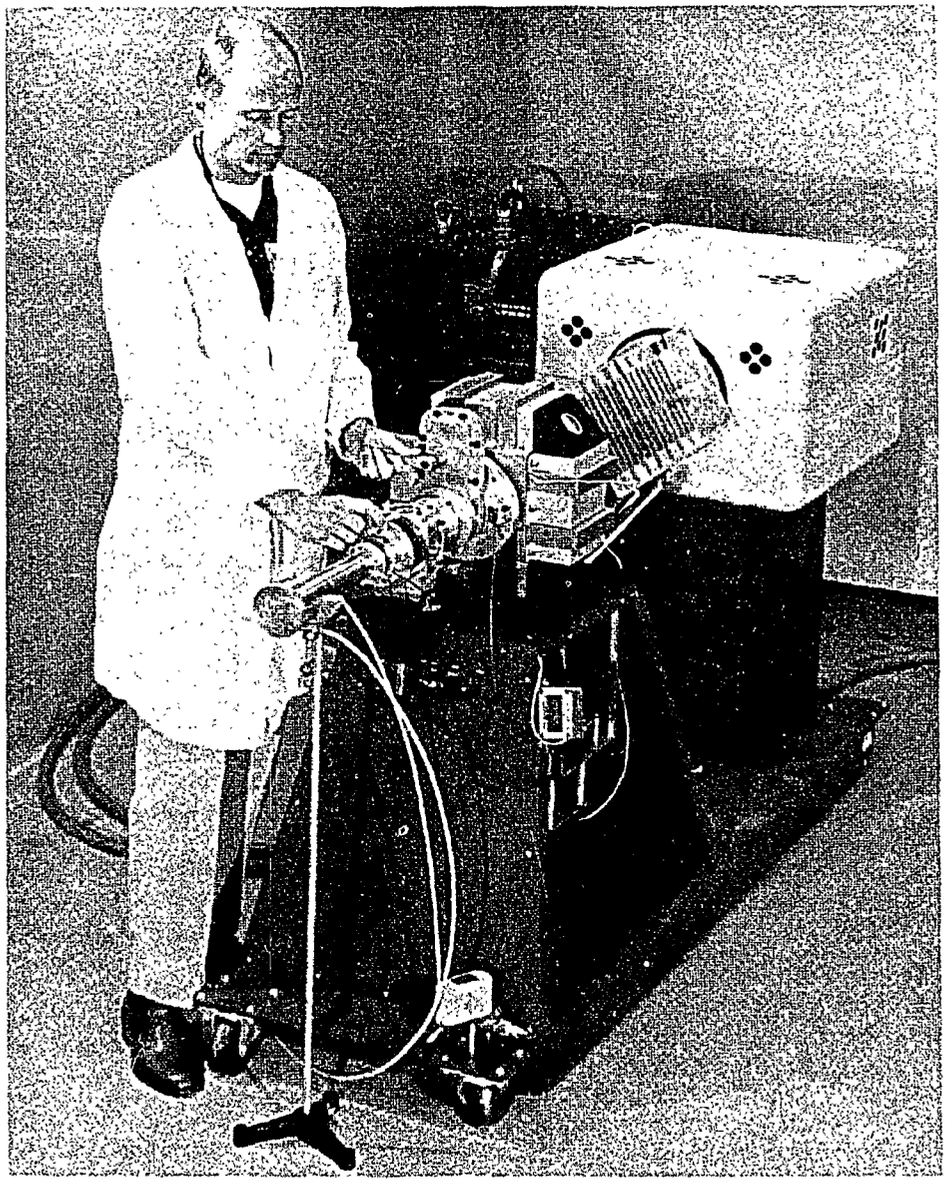


Fig. 4. Accelerator I—A compact, high-intensity source for neutron interrogation and nondestructive assay. (Accelerator potential: 150 kV; maximum beam current: 3.5 mA; beam pulsing by pre- and post-deflection; beam modulation by gating the RF ion source.) Maximum source intensity is $\sim 4 \times 10^{11}$ D-T n/sec; the source can be gated on and off as required for optimum neutron interrogation applications. Two new quick-change target holders have been adapted to Accelerator I for fast, convenient target replacement in laboratory and field installations.

with appropriate scattering and/or moderating materials (e.g., tungsten, beryllium, and polyethylene, in appropriate thicknesses,[11] can shift nearly 90% of the 14-MeV source neutrons to energies below the ^{238}U fission threshold).

Monoenergetic neutrons of variable energies (subthreshold and superthreshold) are provided by a Van de Graaff accelerator as the interrogating neutron source. Neutron energy variability (whether by spectrum tailoring or directly with a Van de Graaff source) becomes important in many practical assay applications. For example, in low-enrichment power reactor fuel elements, the "special nuclear material" (i.e., fissile species such as ^{235}U, ^{239}Pu, or ^{233}U) may be masked by large quantities of fertile material (e.g., ^{238}U or ^{232}Th); however, the relative response of the sparse fissile material can be greatly enhanced simply by shifting the energy of the primary source neutrons to the range below the threshold energy for fission of ^{238}U or ^{232}Th. Thus by the use of "subthreshold" fast-neutron interrogation, good penetrability is preserved and, at the same time, relatively small quantities of fissile material can be detected in the presence of much larger quantities of fertile material; this capability is clearly of practical importance in the quantitative isotopic assay of power-reactor fuel elements, rods, plates, pins, pellets, scrap, etc.

As an alternative to accelerator and generator

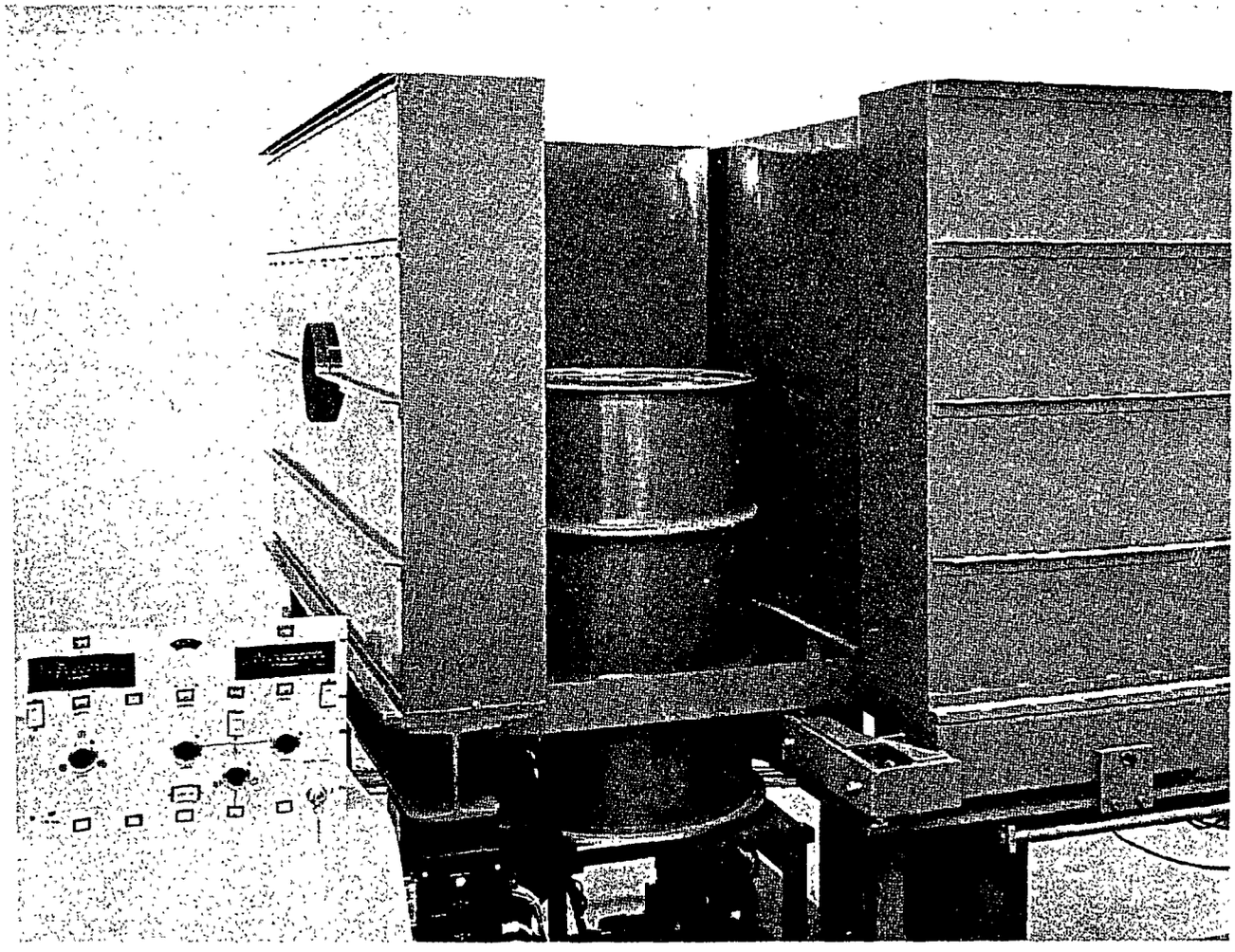


Fig. 5. Apparatus for assaying barrels of nuclear waste (photofission interrogation). The rectangular tanks contain oil (for neutron moderation) and BF_3 thermal-neutron detectors for prompt and delayed neutrons. There are also gamma detectors in the tanks. One tank is easily removable (on tracks as shown) for ease in barrel insertion. The collimated photon beam passes through the reentrant holes in the tanks. The barrel is rotated and translated (vertically and horizontally) across the photon beam to achieve uniform sensitivity to photofissionable materials within the barrel.

neutron sources, a ^{252}Cf spontaneous fission source of suitable intensity (e.g., 3 mg, giving 10^{10} n/sec) offers a very practical, maintenance-free neutron source in extremely compact form, and having the convenient, highly penetrating, fission-spectrum neutron energy distribution. A ^{252}Cf neutron-interrogation assay system is presently under development at LASL.[12]

Among the several types of electron accelerators available for safeguards applications, utilizing the photon interrogation method, are traveling-wave linear accelerators, betatrons, microtrons, and flash x-ray devices. Of these, the traveling-wave electron accelerator ("linac") appears to be the most promising.[13,16] Small commercial accelerators of this type have been developed extensively for radiotherapy treatment of cancer and for radiography. Safeguards research on photo-induced reactions performed at Gulf General Atomic has been oriented toward implementation of a machine of this type having only one section and a continuously variable energy up to about 10 MeV. An apparatus for assaying barrels of nuclear waste using photofission interrogation techniques has been built at Gulf General Atomic; this unit is shown in Fig. 5.

PRACTICAL ASSAY APPLICATIONS

The practical applications of nondestructive assay techniques in the nuclear industry may be divided into three general categories:

1. Reactor fuels (e.g., fabricated fuel rods, plates, disks, pins, feed materials, molten salt fuels).
2. Small test samples (e.g., fissionable material standards, prototype and experimental

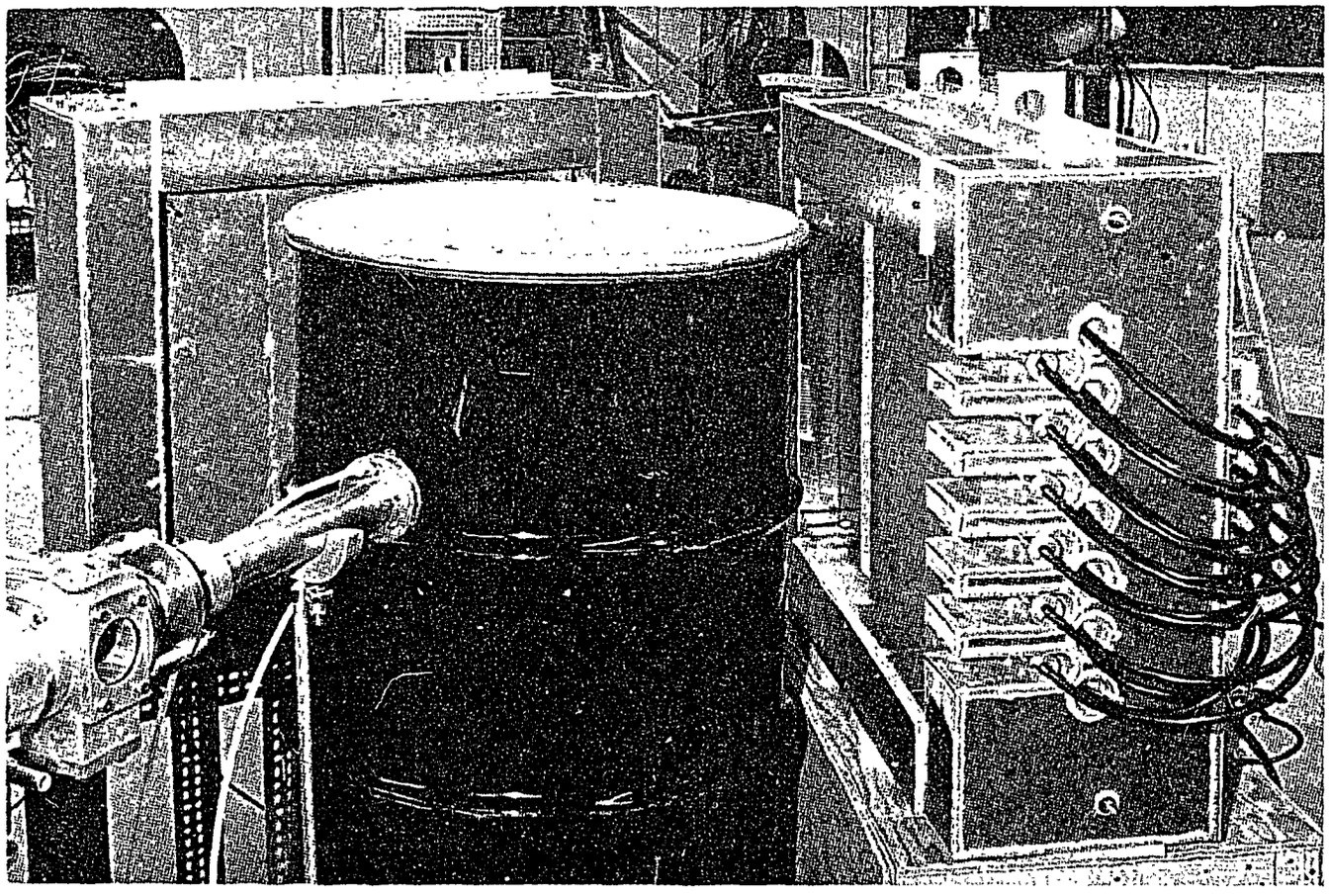


Fig. 6. Experimental arrangement for nondestructive assay of fissile material in scrap barrels using neutron interrogation. The target assembly of the compact, intense neutron source, "Accelerator I," is seen at the left. Two high-efficiency neutron detectors (N-6 slab detectors) can be seen around the barrel being assayed.

materials, compounds, mixtures, process-line samples in various physical and chemical forms).

3. Scrap (e.g., from isotope separation, fuel reprocessing and fuel fabrication plants).

An example of a practical assay application using neutron interrogation is presented in Fig. 6 which shows an experimental arrangement for assay of fissile material scrap in a 55-gal storage barrel. The target assembly of the "Accelerator I" neutron source is seen at the left, and two high-efficiency neutron detectors can be seen around the barrel being assayed.

As already noted, typical assay accuracies using delayed-neutron response techniques are presently ~2% for relative isotopic abundance determination in small samples, thin plates, fuel pins, rods, etc., and 0.5 to 1% accuracy in determining absolute amounts of fissile materials in small or thin samples using appropriate standards. Comparable accuracy for the assay of ^{235}U-^{238}U composite systems by photon interrogation has been indicated in the Gulf General Atomic program using a linear accelerator source.[3,6] Whether using neutron or photon interrogation, some nominal reduction in assay accuracy will be obtained for larger samples where one must take account of the penetrabilities of the interrogating beam (neutrons or photons) as well as the response signal (fission neutrons and/or gammas). The actual assay accuracy in practical situations will depend, of course, on details of the configuration and composition of the particular system under investigation.

Direct empirical approaches to practical inspection and surveillance problems include measuring and cataloguing of delayed-neutron and gamma-ray response signatures for various types of systems and their direct application in development of field-tested isotopic assay systems, optimized for a wide variety of practical assay problems in safeguards and nuclear materials inspection and accountability. Needless to say, before a given nondestructive assay method is adopted for a particular field application, it should be fully proved-in by detailed intercomparison with standard chemical (destructive) assay techniques. It may be noted here that the U. S. Atomic Energy Commission has already initiated an extensive program of detailed intercomparison between chemical assay and delayed-neutron assay techniques. The availability of proven, on-line nondestructive assay systems can be expected to have far-reaching implications, not only for safeguards and worldwide NPT inspection, but also as a means of implementing efficient, safe, and economic management of nuclear materials throughout the nuclear energy industry.

ACKNOWLEDGMENTS

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SAFEGUARDS SYSTEM STUDIES AND FUEL CYCLE ANALYSIS

W. HÄFELE, W. GMELIN, D. GUPTA, J. LARISSE,* and H. WINTER *Institut für Angewandte Reaktorphysik Kernforschungszentrum Karlsruhe, Karlsruhe, Germany*

INTRODUCTION

In the early days of safeguarding, isolated nuclear facilities, such as research reactors and the nuclear material in them, were the concern of safeguards. In the present and future era of commercial nuclear power generation, concern is with the nuclear material flow through the various nuclear facilities in a fuel cycle, and the principle of effectively safeguarding the flow of fissile material by use of instruments and other techniques at certain strategic points.

To assess the requirements of such a safeguards system, a detailed systems analysis is necessary. Besides establishing quantifiable criteria for a safeguards system, such an analysis enables one to set the target values of instruments and methods as well as other objectives of development. Extensive experiments in industrial-scale facilities are also required to demonstrate the feasibility of such a safeguards system.

This paper describes the various phases of activities carried out in this area at the Karlsruhe Research Center, and deals at some length with the system analytical approach we followed to establish a safeguards system based on the above-mentioned principle. The paper also describes in detail the safeguards exercise carried out in the plutonium fabrication plant ALKEM at Karlsruhe. The method of assessing the relative importance of the chosen strategic points, preparation of material balance, and establishment of different types of statements that can be made by an inspection authority, have been discussed. The possibility of estimating the dynamic behavior of the process inventory for a given plant layout has been indicated. It has been shown that this principle can also be effectively realized in existing plants of the ALKEM type.

*On delegation from EURATOM

HISTORICAL BACKGROUND

If one casts a glance over the past two and one-half decades of nuclear energy development, one can distinguish between three different phases.[1,2] The first phase which lasted from 1942 to 1953, was mostly a military-oriented phase. The reactors installed during this phase, for example at Hanford and Windscale, were mainly plutonium-producing reactors. Although fabrication and reprocessing facilities were available for these reactors, they did not operate under commercial aspects. The development of nuclear energy during this phase took place under the shadow of atomic explosions at Nagasaki and Hiroshima. All the major activities in this field were governed by the assumption that a 100% effective and technically feasible control system for the peaceful sector had to be in existence before the nuclear energy could be used for civilian purposes. The Baruch plan and the Atomic Energy Act of 1946 in the USA, which prevented dissemination of any nuclear information or supply of nuclear materials to other countries, were the outcome of this era.

The second phase was initiated by the "Atoms for Peace" program of President Eisenhower in 1953 followed by the passing of the Atomic Energy Act of 1954.[3] This phase was characterized by a worldwide exchange of nuclear information in the peaceful sector, supply of limited amounts of nuclear materials and research reactors to different countries, and the establishment of the International Atomic Energy Agency (IAEA). The European Atomic Energy Community (EURATOM) was also established during this phase. The internationally known safeguard systems of the IAEA and EURATOM were worked out during this period and strongly reflect the characteristic features of the nuclear energy development during this period.

The third phase began in 1963 with the Oyster

Creek event--large-scale commercial use of nuclear energy including full-scale industrial competition. It is also during this phase that the commercial use of all the steps of a nuclear fuel cycle, namely, reprocessing, refabricating, and possibly isotope separation, becomes essential so that the nuclear power stations can produce power economically. The third phase is rapidly expanding to many countries of the world, and the amount of fissionable material which is expected to be required and produced in the civilian sector will be higher by several orders of magnitude than that in the second phase. Any safeguard system which has to be applied during this phase has to be oriented to the conditions pertinent to this phase.

THE PRINCIPLE OF MODERN SAFEGUARDS IN THE FIELD OF PEACEFUL APPLICATION OF NUCLEAR ENERGY

It is vital to ensure that the peaceful energy[4] does not proliferate into the domain of nuclear weapons and that it is solely used for fulfilling man's hope for peace and progress. If one can ensure that all fissionable material, required and produced in the peaceful sector, also remains in this sector, such proliferation cannot take place. Therefore, the only and specific objection of a modern and properly designed safeguard system is to ensure that virtually all fissionable material, which is used in the civilian domain, remains there. Logically, it cannot be the objective of a modern and properly designed safeguard system to control the peaceful application of nuclear energy as such.

If the flow of fissionable material in the civil domain could be entirely and effectively contained in this domain, this would be the only required safeguards measure. In such a case it would be irrelevant to know the amount and the quantity of the fissile material. Therefore, it must be the first safeguards measure of a modern safeguards system to ensure that such a containment measure is realized wherever that is possible. It is important to realize that most of the nuclear facilities require containments of different types because of the requirements inherent in the handling of nuclear material. The reactor vessel of a nuclear power station, the hot cells in a reprocessing plant, and the glove boxes in a fabrication facility are typical examples of such containments.

In practice it might not always be possible to realize a fully effective containment. It is therefore necessary to introduce a second safeguards measure. This measure consists of safeguarding the flow of fissionable material throughout the whole fuel cycle. This can best be executed at certain strategic points. The first safeguards measure, namely, the containment, provides for a kind of conservation of mass flow and it is not necessary to follow the flow everywhere inside a facility. Although a detailed systems analysis is required to determine the location and the number of such strategic points, the entrance and the exits of all nuclear facilities appear to be the more important of these strategic points. If all the safeguards activities are confined to these points, it will suit the commercial nature of the competitive nuclear industry of the third phase in an ideal manner; under such a condition, the industrially sensitive parts of a nuclear facility would then remain untouched.

In any commercial-scale nuclear facility, a process inventory of fissile material is always required to enable the plant to operate under equilibrium conditions. This process inventory cannot be measured directly by measuring the throughput of the fissile material alone, and can only be calculated from the difference between the input and the output flows. If the process inventory would have been negligible compared to the throughput over a given period of time, the first two safeguards measures would have been sufficient. However, this condition is normally not fulfilled in large, industrial-scale nuclear plants, and can only be approximately met with a very large number of strategic points inside a facility. Therefore, to establish a complete material balance, a third safeguards measure has to be introduced; namely, the inventory taking. As will be shown later, there are several ways in which the process inventory can be estimated and established independently of throughput measurement. One of the ways is washing out the plant. In such a case the process inventory is temporarily transformed into a flow and measured at one of the strategic points. Such an inventory procedure should, however, to the greatest possible extent, coincide with one of the operational washouts of a plant to make this measure as unintrusive as possible. Normally, inventory taking, once or twice a year, appears feasible.

The above considerations lead to the following scheme for a modern safeguards system:

1. The objective of a modern safeguards system is to significantly reduce the possibility of diversion of fissionable material from the domain of peaceful use of nuclear energy.
2. It is the fissionable material in the domain of peaceful use of nuclear energy and not the peaceful use of atomic energy, as such, that must be subject to safeguards, which is in view of the ultimate purpose of such safeguards; namely, to prevent the illegal manu-

facturing of nuclear weapons—an indirect approach.

3. The design of a modern safeguards system is governed by a quantified criterion of the following type:

   "The requirements of safeguards are met, if with x o/o confidence level the balance is closed within y o/o."

   Such a criterion can be established with the help of an extended systems analysis and cuts the open-endedness.

4. The first safeguards measure is to materialize the principle of containing the fissionable material to the greatest possible extent. Therefore this first safeguards measure covers, among other things: real containments (buildings) of principal nuclear facilities, gate controls, waste control, and safing and sealing, particularly in the case of transportation.

5. The second safeguards measure is to measure the flow of fissionable material at a finite number of strategic points. The assessment of strategic points, their distance and, therefore, the holdup between two of these strategic points and their required accuracy of flow measurement, shall be such that the quantified criterion in (3) is met. In particular, it will be the amount and the constancy of the holdup between two strategic points that have to be taken into account when this assessment is made.

6. The third safeguards measure is inventory taking—intentionally a rare event—which should coincide to the largest possible extent with the normally expected regular washouts. The type of inventory taking shall be at the discretion of the operator of a principal nuclear facility, provided the accuracy of the chosen type of inventory taking is in conformity with the purpose of that inventory taking.

7. Inspectors shall not interfere with the operation of a principal nuclear facility and shall have access only to the strategic points. If in the course of safeguards experience it can be demonstrated that another area of a principal nuclear facility has to be touched also, this other area shall be identified as another strategic point by proper agreement between the involved parties or authorities.

8. Design details of a principal nuclear facility are of relevance for safeguards purposes only insofar as certain ground rules for the general layout of the building must be implemented. These ground rules are there to make the containment function of the building obvious and to identify, in advance, the strategic points and enhance their efficiency.

9. On a somewhat larger time scale, tamperproof instruments for measuring the flow of fissionable material at the strategic points shall be developed and their readings shall be processed by a suitable automatic data processing system. As these instruments are developed, they shall gradually replace the safeguard inspectors.

IMPLEMENTATION OF THE PRINCIPLE

Work in three major areas is required to implement the principle of the proposed safeguards system in commercially operating nuclear facilities:

1. Systems analysis
2. Containment studies
3. Development of instruments.

A detailed research and development program has been worked out at Karlsruhe[5] in which the different types of activities being performed under the three major areas have been detailed. These activities and their time scales are indicated in Tables I, II, and III.

Systems Analysis

Emphasis has been laid on model simulation, development of statements, and cost-effectiveness study. The results of the first two areas of activities are expected to be available by the middle of 1969. Some more time would be required to obtain the results of the cost-effectiveness studies; however, they should be available not later than the end of 1971 (see Table I).

Containment

The role of the first measure of safeguards in the proposed scheme of safeguards, i.e., containment, cannot be overemphasized. Close collaboration with the operators of different nuclear facilities in the fuel cycle is required to determine the optimum way of laying out a plant so the containment requirements can be fulfilled to the maximum possible extent without significantly affecting the economics of the plant. The first concrete results of these studies should be available by the end of 1969. The development work on sealing and identification of fuel subassemblies should yield definite results by the middle of 1970.

TABLE I

Activities and Time Scale for the Systems Analysis

Activities	Time Scale
1. Model simulation	
(a) Establishment of objectives for safeguards methods	Middle of 1969
(b) Number, location, and relative importance of strategic points	
(c) Process inventory analysis	
(d) Effective use of statistical, probabilistic, and other similar methods	
2. Development of statements	
(a) For the safeguard authority	
(b) For the operators of nuclear facilities	
(c) For safeguards systems designer	
(d) On effectiveness of safeguard system	
3. Cost effectiveness	
(a) Use of operators data	End of 1969
(b) Cost functions for measuring accuracies, containment, and other safeguards measures	End of 1971
(c) Optimization of the whole safeguards system	

TABLE II

Activities and Time Scale for Containment Studies

Activities	Time Scale
1. Containment of nuclear facilities	
(a) Nuclear reactors	End of 1969
(b) Fabrication plants	
(c) Reprocessing plants	
2. Containment of fissionable material	
(a) Sealing and identification of fuel subassemblies	Middle of 1970
(b) Sealing of containers and transport casks	
3. Tamperproof storage and transmission of safeguards information	End of 1971

The ultimate goal of the containment measure as well as the use of instruments is the tamperproof storage and transmission of the information obtained from these measures. However, this goal is not expected to be reached before 1971 (see Table II).

Development of Instruments

Development of instruments is required mainly to implement the second and third safeguards measures; namely, the measurement of the fissile material throughput and the process inventory at the strategic points. In following the flow of fissile material in a fuel cycle (Fig. 1), it becomes evident that two different types of measuring methods are required. After the fissile material is filled in the fuel pins at the final stage of a fabrication plant, it is no longer available in a directly accessible form and remains in this inaccessible and quantified form during its passage through the reactor and until the irradiated subassemblies containing the pins are destroyed in the dissolver stage of a reprocessing plant. At the strategic points in this part of the fuel cycle, indirect, nondestructive methods are required to determine the fissile material flow. On the other hand, direct methods of measurements can be used at strategic points in the rest of the fuel cycle. Some of the indirect and direct methods, e.g., calorimetry for Pu-containing pins and x-ray

TABLE III

Development of Instruments and Their Time Scale

Instruments	Time Scale
1. Indirect methods	
(a) Calorimeter (Pu-containing fuel pins)	End of 1969
(b) Calorimeter (Pu-containing subassemblies)	End of 1970
(c) Slowing-down-time spectrometer (^{2,45}U-containing fuel pins)	Middle of 1970
(d) Slowing-down-time spectrometer (U and/or Pu-containing fuel pins)	Middle of 1972
(e) Methods based on γ-spectroscopy with induced reactions (U and/or Pu-containing fuel pins)	Middle of 1972
(f) Neutron dose measurement (Pu-containing wastes)	Middle of 1970
(g) Delayed neutron (fissile-material-containing wastes)	Middle of 1970
2. Direct methods	
(a) X-ray fluorescence for β, γ active samples (U + Pu)	End of 1969
(b) Isotope dilution by mass spectrometry (U + Pu isotopes)	Middle of 1970
(c) α spectroscopy (^{238}Pu)	End of 1969
(d) Neutron activation of homogeneous waste solutions	Middle of 1969
(e) Improvement of standard methods	Middle of 1970
3. Other methods	
(a) Distance-cum-load measuring instruments for cranes, fueling machines, etc.	End of 1969
(b) Activity measuring instruments for reactor bay, storage pond for active subassemblies, etc.	End of 1969
(c) Control of personnel and material for concealed fissionable material (Pu)	Middle of 1969

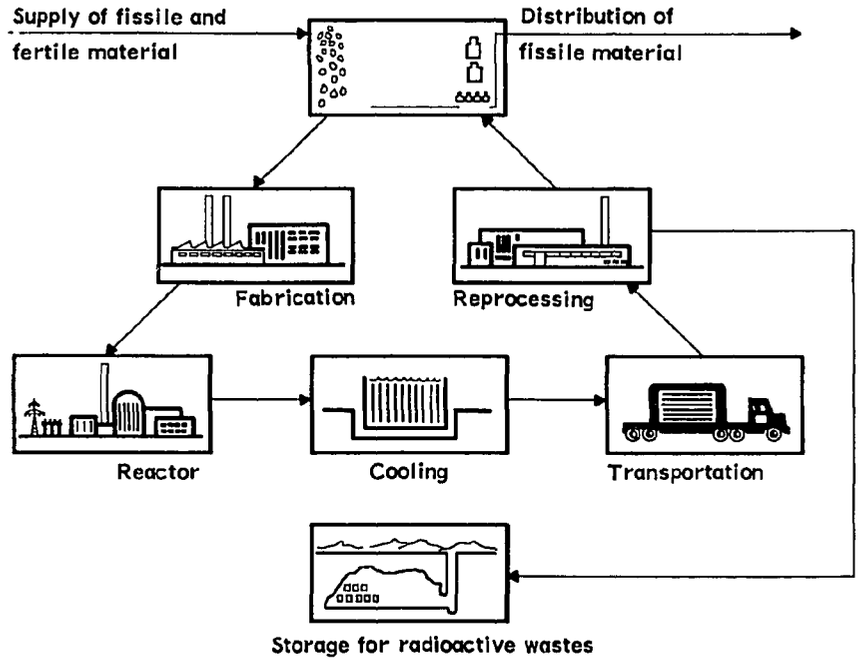

Fig. 1. Fuel cycle industries for the production of nuclear power.

fluorescence for the dissolver solution in a reprocessing plant, should be available in their final industrial form by the end of 1969. Others are expected to be available during the period 1970 to 1972.

Some other instruments, not directly required for the fissile-material flow measurement, but for implementing the containment measure, are listed under point 3 of Table III. All these instruments are expected to be available by the end of 1969.

Experimental Work

In the R&D program at Karlsruhe, one of the important phases of activities is the experimental testing of system analytical results and instruments in industrial-scale nuclear facilities. The main objectives of such testing are summarized below:

Objectives of Experimental Testing. Experimental testing will cover:

1. The proposed safeguards system in existing plants.
2. The validity of system analytical results.
3. The measuring and containment methods during and after development.
4. The final safeguards system with instruments and other techniques in individual nuclear facilities and in the whole fuel cycle.

Implementation at the Karlsruhe Research Center

The principle of the safeguards system as previously discussed can be realized in an effective manner and within reasonable time scales only if the required research, development, and testing program can be carried out and coordinated in an optimum manner. The basic conditions required for the fulfillment of such an objective are present at the Karlsruhe Research Center. Besides the fact that sufficient experience and research facilities in the required fields are available at the Center, there is also a complete, industrial-scale fuel cycle in which the research and system analytical results can be tested without any serious time lag. A rapid flow, exchange, and feedback of information is therefore possible to attain the objectives within a present time schedule. Close collaboration also exists between the Center and international control organizations—essential for the actual implementation of any safeguards system.

RESULTS OF RESEARCH, DEVELOPMENT, AND TESTING

Active work within the framework of the fissile-material control project was started at Karlsruhe in August 1967. Some interesting results obtained during the last year will be discussed.

Systems Analysis

Criteria for Measuring Methods. During the course of the system analytical investigation, it became evident that intensive effort would be required to develop indirect methods for determining fissile material content in fresh and unirradiated fuel pins and subassemblies at the exit of a fabrication plant or at the entrance of a reactor. In the event it would have been absolutely essential to measure the fissile material content in irradiated subassemblies, much larger effort would be required. Fortunately, irradiated subassemblies from most of the presently known reactors are reprocessed so the fissionable material content of these subassemblies can be directly measured there. The direct methods, which are already known, have fairly high accuracies. Therefore, indirect methods for irradiated subassemblies have been allocated a low priority in the Karlsruhe program, and criteria for only freshly fabricated fuel pins or subassemblies have been established. The more important criteria have been shown in Table IV. According to this,[6] the overall measuring accuracy for plutonium-containing fuel pins should be better than ±0.4%, and that for uranium better than ±1.6%. These accuracies are based on throughputs in fabrication plants during the early seventies in Germany.[13-15] The accuracies are so chosen that with such throughputs, the integrated uncertainties in the throughputs reach a value of 10 effective kilograms of plutonium in one year's time.[16]

Relative Importance of Strategic Points. A number of nuclear facilities, like reprocessing plants with different capacities,[7] and fabrication plants with plutonium-containing fuel, were simulated to assess the relative importance of the strategic points. The range of uncertainties in the integrated amount of fissionable material obtained at each of these strategic points after a given amount of fuel has been processed, was taken for the time being as an index for assessing the relative importance of these points. The randomness of the measured results was simulated by using a random number generator. The location and number of the strategic points in the reprocessing plant are shown in Fig. 2, and the results on uncertainties are summarized[7] in Table V. The range of uncertainties at a strategic point is a function of the integrated amount passing through this point, the accuracy of measurement, and the number of samples taken for analysis. For the accuracies considered in this simulation, the feed point shows the highest range of uncertainties. This means that of all the strategic points considered in a reprocessing plant, highest priority has to be given to the improvement of the measuring methods used at the feed point.

Statements. It has been shown[4] that with the information obtained from the second and the third safeguards measures (throughput measurement and inventory taking), three different categories of statements can be made:

1. Probability of diversion (P_D). This is a statement by the safeguards authority. On

TABLE IV

Criteria for Indirect Measuring Methods of Fissile Material Content in Fresh, Unirradiated Fuel Pins

Criteria	Remarks
1. Tamperproofness	Against all conceivable measures, which can simulate the presence or the absence of one of the fissionable elements (inhomogeneity, addition or removal of absorbers, reflectors, and foreign neutron and heat source)
2. Free from systematic errors	Any bias in the measurement should be identifiable and correctable
3. Capacity of discrimination	The method should be capable of discriminating between uranium and plutonium
4. Low measuring time	Depends on the throughput and the number of measuring units used in a plant. For one tonne heavy metal/day capacity fabrication plant and one measuring unit, the measuring time should not exceed 2 to 3 min/pin
5. Accurate	For the same throughput as in (4), the overall measuring accuracy for Pu should be > ± 0.4% and that for ^{235}U ± 1.6% (1 value)
6. Simple, reliable, easy to automatize, and adaptable to continuous operation	
7. Economic	

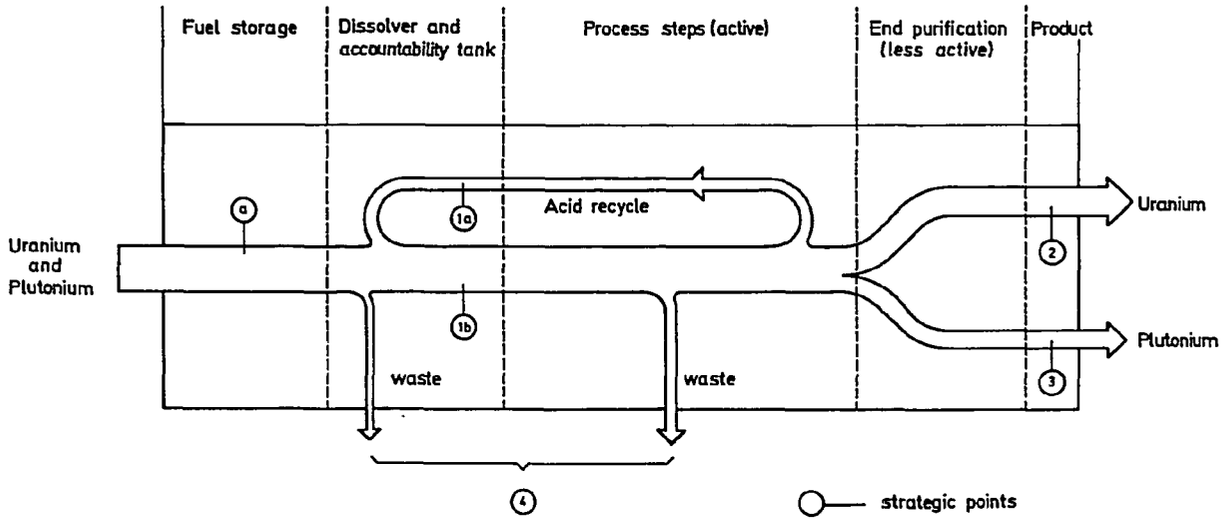

Fig. 2. Fissile material flow and location of strategic points in a reprocessing plant.

TABLE V

Range of Uncertainties in Plutonium Amounts Measured at Strategic Points in a Reprocessing Plant

	Low Pu-Content Fuel			High Pu-Content Fuel		
	Amount of Pu Processed (kg)					
	132			1050		
	Measuring Accuracies (1σ; %)					
Product		0.2 (2)				
Acid Recycle		3.0 (3)				
Waste		10.0 (6)		10.0 (30)		
Feed	0.5 (3)	1.0 (3)	2.0 (3)	0.5 (15)	1.0 (15)	2.0 (15)
	Range of Uncertainties (kg Pu)					
Feed	0.63	1.26	2.53	3.16	6.33	12.68
Acid Recycle	0.45			0.35		
Product	0.27	Same	Same	0.75	Same	Same
Waste	0.10			0.51		
Total	0.85	1.37	2.59	4.05	6.81	12.84

Numbers in parentheses denote number of samples per day

the basis of the two series of measurements, the safeguarding authority can determine the probability with which a minimum amount of fissionable material has been diverted from the plant.

2. Risk of the operator (R_D); this is a statement of the operator. In case an operator plans to divert a certain amount of fissionable material, and knows the accuracies with which the safeguards authority has carried out the two safeguards measures, he can calculate the risk (which can also be expressed as a probability) that the safeguards authority would find out with the probability P_D that he has diverted a minimum amount of fissionable material.
3. Proofing probability (P_B); this is a statement of the safeguards system designer. With this probability he can determine the quality of a particular safeguards system.

For this purpose, he assumes that a certain amount of fissionable material has been diverted by the operator. He can then calculate the chance (which is also a probability) the safeguards authority will have in proving that a fraction of the diverted amount (with a corresponding probability P_D) has actually been diverted by the operator. This particular statement can be extended to determine the effectiveness of a safeguards system.

Because of the inaccuracies inherently associated with the measurement of throughputs and inventory, it is not possible for a safeguards authority to find out with a 100% probability, i.e., with certainty, the total amount of fissionable material diverted by the operator. (See, also, Gmelin et al.[8])

Process-Inventory Functions. It has been indicated that several possibilities exist in determining the process-inventory independently in nuclear facilities, to exercise the third safeguards measure. These possibilities are:

1. Physical measurement of fissionable material inventory in each and every part of the plant during (or after) the process operation.
2. Inventory taking by washing out the fissile material content from the internal parts of a plant to one or more of the strategic points.
3. Determination of the process inventory from the known operational and inventory characteristics of each part of a plant.
4. Measurement of the plant inventory with the help of tracer techniques.

ad 1. The first possibility requires a complete penetration into the plant by the safeguards authority and therefore should be regarded as a rare event and carried out only if the plant operator explicitly agrees to it. Besides, physical inventory is not sufficiently accurate as it is very seldom that all the internal parts are calibrated or that the volumes of the interconnecting pipelines are known with a high degree of accuracy.

ad 2. The second possibility, i.e. washout, was given as an example for exercising the third safeguards measure to emphasize the nonintrusiveness of this measure. If the inventory washouts are allowed to coincide with the operational washouts of a plant and are undertaken once or twice a year, it means that 6 to 12 months would have passed before a diversion by the operator could be detected by the safeguards authority. It is one of the objectives of the proposed safeguards system to reduce the time lag between a diversion and its detection, and the third and the fourth possibilities are being investigated intensively for this purpose.[9] The methods have been analyzed with the fabrication plant where the control experiment (see below) was carried out, as an example; they are similar for a reprocessing plant.

ad 3. A fabrication plant can be divided into a number of unit fabrication cells. These cells can further be divided into storage and machine parts. The flow of the fissionable material through such a cell can be uniquely described with the help of three characteristic functions of time. They are (a) the inventory function $h(t)$, which gives the mass of fissionable material present at time t in the fabrication cell (e.g., in kg); (b) the output function $k(t)$, which gives the rate of mass flow leaving the fabrication cell at the time t (e.g., in kg/h); and (c) the residence time function $T(t)$, which indicates how long the fissionable material entering the fabrication cell at the time t remains in this cell (e.g., in hours). If there is recirculation of fissionable material between some of the cells, the fraction or recirculation $\kappa(t)$ should also be known.

The plutonium fabrication plant ALKEM, at which the control experiment was carried out, was divided into 5 such fabrication units as shown in Fig. 3. The holdup and output functions for all these cells, as well as the fractions recirculated at different points for the experimental campaign, were collected and suitable analytical expressions were developed to fit into the actual data. Typical results of these analytical approximations are shown in Figs. 4 and 5 (output functions) and Figs. 6 and 7 (holdup functions). The total holdup in the plant during the campaign was then calculated with

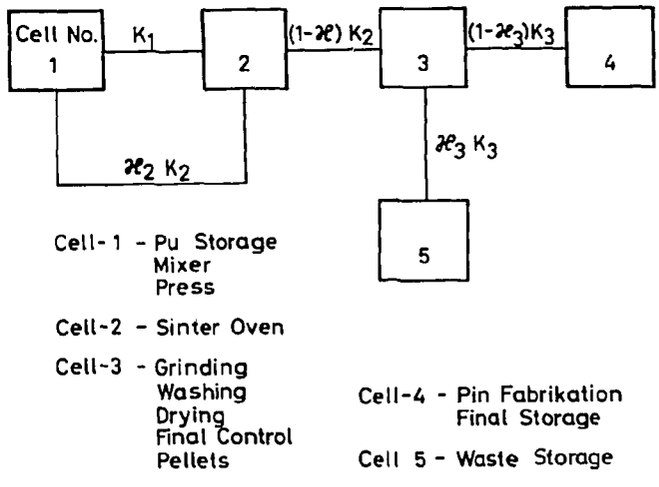

Fig. 3. Division of the fabrication plant ALKEM into five unit fabrication cells.

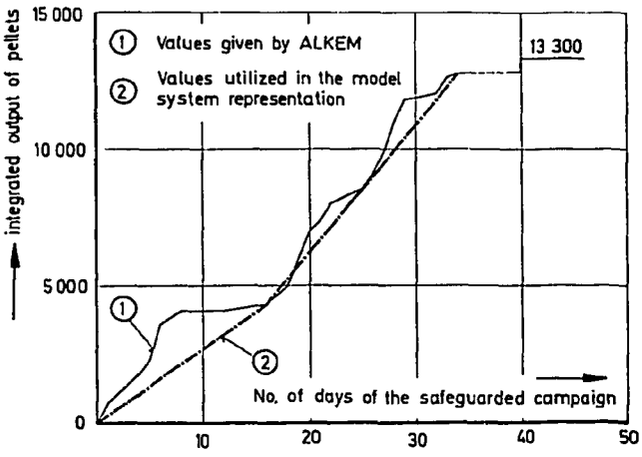

Fig. 4. Output functions for cells 1 and 2 in the fabrication plant.

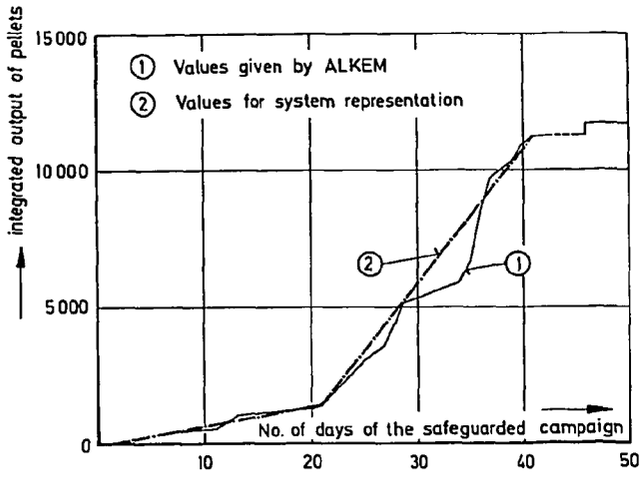

Fig. 5. Output functions for cells 3 and 4 in the fabrication plant.

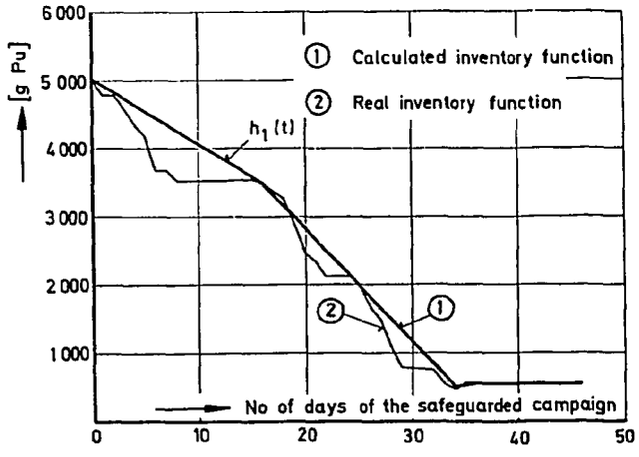

Fig. 6. Inventory function of the first fabrication unit.

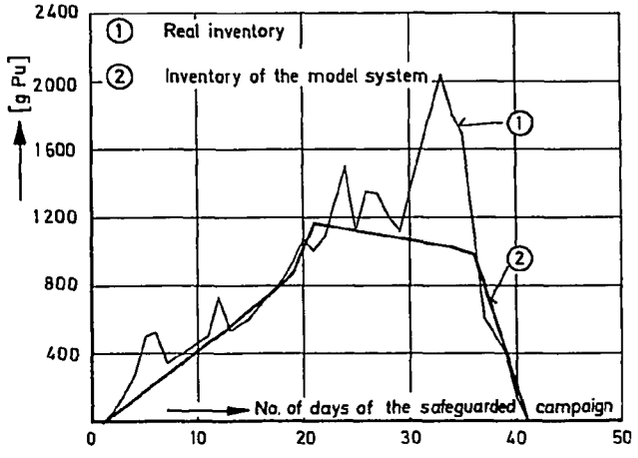

Fig. 7. Inventory function of the third fabrication unit.

the analytical expressions and compared with the actual data obtained. The fitting of the analytical results with the actual values appears to be fairly satisfactory as shown in Fig. 8.

The results of this method indicate it is possible to determine the holdup in a plant at any time if the characteristics of the unit cells are known. However, this method does not appear to be very effective from the point of view of safeguards. As was seen at ALKEM, the holdup functions can vary within a fairly wide range for the same throughput, so the operator can manipulate with his holdup and the manipulation cannot be found out with this method. Besides that, the plant characteristics will vary from plant to plant and the safeguards authority has to have an intimate knowledge of the plant to establish the analytical expressions required. This may not be possible in a large number of cases and is contrary to the safeguards approach proposed here.

ad 4. If fissile material is introduced at the feed point of an operating fabrication plant having a process inventory of $h(t)$, at the rate of $k(t)$ at the time t_0, the same material will appear at the exit of the plant after the whole of the process inventory $h(t_0)$ has been processed out, provided no internal mixing takes place. Therefore, the residence time T of the fissile material, which enters the fabrication plant at the time t_0, is given by

$$\int_{t_0}^{t_0 + T(t_0)} k(t)\,dt = h(t_0) \quad .$$

In the case of a steady-state operation [$k(t) = k_0$], this equation reduces to

$$T(t_0) = h(t_0)/k_0 \quad .$$

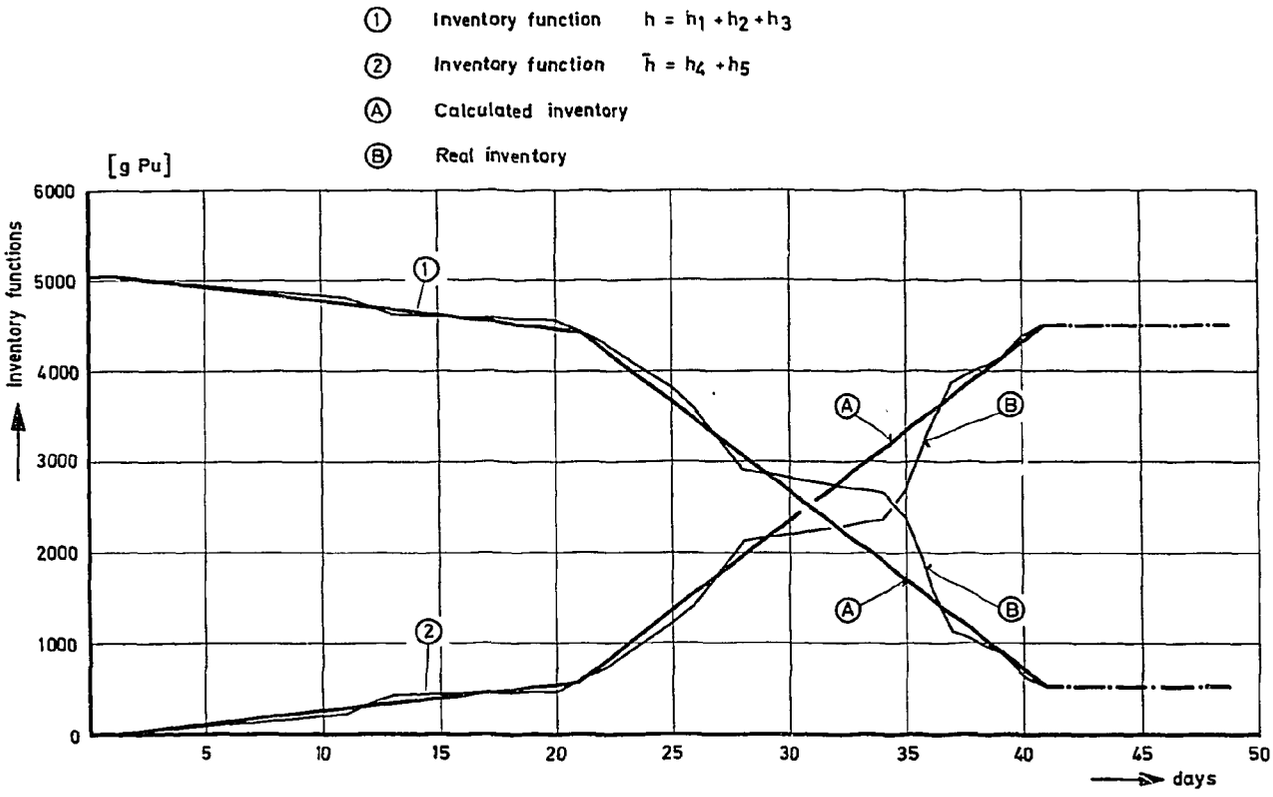

Fig. 8. Behavior of the total inventory as a function of time.

If the fissile material introduced in the plant at t_0 is tagged with a certain amount of tracer isotope and the time T (residence time) the tracer takes to appear at the exit of the plant is noted, the inventory $h(t_0)$ of the plant can be found by knowing the throughput rate k_0. This is an indirect measure of determining the inventory as it is calculated from T and k_0. The holdup can also be determined directly with the help of a traced material. For this purpose, the traced material is fed continuously from the time t_0 into the plant. The untraced material that is still inside the plant is measured at the exit from the same time t_0 until the traced material starts coming out at the exit. The integrated amount of untraced material between these two time limits is the process inventory of the plant $h(t_0)$.

The ACDA-proposed MIST program[19] follows a similar line.

By extension of the indirect tracer method, e.g., by repeating the delta-type signal in a randomly periodic fashion, the rate of change of the process inventory can probably also be determined.

The tracer technique, if properly developed, can be a highly efficient method of nonintrusively determining the inventory during the operation, and the time lag between a diversion and its detection can be reduced significantly. Further and intensive efforts are, however, required for the development of this method.

Control Exercise at the Fabrication Plant ALKEM. So far the most significant result obtained in the framework of the fissile material control project at Karlsruhe is the completion of the first safeguard exercise at the ALKEM fabrication plant. Observers from IAEA, EURATOM, USAEC, and the German Ministry of Scientific Research have visited there. A detailed report on this exercise will be published,[10] and only the important features of the exercise will be presented here.

This fabrication plant is located at the Karlsruhe Research Center but operates under fully commercial conditions. The plant can handle about 200 kg Pu/year, and the fabrication of approximately 1 tonne of UO_2/PuO_2 platelets for SNEAK has been carried out there.

(*a*) *Objectives of the Exercise.* The main objectives of the exercise can be formulated as follows:

1. To determine whether the principle of fissile material flow control at a certain number of strategic points can be realized

in an existing plant, in which the strategic points cannot probably be selected in an optimum manner because of the already existing plant layout.

2. To find out whether the different types of statements developed on the basis of system analysis (discussed earlier) can be made on the basis of the material balance established at the strategic points. For this purpose the owners of the fabrication plant were requested to withdraw a certain amount of Pu (known only to them) from the process stream. The objective was to demonstrate the applicability of the above-mentioned statements.
3. To mainly use the measuring methods already available, but also to use the methods being developed to establish their suitabilities and weaknesses.
4. To prepare an estimate of the total amount of effort required to exercise different safeguards measures.
5. To determine the drawbacks of the existing layout from the safeguards point of view.

(*b*) *Plant Layout and Location of Strategic Points*. The plant layout and the location of the strategic points are shown in Fig. 9. The first strategic point was the Pu and product storage. Since there was no possibility of weighing and sample taking in the storage, and the only weighing and sampling possibility was in the glove box No. 1/85 in the ceramic section, a part of the safeguards activity for the strategic point one (1) had to be carried out at this box. At this point, the input of the plant was measured. The layout of the plant is such that plutonium cannot be introduced into the plant except through this point.

The second strategic point was installed at the final pellet-control stage. At the beginning of the experiment it was not clear whether the calorimeter, which was supposed to be used to measure indirectly the plutonium content of the fabricated fuel pins (which were the final product), would be ready for operation. The final pellet-control stage was the last step at which the product stream could be measured for its plutonium content with the already available method of gamma spectroscopy.

The calorimeter became ready during the exercise and it was put into operation in a corner of

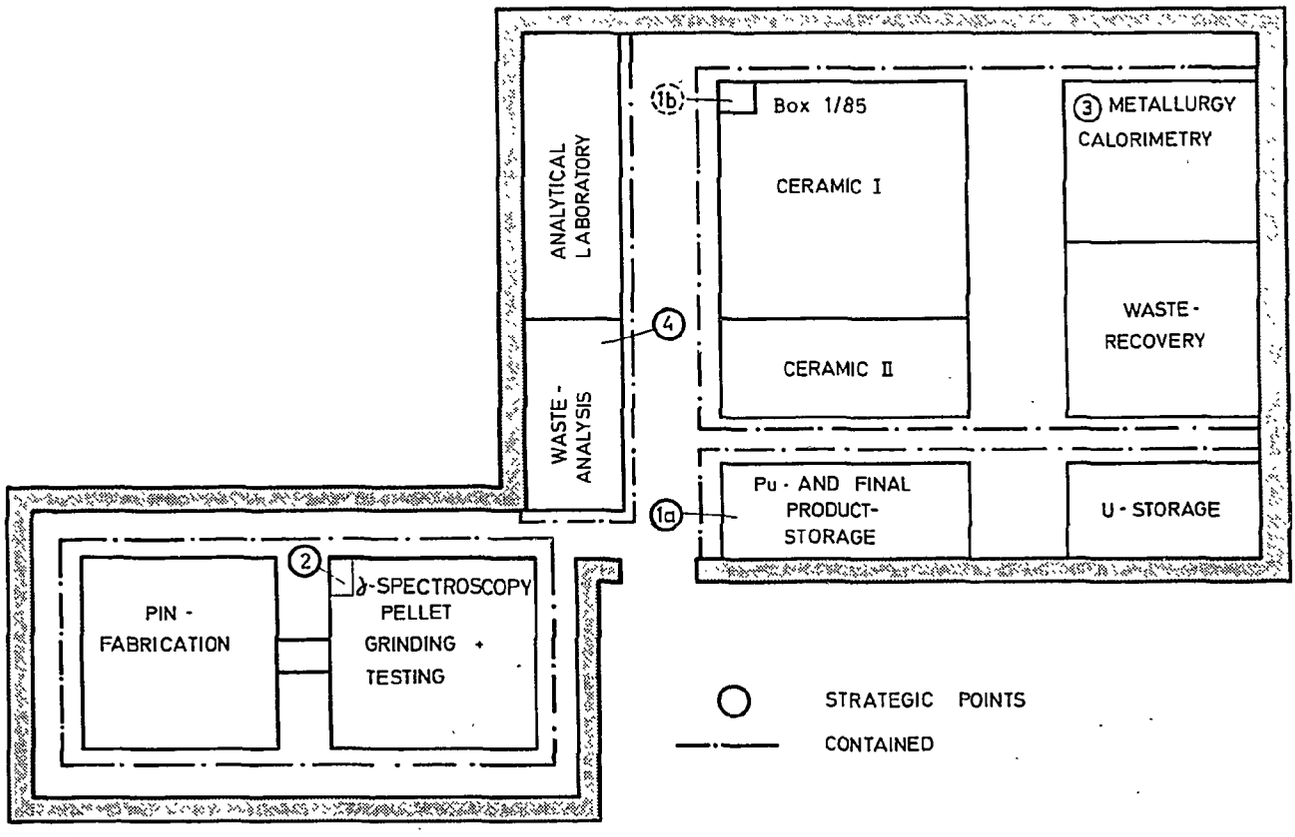

Fig. 9. ALKEM plant layout and the locations of the strategic points for the safeguard experiment.

the metallurgy room, which is air conditioned; the area around the calorimeter was declared as the third strategic point. Logically, the calorimeter belongs in the room for pin fabrication, but there was no space available there and no air conditioning, which is essential for the operation of a calorimeter. The completed pins were measured for their plutonium content in the calorimeter during the end phase of the campaign.

The fourth strategic point was located at the waste analysis room where all the waste streams from the different parts of the plant were collected and the plutonium contents of the waste were measured by a neutron counter. The neutron counter is permanently located in this room and is used regularly during plant operation.

The safeguards measures and activities at these strategic points are shown in Table VI. Some additional safeguards activities were required because of the prevailing conditions in the plant. They are summarized in Table VII. Because of the campaign type of operation during the safeguards exercise, the beginning and the end of the exercise were well defined for the establishment of the material balance. The chemical and isotope analyses required for the material balance calculations were carried out in independent laboratories at the Center, and only these results were used for the exercise.

(*c*) *The Production Campaign.* The specification of the production campaign, which was safe-

TABLE VI

Safeguards Measures and Activities at Strategic Points for the Control Experiment at ALKEM

Strategic point	Safeguards Measures	Safeguards Activities
1a	Containment	Sealing of the Pu storage to identify the in- and outgoing Pu-containing boxes.
1b	Throughput measurement for feed and scraps	Weighing, sample taking, chemical analysis. Known methods.
2	Throughput measurement product stream in the form of pellets	Measurement with the help of γ-spectroscopy. Known method but introduced for the experiment for the first time.
3	Throughput measurement product stream in the form of pins	Measurement with the help of calorimeter. New method introduced particularly for the experiment for the first time.
4	Throughput measurement waste streams	Measurement with the help of n-counter. Known method, standardized for the experiment

TABLE VII

Additional Safeguards Activities in Connection with the Safeguards Experiment at ALKEM

Safeguards Activities	Purpose
Sealing of active waste storage drums	To prevent removal
Sealing of waste storage area	To ensure that no recirculation takes place
Accompaniment during transport of Pu from Pu storage to weighing and sampling box and back (Strategic Point 1a)	To prevent mixing and recirculation
Accompaniment during transport of pellets from final control stage to pin filling stage	To prevent mixing and recirculation
Identification of material under safeguards through mass spectrometric analysis	To prevent mixing between safeguarded and unsafeguarded material which had different Pu isotopic compositions
Marking of finished pins	To prevent recirculation in the calorimeter
Control of cleaning operation of the plant before and after the experiment	To establish well-defined starting and end conditions for establishing material balance
Homogenization of scraps	To determine accurately the Pu content in scraps

guarded, is given in Table VIII. About 200 kg of U + Pu mixed oxide were used to produce 186 fuel pins. The plutonium concentration was 2.3%. The total amount of plutonium supplied to the plant was 4909 g. Only the flow of Pu was safeguarded.

TABLE VIII

Specification of the Production Campaign Safeguarded During the Experiment

Amount of Pu supplied (g)	4909.00, as PuO_2	
Amount of U supplied (kg)	250, as UO_2	
Total amount of ceramic processed (kg)	200, mixed U + Pu oxide	
Pu concentration (%)	2.3	
Pellet specification		
Height (mm)	15	
Diameter (mm)	12.5	
Weight (g ceramic)	18.6	
Pin specification	Type I	Type II
Height (mm)	1325	410
Diameter (mm)	13.5	13.5
Weight (g)	1472	452
No. of pins	113	73

TABLE IX

Results of the Material Balance

Point	Amount (g Pu)	Range of Uncertainties (1 - σ Value in g Pu)
Input	5070.25	5.86
Output, Product; Pellets (γ-spectroscopy)	4209.57	4.91
Output, Product; Pins (Calorimetry)	4213.76	14.48
Output, Scrap	677.63	2.19
Output, Waste	127.95	1.49
Output, Waste (box scrapings)	6.50	0.06
Difference between input and output (based on the γ-spectroscopy)	48.38	8.09

Accuracies of measuring instruments used:

Feedpoint	0.4%
γ-spectroscopy (pellets)	0.5% per batch
Calorimeter	0.8 - 1.2%
n-counter	8%

(*d*) *Results of the Exercise.*

(i) Material balance: The material balance for the safeguarded campaign was obtained by summing up the output and the input streams. The output streams consisted of the product stream (measured as pellets by the γ-spectroscopy and as pins by the calorimeter), the waste streams, and the scrap stream (the scrap is obtained as PuO_2 mixture from different process steps; it is normally recovered and reused in one of the next campaigns or returned to the owner of the material), and the scrapings from the boxes at the end of the campaign. The amounts measured, the range of uncertainty for each of these amounts, and the resulting differences are shown in Table IX. The difference between the input and the output stream was found to be 48.38 g of Pu with a 1-σ range of uncertainty of $\pm$ 8.095 g. No significant difference between the main values of calorimetry and γ-spectroscopy was obtained. This means that all the pellets measured by γ-spectroscopy were also introduced into the pins, which were then measured by calorimetry. However, the difference was calculated with the results of the γ-spectroscopy as it was found to be more accurate (Table IX). In this exercise, the calorimetry was used mainly as a containment measure.

(ii) The probability of diversion (P_D): The probabilities of diversion for different amounts of Pu were calculated according to the principle statements in Gmelin et al.[4] and are shown in Table X. The actual amount of Pu withdrawn during the exercise by the ALKEM authority was 42 g Pu. It was possible to state, for example, that with 95% probability $\geq$ 35.06 g Pu had been diverted from the process stream.

(iii) Risk of the operator (R_D): The risks of the operator for different amounts of plutonium and the corresponding probabilities of diversion have also been indicated in Table X. Gmelin et al.[4] showed that for a given fraction of the diverted amount (which can be declared with P_D as diverted), the risk of the operator is mainly a function of the ratio of the plutonium amount, which he plans to divert, to the total range of uncertainty in the measurement. Since this ratio was fairly high in the safeguards exercise (amount withdrawn = 42 g; 1-σ range of uncertainty 8.09 g), his risk was also high. For example, as Table X indicates, he stood a risk of 91% that 10 g, from a total of 42 g, would be declared with a P_D of 95% as diverted. In actual case, $\geq$ 35 g (i.e., 83.2% of the diverted amount) were declared with a probability of diversion of 95%. For this statement, the operator stood a risk of only 30.9%. This shows clearly that, even with a low risk, the probability of diversion can be very high.

TABLE X

Probabilities of Diversion (P_D) and the Risk of the Operator (R_D) Calculated on the Basis of Material Balance

Actual Amount Diverted - 42 g of Pu
Probability of Diversion

	P_D(%)		
	95	99	99.9
Amount (g Pu)	35.06	29.55	23.37
Percentage of the actual amount diverted	83.2	67.8	55.5
	P_D(%)		
	90	95	99.9
For amounts declared as diverted (g Pu)	Risk (R_D)		
10	97.9	91	61
15	94	84	52.7
20	85.9	73.5	44.4
30	54.7	45.2	29.5
35	35.9	30.9	23

(iv) Effort of safeguards: The man-hours for different safeguards activities have been summarized in Table XI, and the chemical and mass spectrometrical analyses in Table XII. The largest fraction of man-hours was required by the γ-spectroscopy mainly because two persons were required at this point. A large saving in man-hours will be caused by eliminating this point and using only the calorimeter at the final stage of pin production. A fairly large fraction of the chemical and mass spectrometrical analysis was also required for the γ-spectroscopy.

(v) Relative importance of the strategic points: It has been indicated earlier in this paper that the

TABLE XI

Man Hours for Different Safeguards Activities During the Control Experiment at ALKEM

Location	Man Hours	% of Total
Pu storage	32	3.9
Box 1/85	62	6.9
Waste analysis (n-counter)	121	13.4
γ-spectroscopy	484	53.5
Calorimeter	70	7.7
Waiting time	80	8.8
Miscellaneous	56	6.2

TABLE XII

Efforts on Chemical and Mass Spectrometrical Analysis

Purpose	Number	% of Total
1. Chemical		
Input	8	20.5
γ-spectroscopy	26	16.7
Scrap + waste	5	12.8
Calorimetry	0	0
Total	39	100
2. Mass spectrometrical analysis		
Input	12	35.3
γ-spectroscopy	12	35.3
Scrap + waste	5	14.7
Calorimetry	5	14.7
Total	34	100

range of uncertainties can be regarded as an index for the relative importance of the strategic points. For a first approximation, this can be characterized for a strategic point i by a number Z_i, which is a product of the range of uncertainty σ_i and the square root of the total effort A_i spent at that strategic point. For the ALKEM experiment, the total effort is given by the sum of the man-hours, chemical and mass spectrometrical analyses. The number Z_i and the relative weightage of Z_i have been calculated for all the strategic points as shown in Table XIII. The highest number is given by the calorimeter because of the highest range of uncertainties. However, this is not contradictory to our former statement, as it was this particular and first device in which the full potential of the method has not been realized.

(vi) Drawbacks of the instruments and layout: Because of the use of γ-spectroscopy and physical separation of the strategic points 1a and 1b, three safeguards personnel were required continuously during the exercise. Besides, the safeguards personnel had to enter the ceramic and the pellet production area. Absence of adequate space and air conditioning in the pin fabrication room necessitated the location of the calorimeter in the waste reprocessing room. None of the measuring methods used at the plant, namely, the γ-spectroscopy, the calorimeter, and the neutron counter, is tamperproof in its present form. However, the calorimeter and the neutron counter can be made tamperproof with further effort.

(*e*) *Conclusion.* The results of the exercise have shown that the principle of fissile material

TABLE XIII

Relative Importance of Strategic Points

Strategic Point	Measurement of	Range Uncertainties (g Pu)	Man Hours (h)	Chemical Analyses (No.)	Mass Spectrometry Analyses (No.)	Costs (*DM*) A_i	Z_i	Z_i Rel%
1a 1b	Input + Scrap	6.256	94	13	17	7 430	539	27.0
2	Pellets (γ-spectroscopy)	4.91	484	26	12	15 280	606	30.6
3	Pins (calorimetry)	14.48	70	0	5	2 650	744	37.4
4	Waste	1.46	121	5	5	4 710	100	5.0

control can be realized in the existing ALKEM-type plant, with reasonable efforts. Because of the fairly high measuring accuracies obtainable, diversion of relatively small amounts can be detected with a fairly high degree of probability. Valuable experience was gained that can be used in setting the priorities of different development work and system analytical investigations.

Containment Studies

Nuclear Power Stations. A recent study,[11] undertaken to determine the optimum and effective safeguards measures for nuclear power reactors, has shown that nuclear power stations of the presently known heavy-water natural-uranium and light-water slightly enriched uranium types, can be safeguarded mainly with the help of containment measures. A nuclear reactor is the only step in the whole fuel cycle in which fissionable material remains contained in fuel subassemblies during its entire residence time. During normal operation of such a reactor, the fuel subassemblies move through three well-defined containment areas; namely, the dry storage area for fresh fuel subassemblies, the reactor vessel, and the wet storage area for irradiated subassemblies. Three measures are required to safeguard the movement and account for the subassemblies:

1. Sealing and identification of the subassemblies at the dry storage and the wet storage area.
2. Registration of movement and loading of the main cranes and the refueling machines.
3. Measurement of the activity over the reactor bay area.

A combination of these measures can determine the movement and the number of subassemblies in a reactor as a function of time.

Fabrication Plant. A typical layout[12] of a 100 tonne/year of $PuO_2 + UO_2$ fuel for fast breeders is shown in Figs. 10a and 10b. The experience gained from the ALKEM experiment and the trend of the fabrication industries for automation has been incorporated in this reference layout. Figure 10a gives the layout of the cellar and the ground floor at which all the fabrication steps are located. Figure 10b gives the front view of the fabrication building. The plant is laid out in such a way that the movement of fissile material is fully contained inside the plant and the fissile material can enter or leave the plant only through strategic points. The first of these strategic points is in the cellar for the fissile material entrance and for the waste material leaving the plant. It is possible to weigh and take samples at this point. The second strategic point is on the ground floor directly over the first strategic point, and is used for the personnel check. The third strategic point is at the product end. At this point there are possibilities for both pin measurement and sealing of subassemblies. The containment of the fissile material is shown by the dotted line.

In laying out this particular plant it has been assumed that the fuel for the core part of the subassembly will be received in the form of sinterable UO_2 and PuO_2 powder; whereas, that for the axial blanket (which is normally depleted uranium) will be obtained as completed pellets. All these materials will be received at the first strategic point which is located in the cellar. The sealing of the bird cages will be checked at this point and, if necessary, samples can be taken on the basis of random statistical methods. Simple chemical analyses, if necessary, can also be carried out by the safeguards personnel at this point, but samples for independent mass spectrometric analyses will have to be sent to other laboratories. After identification and sampling,

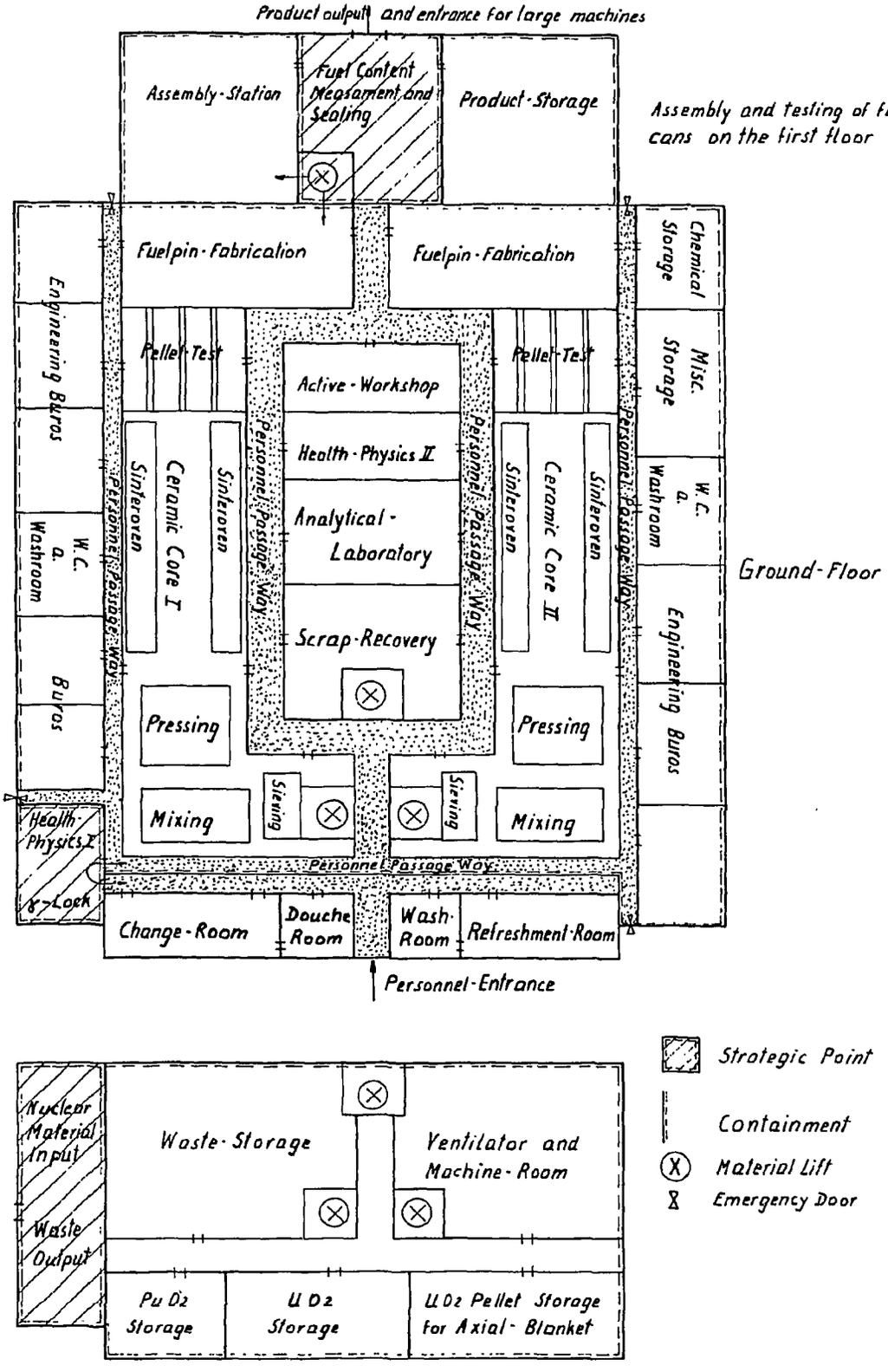

Fig. 10a. Layout of the ground floor and cellar of a reference fabrication plant for fast breeder fuel subassemblies.

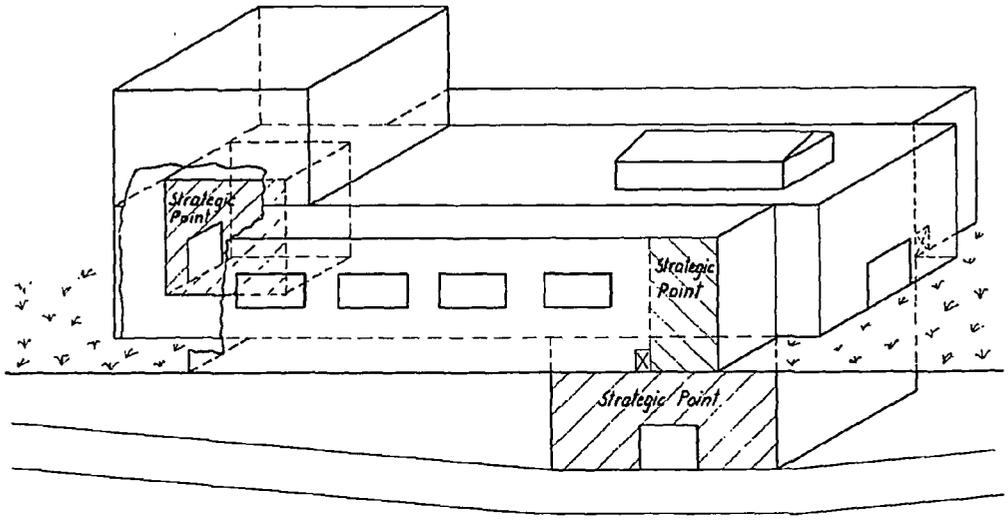

Fig. 10b. Front view of the reference fabrication plant.

the fissile material will be stored in the respective storage areas.

The material from these areas is transferred to the ground floor with the help of lifts provided for this purpose. All the operational personnel can enter or leave the fabrication area through the personnel lock only, which functions as the second strategic point. The fabrication area has been divided into two parallel lines to fabricate fuel separately for the two core zones of a fast breeder, each of which has a different plutonium concentration. The two lines join at the third strategic point. At this point the finished pins or the subassemblies can be tested for their plutonium content. If necessary, sealing of the pins and subassemblies by the safeguards authority can also be carried out here. The completed subassemblies can leave the plant only at the third strategic point. This point is used also for the supply of structure and canning materials and other inactive materials required for the plant.

The scraps and analytical wastes are reworked and the reworked plutonium is sent back to the first stage of the fabrication. Only the waste from this stage is sent to the waste storage cellar with the material lift. The waste can leave the plant only through the first strategic point.

The fabrication area is flanked by two wings of the building in which the technical offices, storage for inactive materials, etc. are located. All the areas surrounded by the dotted line are contained. Different measures can be taken for ensuring this containment.

On the first floor, assembly and testing of the canning material is carried out. The tested and partly assembled canning materials are sent to the pin fabrication station with the lift located at the third strategic point. The inactive workshop is also located on the first floor.

Several such layouts, their drawbacks and advantages, have been discussed in detail by Richter et al.[12]

Instruments

Important progress has been made on slowing-down spectrometry on the basic research on n,γ reactions, and on the calorimeter. Work on the first method is being referred to by Stegemann and Seufert,[16] and on the second method by Michaelis[17] at this Conference.

Calorimeter. The radiometric calorimeter is a well-known device for determining the heat generated by the α-decay of Pu in Pu-containing fuel. If the isotopic composition of Pu is known, the total amount of Pu present in the fuel can be calculated from the heat generated by the different isotopes of Pu and by the ^{241}Am which is present in the fuel at the time of measuring the heat.

The principle of the calorimeter is shown in Fig. 11. Fuel pins with unknown Pu content are introduced into the α-calorimeter which is surrounded with thermocouples. The heat flux obtained by α-decay of Pu and ^{241}Am in the pin generates the potential difference in the thermocouples and can be measured accurately by a micro-voltmeter. This calorimeter is immersed in a constant-temperature water bath. A second identical reference calorimeter connected to the α-calorimeter is also immersed in the same bath. The reference calorimeter contains an electrical resistance which is heated up simultaneously with

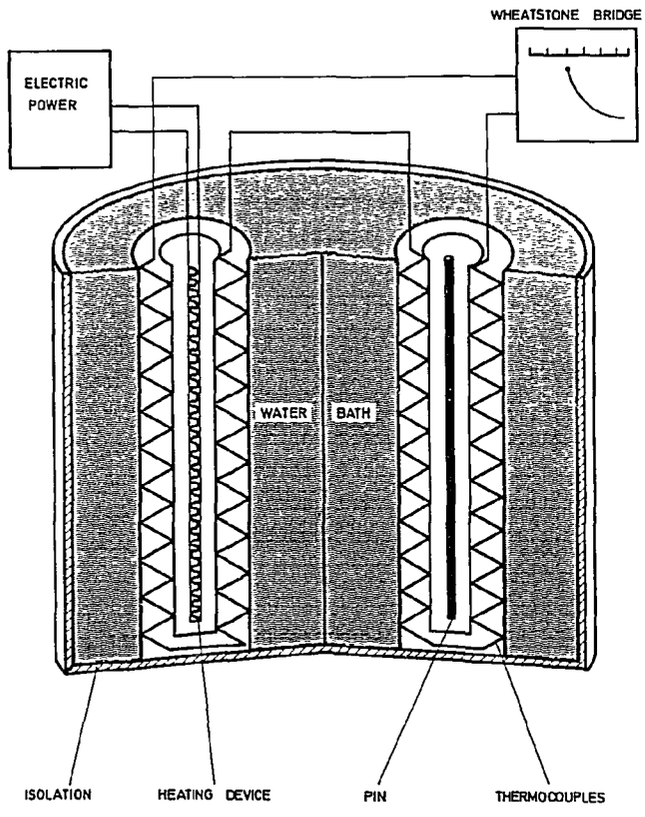

Fig. 11. Principle of the calorimeter used for the safeguards exercise.

the heating of the α-calorimeter, and the heat input is registered. The voltage of the two calorimeters is balanced in a Wheatstone bridge. From the accurately measured voltage supplied to the reference unit, the heat production rate of the α-calorimeter, and therefore the amount of Pu inside the pins, can be calculated once the potential difference and the heat flux relation has been standardized.

The main advantage of a calorimeter of this type lies in the fact that the method is simple, reliable, and easy to automatize. In principle, it is possible to estimate the plutonium content in fuel subassemblies also. It is not fully tamperproof, as plutonium in the fuel pins could be replaced by some other α-producing element. However, the method can be made tamperproof if the neutrons produced by the isotopes, due to spontaneous fission, are also measured simultaneously and the ratio of neutrons-to-watt is determined.

The inaccuracies in this measurement result from two sources: (a) the inaccuracies in the measurement of the Pu isotopic ratio and the ^{241}Am content; and (b) inaccuracies caused by the reproducibility of the measurement error (because of the first source) obtained for the safeguards exercise, was found to be 0.45% as shown in Table XIV. The major part of the total error (found between 0.8 to 1.2%, Table IX) was from the reproducibility of measurement and varied between 0.6 to 1.0%. With further development, the total percentage error from all the sources is expected to be reduced to around 0.4 to 0.5%.

TABLE XIV

Error in the Measurement of Calorimetry Because of Isotope Measurement Errors

Isotope	Percent	Error	Heat Val Rel (W/g)	W
^{238}Pu	0.27099	1.3	0.569	0.001542
^{239}Pu	75.492	0.21	0.001923	0.0014517
^{240}Pu	17.9703	0.56	0.00703	0.0012633
^{241}Pu	4.8261	0.97	0.0045	0.0002172
^{242}Pu	1.0704	1.33	0.00012	$1.28\ 10^{-6}$
^{241}Am	0.3699	1.5	0.1084	0.000401
Total		0.45		0.00487644 W/g of safeguarded Pu

Slowing-Down-Time Spectrometer. The slowing-down-time spectrometer has also reached an advanced stage of development.

A pulse of fast neutrons is allowed to pass through a lead pile. Because of the slowing-down process, the average energy of neutrons can be calculated as a function of time. If a fuel pin containing uranium or plutonium is placed in the lead pile in the path of these neutrons, fission of ^{235}U or ^{239}Pu is initiated by the impinging neutrons, provided they have energies in one of the resonance regions. Knowing the energy of the neutrons and the cross sections of the fissile material at this energy, the amount of fissile material can be determined by measuring the resulting fission neutrons. For the determination of ^{235}U alone, the resonance energy level of 0.28 eV may be chosen.

As indicated before, an industrially finished instrument based on this principle is expected to be ready by the middle of 1970. This instrument will be in a position to measure the ^{235}U content in fuel pins for light-water-type reactors ($\sim$3% ^{235}U concentration) with an accuracy of $< 2\%$. The capacity of this instrument will be around 600 pins/day. This corresponds to a fabrication plant of 1 tonne/day capacity.

Further details of this method are given by Stegemann and Seufert.[16]

CONCLUDING REMARKS

The modern safeguards system involves a number of complex and interrelated problems. They range from the intangible political feelings, human relations, and other apparently unquantifiable areas, to the development of highly sophisticated methods. Experience and results gathered during the past year indicate, however, that most of these problems yield solutions if handled in a rational manner; that most of the areas, hitherto considered unquantifiable, can be quantified; and, finally, that the whole development of a modern safeguards system is a fully rational venture. These experiences and results also indicate that such a system is not a long-term hope, but a short-time reality. It can be realized and implemented in existing plants in the near future.

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SAFEGUARDS, INTERNATIONAL COOPERATION, AND TRADE

MYRON B. KRATZER
Assistant General Manager for International Activities
U.S. Atomic Energy Commission, Washington, D.C. 20545

Just a year and a half ago, in a paper on safeguards presented before a meeting of the American Assembly at Airlie House, I felt justified in saying that safeguards was a subject that had received far less attention than it deserved. That statement could hardly be made today.

This year has been marked by intensive discussion of safeguards in national and international forums and in the press. The intensity of this discussion has, however, not always been matched by the depth of understanding and information on the subject by participants and public alike. I say this not in a critical manner but rather in recognition of the fact that the complexity, the newness, and the highly technical nature of safeguards create formidable and understandable difficulties in conducting discussions at the international level.

The subject assigned to me is not safeguards alone, but three topics: safeguards, international cooperation, and trade. This assignment reflects the close relationship these subjects bear to each other, a relationship which is apparent in the Non-Proliferation Treaty itself. In that Treaty, the proximity of Article III (which deals primarily with safeguards) with Articles IV and V (which deal with international cooperation and the peaceful uses of atomic energy) is not accidental. Indeed, even within Article III itself, the principle is laid down that the safeguards required by the Treaty are to be designed to avoid hampering international cooperation and trade. I will speak first about safeguards—which, as many of you know, is a subject on which Del Crowson, who preceded me as a speaker, has the general AEC responsibility for technical policy. I will then conclude my remarks with some observations about their interrelationship with international cooperation and trade.

I believe that one of the overriding facts regarding safeguards is their novelty, both in political and technical terms.

Chairman Seaborg has stated that the program of international cooperation in peaceful uses of atomic energy, the "Atoms for Peace" program as it is often called, is unprecedented in the history of scientific cooperation among nations. He was referring to the fact that never before had nations with a position of leadership in a broad new field of technology deliberately undertaken to share this knowledge so thoroughly with the rest of the world.

I believe that safeguards—the measures designed to ensure that this program of international cooperation remains peaceful—are no less unprecedented.

The first arrangements in which nations agreed to open at least some of their peaceful nuclear programs to outside inspectors with broad rights of access and investigation came into being in 1956. They represented then, as they do now, a significant departure in the means by which voluntary international undertakings are enforced. The first international inspection in implementation of these arrangements took place, to the best of my knowledge, in 1957.

Today, international safeguards arrangements—by which I mean any arrangements under which nations agree to the application of safeguards by nationals of another country or countries—are in effect in more than 40 nations. Three international organizations—the IAEA, the European Atomic Energy Community, and the European Nuclear Energy Agency—are authorized by their respective statutes to apply safeguards under certain conditions. Many more than 1000 international inspections have actually been conducted, either on a bilateral or multilateral basis. In nearly every case these safeguards have worked smoothly and to the satisfaction of the parties

directly concerned. The record, I believe, is a creditable one.

In view of this sustained, substantial, and favorable experience with international safeguards in practice, it is perhaps a bit surprising that much of the discussion of safeguards over the past year and a half has been directed toward supposed problems and difficulties. The explanation for this, in large part, is the concern generated by the tremendous increase in safeguards activities to be anticipated as a consequence of the growth in the application of nuclear power around the world and the spotlight of publicity placed on safeguards by the Non-Proliferation Treaty.

It is not the Non-Proliferation Treaty alone that will bring about the vast increase in the future scale of the international safeguards effort. For some time to come, many, indeed most, of the peaceful nuclear programs that will be subject to safeguards under the Non-Proliferation Treaty would also have been subject to safeguards under already existing arrangements. The Treaty will fill the important, and increasingly large gap, of requiring safeguards on the *entire* peaceful nuclear programs of its non-weapons signatories, including entirely indigenous efforts, and not merely those undertaken in cooperation with outside supplier nations who require safeguards as part of their supply arrangements.

Whatever its origins, the dialogue of the past year on safeguards has been, on the whole, a useful and constructive one. If we have learned one thing about international safeguards since their inception, it is that they are a difficult technical problem, and like other such problems, can benefit from informed discussion as well as from vigorous research and development.

There are, nevertheless, a number of questions that have been raised concerning international safeguards over the past year and a half which I believe have grown somewhat out of proportion. I hope that my remarks will help in some measure to place these questions in better perspective. Here are some of the principal problem areas that have been discussed.

1. The view is sometimes expressed that the Agency safeguards system needs substantial revision to reflect the new situation under the NPT, and to incorporate technical improvements that would make the system more efficient and less intrusive.

I believe that much of the discussion of this type stems from some misunderstanding of the nature of the present IAEA system, and even from a certain confusion in terminology.

The IAEA's safeguards are defined, first and foremost, in the Agency's Statute itself, and additionally, in a document entitled "Information Circular 66—The Agency's Safeguards System (1965)."

If one studies the latter document, however, it is quite clear—despite its title—that it describes not an explicit and detailed safeguards system, but rather a series of broad principles and a somewhat more detailed assertion of the Agency safeguards rights than those that appear in the IAEA Statute itself. In short, much of Information Circular 66 is a statement not of what the Agency *must* do in conducting safeguards, but of what it *may* do.

Much of the concern expressed over the past months has been directly related to the effect of safeguards on trade. For example, some are concerned over the possibility of excessive application of safeguards, and consequent unwarranted intrusion of inspectors into peaceful nuclear activities of a commercial nature. Yet, both the Statute and Information Circular 66 repeatedly enjoin the Agency to restrict its safeguard activities to those *necessary* to provide assurance against the diversion of nuclear materials. Indeed, the very language of Article III of the NPT, that the safeguards system shall be designed so as to avoid hampering technological development, is derived from the IAEA safeguards document. Moreover, the obligation to give effect to improvements made possible by technological developments is itself an important principle of the existing Agency safeguards document.

The question, therefore, is not whether the Agency's safeguards system should be improved; it clearly can and must be. The question is how this can best be accomplished, and whether improvement calls for an early and general revision of the Agency's safeguards document, Information Circular 66.

I believe we would be well served in discussing the future improvement of Agency safeguards, to make a clear distinction between the Agency's safeguard *system*—that is, what the Agency does in practice—and the Agency's safeguard *document*, Information Circular 66, which, as I have noted, does little more than lay down a series of broad general principles and safeguards rights. If this distinction is made, I believe we may well conclude that progress can be made fastest by the evolutionary improvement of the Agency's system within the broad and flexible provisions of the current document, rather than through a time-consuming debate to change the already permissive provisions of this document itself.

2. A special case of the view that the Agency's safeguards system must be modified is the contention that only in this way can effect be given to

the principle referred to as a goal by the NPT itself—that of concentrating safeguards on the flow of materials at strategic points, using instruments and other advanced techniques insofar as possible.

I believe this principle represents a logical evolution of concepts already embodied in the Statute of the IAEA and in its safeguards document. Much of the effort of the general review of the Agency's safeguards, which culminated in the adoption of Information Circular 66, was devoted to making it clear that materials are, indeed, the focal point of safeguards, and a careful reading of the document will show that only materials are subject to Agency safeguards. Obviously, nuclear materials are not used simply in a vacuum; they are employed in equipment and facilities and, therefore, a realistic and useful safeguards document cannot avoid reference to the safeguards procedures applicable at various types of facilities in which safeguarded materials may be employed. Such references do not detract from the fact that it is materials, and not facilities, that are subject to safeguards.

The principle of maximum use of instrumentation has also had a lengthy tradition in safeguards thinking. As early as 1959, the USAEC undertook work specifically directed toward the development of safeguards instruments that would reduce the need for, and intrusion by, inspection personnel.

The objective of these references to early support for the principles of concentrating safeguards on the flow of materials and for the use of instruments, is not in any way to detract from the value these concepts have had in defining the future direction of safeguards improvements, but rather to underline the fact that US support for these principles has, in fact, been one of long standing, and not simply a response to NPT provisions.

It is important, of course, to bear in mind that the principle of safeguards at strategic points is, as the Treaty itself makes clear, subject to the important qualification that these safeguards must be effective ones.

Today, our technical capabilities for accurate measurement at several points in the fuel cycle are not sufficiently advanced to permit the general application of the concept of safeguards at strategic points, to the exclusion of more conventional techniques. It is for this reason that the Treaty recognizes the need for further research and development in this field. In addition, of course, the principle is itself a flexible one which permits changes in the selection of strategic points and the degree of application of instruments as our technical capabilities improve through research and development.

3. Another of the concerns, expressed with regard to the intrusiveness of safeguards, relates to continuous inspection. Since an early date there has been sensitivity toward the possibility that effective safeguards on certain peaceful nuclear activities would require the presence of inspectors on a full-time basis.

Continuous inspection means inspection of sufficient intensity to make it impossible for the plant operator to know with certainty that a particular measurement or material flow operation was not observed or subject to verification. In the present stage of technology, the capability of international inspectors to make independent measurements of some types of plant inputs, outputs, and inventories is limited. Some examples of these difficult cases are the irradiated fuel input to a reprocessing plant, the fabricated fuel output of a fabrication plant, and some types of scrap. In these cases, continuous inspection can afford greatly increased opportunity to enable inspectors to verify the validity of plant materials accountability as well as to strengthen the two other important elements of safeguards—containment and surveillance. Thus, under present conditions, continuous inspection is felt to be essential to the application of effective international safeguards to certain types of activities.

Even though continuous safeguards are unique in terms of their effectiveness in certain circumstances, I am unable to agree with the contention that they are especially intrusive. The intrusiveness of an inspection regime depends not on the frequency of safeguards visits alone, but also on the intensity of access by the inspectors when these visits take place. Highly detailed inspections undertaken by a number of inspectors seeking access to every record and every corner of, let us say, a power reactor, during periodic visits may involve much more intrusion into the operation of the facility and lead to much more knowledge of its design and performance, than would inspection on a continuous basis, concentrated on the transfer of irradiated fuel elements to and from the reactor cooling basin.

Increased frequency of inspection, including particularly continuous inspection where appropriate, can reduce the required intensity of access and thereby bring about a reduction in the intrusiveness of the inspection procedure as well as other important simplifications in safeguards techniques.

4. The last problem area I want to discuss is that of safeguards costs and manpower requirements.

The public attention that has been called to the future cost of safeguards in dollars and people has

been useful. These requirements cannot be met without recognition of their existence and careful planning.

Nevertheless, I believe it is a valid observation to make concerning these needs that even the highest of these estimates represents only a small fraction—on the order of 1%—of the cost of the nuclear power produced in the complex to which the safeguards would be applied.

A reasonable estimate of the world's installed nuclear power capacity in 1980 is 300,000 MW. Such a capacity would produce electrical power, valued at 4 mills/kWh, worth $8 billion annually, and might require direct operating manpower of 60,000 persons, not including those in an administrative capacity or those engaged in the development, design, and construction of additional nuclear plants and facilities. If we are able to accept that these personnel requirements, many of them in highly skilled categories, can be met, I see no basis for feeling that with proper planning, recruiting, and training, safeguards personnel needs cannot also be met. The point to be made here is that we should view safeguards manpower needs as one of the many specialties required by a vast new industry, and plan and act accordingly. In this perspective, the safeguards requirements loom no larger than many others which must—and will—be met by the world's burgeoning nuclear power industry.

I would now like to specifically stress the interrelationship between international cooperation, trade, and safeguards.

It has been a source of some puzzlement to me that the early discussion of the Non-Proliferation Treaty reflected a concern that it might interfere with international cooperation in atomic energy, and thereby hinder technological progress in this field, which has depended so heavily on such cooperation.

On reflection, it seems to me that the conclusion can only be reached that the Treaty is intrinsically favorable to international cooperation. The reason for this is a simple one. For countries with advanced nuclear energy programs, the concern in sharing their progress with other nations has always been that their assistance might be turned to military purposes. Despite this concern, and thanks to arrangements such as safeguards, there has been an unprecedented degree of international cooperation in atomic energy.

Limitations on cooperation, where they exist, derive from the concern that the cooperation might be turned to military purposes; in other words, that it might lead to proliferation of nuclear weapons. By increasing the assurance that additional nations will not acquire nuclear weapons, either through their own efforts or with the direct or indirect assistance of others, the Non-Proliferation Treaty can only improve the atmosphere for even broader and more intensive cooperation in the peaceful uses of atomic energy.

However, as we are all aware, the Treaty does not limit itself to implicit assurances of continued and strengthened international cooperation. Explicit provisions to this effect are included in the Treaty in the form of Article IV, which deals with international cooperation in general peaceful uses of nuclear energy, and Article V, which deals with the special case of peaceful uses of nuclear explosions. These provisions give effect to the principle that nations who agree to deny themselves nuclear weapons should enjoy the fullest possible access to the peaceful benefits of nuclear energy.

Finally, let me touch on the subject of international trade. In nuclear terms, international trade means, above all, trade in nuclear fuel. I believe there is a general consensus that, given the growing capabilities by many nations to manufacture nuclear power equipment they require, trade in reactors themselves and in related equipment, while by no means insignificant, will not be of comparable importance. On the other hand, trade in nuclear fuel, both natural and enriched, will remain an impressive item of commerce until breeder reactors reduce fuel costs to a minor factor and permit the use of even high-cost natural uranium, which is likely to be available in nearly every nation.

It was, of course, in relation to trade in nuclear materials that safeguards first acquired their importance, and once again, the strengthened assurances of peaceful use that will result from the NPT can exercise a salutary effect on this trade in the future. Competition for valid commercial reasons—competition in price, performance, and service—can and should play a major role in bringing the peaceful benefits of nuclear energy to an ever-widening segment of the world's people. To give this constructive sort of competition free reign, and to avoid irreparable harm to world security, competition in safeguards must be avoided. It is toward this end that the Non-Proliferation Treaty, by assuring the widespread application of effective international safeguards, may make one of its most vital contributions.

THE NEGOTIATION OF SAFEGUARDS IN THE NON-PROLIFERATION TREATY

HERBERT SCOVILLE, Jr., *Assistant Director, Science and Technology*
U.S. Arms Control and Disarmament Agency, Washington, D.C. 20451

INTRODUCTION

The Non-Proliferation Treaty represents the end product of several years of intensive diplomatic negotiations in many forums, especially the Eighteen-Nation Disarmament Conference (ENDC) and the United Nations General Assembly. It also represents the culmination of a long-standing, major and bipartisan foreign policy objective for the United States; an objective that seeks to increase global stability and security, as well as to provide opportunities for disarmament, by preventing the further spread of nuclear weapons.

Of course, this objective was shared by many other countries and, indeed, the General Assembly commended the Treaty by the overwhelming majority of 95 to 4 in June 1968. At that time, the General Assembly urged that the Treaty be opened promptly for signature and this was done on July 1 in Washington, London, and Moscow. Over 80 countries have now affixed their signatures to it. Although the Treaty was reported favorably by the Foreign Relations Committee by a vote of 14 to 3, and has always been popular in Congress, the US Senate's advice and consent has been delayed. However, President Johnson, as well as the leadership in the Senate, have indicated their desire to move the treaty promptly. Also, I should note that President-Elect Nixon has publicly endorsed the Treaty and has expressed the hope that it can be universally adopted. Although Mr. Nixon supported a pause in US ratification because of events in Czechoslovakia, he has stated that he fully expects to implement the Treaty in his new Administration.

In essence, the Treaty is intended, first, to obligate those countries not now having nuclear weapons to neither produce nor receive them in the future; second, to obligate nuclear-weapon countries to not help non-nuclear countries in obtaining such weapons; third, to provide assurance, through international safeguards, that the nuclear materials used in peaceful nuclear activities of these countries are not diverted to making nuclear weapons; fourth, to facilitate the sharing of peaceful benefits of nuclear energy with these countries; and fifth, to give recognition to the determination of all the parties to the Treaty that it should lead to further progress toward arms control and disarmament.

My purpose in this presentation is to elaborate on the third intention–providing assurance through international safeguards—which was embodied in Article III of the Treaty—the safeguards article.

ARTICLE III—SAFEGUARDS

It is difficult to overemphasize the importance of this safeguards article. It is designed to verify important Treaty obligations and thereby serve in itself as an important instrument for dispelling suspicions, reducing tensions, and increasing trust. The extensive application of mandatory safeguards will reduce concerns about providing nuclear material, specialized equipment, and information to non-nuclear-weapon States. It will thus greatly facilitate and accelerate cooperation among all Parties in the peaceful development of nuclear research and industry. With its fissionable material safeguarded, a country can carry out its peaceful programs without fear of accusation that it is secretly making weapons.

In broad outline, the four paragraphs of Article III provide the following:

Paragraph 1 provides for verification, by means of international safeguards, of compliance with the obligations assumed by non-nuclear-weapon parties to the Treaty to ensure that nuclear energy is not diverted from peaceful uses to nuclear weapons or other nuclear explosive de-

vices. Such safeguards are to be applied on all source or special fissionable material in all of the country's peaceful nuclear activities. Furthermore, the safeguards are to be as set forth in agreements to be negotiated and concluded with the IAEA in accordance with the IAEA's statute and safeguards system.

Paragraph 2 prohibits the provision by any of the parties of (a) source or special fissionable material, or (b) equipment or material especially designed or prepared for the processing, use, or production of special fissionable material, to any non-nuclear-weapon State for peaceful purposes, unless the source or special fissionable material shall be subject to the safeguards required by Article III.

Paragraph 3 prescribes that the safeguards be implemented to comply with Article IV of the treaty, which deals with furthering the peaceful uses of nuclear energy, and to avoid hampering either the economic and technological development of the parties or international cooperation in the field of peaceful nuclear activities.

Finally, paragraph 4 of the safeguards article concerns the manner in which the agreements called for in paragraph 1 shall be concluded. Non-nuclear parties may conclude such agreements either individually or together with other States in accordance with the statute of the IAEA. Euratom States, for example, could therefore work out safeguards arrangements as a group. Paragraph 4 also provides that negotiations of safeguards agreements shall begin within 180 days from the date of original entry into force, and allows 18 months for concluding these negotiations. This schedule provides for a transition period during which the arrangements for treaty safeguards can be worked out and put into operation.

The negotiation of the safeguards article turned out to be one of the main sticking points in the NPT. It had always been United States policy to work toward internationally administered safeguards; safeguards in which all countries could be confident that no diversion of nuclear material was taking place. However, in addition to the IAEA safeguards system, the Euratom countries already had their own international system and were concerned about allowing "superposition" of the IAEA safeguards system in their countries. They feared that duplication could result in abandonment of the entire Euratom system and thus would have very unfavorable effects on progress toward European unity.

The Soviet Union agreed to mandatory safeguards for non-nuclear-weapon signatories of the Treaty, but considered that such safeguards should be administered by the IAEA. They opposed acceptance of Euratom safeguards as a total substitute for IAEA safeguards under the Treaty because they claimed Euratom safeguards amounted to self-inspection of Euratom members. The problem, therefore, was how to specify international safeguards of such a nature that all parties could have confidence in their effectiveness while, at the same time, avoiding unnecessary duplication by the IAEA of existing records and safeguards.

There was a lengthy impasse in the negotiations over the safeguards article until the early Fall of 1967. Then, after lengthy discussions in Geneva, in the capitals of the NATO alliance members, in the North Atlantic Council, and in Euratom, the United States, on November 2, presented a revised compromise draft as a basis for continued negotiations. Throughout November and December we urged Soviet acceptance of the November 2 draft.

Finally, when the ENDC reconvened in Geneva on January 18 of this year, the Soviet Union agreed, at the last moment, to the November 2 draft. A complete draft treaty, including the safeguards article, was submitted to the ENDC on that same day and it is this safeguards article—the result of very complex and difficult negotiations—that appears in the final version of the Treaty. This article indeed provides for safeguards of such a nature that all parties can have confidence in their effectiveness and it enables the IAEA to carry out its responsibility of providing assurance that no diversion is taking place. Furthermore, the article does not interfere with the continuation of the Euratom safeguards system under the NPT, under arrangements whereby the IAEA can satisfy itself that no diversion of nuclear material is taking place.

There were also other problems in negotiating Article III. For example, it was noted adversely by many States in the ENDC, in the UN, and in consultations in the North Atlantic Council, that Article III formulations being considered did not require safeguards on the peaceful nuclear activities of nuclear-weapon parties. Non-nuclear-weapon States feared this would discriminate against them since safeguards inspections might disclose industrial secrets. The US does not think this a legitimate concern since safeguarding does not need to involve such disclosure. However in response to this, and to remove the grounds for claims that the safeguards required by the NPT would unfairly place industrial, economic, or other burdens on non-nuclear-weapon parties, President Johnson, on December 2, 1967, stated that the United States is not asking any country to accept safeguards that we are unwilling to accept ourselves. He announced that "when such safeguards are applied under the treaty, the United States will permit the International Atomic

Energy Agency to apply its safeguards to all nuclear activities in the United States—excluding only those with direct national security significance."

The United Kingdom made a parallel offer, but the Soviet Union has so far declined to do so, questioning the relevance of such safeguards in countries that continue to produce nuclear weapons as permitted by the NPT.

Many other questions were raised and answered in diplomatic discussions related to the provisions of Article III: questions, for example, relating to the applicability of treaty safeguards to mines and ore processing plants, and to prohibitions on fast breeder reactor experiments or on stockpiling of nuclear material. For those who are interested, I would say that the answer was, and is, that the NPT requires no change in the present IAEA practice of not applying safeguards to uranium mines and ore-processing plants. There were also questions as to what activities are permitted or forbidden under the Treaty. For example, fast breeder experiments or nuclear material stockpiles for peaceful uses are not prohibited as long as these are subject to the safeguards requirements prescribed in Article III. On the other hand, manufacture of components that could only have relevance to a nuclear explosive device would be forbidden.

Furthermore, the Treaty recognized the basic scientific fact that safeguards are not static and that technology will be continually changing. As a result of discussions with allied States, the US incorporated a preambular paragraph into the NPT in which the parties express "their support for research, development, and other efforts to further the application, within the framework of the International Atomic Energy Agency safeguards system, of the principle of safeguarding effectively the flow of source and special fissionable materials by use of instruments and other techniques at certain strategic points." Of course, it was recognized that this principle—which is also referred to in Article III—was a goal, but that efforts should continue for its fullest implementation as the pertinent techniques become available. It was not intended to imply that existing safeguards systems could not now be satisfactorily applied under the NPT.

It is the intent of this paragraph not only to enlist the support of the signatory States in safeguards research and development, but also to suggest that such research should be directed toward the evolution of the most effective safeguards system which, by the use of instruments and other techniques, will be as nonintrusive as possible. The United States is firmly committed to this policy.

Specifically to support the objectives of this preambular paragraph, concerned US government agencies, and in particular the USAEC and US Arms Control and Disarmament Agency (ACDA), have instituted R & D safeguards programs on an expanded and priority basis. For its part, ACDA has a statutory as well as a special interest in the implementation of the NPT. It is essential that, under the NPT, IAEA safeguards procedures have a high degree of credibility and yet not be more intrusive than is absolutely necessary. These diverse requirements urgently underscore the immediate tasks that our research and development programs must face.

You have heard today some of the elements of the AEC research program as well as those of several other countries. Three examples of ACDA's R & D efforts in support of NPT safeguards might be cited. These activities, it should be stressed, have been undertaken in close coordination and cooperation with the safeguards research program of the AEC, the IAEA, and other countries.

Portable Instrumentation

To assist an international inspector to perform more effectively the task of taking field inventories of plutonium placed under IAEA safeguards, we have, with the assistance of NRL and Brookhaven, made prototype portable instrumentation available for field test and evaluation to identify the presence of plutonium by detecting spontaneous fission neutrons from ^{240}Pu, which is always associated with ^{239}Pu. This coincidence neutron detector should thus be of value in the field inventory and verification of plutonium. We have also made available to the IAEA for evaluation, a portable multichannel gamma-ray spectrometer. This instrument is to be used in the inventory and field assay of the enriched uranium in unirradiated reactor fuel elements now under international safeguards. Ultimately we would hope to see the IAEA equipped with a variety of portable instruments specifically designed to meet the needs of international safeguards.

Tamper-Resistant Instrumentation

To ensure that an international inspection system will have the option of installing unattended safeguards instrumentation where it is appropriate, we have undertaken, in cooperation with the AEC and the IAEA, a joint program to develop a tamper-resistant/tamper-indicating instrumentation system. This unobtrusive safeguards system will include the devices and the "secure volume" technology required to prevent tampering with both sensors and data recording units. Quite

apart from the unobtrusive features of such safeguards instrumentation, the use of this type of equipment should contribute to the most efficient use of the inspectors' time and effort and thus directly to lowering the total costs of international safeguards inspection.

Minor Isotope Safeguards Techniques (MIST)

The prospective use of the minor uranium isotopes in safeguards applications is based on the abundance of the natural isotopic tracer ^{234}U found in all uranium samples, and the artificial isotopic tracer ^{236}U found in those samples that contain at least some uranium that has been exposed to neutron irradiation. It is the premise of the MIST program, presently being undertaken, that these additional tracers can be used in the following ways to supplement existing safeguards procedures:

1. Characterize or "finger print" individual or specific batches of uranium so they may be followed and identified at different points in the fuel cycle.
2. Identify the processes that have altered the isotopic composition of the uranium; for example: enrichment and irradiation.
3. Verify that certain safeguarded industrial operations have been performed as specified; for example:
   (a) The blending or mixing associated with reactor fuel element fabrication.
   (b) The dissolution associated with the chemical processing of irradiated reactor fuels.
   (c) The recovery of fissionable scrap and waste materials.
4. Monitor overall the operation of certain major nuclear facilities; for example: reactor operations and chemical separation plants.

It should be noted that many of the applications of MIST cited above are also directly applicable to plutonium using similar ratios of the minor isotopes ^{241}Pu and ^{242}Pu to the more abundant plutonium isotopes ^{239}Pu and ^{240}Pu.

ARTICLES IV AND V

I should not discuss the NPT in this forum without at least touching upon two other articles of that Treaty in which many of you will be greatly interested.

Articles IV and V of the Treaty encourage the peaceful uses of nuclear energy. Article IV provides that nothing in the Treaty will be interpreted as affecting the right of all parties, without discrimination, to use nuclear energy for peaceful purposes.

Article IV also contains an undertaking by all parties to facilitate, and affirms their right to participate in, the fullest possible exchange of equipment, materials, and scientific and technological information for the peaceful uses of nuclear energy. Finally, it requires those parties in a position to do so to cooperate in contributing to the further development of peaceful applications of nuclear energy, especially in the territories of non-nuclear-weapon States and with due consideration for the needs of the developing areas of the world.

Article V was developed in the context of the undertaking by non-nuclear-weapon parties in Article II not to acquire nuclear explosive devices even for peaceful purposes. It provides assurance to such parties that they will not lose, but will in fact gain by such renunciation, the potential benefits from peaceful applications of nuclear explosions. It is also designed to make it completely clear that there would be no economic incentive for them to try to develop their own nuclear explosive devices for such purposes. Specifically, the parties to the Treaty agree to take appropriate measures to ensure that the potential benefits of such peaceful applications will be made available to non-nuclear-weapon parties on a non-discriminatory basis and that the charge to such parties for the nuclear explosive devices used will be as low as possible and exclude any charge for research and development on those devices. The Article requires that such benefits shall be made available in accordance with the Treaty—which would preclude non-nuclear-weapon States from acquiring the nuclear explosive devices themselves or control over them. Thus, the devices would remain under the custody and control of a nuclear-weapon State, which would, in effect, provide a nuclear explosion service.

Finally, I would like to note that the NPT obviously is but a step in the difficult and long road toward lasting global stability and peace. As President Johnson said before the UN General Assembly on the day last June when he endorsed the NPT:

"I believe that this treaty can lead to further measures that will inhibit the senseless continuation of the arms race. I believe that it can give the world time—very precious time—to protect itself against Armageddon. And if my faith is well founded, as I believe that it is, then this treaty will truly deserve to be recorded as the most important step toward peace since the founding of the United Nations."

DISCUSSION

G. ROBERT KEEPIN

University of California, Los Alamos Scientific Laboratory
P.O. Box 1663, Los Alamos, New Mexico 87544

N. OTERO *(Spain)*

Can you tell us what Eastern European countries actively participate in the safeguards system of the IAEA, or have accepted the Agency's safeguards system?

S. EKLUND *(IAEA)*

At the present, no Eastern country has opened itself to inspection under the IAEA safeguards system, but the European countries participate in the Non-Proliferation Treaty, and in this way they will become partners in the Agency's safeguards system.

OTERO

How many projects of nuclear power stations or related fuel cycle installations have been put under IAEA inspection to check their fitness, when in operation, for the implementation of the Agency safeguards system? Are some of them in nuclear weapons countries?

EKLUND

There are at present three nuclear power stations under IAEA safeguards. The Yankee station in the United States, the Bradwell station in the United Kingdom, and the Tokai Mura station in Japan.

OTERO

These are already-existing stations, and I don't think either the stations or the associated projects were designed to implement IAEA safeguards. My question is if future projects under consideration are requesting your services in order that, in the future, IAEA safeguards could be applied effectively to them.

EKLUND

Well, I am not aware of any nuclear power station under construction where designs, for example, were submitted to the Agency in order to facilitate future safeguarding. I would, however, like to touch upon a border-line case, and that is the heavy-water-moderated station in Czechoslovakia, where we have an Agency research contract with that country to study the implementation of safeguards in heavy-water-moderated stations. A second case, and a new case, is the plant in Argentina which is now under construction, where we have the same arrangement as with regard to the station in Czechoslovakia just mentioned.

W. P. DONOVAN *(IAEA)*

I realize it may be an imposition, but could the panel be polled as to whether their outlook for for the future is optimistic or pessimistic with regard to the future application of international safeguards around the world?

EKLUND

Well, if I may answer that question. My personal position is one of great optimism with regard to the development of international safeguards; I think safeguards is very necessary simply because I don't believe that the world will become safer if a large number of countries get access to fissile material without control.

D. GUPTA *(Germany)*

I have a question for Dr. Keepin. You have mentioned the accuracy of 0.5% for the MTR elements. Have you checked this method for light-water-reactor-type elements or elements with low concentration of uranium and plutonium, as for example is found in some recycle fuel pins?

G. R. KEEPIN *(USA)*

Not with actual fuel elements, but we have done assay with various ratios of ^{235}U to ^{238}U in idealized disk-geometry systems. We intend to go to a Yankee-type fuel element ($\sim 3\%$ ^{235}U enrichment), and see no real problem there. The figure of half-percent accuracy which I mentioned is obtained using very careful standards. And the 2% accuracy figure applies to relative isotopic abundance as determined from kinetic response measurements based on response

data from disks of pure fissionable material. These methods will be described in some detail in the paper by Dr. Augustson in tomorrow's session on Safeguards Technology, and I would like to refer you to that session. Our measurements on somewhat idealized systems, including the MTR element, have given assay accuracies in the neighborhood of 0.5%, and we expect that this kind of accuracy would be applicable to the assay of low-enrichment light-water-moderated fuel elements. Thus far we have not yet actually done assay measurements on the fuel elements themselves, but we plan to do so in the future.

W. FINKE *(Germany)*

Could you indicate where the areas of disagreement lie for the verification of the EURATOM system by the International Atomic Energy Agency? I presume there are areas of disagreement, or I expect we would already have a general agreement on verification of the EURATOM system by the International Atomic Energy Agency.

EKLUND

Well, I wonder if I can turn this question over to Mr. Nakicenovic of the IAEA safeguards staff?

S. NAKICENOVIC *(IAEA)*

To my knowledge, there exists no exhaustive, comprehensive comparison, but I would prefer to identify some of the points on which both systems were in agreement rather than emphasize disagreement, as the latter requires comprehensive study. I had noted from Dr. Spaak's paper on EURATOM Safeguards several points where agreement exists. As several speakers have pointed out, safeguards terminology is sometimes confusing; for example, what EURATOM calls the "technical characteristics" of safeguards, IAEA calls "design information." Both systems seem to consider this necessary. What IAEA calls "the system of records and reports" I understood Dr. Spaak calls "the declaration of all information, which must be available to the community." Inspections are also carried out in both systems. In connection with the frequency of inspection, I understood that the Community has a right to inspect any facility any time, regardless of size, type of power, or conditions. I understood by implication that this right is not applied to every facility. The frequency of the Agency's inspections is based on the quantity of safeguarded material, and the Agency has the right to access at all times when this material is in excess of a certain quantity. I also understood that another common practice was that inspectors may be designated after consultation with the State involved and with its agreement. The EURATOM system is being technically developed using joint technical working groups; the Agency is also doing this. Both systems adopt new, more effective techniques while keeping the burden on the operators to the minimum possible. In both systems we have the same emphasis: to apply minimum inspections with a minimum of inspectors, and to use techniques that are available today.

Unlike the Agency, EURATOM's safeguards system does not contain a termination clause. In conclusion, I should like to mention that the Agency's safeguards system must be as effective as possible but also acceptable to all States. Thus, as has already been mentioned, the provisions of the Safeguards System[1] are valid if incorporated in a safeguards agreement, which determines the basic obligations, and the basic rights, of both sides.

EKLUND

May I ask if Mr. Spaak wants to comment on this?

F. SPAAK *(EURATOM)*

Mr. Chairman, as to the existing systems, I don't think there are many great differences in the application of the systems for a very good reason. And that is that we have always developed the EURATOM system in such a way that it would be entirely compatible with the IAEA system. And this, I think, has been generally recognized. As to the other question of knowing what type of difficulties we should expect to meet when we start working in closer harmony, which would be the next step in our relationship, that, I'm afraid, would be a bit early to say, as we haven't started. We haven't even had a chance of talking together, for a very good reason—that member countries haven't yet all signed the treaty of nonproliferation.

EKLUND

Thank you. Are there any other questions? If not so, before closing the meeting I would like to thank all of you for your participation. I think the large audience we have here today

1. IAEA Statute: "Information Circular 66—The Agency's Safeguards System (1965)."

clearly testifies to the greatly increased interest in the timely questions of international safeguards. I would also like to thank my fellow members of the panel very much for their contribution. And especially I would like to address myself to Dr. Keepin, who as Technical Secretary has shown an unfailing interest and effort in order to organize this meeting; I suggest that we end the session by giving him an applause.

FISSIONABLE MATERIALS SAFEGUARDS TECHNOLOGY

Papers presented on November 13,1968 at the Winter Meeting of the American Nuclear Society held in Washington, D.C.

APPLICATION OF THE SLOWING DOWN TIME SPECTROMETER FOR THE CONTROL OF FISSIONABLE MATERIAL

D. Stegemann and H. Seufret
Kernferschungszentrum Karlsruhe, Germany

The slowing down time spectrometer has been applied for the quantitative and non-destructive determination of fissionable material in fuel pins. For nuclear safeguards such a control instrument should meet the following conditions:

1) The total amount of fissionable material within a fuel pin or subassembly has to be determined;

2) A discrimination between different fissionable isotopes, in particular U^{235} and Pu^{239}, must be possible;

3) The method must be tamperproof, which means in this case that adding absorber materials--in particular resonance absorbers--to simulate a lower content of fissionable material should be detectable;

4) The measurements should be fast because of the high throughput in fuel fabrication plants and

5) The cost of the instrument should be low for economy of the fuel cycle.

These conditions can be met by application of the slowing down spectrometry technique. The assembly used for these investigations is shown in Fig. 1. The main components are a pulsed 14 MeV accelerator and a lead cube of 1.3-m side length. The lead cube contains channels for the accelerator target and for insertion of fuel pins and detectors. Pulses of 14 MeV neutrons are shot into the lead cube repetitively. Within the first microsecond the

neutrons undergo inelastic collisions and their average energy is degraded to about 100 keV. Thereafter elastic collisions take place, the neutron energy loss per collision being relatively small due to the heaviness of the lead nuclei. Below 100 keV there exists a relation between the time after occurence of the neutron pulse and the average neutron energy in the lead pile. As a result of this, quasi monoenergetic neutrons with an energy resolution of about 30% are available in the energy range from about 20 keV down to thermal. More details of this spectrometer have been published.[1-2]

The measurements are performed by introducing the fuel pin, to be investigated, into the experimental channel shown in Fig. 1. The fuel pin is surrounded by a multidetector array for the detection of fission neutrons. The fission events are induced by the neutrons in the lead cube impinging onto the pins. The time-dependent fission neutron counting rate registered by a multichannel time analyzer is proportional to the neutron flux $\phi(E)$ and $\Sigma_f(E)$ of the fissionable material. Relating this to the neutron counting rate of reference pins with known masses of fissionable material, one obtains the content of fissionable material of the pin investigated.

The discrimination between U^{235} and Pu^{239} and higher Pu isotopes makes use of the different values of the fission cross sections as a function of neutron energy preferably in the resonance region. Due to resonance self-shielding effects within the pin, the variations in the fission ratio of U^{235}-to-Pu^{239} vs. neutron energy are, however, decreased; the higher the enrichment of the fuel pin, the stronger the decrease. Whereas preferably the lower fission resonance are used for discrimination between U^{235} and Pu^{239} for enrichments up to 3 or 4% in U^{235} and Pu^{239}, typical for LWR fuel pins, the upper resonances and the unresolved resonance region must be used for highly enriched fuel pins, typical for fast breeder reactors.

To detect absorbers, in particular, resonance absorbers eventually added to the fissionable material, capture gamma rays can be measured in a second experimental channel simultaneously. The gamma counting rate vs. slowing down time shows resonances for this case as can be seen from the results given before.[1,2] By this procedure, the third condition, given above, is fulfilled also.

A set of pins recently measured had the following composition.

TABLE OF TESTPINS

PIN DIMENSIONS: LENGTH = 1000 mm, DIAM = 13 mm

PIN NO.	U - 235 ENRICHMENT (%)	PU - 239 ENRICHMENT (%)	U - 235 Mass [g]	PU - 239 Mass [g]	PU - 240 Mass [g]
ST 1	2	0	11.95	---	---
ST 2	2	0.5	12.00	2.75	0.25
ST 3	2	1	11.89	5.48	0.50
ST 4	2.5	0	15.05	---	---
ST 5	2.5	0.5	14.99	2.74	0.25
ST 6	3	0	18.08	---	---

The composition is typical for thermal reactor fuel pins. Another set of pins typical for fast reactors is available for measurement in the near future.

Fig. 2 shows the response curves for three of the test pins. The counting rate of fast fission neutrons is plotted against slowing-down-time or average pile neutron energy. These measurements have been performed primarily in the eV and tenth eV energy range because the difference between U^{235} and Pu^{239} fission resonances is most pronounced there. This is clearly visible in the response curves. The lower two curves are for pins with 2% and 3% U^{235} enrichment, containing no plutonium, whereas the upper curve is for 1% Pu^{239} together with 2% U^{235} enrichment. The discrimination between U^{235} and Pu is clearly visible.

The evaluation procedure is illustrated in Fig. 3. The counting rate is integrated for the shaded regions. Ru-235 is the U^{235} contribution to the response curve and R_{Pu-239} is that of Pu^{239}. These two quantities as well as their ratio are of interest.

Fig. 4 shown Ru-235 as a function of U^{235} content of the test pins. The deviation from linearity is apparantly due to selfshielding effects within the pins. Fig. 5 shows the response function for Pu^{239} for inpins having two different enrichments of U^{235}. This response was obtained by subtracting the U^{235} contribution to the total response curve as determined from U^{235} test pins of equal enrichments. The selfshielding effects become here even more important. Finally, the ratio of Pu^{239} response to that of U^{235} versus Pu^{239} enrichment is shown in Fig. 6. This ratio is effective for discriminating between the two isotopes. From these results, for instance, it can be concluded that replacement of 5 to 10% U^{235} by Pu^{239} in a fuel pin with 2-3% of U^{235} could easily be detected.

These preliminary results are encouraging. The next steps in the program will be investigation of test pins typical of fast reactors and examination of the response curves at higher energies. Also error estimates for determination of the fissile isotopes will be further refined.

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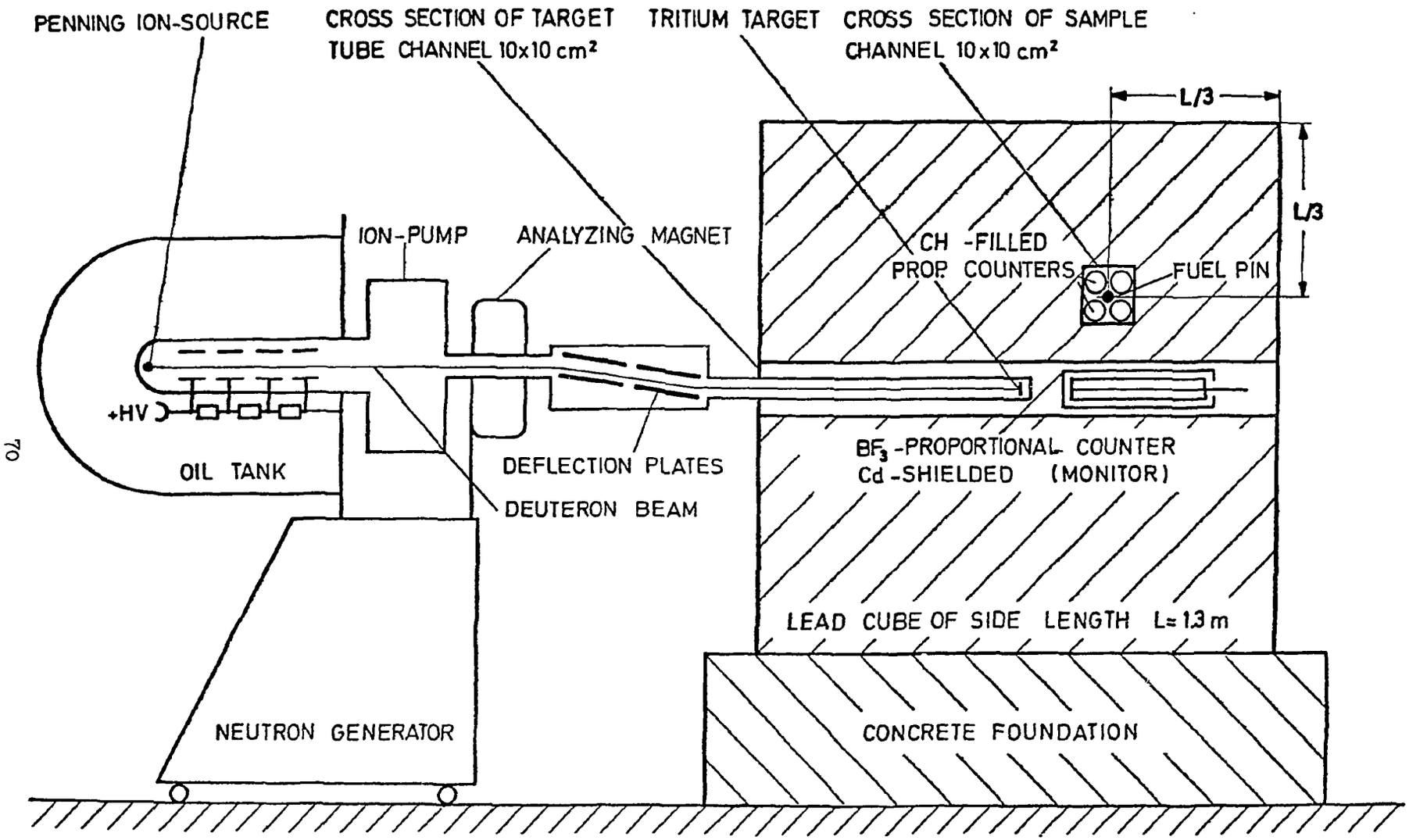
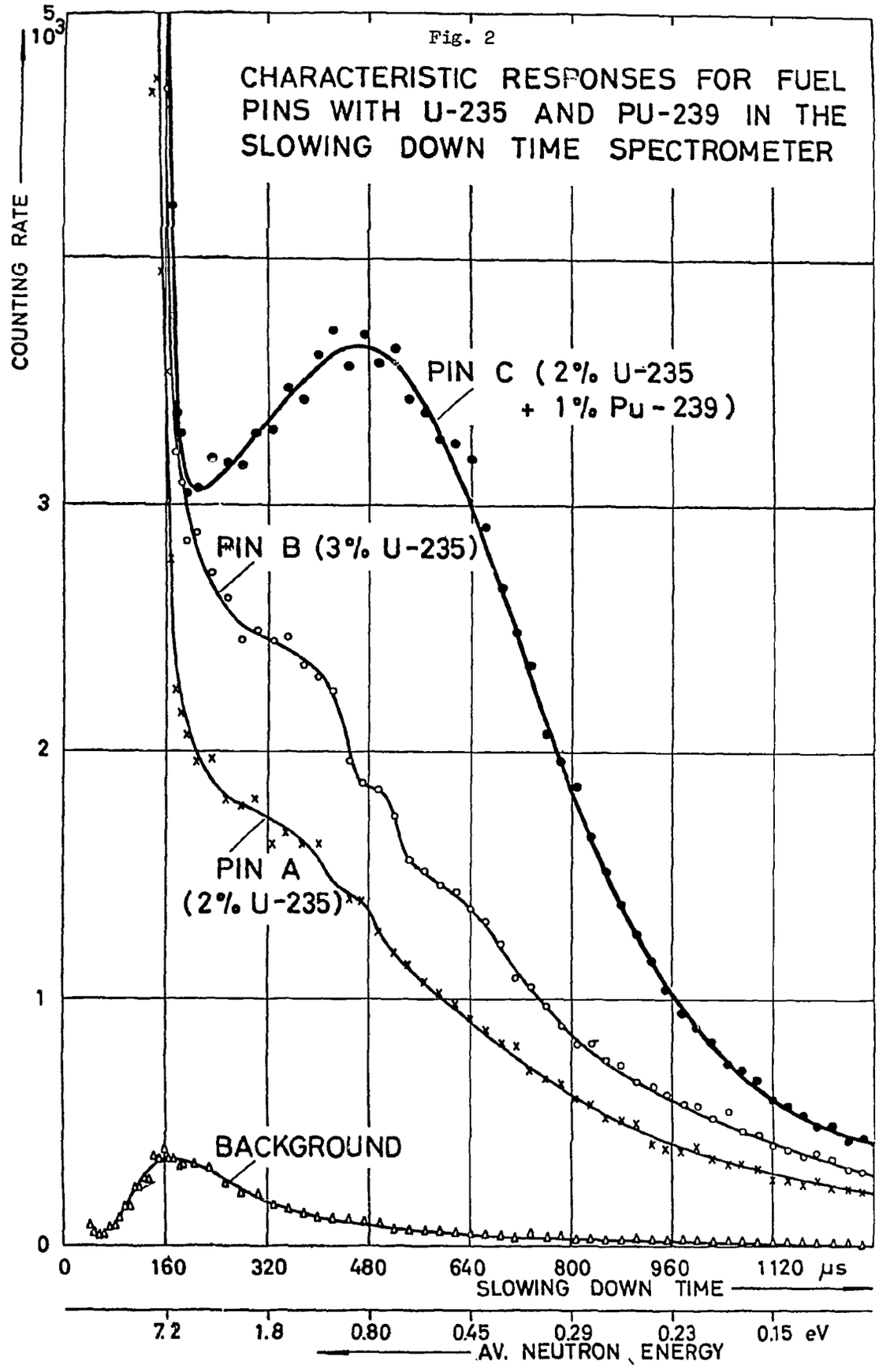


FIG. 1 SETUP OF NEUTRON SLOWING DOWN TIME SPECTROMETER

Fig. 2


CHARACTERISTIC RESPONSES FOR FUEL PINS WITH U-235 AND PU-239 IN THE SLOWING DOWN TIME SPECTROMETER
COUNTING RATE
5
10^3
3
2
1
0
PIN C (2% U-235 + 1% Pu - 239)
PIN B (3% U-235)
PIN A (2% U-235)
BACKGROUND
0
160
320
480
640
800
960
1120 μs
SLOWING DOWN TIME
7.2
1.8
0.80
0.45
0.29
0.23
0.15 eV
AV. NEUTRON ENERGY

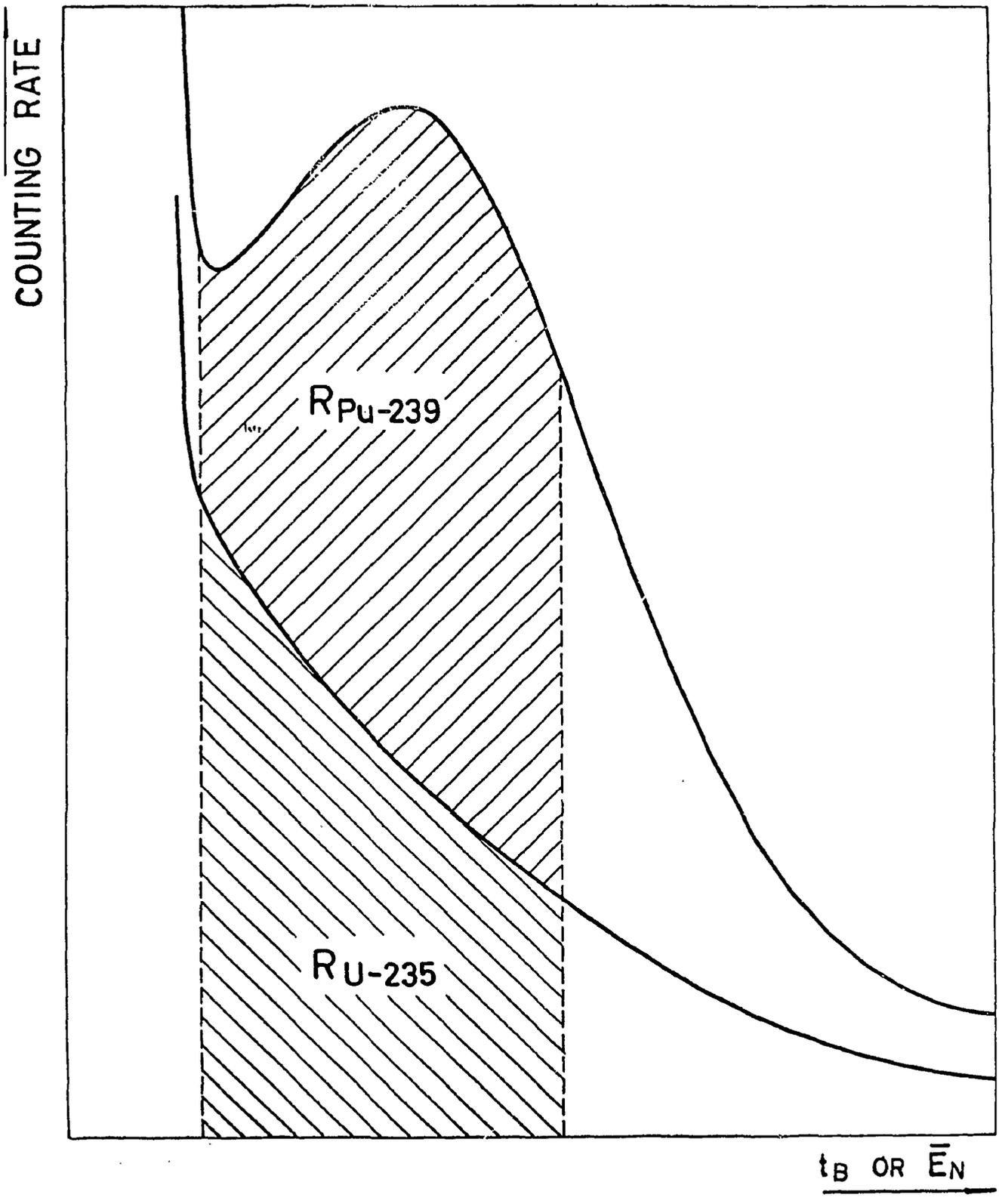


Fig. 3
EVALUATION PROCEDURE

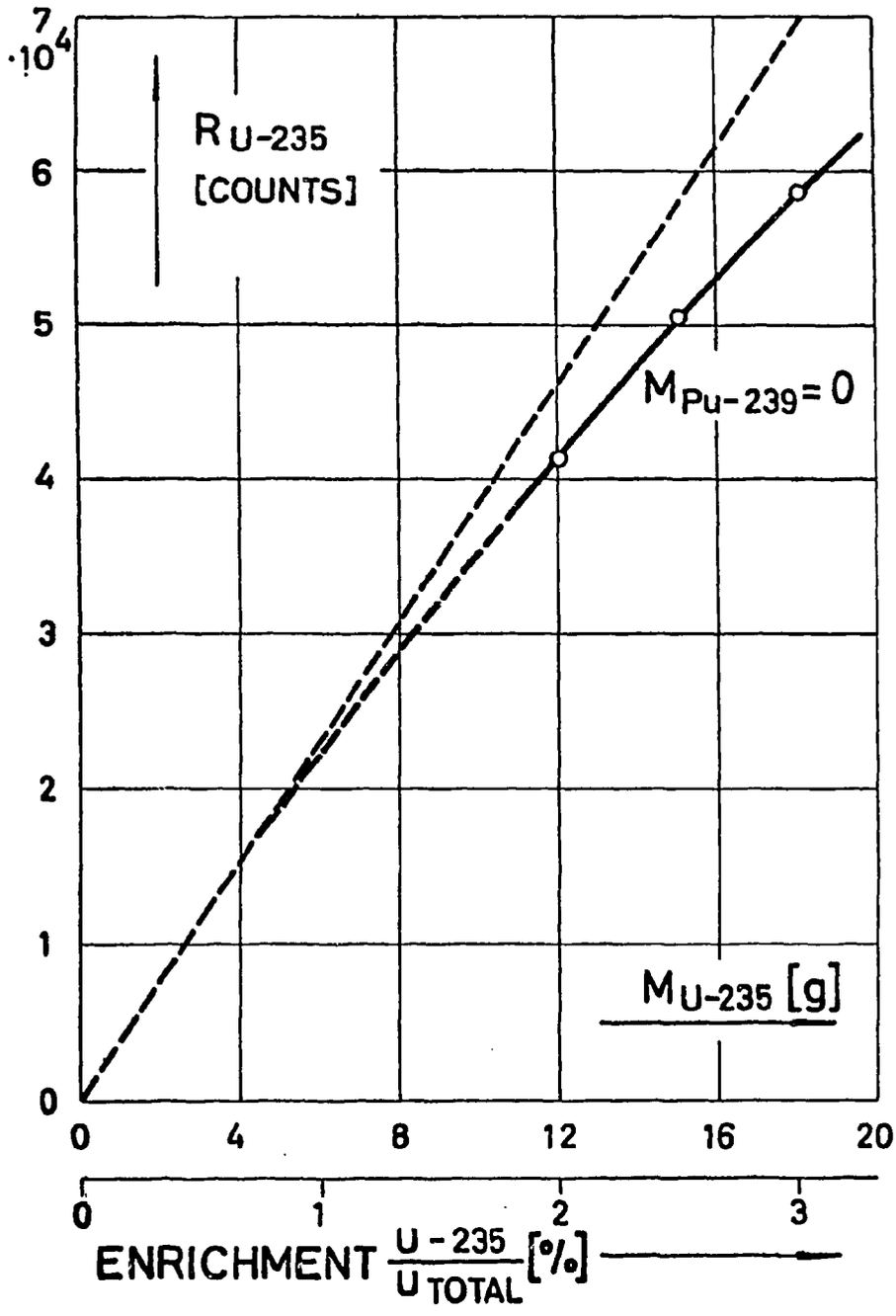


Fig. 4

U-235 RESPONSE, R_{U-235}, VERSUS U-235 ENRICHMENT

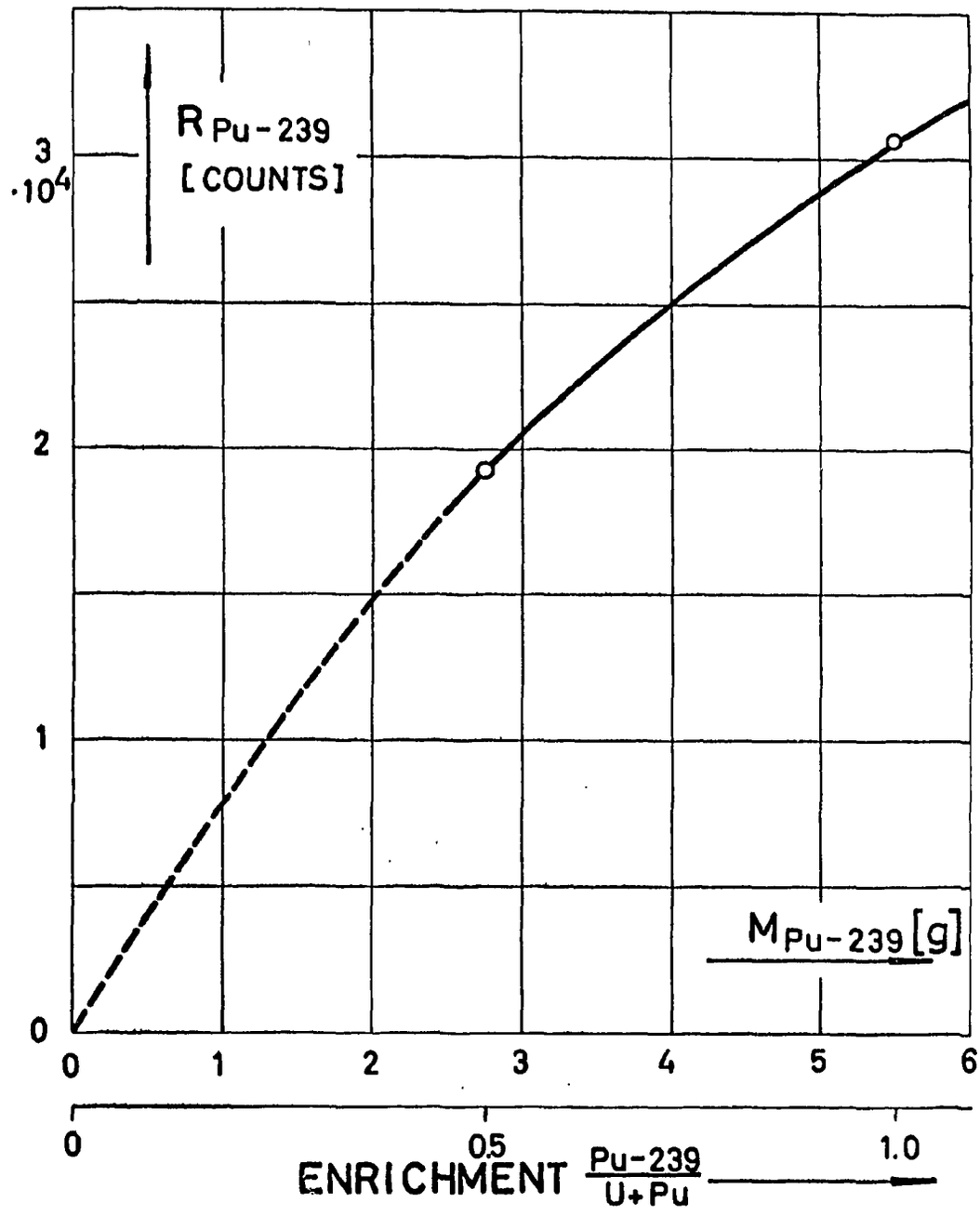


Fig. 5

Pu - 239 RESPONSE, R_{Pu-239}, VERSUS Pu - 239 ENRICHMENT

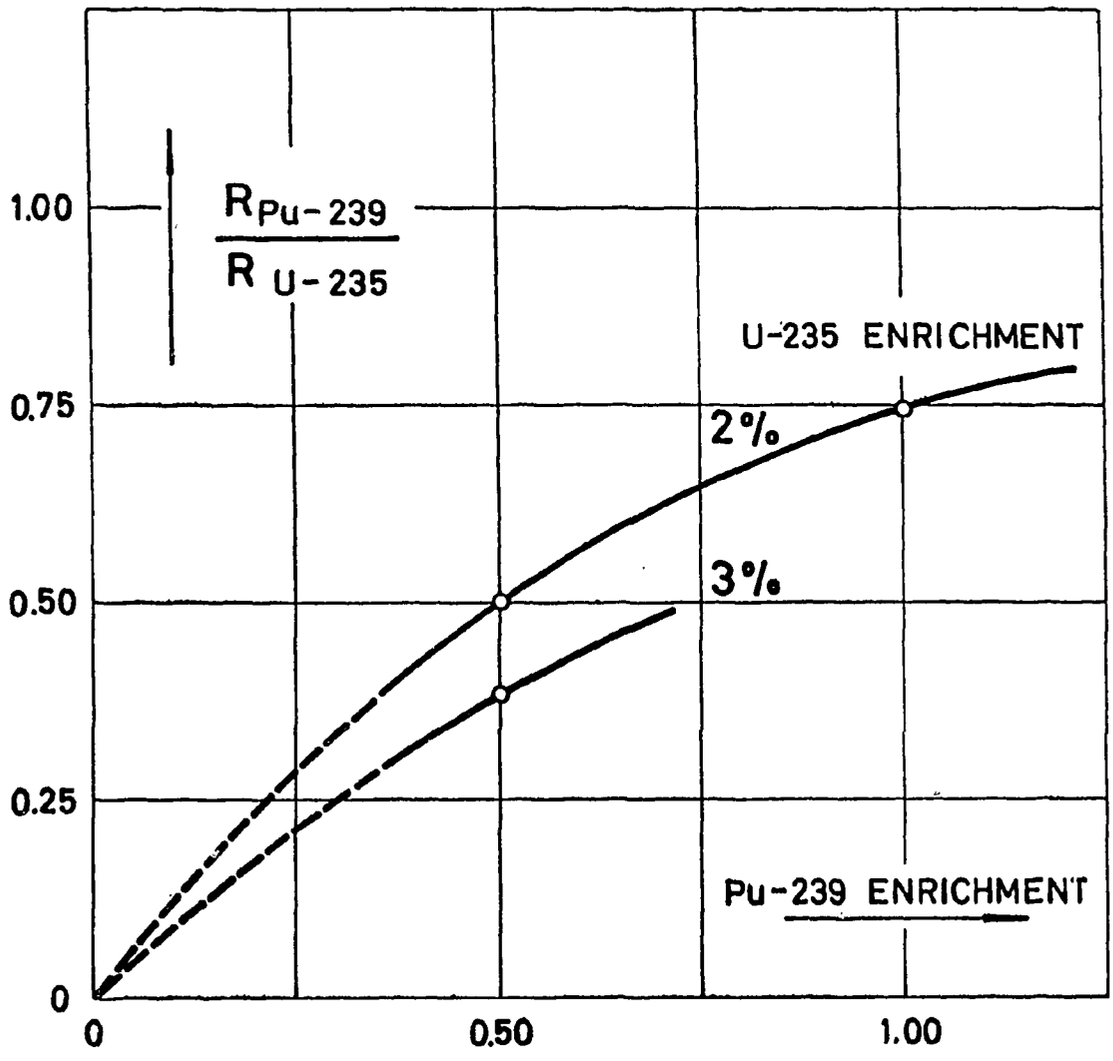


Fig. 6

$\frac{\text{Pu-239}}{\text{U-235}}$ RESPONSE RATIO, $\frac{R_{Pu-239}}{R_{U-235}}$ VERSUS Pu-239 ENRICHMENT

PROMPT AND DELAYED NEUTRON EXPERIMENTS

T. Gozani
Gulf General Atomic

Experimental Apparatus

The rapid increase in the prompt and delayed neutron yields in the energy range below about 8 MeV electron energy necessitates a good electron energy resolution in order to determine the basic discrimination ratios for the various fissile materials. However, for many practical applications, once these values are determined, one can relax the requirement of high energy resolution and use relatively broad energy spectrum, provided this spectrum stays unchanged. For some applications, such as the study of gamma lines following beta decay of fission products or the study of the time dependence of the delayed neutron population, even the last requirement is not essential. In the course of our research we have encountered all these different requirements.

In order to get good energy resolution, $\frac{\Delta E}{E}$, of about 1% FWHM an improved system, which is discussed briefly in the last annual report,[1] was used. The main features of this system compared to previous arrangements are: (1) improved and better defined energy resolution (while maintaining adequate neutron count rate with our low efficiency detector-system), (2) lower extraneous background even at higher electron energies (E > 7 MeV) obtained by using in all sensitive locations copper collimation and shielding instead of lead, which has a lower (γ, n) threshold. This improvement allows one to use the same system for the entire energy range of 5 to 10 MeV.

Further studies which were aimed specifically at reducing the background at energies $\leq$ 5.5 MeV, (at higher energies it is practically negligible) showed that this

background does not come from (γ, n) reactions in the carbon electron collimator (carbon-13 has a threshold of 4.9 MeV). In fact it seems that carbon would not be an "offensive" material at all for photoinduced interrogation technique. A substantial reduction in the background was obtained by adding a 2-in.-thick lead shield (and 1 in. of highly borated polyethylene) in front of the detector tank and extending the copper collimation around the beam channel. All these modications have probably reduced the (γ, n) reaction in the small amount of heavy water present in the moderator tank. The performance of the present modified system, as shown in Fig. 1, has been extensively used as guidance in the design of the "second generation" high efficiency detector assembly, which is now being built.

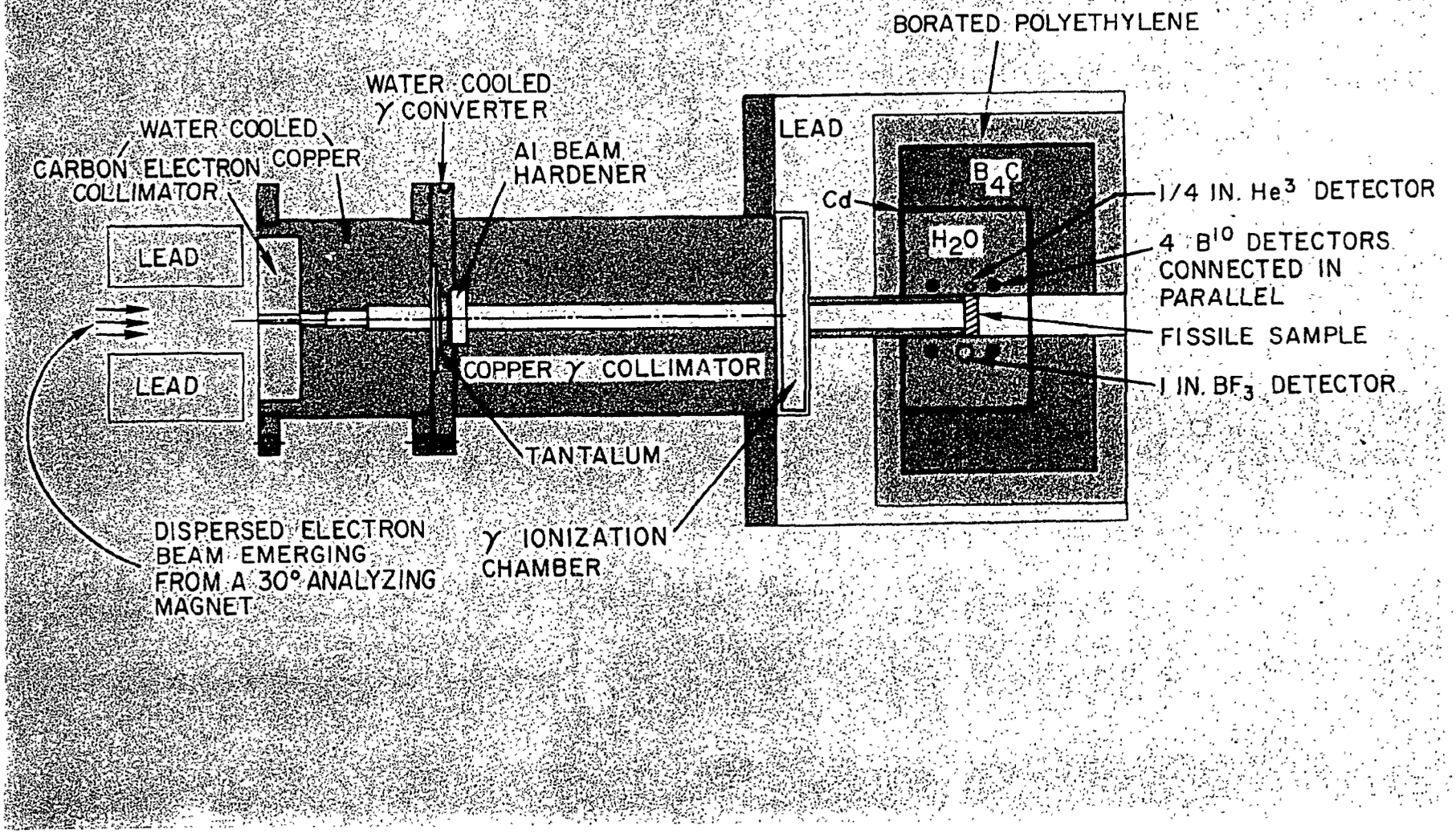
Before proceeding in the "screening experiments" (see previous progress reports) several auxiliary measurements were done. The main purpose of these was to determine the effect of thermal neutron multiplication in the fissile sample when placed in the center of the detector tank. This effect was found to be rather small (about 3%). At the same time the use of Cd lining in the central channel in the tank was found to reduce by 21% and 15% the counting efficiency of a central BF_3 detector and the four ^{10}B detectors (see Fig. 1), respectively. In addition, the perturbation effect of the BF_3 and ^3He detectors on the ^{10}B detectors was measured and found to be rather small. It should be noted that both the BF_3 and the ^3He detectors could be used in these experiments as a result of the substantial reduction of the gamma flash by the additional lead shielding.

Results

Using the setup shown in Fig. 1 the prompt and delayed neutron yields were measured over the electron energy range of 9 to 5.5 MeV for the following materials: ^{238}U, ^{239}Pu (plus 7% wt ^{240}Pu), ^{235}U, Pb, H_2O, C, GA-fuel slug (0.84 gr ^{235}U 93% enriched plus thorium) and Cu (for background measurements). The electronic counting system is shown in Fig. 2. The constant neutron background of the Pu sample (from the spontaneous fission of ^{240}Pu) was used to check for drifts in the electronic system and to check the reproducibility of sample positioning. The energy spectrum of the electron beam emerging from the

Fig. 1

NEUTRON DETECTOR TANK AND BREMSSTRAHLUNG CONVERTER-TARGET ASSEMBLY FOR PROMPT NEUTRON YIELD MEASUREMENT



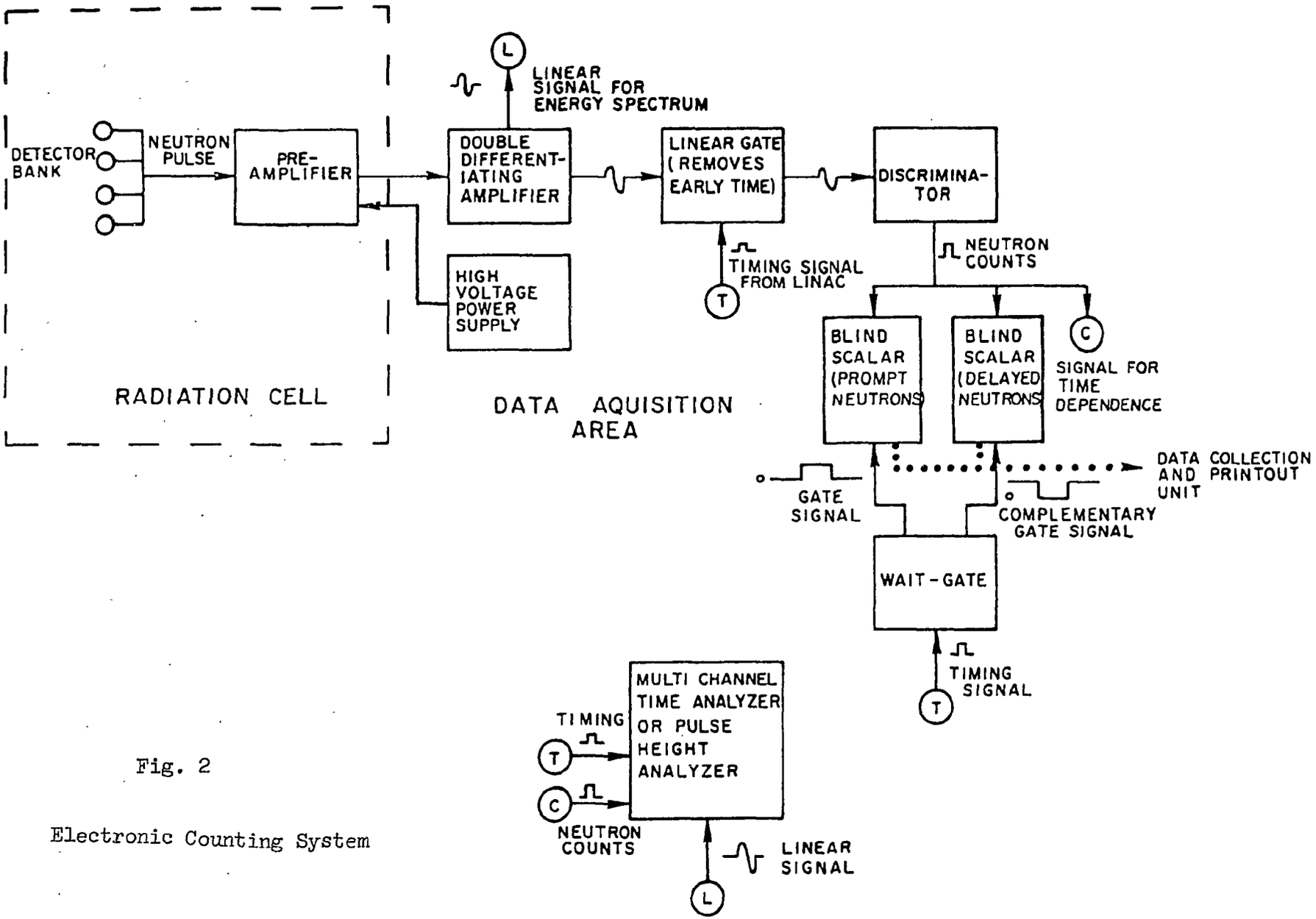


Fig. 2

Electronic Counting System

LINAC was measured at the beginning and end of each set of runs for each energy. The energy acceptance bin of the slit in our setup was ($\Delta E/E$) = 1% FWHM. Two monitors were used: the integrated electron current and the integrated gamma ionization chamber reading. The results presented here were normalized to the ionization chamber which is proportional to the incident gamma dose within 20%.

Figure 3 depicts the prompt and delayed neutron yields from the ^{238}U sample normalized to 1 volt of integrated response of the gamma-ionization chamber. The conversion factor from total count/volt to total neutrons produced per volt per gram sample is about 200 (using detection efficiency of 10^{-3}). The fitted curves agree very well with another independent set of measurements over fewer energy points but with better statistical precision. The delayed neutron yield follows the fission yield except for possible variations of the fractional yield of delayed neutrons (β) with energy. The prompt neutron yield contains both ($X\gamma$, f) and ($X\gamma$, n). However, below 6 MeV the curve describes the fission yield. It is obvious that in the lower energy range below the fission barrier ($E < 6$ MeV) the slope is very large, typically there is a factor of 10 change in yield for 200 keV energy change. Since the delayed neutron yield represents, to a reasonable approximation, the fission yield, a comparison was made between this curve and the theoretical curve[2] based on a simple barrier penetrability model with fission barrier $B_f = 5.8$ MeV, barrier "width" $E_p = 0.085$ MeV and neutron temperature of $T = 0.6$ MeV. The comparison is shown in Fig. 4. Both curves are normalized at 8 MeV. They coincide down to 6.75 MeV with increased deviation at lower energy. The over-all agreement between the measurements and the simple model is rather gratifying, and in the future we shall proceed to refine both, specially in the low energy region.

Measurements similar to those shown in Fig. 3 were also done for ^{235}U and ^{239}Pu. The ratios of the prompt and delayed neutron yields of ^{238}U to ^{235}U and of ^{239}Pu to ^{235}U are shown in Fig. 5. These ratios constitute the discrimination ratios which can be obtained from total prompt and delayed neutron yields. Figure 5 shows that discrimination ratios based on prompt yields of ^{239}Pu and ^{235}U range from 1.2 at $E > 6$ MeV to 9 at $E \simeq 5.5$ MeV. Entirely different behavior is exhibited by the ^{238}U to ^{239}Pu discrimination ratio which changes by a factor of

Fig. 3

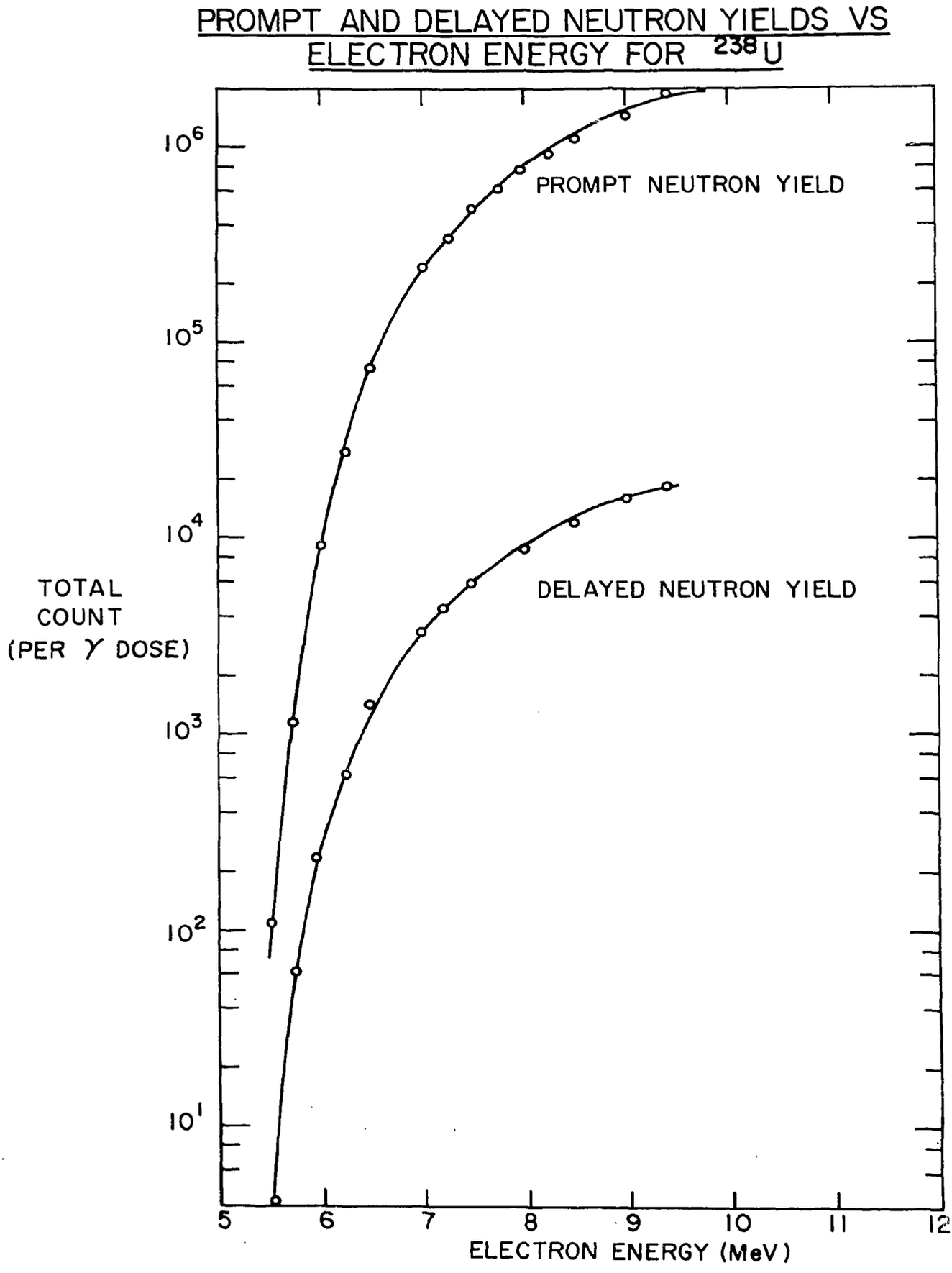

PROMPT AND DELAYED NEUTRON YIELDS VS ELECTRON ENERGY FOR 238U
PROMPT NEUTRON YIELD
DELAYED NEUTRON YIELD
TOTAL COUNT (PER γ DOSE)
10^6
10^5
10^4
10^3
10^2
10^1
5
6
7
8
9
10
11
12
ELECTRON ENERGY (MeV)

Fig. 4

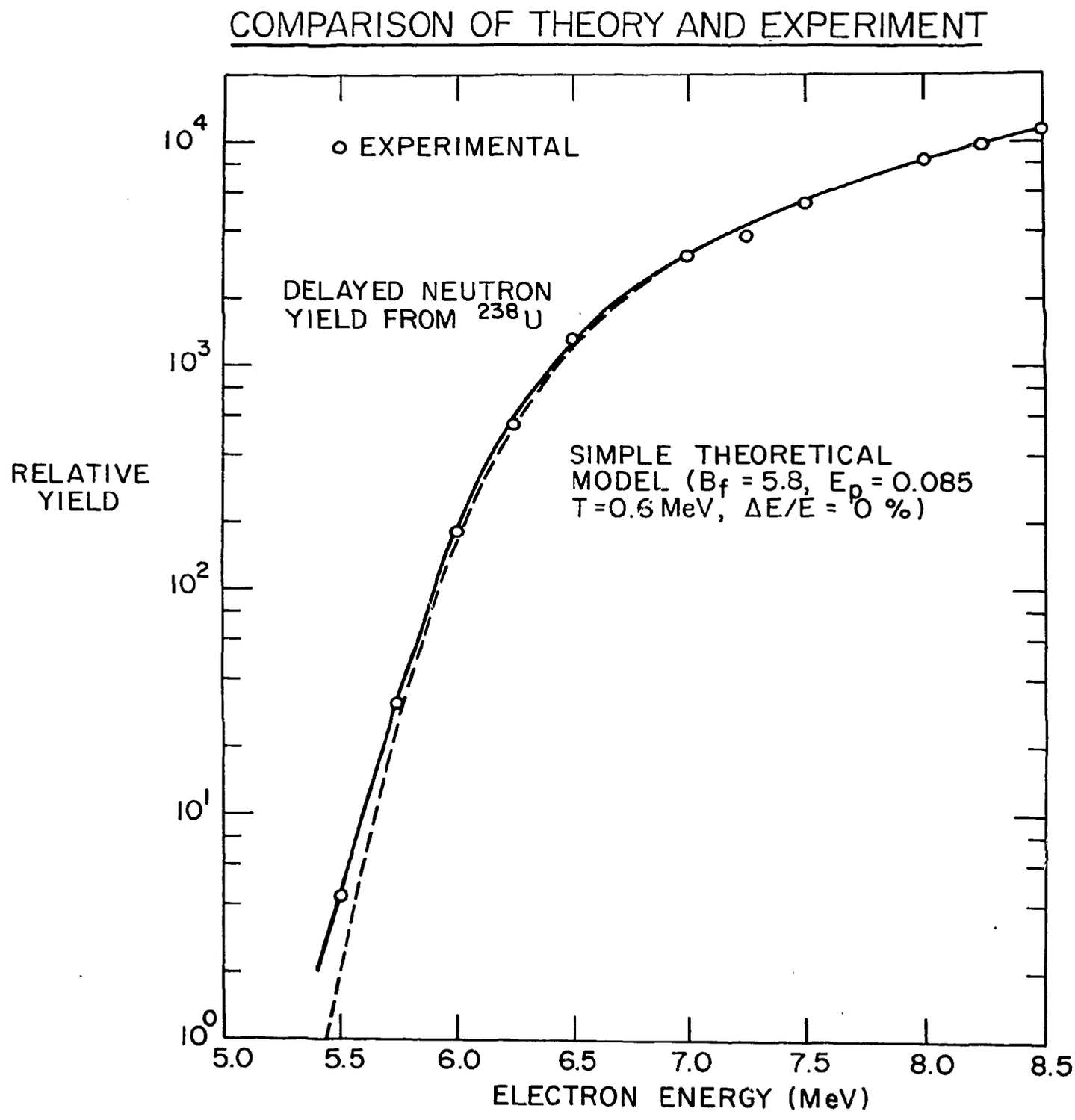
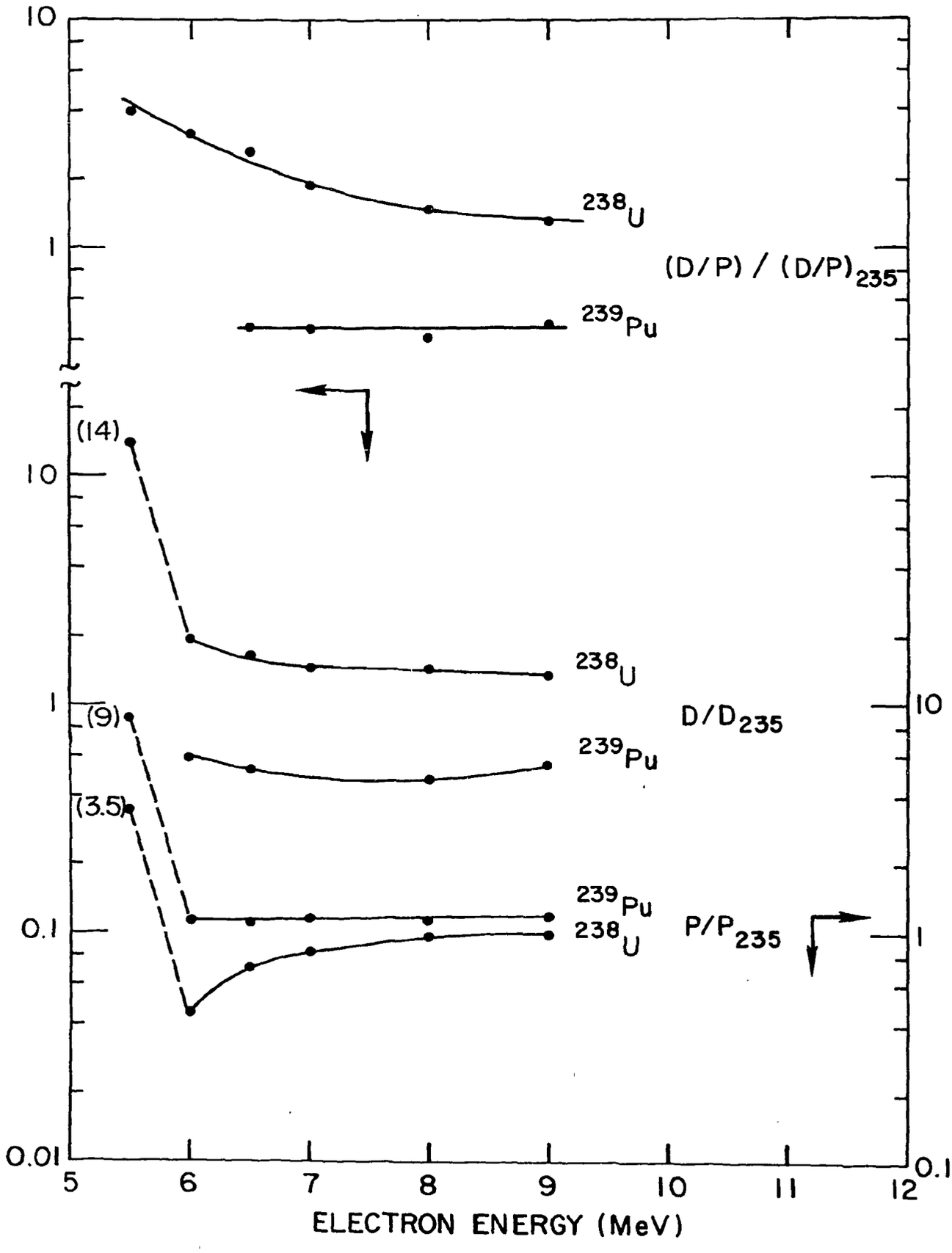

COMPARISON OF THEORY AND EXPERIMENT
o EXPERIMENTAL
DELAYED NEUTRON YIELD FROM ^{238}U
SIMPLE THEORETICAL MODEL (B_f = 5.8, E_p = 0.085 T = 0.6 MeV, ΔE/E = 0 %)
RELATIVE YIELD
10^4
10^3
10^2
10^1
10^0
5.0
5.5
6.0
6.5
7.0
7.5
8.0
8.5
ELECTRON ENERGY (MeV)

Fig. 5

MEASURED DISCRIMINATION RATIOS



8.7 between 6 to 5.5 MeV. The main reasons for this are that the fission barrier of ^{235}U is higher than that of ^{238}U and ^{239}Pu and the fact that the (γ, n) threshold in ^{235}U is lower than that of ^{238}U. The ratio of total delayed neutron yields also offers a very useful discrimination ratio. For ^{238}U to ^{235}U it varies from 1.3 at 9 MeV to 1.6 at 6 MeV and then climbs up to about 14 at 5.5 MeV. The prompt discrimination ratio of ^{239}Pu to ^{235}U seems to be rather independent of energy between 9 and 6 MeV and its value is around 0.5. Similar behavior is seen in the relative signature "delayed/prompt." Thus again, these measurements have confirmed the existence of large discrimination behavior and strong linear independence (evidenced by different shapes) of the various yields as a function of electron energy.

Determination of the Relative β_i

Earlier reports contain a description of the time behavior of delayed neutrons from the photofission of ^{238}U, ^{235}U, and ^{239}Pu for different irradiation times and at endpoint energies (E_e) of 7, 8, and 10 MeV.[3, 4] The data from the ^{238}U and ^{235}U runs at 8 and 10 MeV have been analyzed to determine the relative fractional yield (β_i) of the six delayed neutron groups which contribute to the total yield. The half-lives for the six groups used in the analysis were taken from a similar analysis at E_e = 15 MeV.[5] The results of the analysis are shown in Figs. 6 and 7 together with results from Ref. 5.

The β_i for the 8 and 10 MeV data are very similar. There is some variation with endpoint energy for the 0.2 and 0.5 sec groups of ^{235}U, however, the total fractional yield for the combined fast groups is approximately the same for 8 and 10 MeV data. The 8 and 10 MeV ^{238}U β_i agree quite well with the ^{238}U data at 15 MeV. In the case of ^{235}U, however, there are large differences between the data at 8 - 10 MeV and the data at 15 MeV. At the present time, it is not clear whether these differences are physical or due to differences between the two analyses.

The determination of these β_i allow one to calculate irradiation times and counting period which are optimum for assay experiments. It is also clear that since there are no important variations in the β_i with energy in the 8 to

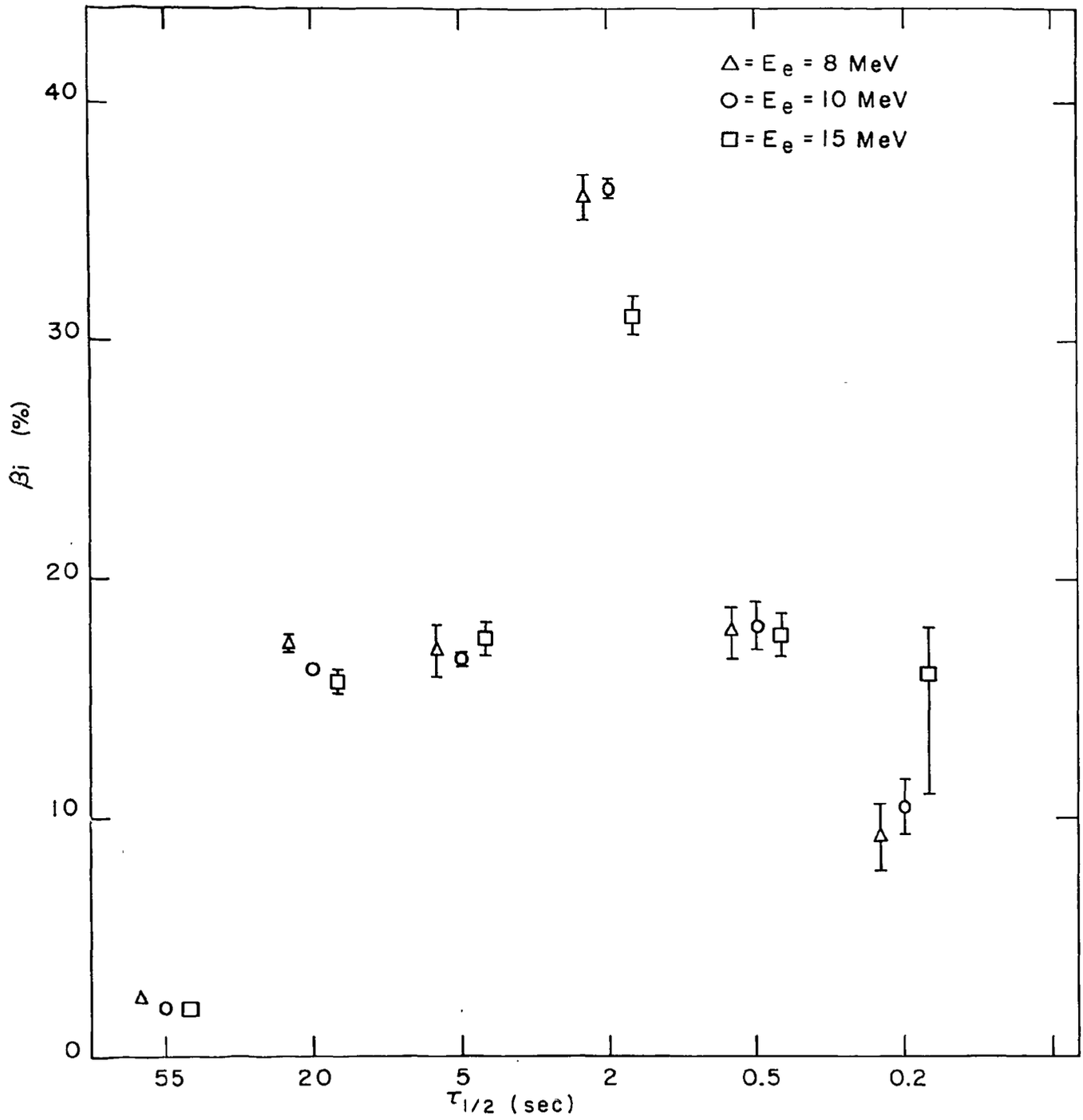


Fig. 6
Relative β 's for ^{238}U (x γ,f)

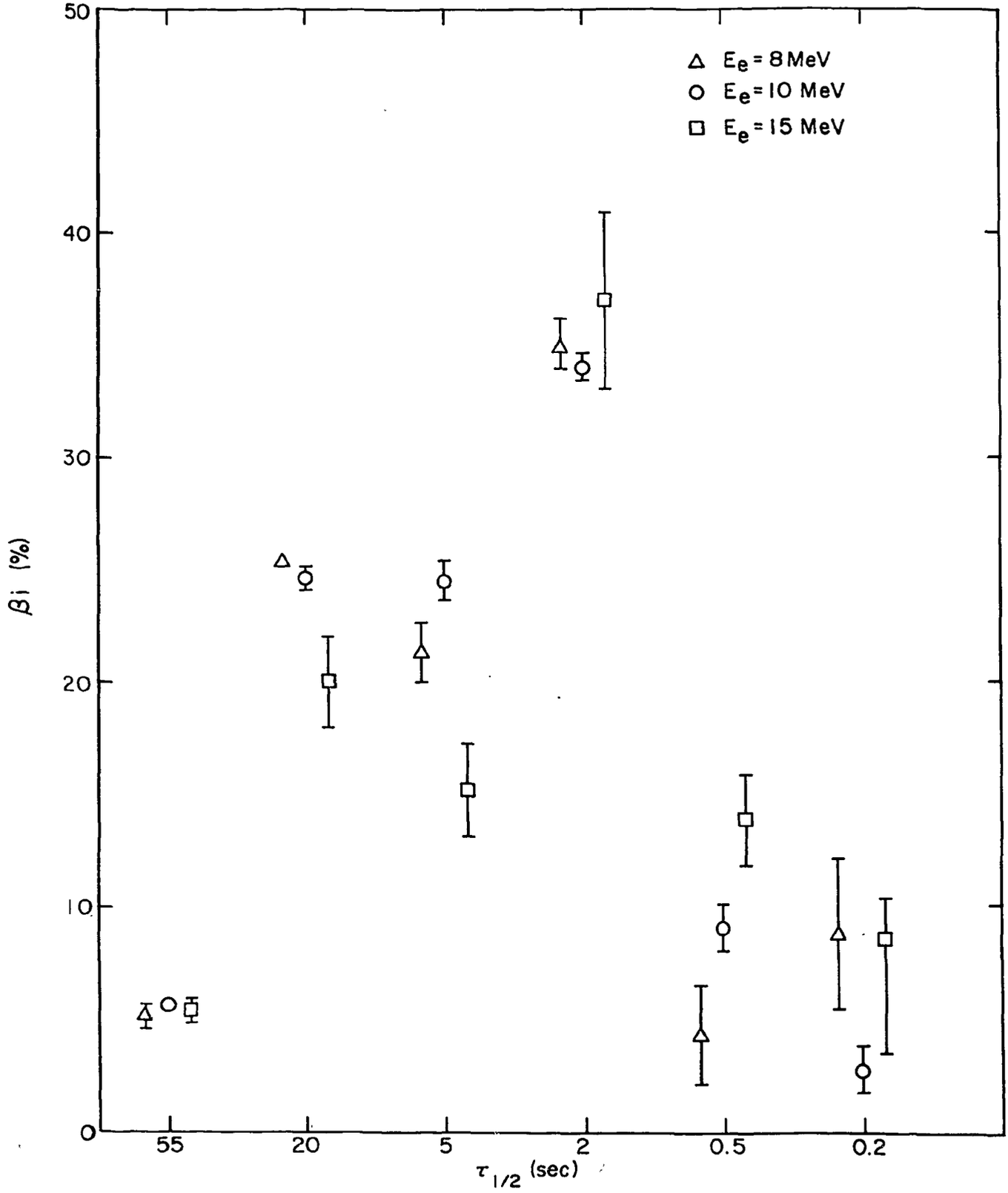


Fig. 7
Relative β 's for ^{235}U (x γ ,f)

10 MeV range, the optimum endpoint energy for delayed neutron signatures is that where the delayed neutron yield is greatest, e. e., the highest energy possible.

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A RESONANCE SELF-INDICATION TECHNIQUE FOR ISOTOPIC ASSAY OF FISSILE MATERIALS*

H. O. Menlove, C. D. Tesche, M. M. Thorpe, and R. B. Walton

University of California
Los Alamos Scientific Laboratory
Los Alamos, New Mexico

ABSTRACT

A resonance self-indication technique has been developed to nondestructively measure the thickness of fissile materials. This method utilizes the resonance structure in the neutron fission cross section by passing an epithermal beam of neutrons through the sample and then to thin fission detectors which are sensitive to the resonance absorption lines in the transmitted flux corresponding to the resonance reaction peaks. The measurements included samples of ^{233}U, ^{235}U, and ^{239}Pu with thicknesses ranging from 5 to 270 mils. The present measurements indicate that this technique could be used to measure the thickness of ^{239}Pu with an accuracy of 1-3%, and an accuracy of 2-10% for ^{233}U and ^{235}U. The influence of extraneous material in the samples has been greatly reduced by using ratios of different fission detectors in the measurement. Computer calculations of the fission rates have been made, and the theoretical results are in good agreement with the measurements.

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*Work performed under the auspices of the U.S. Atomic Energy Commission.

I. Introduction

The distinctive resonance structure in the neutron fission cross sections of the different fissile isotopes, which is most pronounced in the neutron energy range from 0.3 eV to roughly 10 keV, offers a promising technique for nondestructive assay applications for nuclear safeguards and materials management. One method of utilizing this resonance structure is to pass a beam of epithermal neutrons through a sample of fissile material and then to beam monitors consisting of thin-foil fission detectors containing the same fissile isotope(s) as the sample. The sensitivity of this method depends on using the same fissile materials in the fission detectors as are under investigation in the sample, since the selective resonance absorption in the sample is amplified by the (n,f) resonance reaction in the fission foil with the same resonance structure.

Shown in Fig. 1 is the neutron fission cross section of ^{239}Pu for neutron energies up to 30 eV, along with a 1/E neutron spectrum after transmission through two different thicknesses of ^{239}Pu. Curve (a) corresponds to a 10-mil ^{239}Pu sample, and curve (b) to a 100-mil ^{239}Pu sample. It can be seen that the resonance absorption lines in the transmitted flux, corresponding to maxima in the total neutron cross section, occur at the same energies as the resonance peaks in the fission cross section. These curves show that as the sample thickness increases, the absorption from the larger resonances tends to saturate, but the smaller resonances continue giving the self-indication effect.

When the fission detectors are placed in a collimated beam of neutrons, the fission rate in the i-th fission detector is given by

$$R_i \propto \int_{E_{min}}^{E_{max}} \sigma_f^i(E)\varphi(E)\,dE \qquad (1)$$

where $\sigma_f^i(E)$ is the cross section for the (n,f) reaction in isotope i, and $\varphi(E)$ is the neutron flux after transmission through the fissile sample and filter. The neutron flux emergent from the thermal reactor was assumed to have the form

$$\varphi_0(E) \propto \frac{1}{E} + \frac{C\ 2\sqrt{E}}{\sqrt{\pi}\ T^3} \exp\left(-\frac{E}{T}\right) \qquad (2)$$

where the relative magnitude of the 1/E component and the fast fission component had been previously estimated[1] using threshold detector techniques, and the temperature T of the Maxwellian distribution was taken as 1.29 MeV[2] After transmission through a thin Gd (or Cd) filter and a fissile sample (such as Pu), the neutron flux is

$$\varphi(E) = \varphi_0(E) \exp\left(-\Sigma_{Tot}^{Pu}(E)\ x(Pu)\ -\Sigma_{Tot}^{Gd}(E)\ x(Gd)\right) \qquad (3)$$

where $\Sigma_{Tot}^{Pu}(E)$, x(Pu), $\Sigma_{Tot}^{Gd}(E)$, and x(Gd) are the total macroscopic cross sections and thicknesses of Pu and Gd, respectively. The purpose of the thin Gd filter is to absorb the neutrons with energies below the resonance region (~0.3 eV).

Equations (1), (2), and (3) have been combined and numerically integrated using a computer, where E_{min} was taken as 0.01 eV and E_{max} was 15 MeV. The energy dependent cross sections were obtained from the UK (1966), AWRE (1965), and LRL (1966) evaluated data tape files. The resolution of the cross section input was such that roughly 2500 energy intervals were used for the numerical integration, giving adequate energy resolution in the resonance region, as can be seen from the computer plot in Fig. 1. The calculations for the various fissile isotopes (^{239}Pu, ^{235}U, and ^{233}U) are compared with the present measurements in the Results Section of this paper.

Putz, et al.[3] calculated the fission detector response for the neutron transmission through ^{239}Pu where the neutrons were all assumed to be grouped in the region of the large 0.3 eV resonance, and he estimated that the error in the ^{239}Pu thickness using his procedure would be about 1% for a transmission measurement with 3% accuracy. Consequently, Weinzerel, et al.[4] measured the neutron transmission through a 0.007-cm Pu sample using a reactor and a chopper to bunch the neutrons around 0.3 eV; however, their measurements were only of a preliminary nature.

II. Experimental Procedure

A schematic diagram of the experimental arrangement used in the measurements is shown in Fig. 2. A collimated neutron beam from the LASL Water Boiler Reactor was first filtered by a foil of Gd or Cd to remove the thermal neutrons. Gd was used for the Pu sample since its neutron cutoff energy is just below the peak of the 0.3-eV resonance of ^{239}Pu. The epithermal neutron beam then passed through the fissile sample under investigation and thence through a sandwich of four parallel-plate ionization chambers containing back-to-back thin foils of ^{239}Pu, ^{233}U, ^{235}U, and ^{10}B. The counting rates in all four detectors were recorded both with and without the various fissile samples and absorbers in the beam. Counting statistics of better than 1% were obtained in irradiation times of approximately 2 minutes with the reactor power < 10 kw. The Cd ratios in the reactor beam were 44, 16, 40, and 41 for the ^{239}Pu, ^{233}U, ^{235}U, and ^{10}B ionization detectors, respectively.

The fissile samples used in the measurement consisted of 2-in. dia. metallic discs of ^{239}Pu (94.2% enriched), ^{235}U (93% enriched), and ^{233}U (99% enriched) with thicknesses ranging from 5 to 270 mils. The ^{239}Pu and ^{233}U samples were hermetically sealed in 7-mil thick Cu cans to avoid radioactive contamination.

During the irradiations, the ionization chambers were operated as gas flow counters using a gas mixture of 90% A and 10% CH_4 at a pressure of about 4 psi. The sides and back of the detector packet were covered with 16-mil Cd to reduce the background from room-return neutrons. The fission detectors were positioned approximately 3 in. from the exit of the 1-in. dia. collimator

in the reactor shield. The four parallel-plate ionization chambers are shown in Fig. 3. The 1.38-in. dia. deposits of ^{239}Pu (150 $\mu g/cm^2$), ^{233}U (380 $\mu g/cm^2$), ^{235}U (406 $\mu g/cm^2$), and ^{10}B (40 $\mu g/cm^2$) were evaporated on 5-mil thick Pt backings. The spacing between the deposit and the collector plate was 0.25 in., and a positive bias of 100V was applied to the collector plate.

To investigate the potentialities of using a harder energy neutron spectrum in the self-indication technique, a Cockcroft-Walton accelerator was used to produce 14-MeV neutrons from the (D,T) reaction, and these neutrons were then moderated to form a slowing-down spectrum. Computer calculations[5] were performed using a DTF (neutron transport) code to find a moderator that would give a maximum number of neutrons in 1-10,000 eV energy region. Moderators consisting of concentric layers of W, Pb, Be, and CH_2 were used where the 14-MeV neutron source was placed at the center of the assembly. The escaping neutrons were collimated by a 1.38-in. dia. hole through a 6-in. thick B_4C shield. The irradiation and counting procedure was similar to that previously described for the reactor, and the moderating assembly was altered to yield Cd ratios of 10, 20, and 26 for the ^{235}U (n,f) reaction. With each of these neutron spectra, the self-indication effect was less than for the reactor spectrum, which had a corresponding Cd ratio of 40.

III. Results

In the following results, the thicknesses of the fissile samples have been corrected for the ^{240}Pu (5.8%) in the Pu samples and the ^{238}U (7%) in the ^{235}U samples. The densities and enrichments of the present samples yield the following conversion factors:

$$1\text{-mil } ^{239}Pu = 1.01 \times 10^{20} \text{ atoms/cm}^2$$

$$1\text{-mil } ^{235}U = 1.21 \times 10^{20} \text{ atoms/cm}^2$$

$$1\text{-mil } ^{233}U = 1.24 \times 10^{20} \text{ atoms/cm}^2.$$

The response of the fission detectors to the variation in the thickness of the corresponding fissile samples is shown in Fig. 4. These curves represent the transmission of epithermal neutrons through the samples weighted by

the fission cross sections of the detectors. A Gd filter (4.5 mil) was used for the ^{239}Pu and ^{235}U samples, and a Cd filter (30 mil) was used for the ^{233}U samples. If there is no extraneous material in a fissile sample, then a simple transmission measurement of this type could be used to determine the sample thickness. The assay accuracy under these conditions would be quite good, since an error of 1% in the measured response would correspond to errors of approximately 1.0%, 1.7%, and 2.0% in the thickness determinations (in the region of ~ 30 mil) of ^{239}Pu, ^{233}U, and ^{235}U, respectively. With the present experimental setup, a measurement time of 2 min. yielded counting statistics of better than 0.7%.

The counting rate ratios of the fission detector to the ^{10}B detector for different thicknesses of ^{239}Pu are shown in Fig. 5. A 3-mil Gd filter was used for this set of measurements. The upper curve corresponds to the response of the ^{233}U fission detector, the middle curve to the ^{235}U detector, and the lower curve to the ^{239}Pu detector. The three fission detector rates have been normalized to the counting rate in the ^{10}B detector, which makes the measured response relatively insensitive to extraneous materials mixed in with the fissile sample, since both the ^{10}B and the fission response rates are decreased roughly the same amount by the extraneous material. In addition, the multiplicity of curves tends to uniquely specify ^{239}Pu as the sample material since no other combination of materials of any thickness would likely yield the three given fission response ratios. The accuracy of the ^{239}Pu thickness determination using the ratios given in Fig. 5 is good, since an error of 1% in the $^{233}U/^{239}Pu$ ratio (at ~ 30 mil) yields an error of only 1.3% in the ^{239}Pu determination.

The dashed curves in Fig. 5 correspond to the theoretical calculations of R_i using Eq. (1) where curves (a), (b), and (c) correspond to R(233)/R(10), R(235)/R(10), and R(239)/R(10), respectively. For these calculations the coefficient (c) of the fast flux term was taken to be 20 and T was 1.29 MeV. The calculations were relatively insensitive to these coefficient since vaying C from 20 to 40 and T from 0.3 to 1.3 MeV resulted in only small changes (~ 10%) in the calculated curves. This insensitivity to the fast flux component is to be expected, since the fissile cross sections are much larger for lower energy

neutrons. The calculated curves (b) and (c) agree well with the measured responses; however, curve (a) drops below the ^{233}U measurement by roughly 20%. Some of these discrepancies could be caused by neutron scattering and fissioning in the sample, which alters the energy of the primary neutron beam; however, the geometric coupling between the fission detectors and the fissile sample was only ~ 2%, so this effect should be small. When appropriate, the ^{240}Pu cross section was included in the exponential term in Eq. (3).

Figure 6 shows the fission detector ratios for samples ^{233}U and ^{235}U using a 30-mil Cd filter. The upper curve corresponds to the $^{235}U/^{233}U$ ratio with ^{233}samples and the lower curve gives the same ratio using ^{235}U samples. The self-indication effect for these two fissile isotopes is less than ^{239}Pu, since ^{233}U and ^{235}U do not have a large resonance such as the 0.3 eV resonance in ^{239}Pu. The dashed curves correspond to the calculated ratios using Eq. (1) with the same fast flux component as previously mentioned. The calculations are in very good agreement with the measured curves, with a discrepancy of roughly 5% or less. The lack of additional ^{233}U samples prevented a continuation of the ^{233}U measurements beyond 90 mils.

In order to investigate the effect that various extraneous materials have on the fission detector responses, a series of measurements were made wherein a Pu sample (0.010 in. thick) was sandwiched between relatively thick (~ 0.5 in.) layers of common materials (C, Al, Pb, ^{238}U, ^{232}Th, and CH_2). Table I gives the thickness of the added materials, along with the changes in the ^{239}Pu fission detector response as compared with the ($^{235}U/^{239}Pu$) ratio response. It can be seen that the ordinary transmission measurement is very sensitive to the added material (factor-of-2 changes are typical), whereas the changes in the detector ratios are typically only a few percent. Polyethylene (CH_2) has a large scattering cross section for low-energy neutrons, and hence tends to mask the self-indication technique. Since ^{233}U is often found in combination with ^{232}Th, the ^{233}U sample was surrounded by 0.25 in. of ^{232}Th, and the fission detector ratio (upper curve in Fig. 6) changed by less than 1%.

To determine the effect of the neutron cutoff energy on the fission detector responses, the measurements were

carried out using different neutron filters of Gd and Cd; the results are shown in Fig. 7. The differences exhibited by the five curves in Fig. 7 are due primarily to the large 0.3-eV resonance in ^{239}Pu. The 30-mil Cd filter absorbs most of the neutrons in the energy region containing this resonance, whereas the 20-mil Cd filter transmits an appreciable fraction of the neutrons in the resonance wing. Thus the fission response ratio ($^{235}U/^{239}Pu$) increases, with Pu sample thickness, faster for the 20-mil filter than for the 30-mil filter. The 4.5-mil Gd filter attenuates the neutron spectrum on the low-energy side of the 0.3-eV resonance, whereas the 1-mil Gd filter transmits an appreciable number of lower energy neutrons, which in turn tends to decrease the ($^{235}U/^{239}Pu$) response.

The influence of the low-energy neutron cutoff on the fission response ratios ($^{235}U/^{239}Pu$) has been calculated for fissile samples of ^{239}Pu, and the results are presented in Fig. 8. The minimum neutron energies were varied from 0.01 to 6.0 eV, and the maximum discrimination occurs for a cutoff energy of approximately 0.20 eV. For cutoff energies greater than 0.20 eV, the curves are shown as dashed lines to avoid confusion. The curves reach their maximum at 0.20 eV, since at this energy most of the large 0.3-eV ^{239}Pu resonance is utilized and most of the lower energy neutrons, which do not contribute to the self-indication effect, are omitted. The calculated curves in Fig. 8 can be compared with the measured curves in Fig. 7, in which case the cutoff energies of 0.05, 0.10, 0.20, 0.30, and 0.40 eV roughly correspond to the filters of 1-mil Gd, 3-mil Gd, 4.5-mil Gd, 20-mil Cd, and 30-mil Cd, respectively. As might be expected, the calculated curves with their discrete minimum cutoff energies reach a slightly larger discrimination ration ($^{235}U/^{239}Pu$) than the measured curves using Gd and Cd filters to absorb the low-energy neutrons, since these filters have a gradual neutron energy cutoff. Thus even for an optimum thickness of Gd, some neutrons are still absorbed out of the 0.3-eV resonance region and some low-energy neutrons (< 0.2 eV) are transmitted, resulting in a decrease in the self-indicated effect. The cutoff energies of the curves in Figs. 7 and 8 make it clear that roughly 60% of the self-indication effect in ^{239}Pu originates from the large resonance at 0.3 eV.

IV. Conclusion

The results of the present measurements indicated that the resonance self-indication technique can be used for quantitative assay of ^{239}Pu to an accuracy of 1-3% and an accuracy of 2-10% for ^{233}U and ^{235}U. The method is applicable only to fissile materials in homogeneous mixtures or in parallel plate geometry, where the effective thickness of the fissile material is less than ~ 150 mils (15×10^{22} atoms/cm^2). Such a technique should be useful in many applications to on-line isotopic assay (e.g., fissile material moving through a fuel fabrication or reprocessing plant). Since the fission detector ratios are dependent on the shape of the neutron spectrum, calibration curves (such as Fig. 5) should be measured for a given neutron source. The measurements using the harder neutron energy spectra from the moderated 14-MeV source indicated that the self-indication effect decreases as the average neutron energy increases. Calculations of the fission response ratios where the 1/E neutron flux has been replaced by a 1/E component plus a fast component have corroborated these results.

It should be emphasized that the measurement of fission detector ratios has the advantage over a single detector measurement of being relatively insensitive to extraneous material in the fissile sample mixture. For example, the assay of low enrichment ^{239}Pu in a mixture of fertile materials (e.g., ^{238}U or ^{232}Th) would not be greatly influenced by changes in the amount of the fertile materials.

An alternate application of the resonance self-indication principle described in this paper would be to include relatively thick (~ 150 mil) layers of ^{239}Pu or ^{235}U in the neutron beam filter. Then, due to the strong resonance absorption lines in the transmitted epithermal neutron beam, there would be preferential discrimination against that fissile material in the assay sample which is the same as in the neutron filter. For example, neutron activitation analysis using delayed fission gamma rays (e.g., characteristic hard gamma lines for penetrability) can be used to determine the amount of fissile material in an assay sample. If the fissile material in the sample consists of both ^{235}U and ^{239}Pu, the irratiating neutron beam could first be filtered by ^{239}Pu enhancing the ^{235}U over the ^{239}Pu by a factor of ~ 6 (see Fig. 7).

Then the Pu filter could be replaced by a thick ^{235}U filter enhancing the ^{239}Pu over the ^{235}U by a factor of ~ 1.3, thus giving a combined discrimination factor of ~ 8 between the ^{235}U and ^{239}Pu in the sample. As the thickness of fissile material in the sample increases, the effective discrimination factor would decrease because of resonance self-shielding in the sample itself.

TABLE I

EFFECTS OF ADDING EXTRANEOUS MATERIALS

TO FISSILE SAMPLE

Fissile Sample and Absorber[a]	Change in ^{239}Pu Detector Response[b]	Change in Fission Detector Ratio ($^{235}U/^{239}Pu$)[b]
^{239}Pu in C (0.50 in.)	1.52	-0.9%
^{239}Pu in Al (1.00 in.)	1.41	-0.3%
^{239}Pu in Pb (0.50 in.)	1.73	-2.4%
^{239}Pu in ^{238}U (0.50 in.)	2.28	-2.6%
^{239}Pu in ^{232}Th (0.25 in.)	1.34	-0.2%
^{239}Pu in CH_2 (1.00 in.)	10.02	-36%

[a]The 10-mil (0.010-in.) thick ^{239}Pu sample was sandwiched in the center of the absorbers. The neutron beam was filtered by 4.5-mil Gd.

[b]Represents change caused by the absorber compared with the pure Pu measurement.

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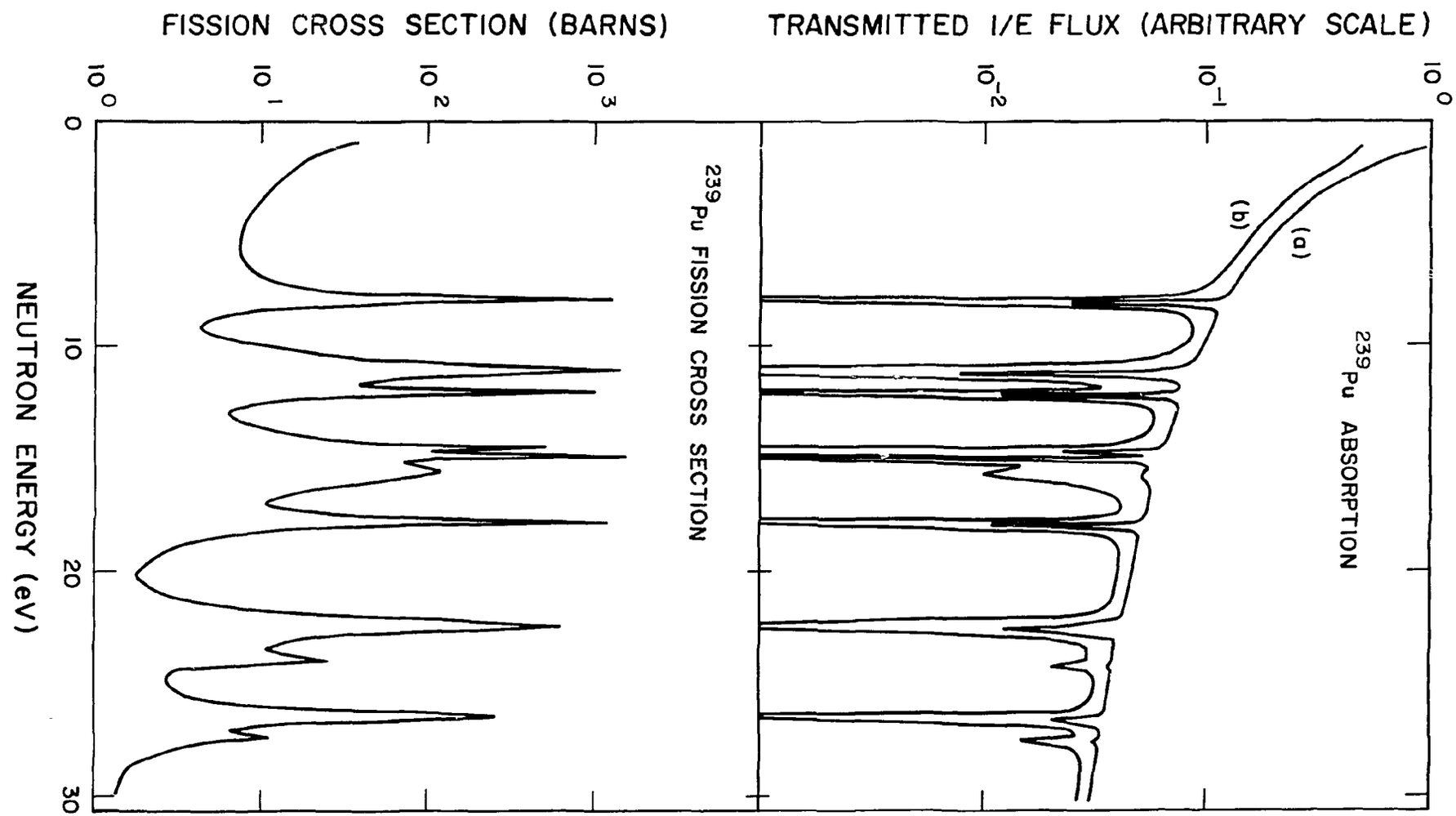

Fig. 1

The ^{239}Pu fission cross section from 0 to 30 eV compared with the absorption lines in a 1/E neutron spectrum after transmission through 10-mil (curve a) and 100-mil (curve b) of ^{239}Pu.

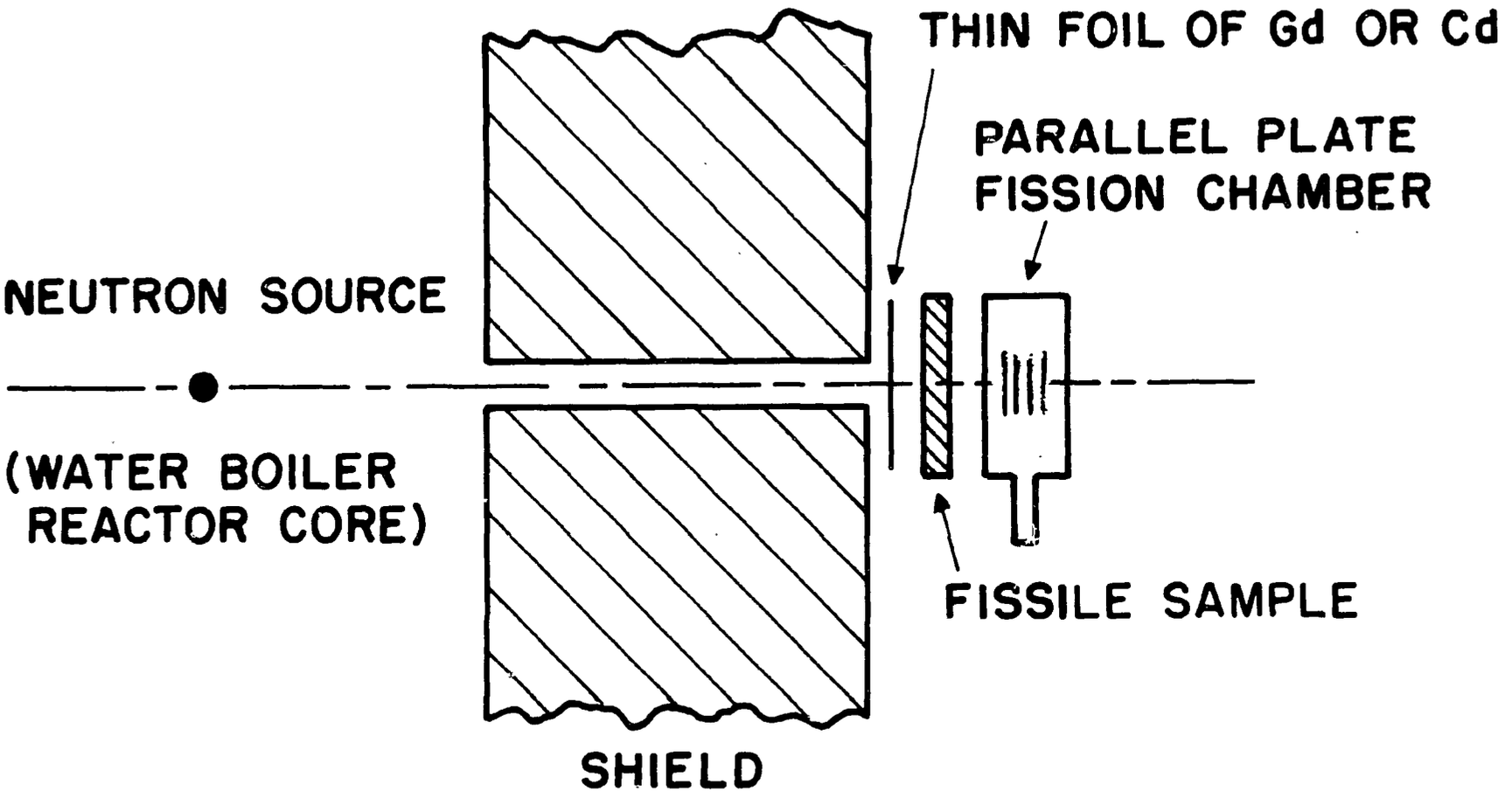

Fig. 2

The experimental setup used for the self-indication measurements.

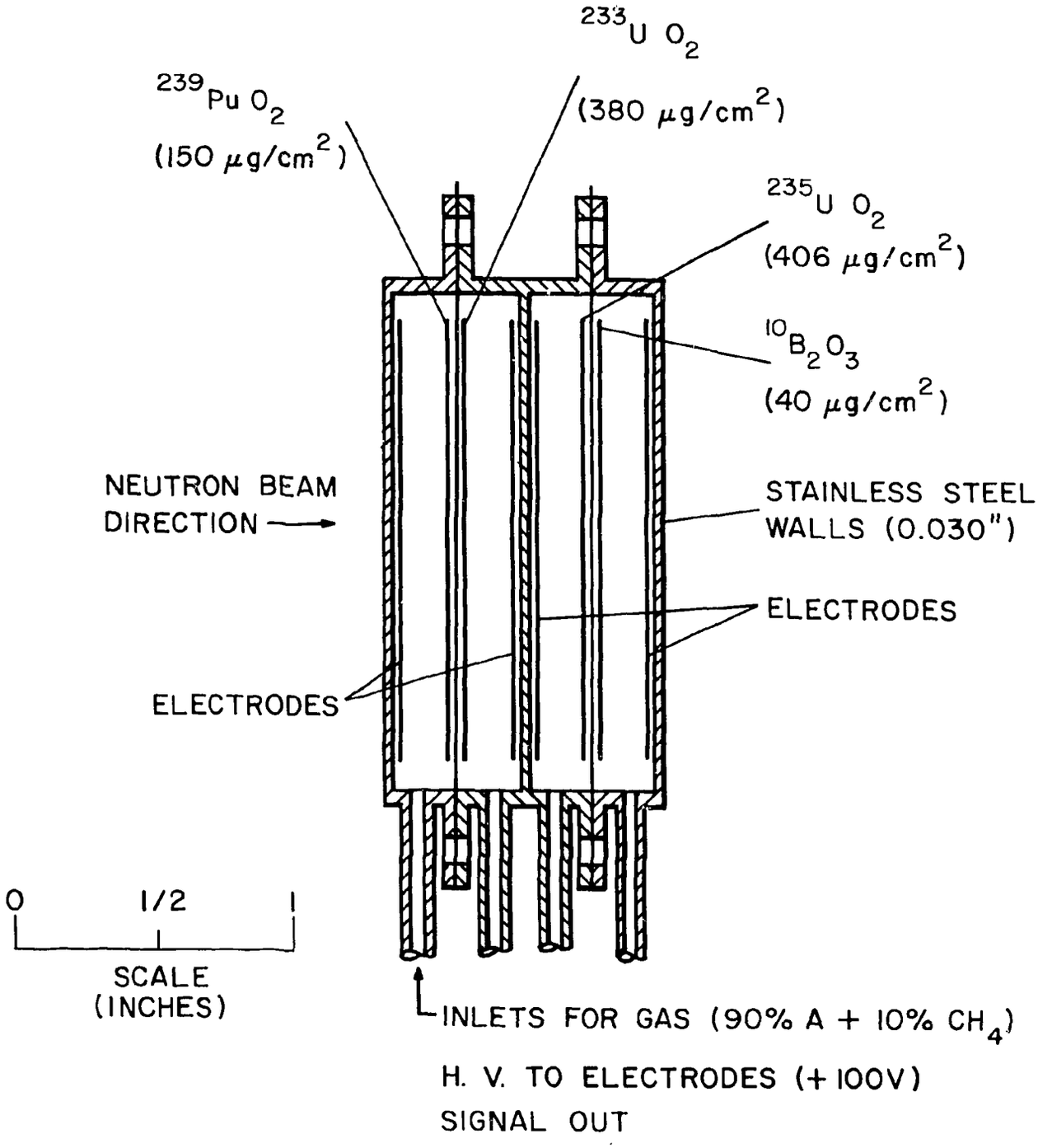

Fig. 3

The ionization chambers used for the self-indication measurements.

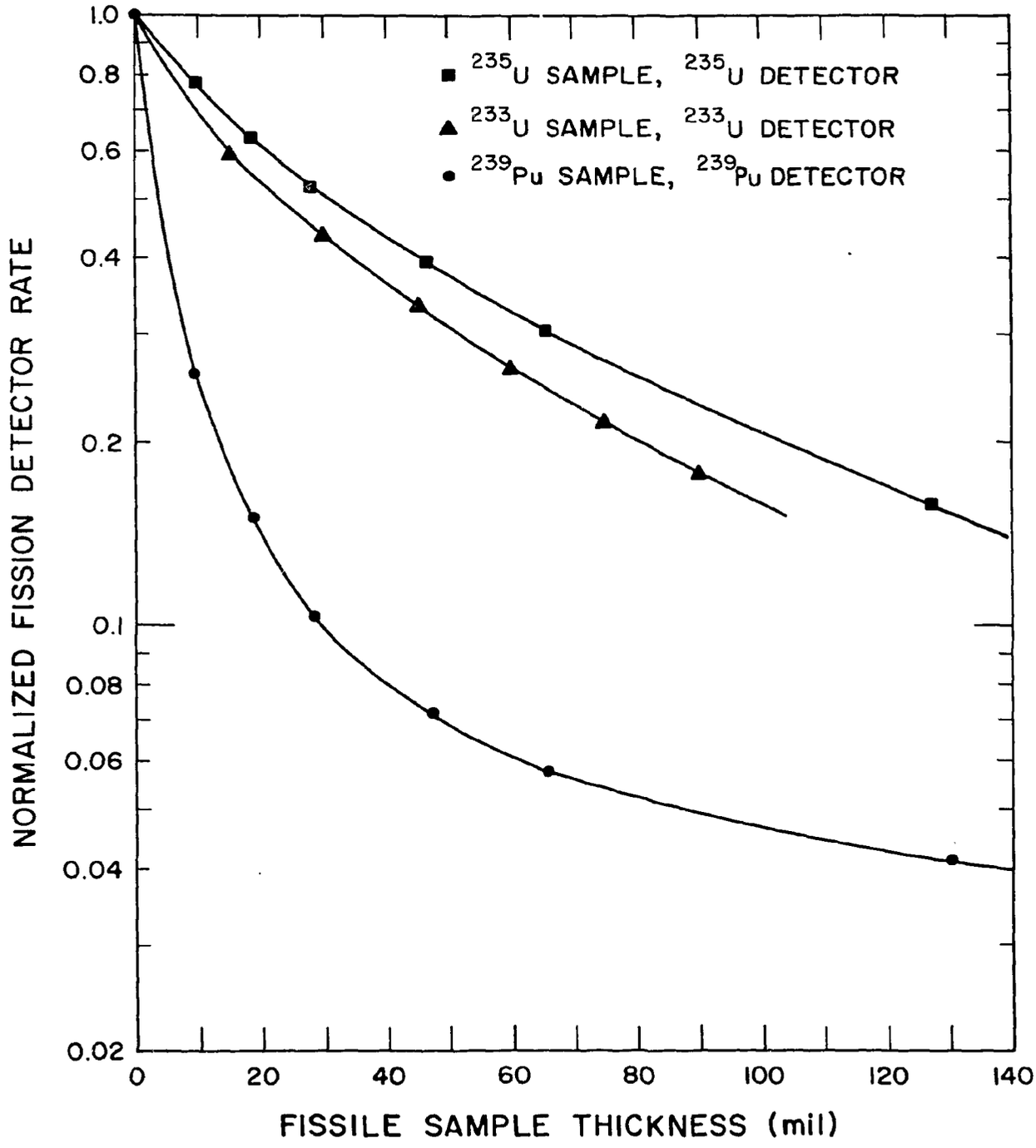

Fig. 4

Transmission curves for epithermal neutrons through ^{235}U, ^{233}U, and ^{239}Pu.

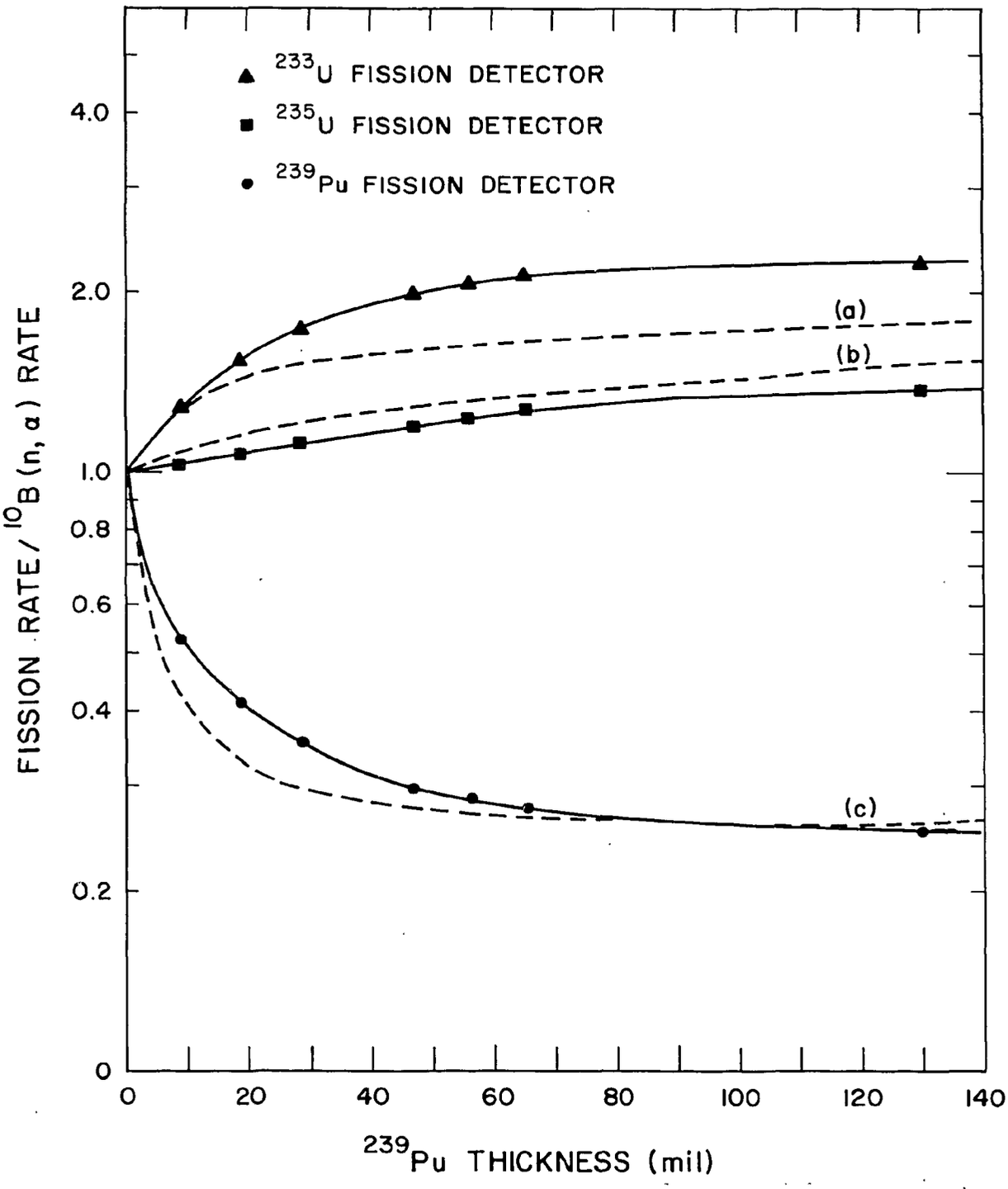

Fig. 5

Counting rate ratio of fission detectors to ^{10}B detector as a function of ^{239}Pu sample thickness (using a 3-mil Gd filter). The dashed curves correspond to calculated fission response ratios using Eq. (1).

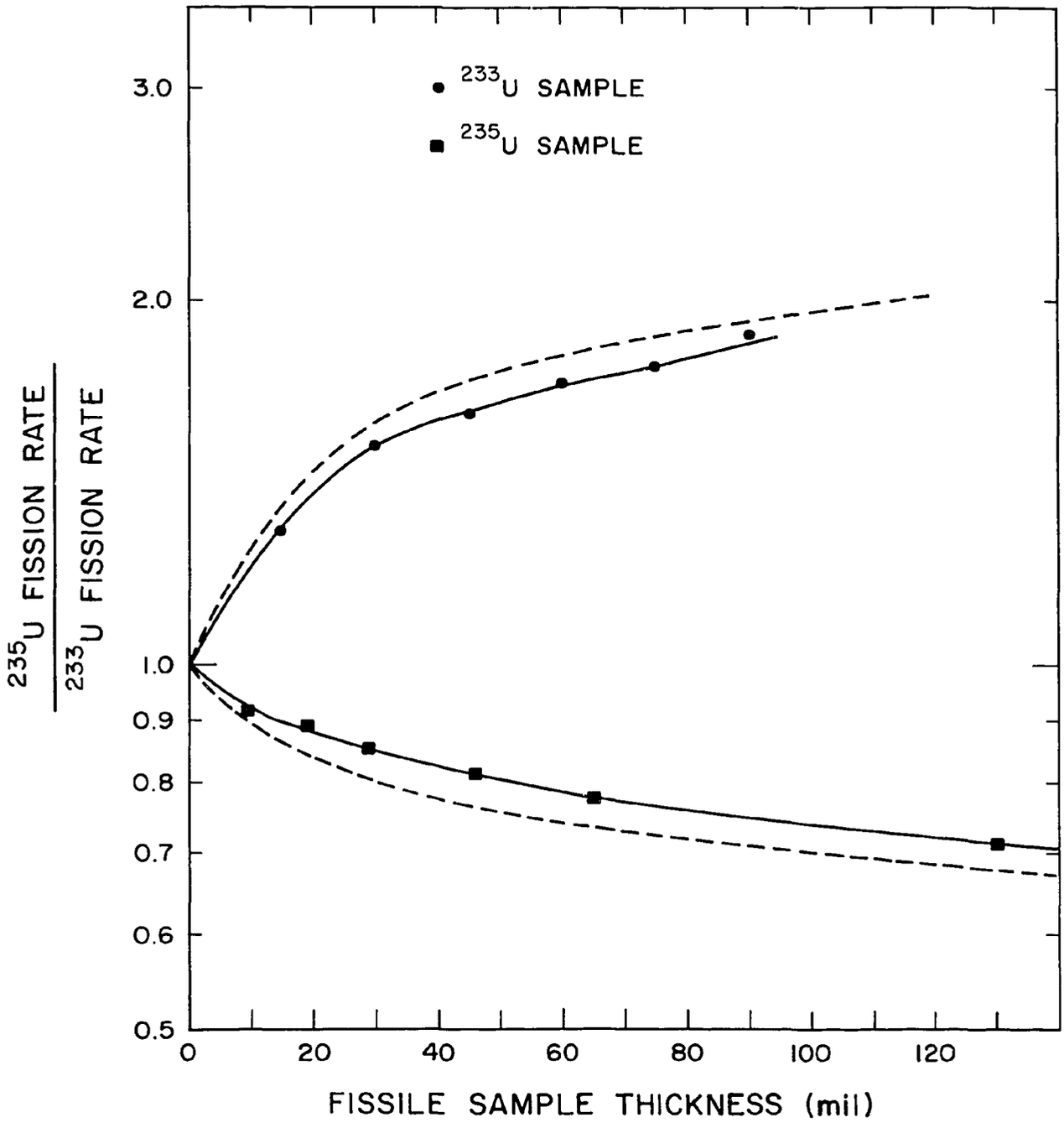

Fig. 6

The fission rate ratio ($^{235}U/^{233}U$) as a function of fissile sample thickness. The upper curves correspond to using ^{233}U samples and the lower curves correspond to using ^{235}U samples. The dashed curves represent the calculations using Eq. (1).

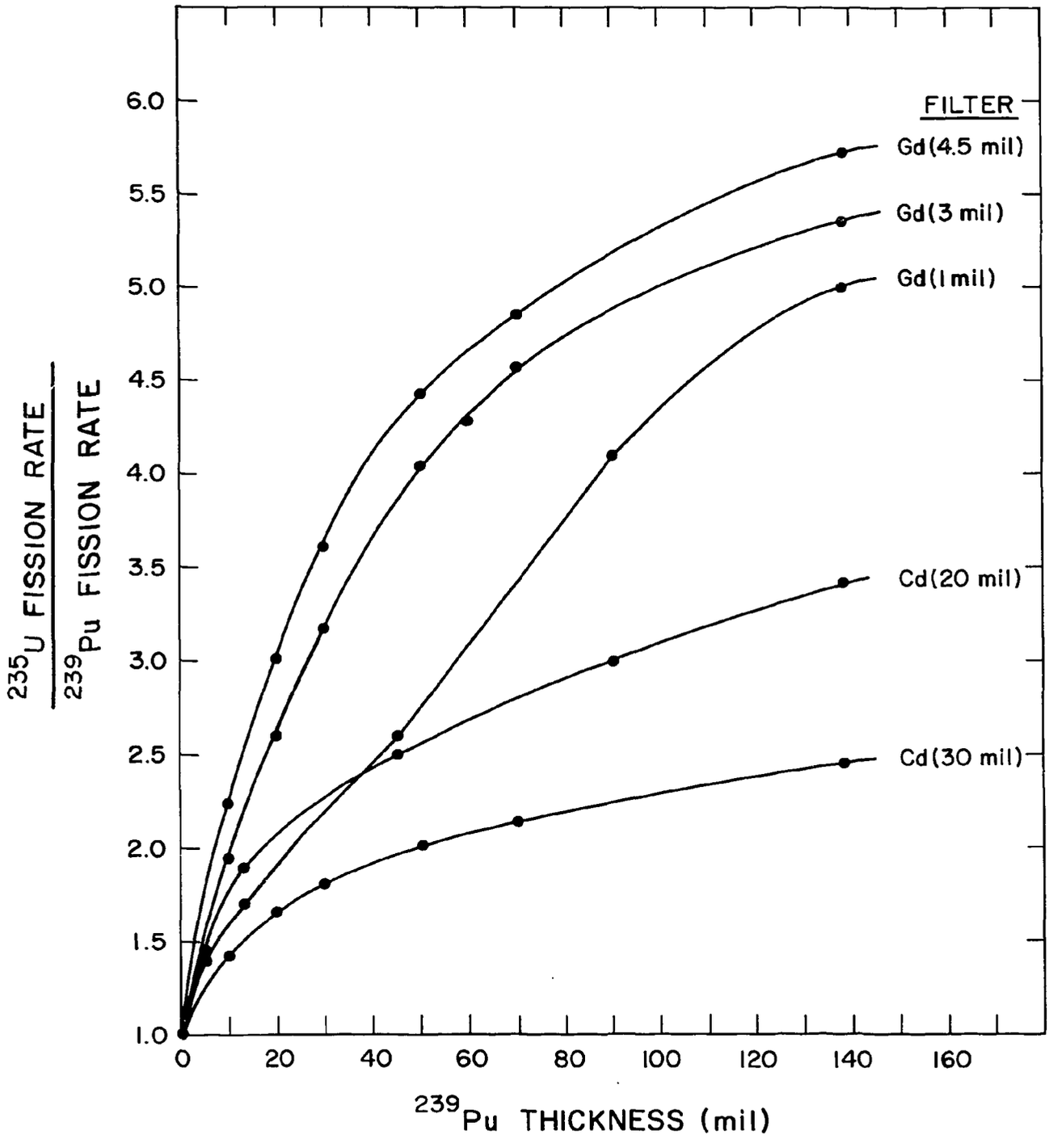

Fig. 7

Fission detector ratio ($^{235}U/^{239}Pu$) versus ^{239}Pu sample thickness for different neutron filters (Gd or Cd).

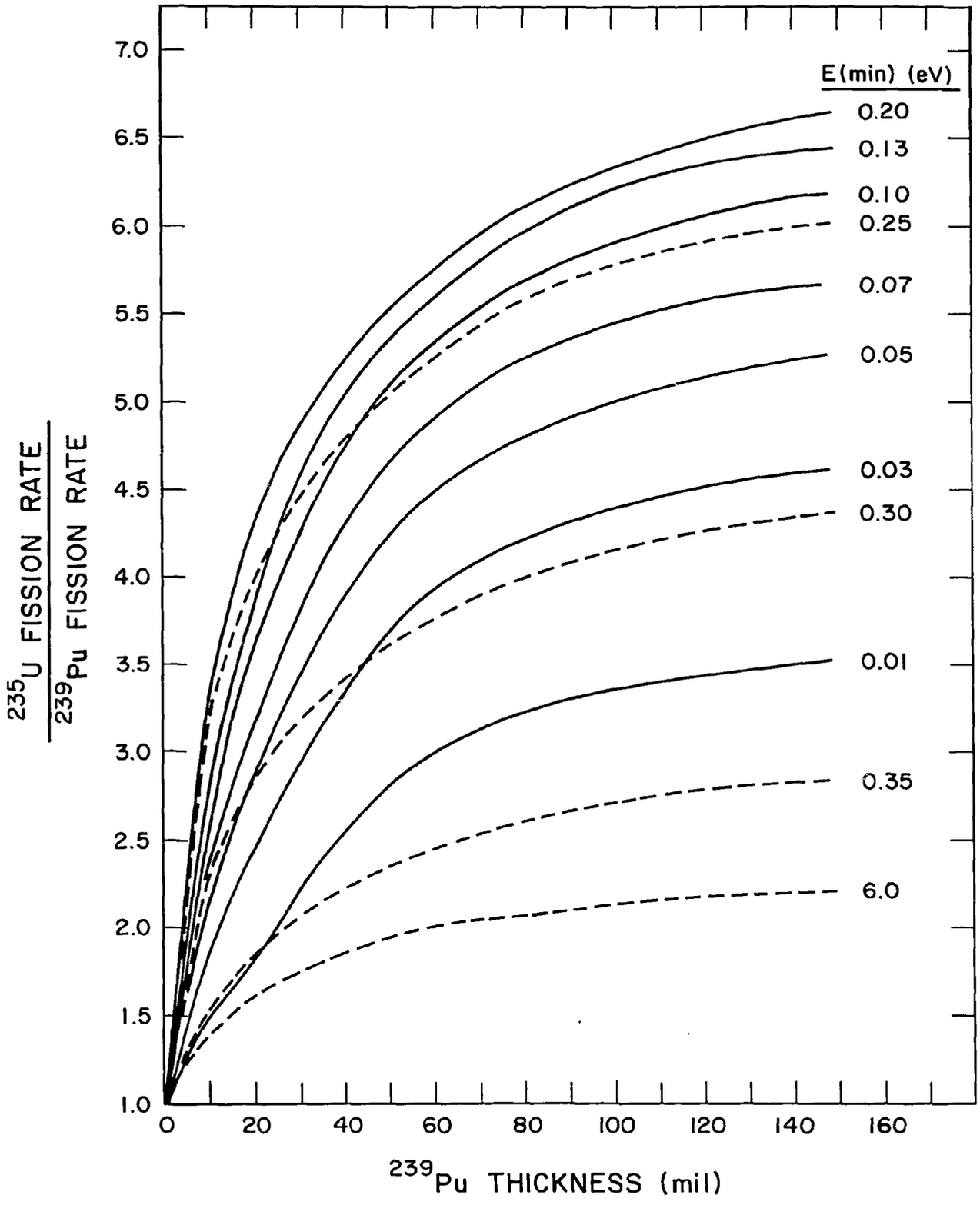

Fig. 8

The calculated fission detector ratio ($^{235}U/^{239}Pu$) as a function of ^{239}Pu sample thickness for different neutron cutoff energies (E_{min}) using Eq. (1).

DELAYED NEUTRON KINETIC RESPONSE METHODS OF NONDESTRUCTIVE ASSAY

R. H. Augustson, H. O. Menlove, C. N. Henry,
C. F. Masters, and G. R. Keepin
University of California
Los Alamos Scientific Laboratory
Los Alamos, New Mexico

I. Introduction

Figure 1 illustrates the physical processes which occur as a result of neutron induced fission, and which might provide signatures distinguishing one fissionable isotope from another. The scope of my presentation will be limited to only that part of the Los Alamos program shown with the double line--namely, the emission of delayed neutrons following a fission caused by a neutron of energy E_n. Our principal effort so far has been to apply what we know about delayed neutrons to developing nondestructive assay techniques; i.e., given an unknown sample, what fissionable isotopes are present and how much of each. Two approaches are being developed which make use of two properties of delayed neutrons. The first involves the differences in time decay behavior between isotopes; this is the kinetic response method. The second measures the differences in yield from isotope to isotope as a function of incident neutron energy.

II. How are the measurements made?

To give a better perspective for the discussion to follow, Figure 2 shows how the measurements are actually made. This is one of our accelerators--a 150-keV, 3-ma machine producing 14-MeV neutrons from the (D,T) reaction and capable of a wide range of pulsing modes. The incident flux on the sample is usually measured by a pair of fission chambers straddling the sample. This

method of monitoring tends to correct for changes in the flux due to the presence of the sample. If these flux changes are negligible, a neutron beam monitor is sufficient. The delayed neutrons are counted by high-efficiency detectors such as the slab detector. A more detailed description will be given in the Thursday session dealing with instrumentation and related topics by Drs. L. V. East and R. B. Walton. It suffices for now to say its intrinsic efficiency is 13%, its energy response is flat from 20 keV to 2 MeV, and, by taking the ratio of counts in the front bank of detectors to the back bank, it can be used to give a rough measure of the average energy of the detected neutrons.

The incident 14-MeV neutron energy can be tailored by means of a moderating assembly surrounding the tritium target to give a spectrum with most of the neutrons below the ^{238}U and ^{232}Th fission threshold, and yet not so soft as to be non-penetrating. At present, guided by DTF transport calculations, we are using a cylindrical assembly of 3" W, followed by 3" C and 1" of polyethylene, as shown in Figure 3.

III. The Kinetic Response Method.

The most consistent analysis of delayed neutron time behavior is in terms of six neutron groups, each characterized by an abundance, a_i, and a period, λ_i. Because the λ_i's (half-lives ranging from .2 sec to 50 sec) are similar for all the fissionable isotopes, it is the a_i's which provide the signatures which distinguish one from another. For example, the a_i's of the 50-sec group for ^{235}U and ^{238}U are nearly equal, 6.3×10^{-3} and 5.4×10^{-3} delayed neutrons/fission, respectively. Their abundances at .2 sec are a factor of seven different, with ^{238}U having a much greater percentage at early times.

Keepin has formulated the time behavior in terms of experimentally measurable functions, $R_{f\pm}$ and $S_{f\pm}$, which bring out the maximum discrimination between two isotopes. The R_f functions represent a measurement using an irradiation short compared to the shortest delayed neutron period (.2 second), and S_{f+} is for a saturation irradiation of the longest half-life. These two emphasize the early

and late portions of the decay curve. Discrimination ratios for ^{235}U-^{238}U-^{239}Pu systems have been calculated from known $a_i\lambda_i$ data and experimentally verified by us.

The R_f discrimination ratio is 3.8 between ^{238}U and ^{235}U, and 5.0 between ^{239}Pu and ^{238}U. In addition, parametric studies have been performed on pulse widths and pulse periods to maximize the accuracies attainable for a given counting time; i.e. with a given statistical accuracy. The significance of the results is in its implication of the ultimate accuracy attainable in an assay of, for example, fast breeder fuels which will contain these fissile-fertile combinations.

To test the ability of this approach to determine the components of a composite system, an irradiation counting cycle of one second-twenty seconds was used to determine the relative isotopic abundance of 2" diameter discs of ^{238}U and ^{235}U stacked to form various percentage mixtures. Figure 4 shows the data. Curve (a) is the pure ^{238}U standard (note increased early time response); (b) 69% ^{238}U; (c) 52% ^{238}U; (d) 35%; (e) 100%. A code was written which extracted the relative isotopic abundance from the composite system decay curves by fitting them to a linear combination of the two normalized pure isotope curves. At the present stage of development, the measured abundances are accurate to within 1-3% of the major component. This is a relative isotopic assay, but the absolute results are obtainable with this procedure by monitoring the neutron flux for the mixture curves--something not necessary for the relative determination.

Moving on to the second approach:

IV. Delayed Neutron Yield

For a system with one fissionable isotope or with a known isotopic composition, the most straightforward way of measuring how much material is present is by measuring the delayed neutron yield per flux monitor and comparing with the yield of a standard. This method is a direct outgrowth of the Los Alamos measurements of absolute delayed neutron yield per fission, as reported at the Toronto ANS Meeting by Masters, et al. For these analyses, the sample is repetitively pulsed with

short pulses (100 ms-1 sec), and the delayed neutrons are counted for an equal time; i.e., 50% duty cycle. An example is an assay for thorium on three samples, received from Oak Ridge, of fuel salt for the Molten Salt Breeder Reactor. The disc samples, ½" thick by 1-½" in diameter, did not contain ^{233}U. In addition to the thorium in the form of fluoride, the samples contained lithium and beryllium fluorides. All three of the samples were chemically analyzed by sampling while molten. The results of one of these analyses was communicated to LASL, and this disc sample with known thorium content was used as a calibration standard for measuring the total thorium content in each of the other two discs. In addition to the three discs, four similar discs were prepared at LASL from pressed powders. The yield technique was used for these measurements. After the thorium determinations were completed at LASL, the assay values were communicated to ORNL, where they were compared with ORNL chemistry analyses. Table I presents the results. The uncertainties in the ORNL analyses are as yet undetermined, but are believed to be equal to or less than 1%. The table shows quite good agreement between the values for weight percent of thorium as determined by delayed neutron nondestructive assay and chemical analysis. The bottom half of the table presents the results of similar measurements utilizing the LASL-prepared pressed powder samples. The thorium weight for sample 6A was assayed independently, treating LASL sample R[1] as the known standard, and was found to agree with the quoted ORNL value for sample 6A to within 0.5%. A second example utilizing the technique will be presented in the paper of M. Thorpe later in the session.

This technique is being applied to two-component systems with one of the components being a threshold fissioner. In many reactor fuels the fissile material is masked in a large amount of fertile material. To enhance the fissile response, we use the moderating assembly to obtain subthreshold neutrons. The yield of the unknown sample is measured at 14 MeV (superthreshold) and subthreshold. By comparison with standards measured under the same conditions, the relative or

absolute amounts can be obtained. In a series of measurements with the 2" metallic discs, a 10% enriched ^{235}U-^{238}U mixture was analyzed for the absolute amount of ^{235}U to within 2% with careful choice of standards.

In summary, nondestructive assay techniques based on delayed neutron response have been and are being developed and refined at Los Alamos as one part of its safeguards program. The applications of these methods to practical situations will be presented in the succeeding paper. The present accuracies (not the ultimate) are:

1. 1-3% for a relative isotopic abundance measurement.

2. Better than 1% for an absolute amount determination in a situation with known isotopic composition and a standard for comparison.

3. 2% absolute determination for a sub-and super-threshold measurement--practically applicable to low fissile enrichment systems.

While these accuracies have not attained those of chemical analysis, the techniques have at least three important advantages:

1. They are nondestructive.

2. They are relatively independent of most other material in the system and inhomogeneity in the system, making sampling less critical.

3. The results are available immediately.

TABLE I

RESULTS OF DELAYED NEUTRON ASSAY OF MSBR FUEL

ORNL Sample No.	Th Weight (grams)	Wt. % of Th	
		LASL	ORNL
6A	13.80	Standard	34.9
7	18.19 ± .13	45.6 ± .3	45.0
8	16.33 ± .11	40.9 ± .3	40.6

LASL Standard No.	Th Weight (grams) Measured by Delayed Neutron Yield	Known Weight
R^1	Standard	16.85 ± .01
R^2	18.91 ± .13	18.98 ± .01
R^3	21.12 ± .15	21.44 ± .01
R^4	24.02 ± .20	24.25 ± .01

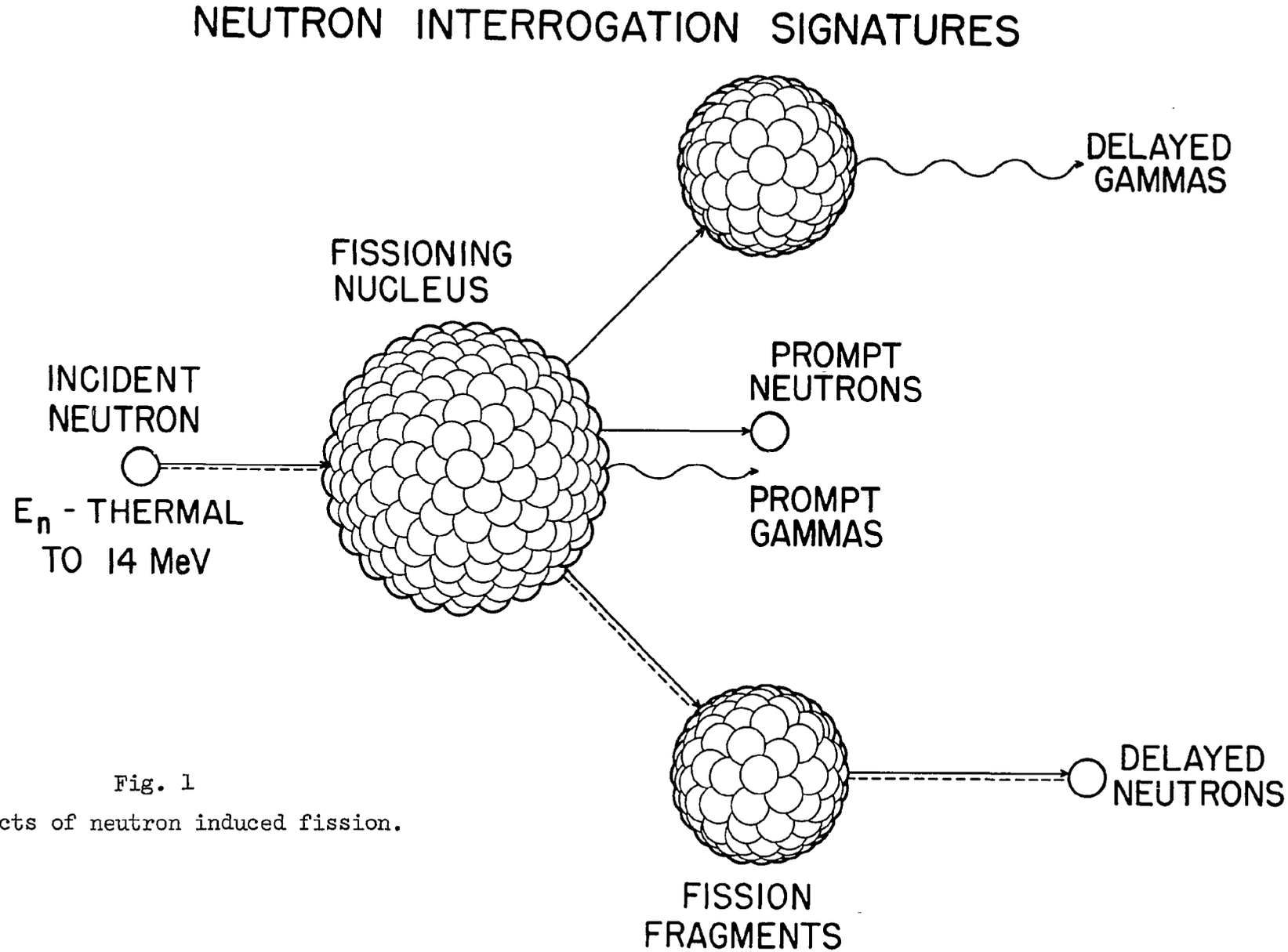

Fig. 1

Products of neutron induced fission.

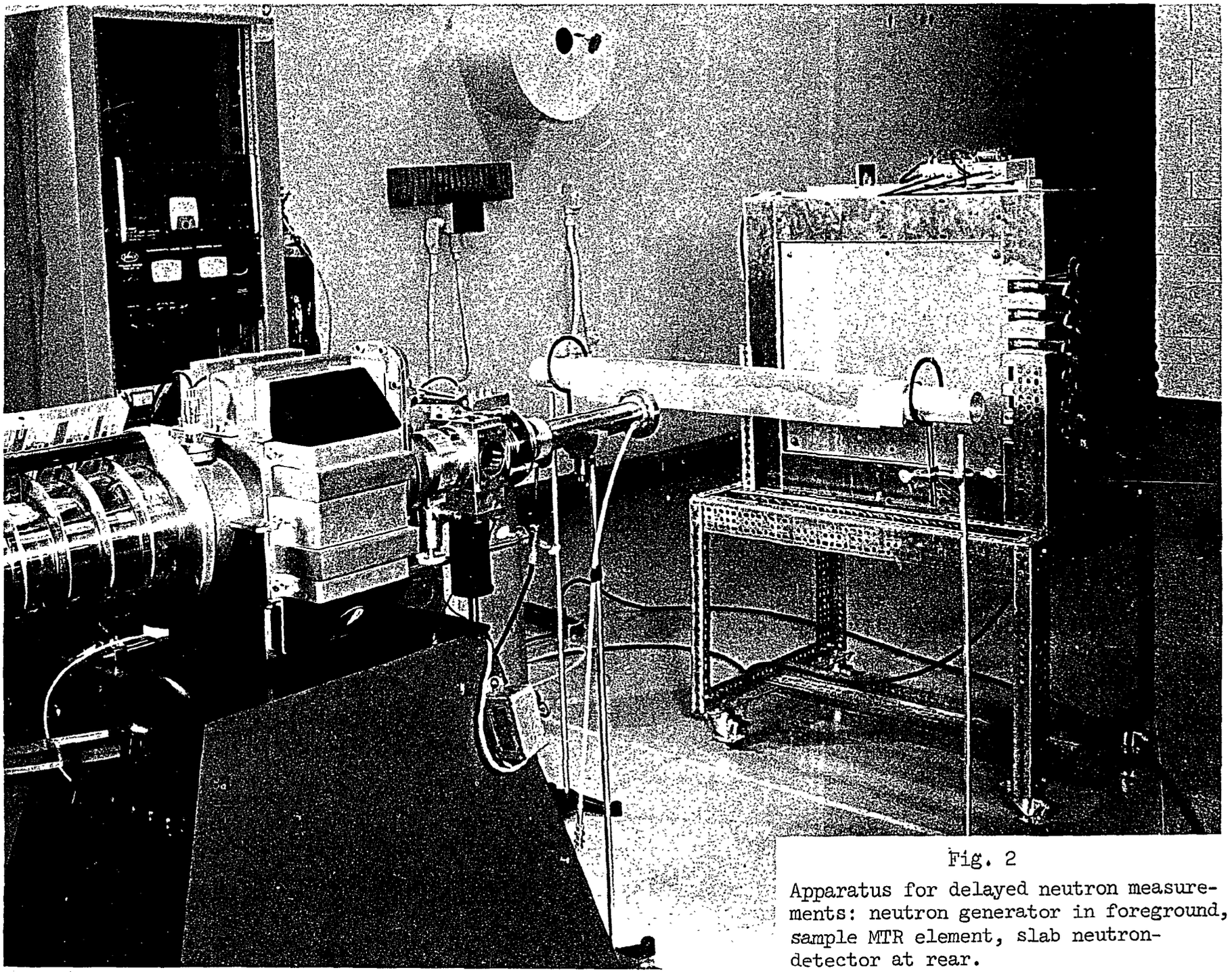

Fig. 2

Apparatus for delayed neutron measurements: neutron generator in foreground, sample MTR element, slab neutron-detector at rear.

Fig. 3

Moderating assembly for neutron generator to peak spectrum below fission threshold of ^{238}U.

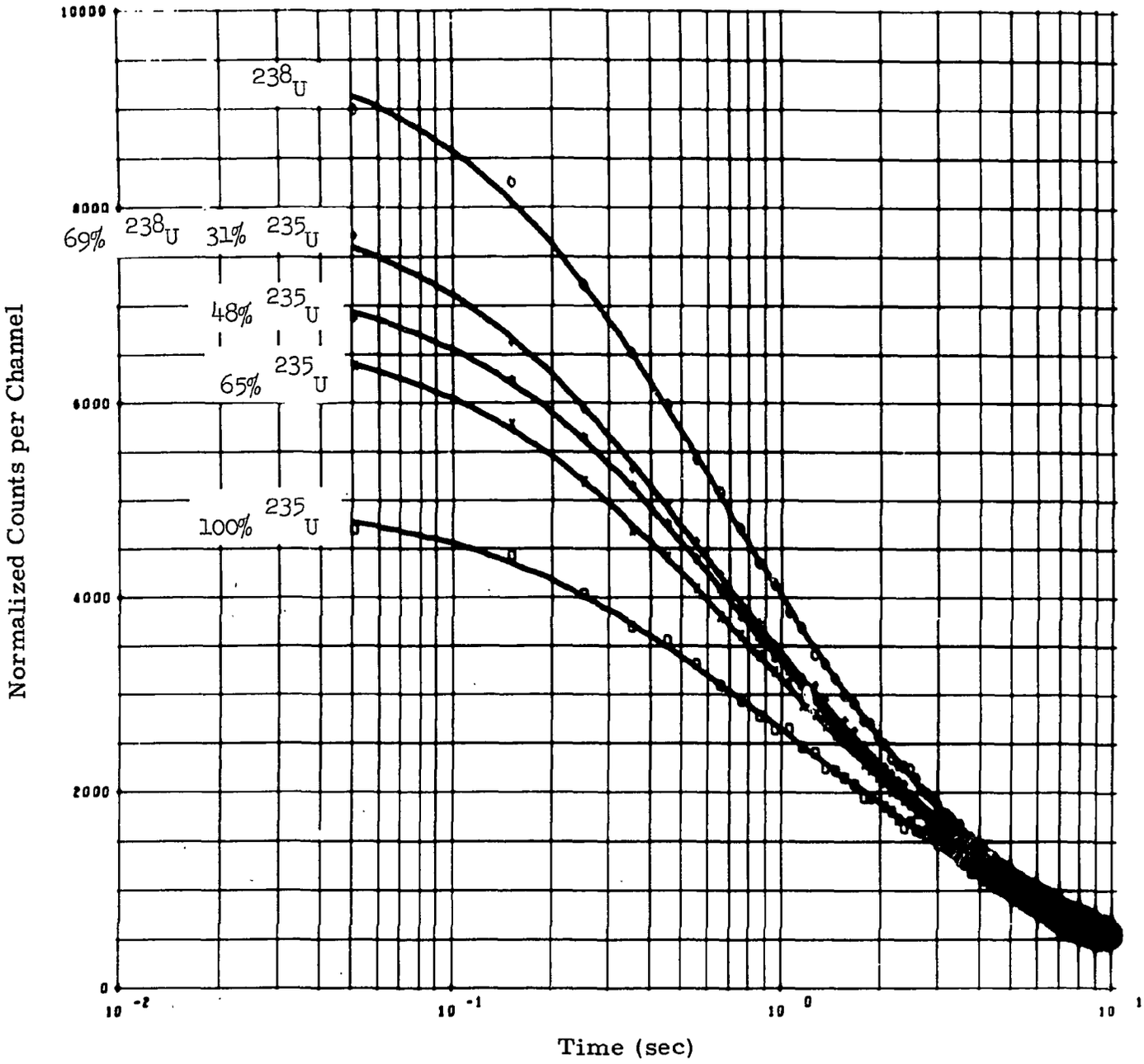

Fig. 4
Delayed neutrons as a function of time after excitation.

APPLICATIONS OF NEUTRON INTERROGATION TO ASSAY OF FISSIONABLE MATERIAL

M. M. Thorpe, C. N. Henry, Darryl B. Smith
H. O. Menlove, and R. H. Augustson

One area of interest for nondestructive analysis is an assay performed under the condition that little if anything is known about the nature of the extraneous material. At Los Alamos a program of neutron transport calculations has been undertaken to guide experimental investigations of nondestructive assay techniques. In particular, calculations using the one-dimensional DTF-IV transport code have been concerned with simulated scrap barrels and the expected delayed neutron response from small amounts of fissile material interspersed in large amounts of other nonfissile material. The standard 55-gallon barrel is simulated by a sphere of radium 30 cm ~ equivalent to the 12-inch radius of the 55-gallon drum. Representative moderating or matrix materials were chosen as follows: (1) hydrogenous material (polyethylene); (2) low-Z material (carbon); (3) medium-Z material (iron); and (4) high-Z material (lead). Several average material densities were assumed in the calculations. Each calculation consisted of two parts. The first was a direct calculation to compute the flux as a function of radius in the barrel due to the external 14-MeV surface source. This was followed by an adjoint calculation to determine the leakage probability for a single delayed neutron as a function of radial position of emission. The results of these two calculations were then used to compute the desired delayed neutron response of a gram of ^{235}U at any radial position within the barrel.

Figure 1 shows the results of some of these calculations. The results shown are for one gram at 15-cm radius. For the polyethylene, the response is seen to increase quite strongly with increasing density. The repsonse increase provides

greater detection sensitivity, but introduces undesirable extraneous material dependence. The increase is caused by the moderating effects of the polyethylene, which increases the average fission cross-section and delayed neutron yield. The response increases with density until the delayed neutron attenuation becomes the dominant effect. The iron-polyethylene mixture curves represent approximately equal weights of polyethylene and iron, and show the effects of roughly doubling the thermal neutron absorption cross section.

One method of reducing the response variation is to normalize to the response of a small amount of material placed at the surface of the barrel. This small amount of material samples the magnitude and energy spectrum of the surface flux. Insofar as the surface flux is representative of the radiation environment within the barrel, this technique, nicknamed the "add-a-gram" method, calibrates or normalizes out change in effective cross section due to moderation in the matrix material as well as the energy dependence of the delayed neutron yield. Other calculations indicated that a 2.5-cm polyethylene reflector would also reduce the variation in response. The results of calculations combining the effects of reflector and the add-a-gram method are shown in Fig. 2. The response curves are now considerably compressed, and lie within ±20% out to a net barrel weight of a little over 200 lbs. One of the reasons which accounts for this small variation in material dependence is the spatially compensating nature of the flux within the barrel and the delayed neutron leakage. Figure 3 illustrates this point. The delayed neutron leakage is least at the center of the barrel, and increases as the surface of the barrel is approached. On the other hand, the flux decreases from the center of the barrel. The net response is shown as the heavy line. For barrel weights of about 200 lbs. of the common material compositions thus far considered, the radial variation of response is about equal to the material response variations. The preceding considerations indicate that the interrogation by 14-MeV neutrons and detection of delayed neutrons should provide a useful technique for barrel assay. When the extraneous material is not highly radioactive, it may also be possible to utilize the fission gamma rays, both prompt and delayed. This possibility has not as yet been investigated.

The large barrel is not the only area of interest. Scrap, for example, is encountered in a variety of containers. In order to become familiar with the properties of systems much smaller than the barrel, it was decided to measure the neutron response, utilizing the yield technique, of a 2-inch diameter disc sample of ^{235}U both as a function of the amount of hydrogenous material present and as a function of axial position within a 4"-thick by 12" x 12" volume. The hydrogen content was varied by alternating slabs of polyethylene with void, thus obtaining an approximation to 1/4, 1/2, and 3/4 polyethylene density. To provide an approximation to the "add-a-gram" technique mentioned previously, the response was measured relative to two ^{235}U fission counters placed on either side of the slab. The fission counters respond to changes in effective fission cross section, but do not take into account the energy dependence of delayed neutron yield. The first measurements indicated a strong spatial variation in response at the edges. For this reason, an extra half-inch slab of polyethylene was added to each side to act as a reflector. The fission counters were imbedded in the slab so that they still sampled the surface of the volume. In practical assay applications, the sample may be inverted with respect to the target and detector (that is, rotated by 180^{o}), and the results of the two measurements averaged. The averaged fission counter normalized results with reflector are presented in Fig. 4. The neutron source target is located approximately 9" from one face of the volume, and the 20" x 24" cadmium covered sensitive area of a high-efficiency neutron detector is located about three and one-half inches from the opposite edge. That the fission counter normalization is indeed quite effective is attested to by the fact that for the entire range of densities the response variation is only about ± 50%, and less than this--about ± 15%--at the edges. It is to be noted that the response decreases with increasing density. This corresponds to the delayed neutron attenuation-dominated portion of the unnormalized barrel response curves. The spatial variation of the response curves is fairly flat up to half density. These response curves, obtained from a relatively simple arrangement of source and detectors, represents a beginning to the problem of assay in small bulk systems when little, if anything, is known about the nature of the extraneous material.

I should now like to turn attention to some specific applications where the extraneous material is known, so that it is possible to employ comparison assay to at least a rudimentary standard. Augustson, in the previous paper, has given an example of the yield technique as applied to a small disc sample. That the yield technique of delayed neutron assay can be effective under the proper conditions, for much more extended geometry than a small disc, is illustrated by the results of measurements performed on a mock fuel element. Figure 5 is a photograph of the element. The element resembles an MTR-type fuel element, constructed so that the amount and isotopic composition of the fissionable material can be varied by adding or subtracting material, in the form of thin strips between .030" sheets of aluminum held in place by slots at either end of the box. The dimensions of the box are approximately 4" x 4" x 18". To eliminate unwanted geometric effects, the element was positioned between the 14-MeV neutron source and a high-efficiency neutron detector so that the addition of a foil to either the front or the back side of the element gave an equal increase in delayed neutron response. Figure 6 shows that the response is quite linear, even though the amount of material is changed by about 40%. The maximum deviation was a little over 1%, with the average deviation being comparable to the measurement statistics of 0.5%.

For those applications of delayed neutron assay to the case where the sample is highly radioactive (for example in fuel elements or pieces thereof), it would be convenient both for the protection of personnel and for shielding of conventional neutron detection instrumentation if it were possible to perform assay through fairly massive quantities of lead. Recent preliminary measurements have indicated that assay through lead is not only possible but is an advantage in those instances when it is necessary to utilize a moderated or tailored spectrum to provide discrimination between the fissile and fertile species. In this instance the lead serves not only as a shield, but provides part of the material necessary to degrade the neutron source energy below the fission threshold of the fertile isotope. The physical arrangement of source, detector, and lead cask is shown in Fig. 7. The lead cask is 12" x 12" x 24" high with a 4" square hole in the middle, thus providing 4" of lead all around. When the ^{235}U loaded mock fuel element was surrounded by lead, the lead had the effect of increasing the response by a factor of

2.3. The trend of the deviations from linearity seemed to be a little poorer with the lead. The maximum change of the amount of material was 20%, and this gave a measured deviation of 1.3%. But this deviation includes a statistical uncertainty of 0.5%. Figure 8 shows the effects of adding two inches of polyethylene around the outside of the lead cask to provide considerable, although non-optimum, moderation of the neutrons from the source. The side facing the detector is left free of polyethylene. The curves illustrate the ability to detect small amounts of ^{235}U in the presence of a large amount of the fertile isotope ^{238}U. The difference between the upper and lower curves is not due only to the response of ^{238}U, since the depleted uranium added also contained some ^{235}U. The discrimination ratio is better than 100:1.

In conclusion, I should like to point out that even though, in principle, comparison to a standard that closely resembles the unknown offers the greatest accuracy, the incorporation of such techniques as add-a-gram or a variation of it, such as fission counter monitoring, might provide acceptable accuracy when used with only rudimentary standards.

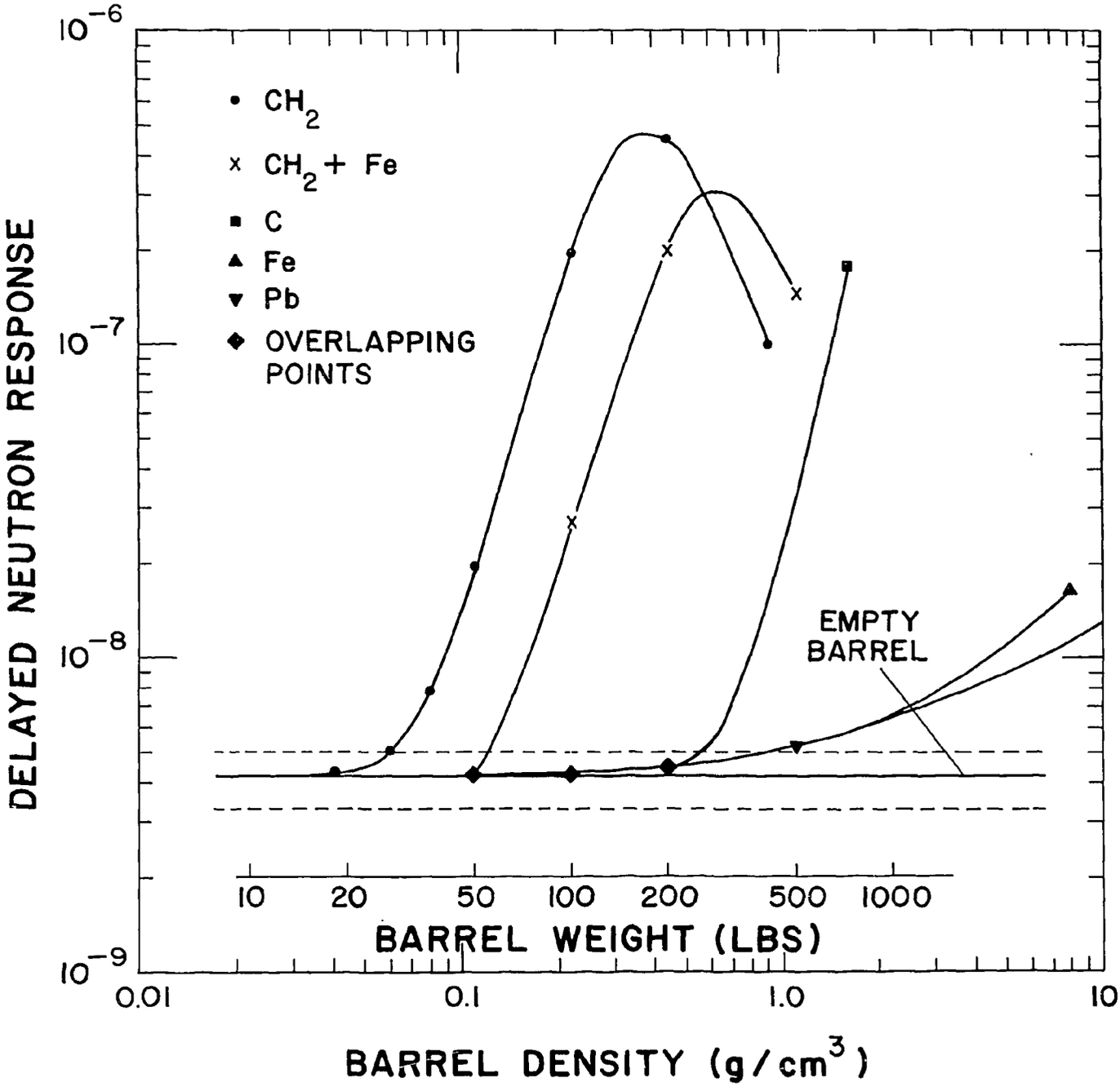

Fig. 1

Delayed neutron response of a simulated scrap barrel containing one gram of ^{235}U at 15 cm in representative moderator or matrix materials--14 MeV source. The dashed lines indicate empty barrel (no moderator) response ± 20%.

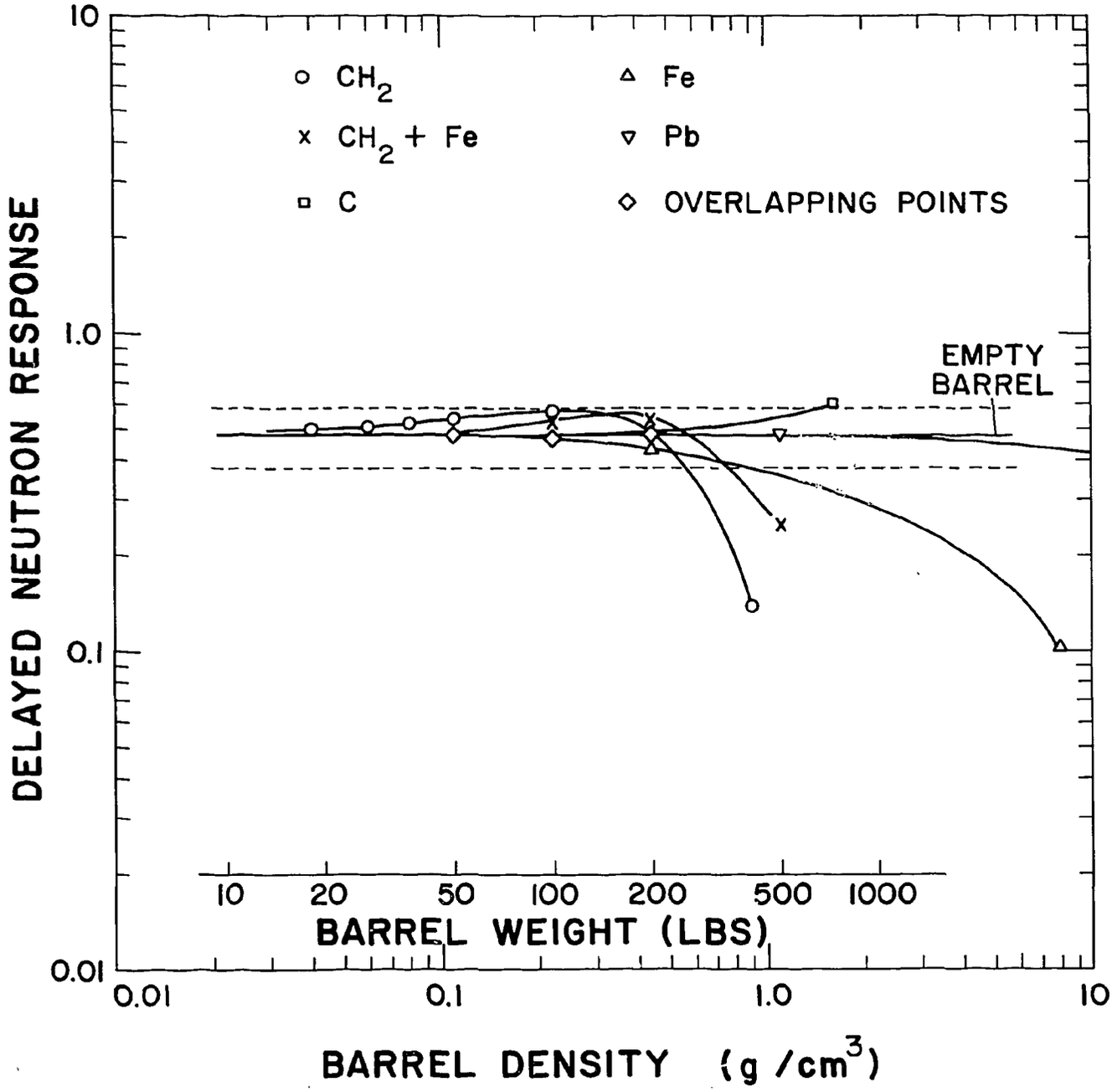

Fig. 2

Delayed neutron response of a simulated scrap barrel surrounded by a 2.5 cm polyethylene reflector. The response has been normalized by the add-a-gram method. The dashed lines indicate empty barrel (no matrix material) response ± 20%.

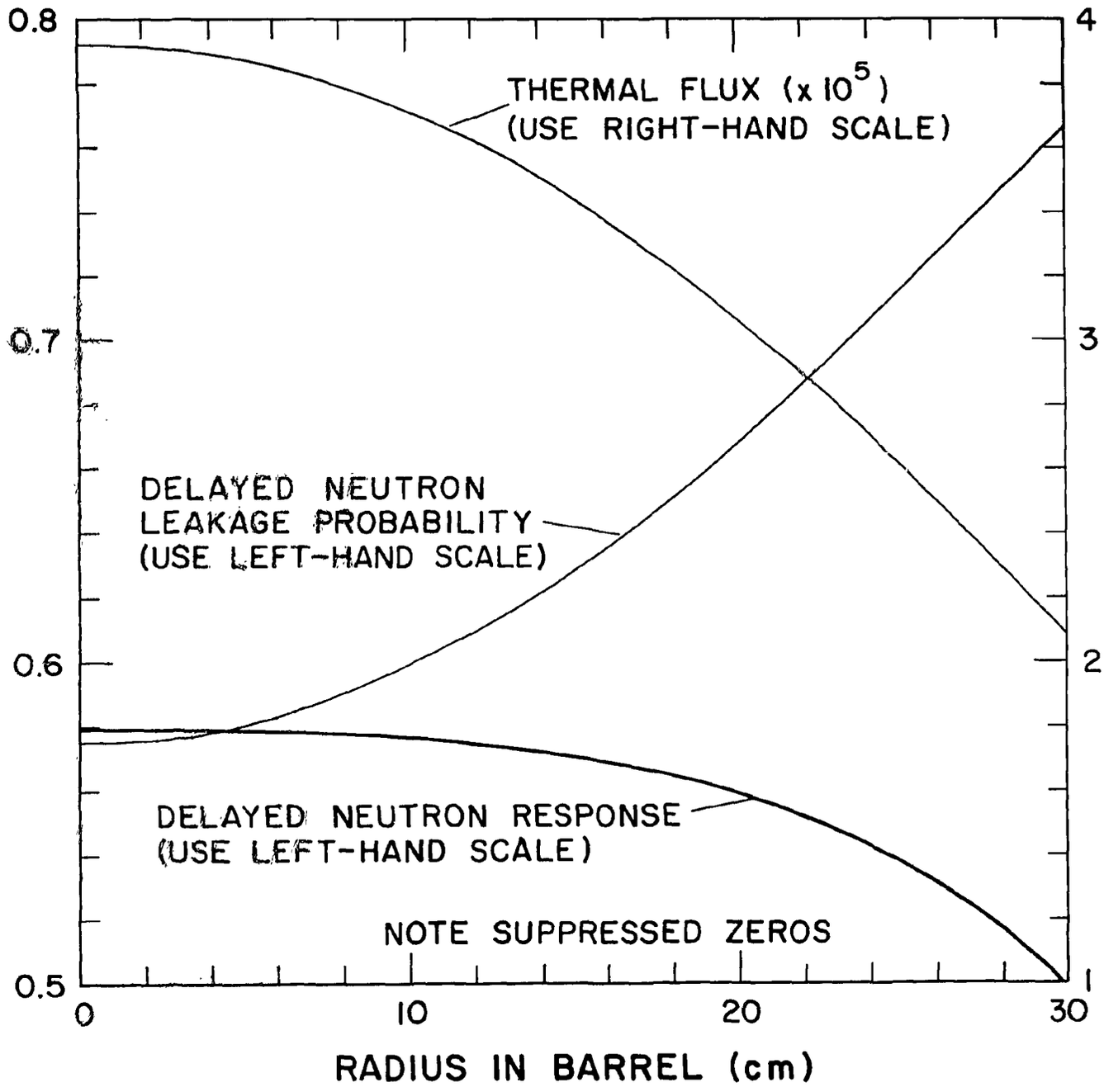

Fig. 3

Thermal flux, delayed neutron leakage probability, and resulting delayed neutron response (heavy line) as a function of the radial position of one gram of ^{235}U in a matric of CH_2 (average density 0.22 gm/cm^3).

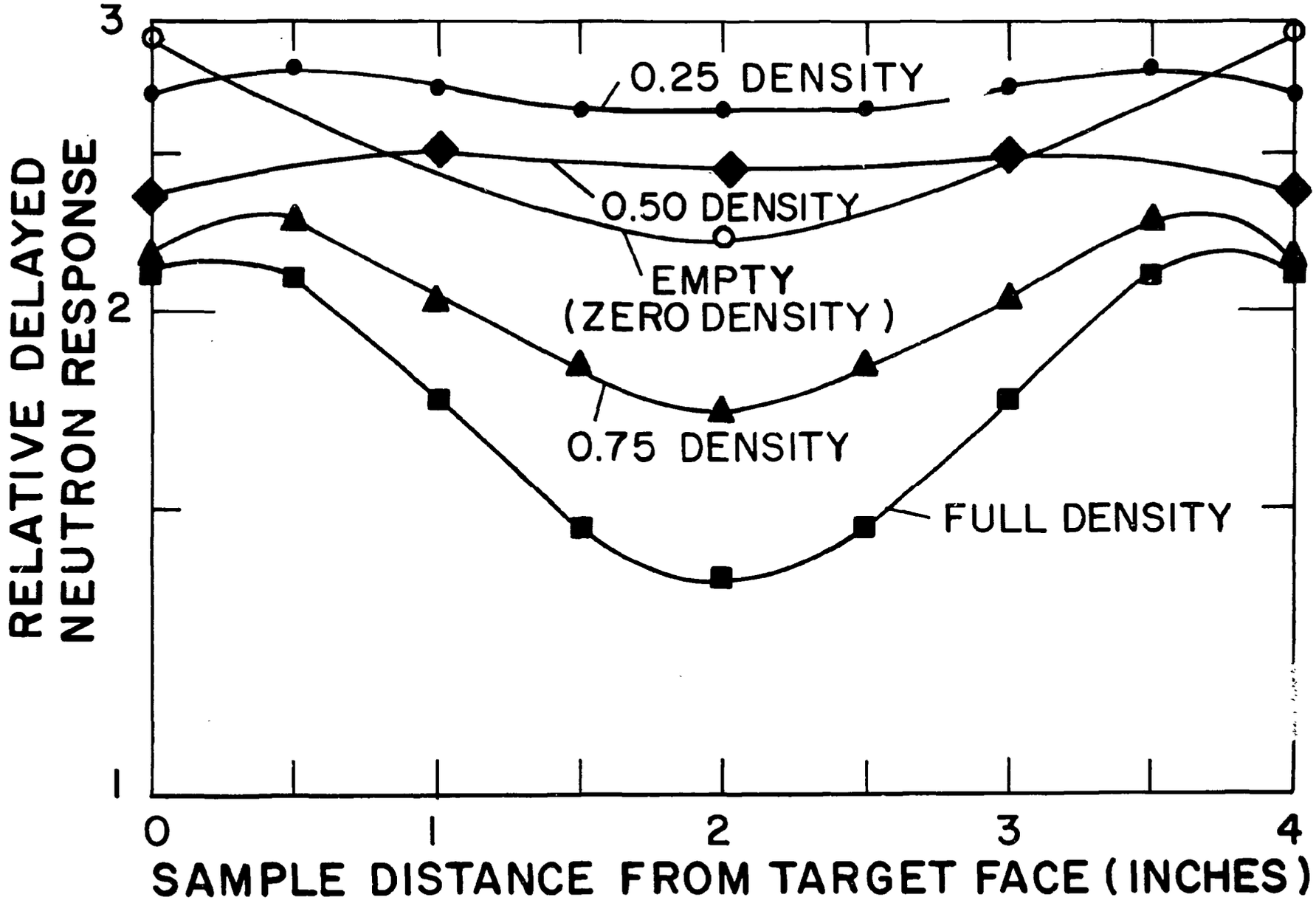

Fig. 4

Averaged delayed neutron response with half-inch polyethylene reflector.

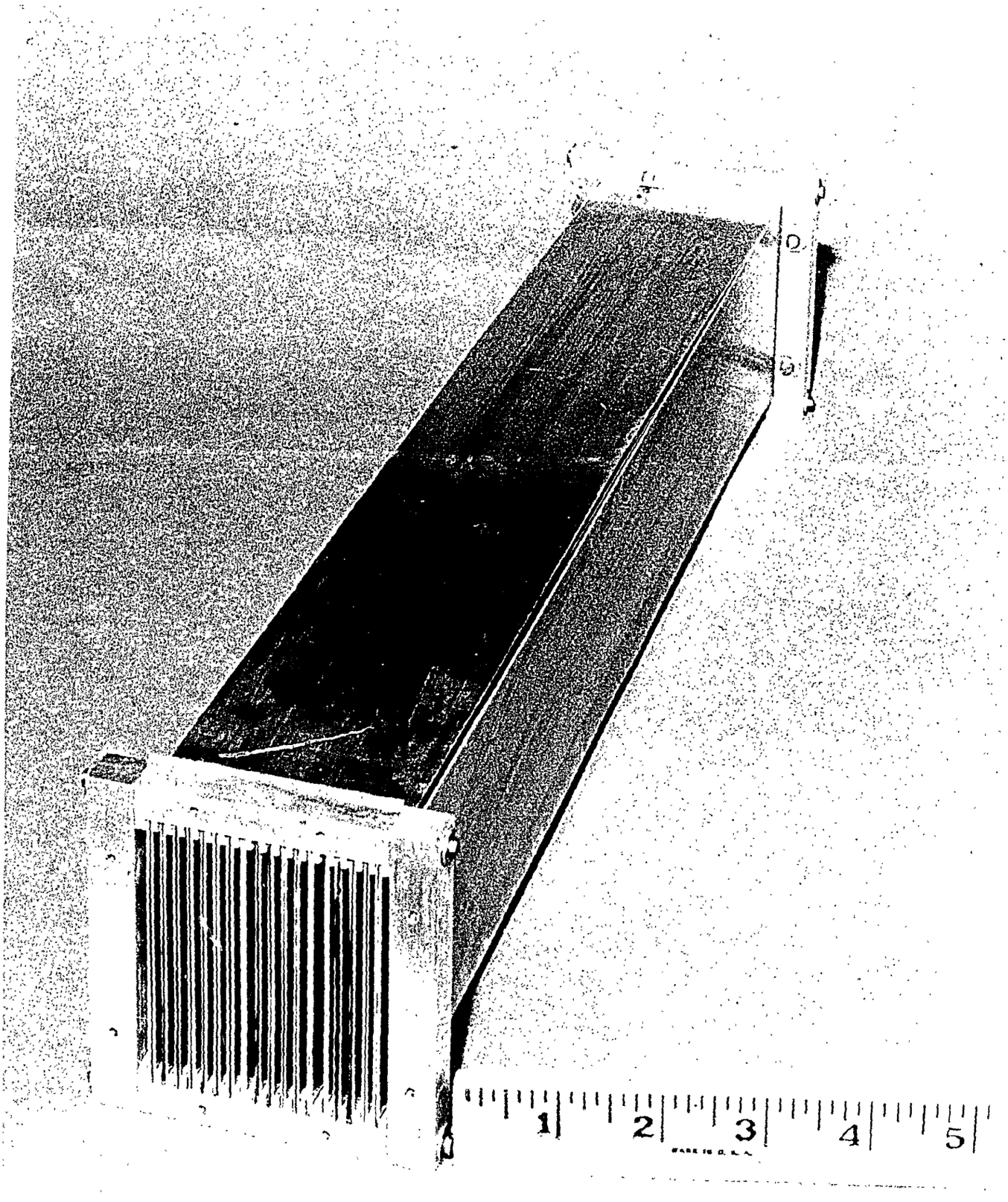

Fig 5.
Mock MTR fuel element.

DELAYED NEUTRON ASSAY OF ABSOLUTE
AMOUNTS OF FISSILE MATERIAL
IN MOCK MTR-TYPE FUEL ELEMENTS

14-MeV (D,T) Interrogating Neutrons

Actual Weighed ^{235}U Content (grams)	Delayed Neutron Assay Determination (grams)
339.95 (fully loaded)	Calibration Point
320.95	320.9
302.15	303.2
283.95	283.5
265.45	266.5
247.05	249.9

Average Deviation: 0.4%

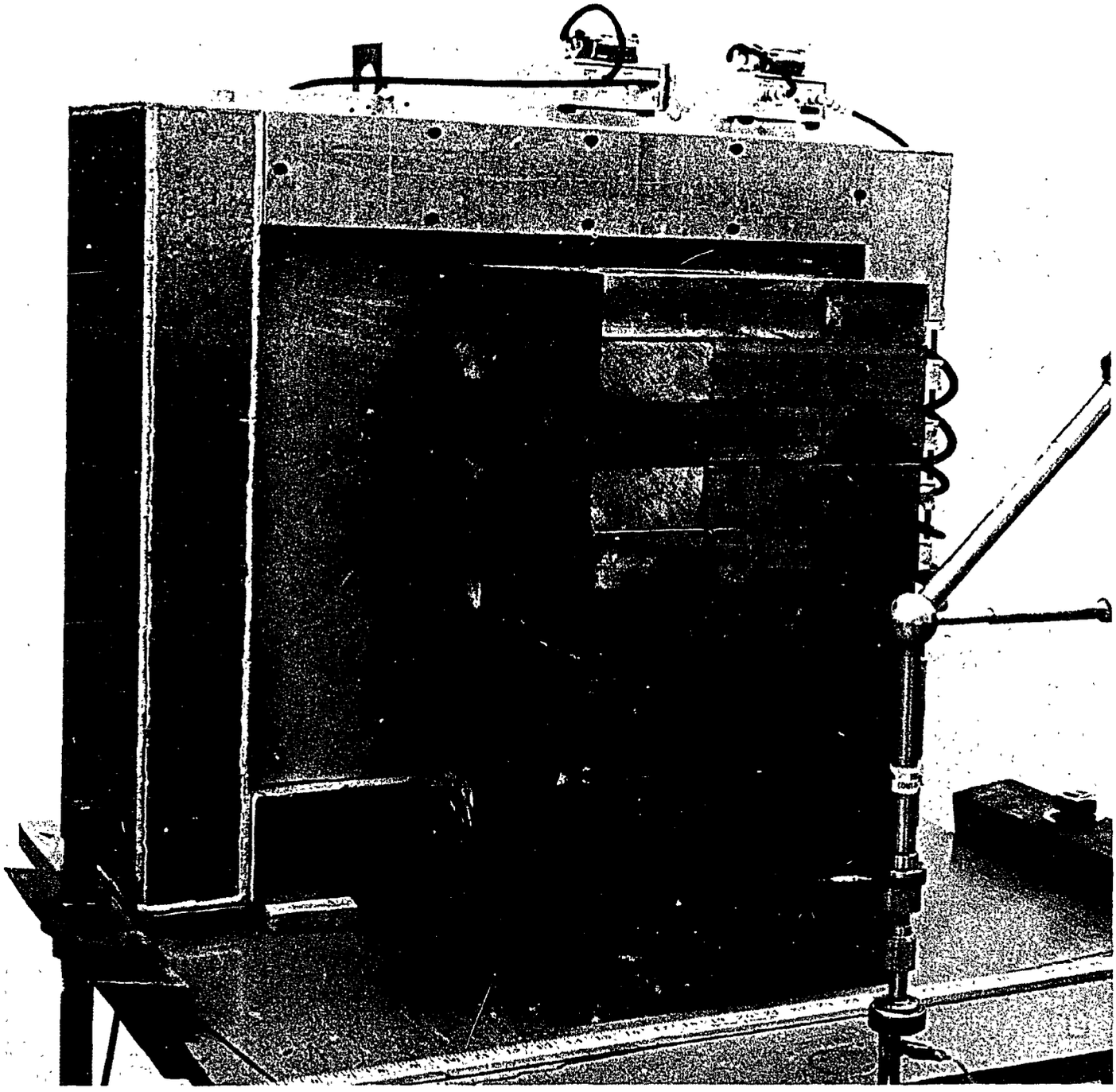

Fig. 7

Experimental arrangement for delayed neutron assay of MTR-type fuel element completely enclosed in a lead shield. The D, T neutron source is in the stainless steel sphere seen in the foreground, and the high-efficiency neutron detector is located behind the large lead shield.

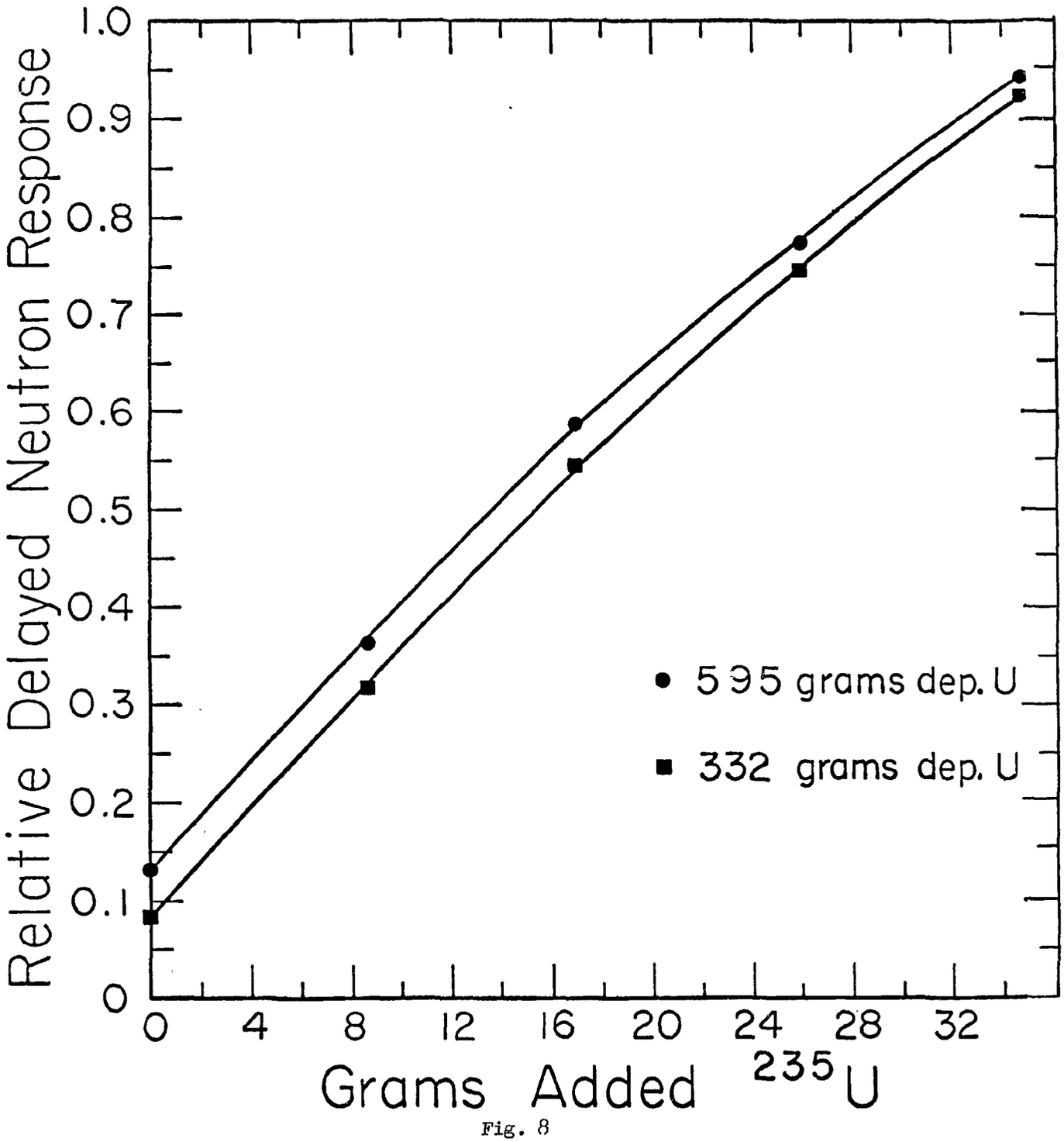

Fig. 8

The change in response due to adding depleted U. (Moderated interrogating neutron spectrum.)

PRECISION OF NONDESTRUCTIVE NUCLEAR MATERIALS ASSAYS

R. L. Bramblett

The nondestructive assay work now being done at Gulf General Atomic is based on the use of a low energy electron accelerator as a radiation source. There are several nuclear reactions that can be produced with such a source, and because of the variety of reactions, the electron accelerator approach can be applied to most safeguards assay problems. A very important aspect of the photoneutron and photofission reactions which occur is their very strong dependence on the end point of energy of the bremsstrahlung. The energy dependence adds a dimension to the space of measurements that can be used for assays. Another important aspect is that the yields from photoreactions can be very large.

The yield and energy dependence are shown in Fig. 1. The curves are bremsstrahlung yields from ^{238}U in a specific geometry: 1 g at 20 cm for a 10-μA electron current. Currents as large as 200 μA have been obtained on the Gulf General Atomic LINAC. Basically two curves are used to make the figure: photofission and photoneutron yields. The bremsstrahlung produced yield at 6.5 MeV is approximately the same as that from a 14-MeV neutron generator with 10^{11} neutrons/sec. The description of how one extracts assay data from the prompt neutron and the delayed neutron, and timed delayed neutron results has been given in much the same way as time delayed neutrons except that low resolution gamma detectors are used instead of neutron detectors and the time scale may be different.

Fission product gamma rays resulting from photofission by the bremsstrahlung have not been discussed. Consequently, some of our results are shown in Fig. 2. In this figure we

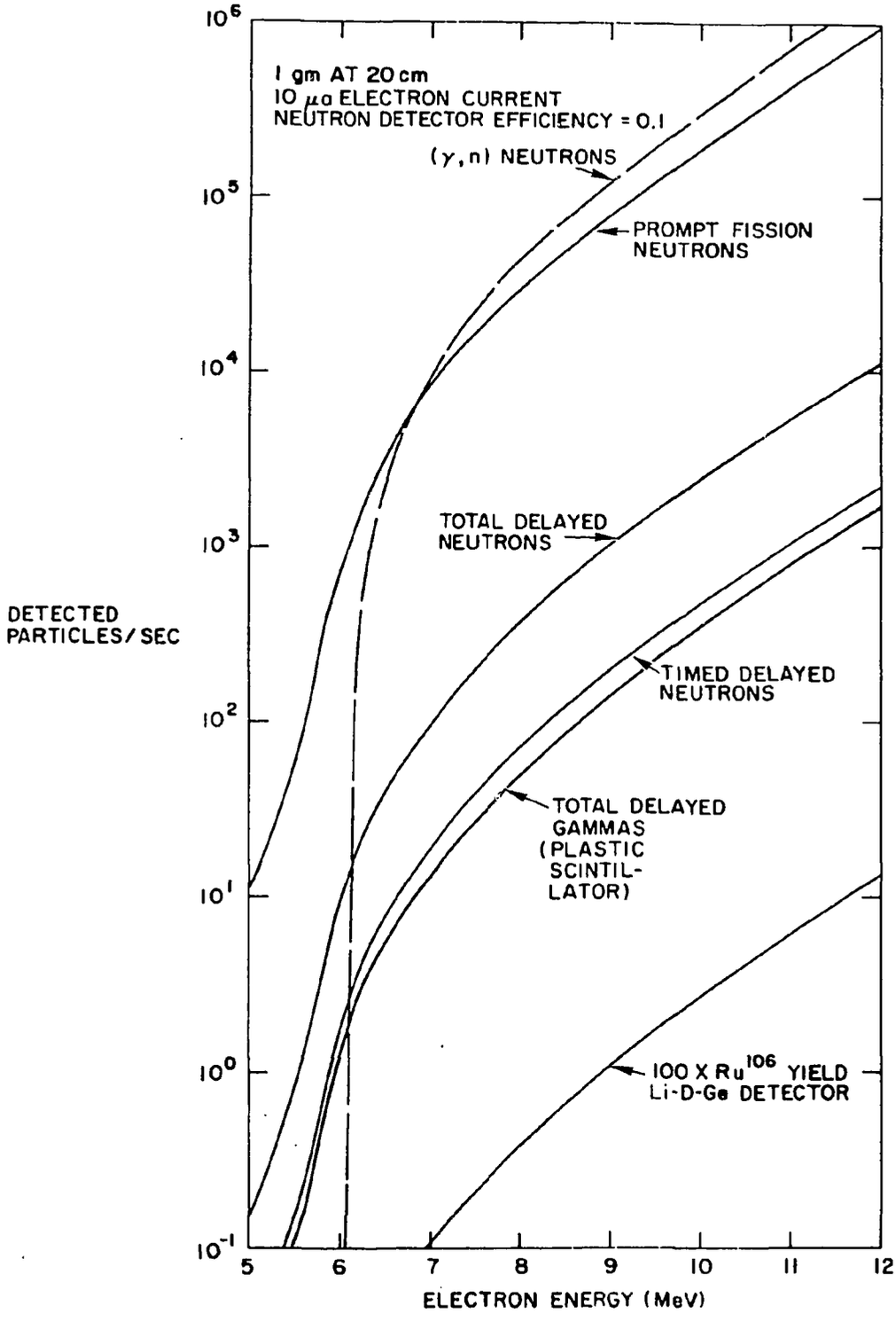

Figure 1. Photo produced yields in ^{238}U.

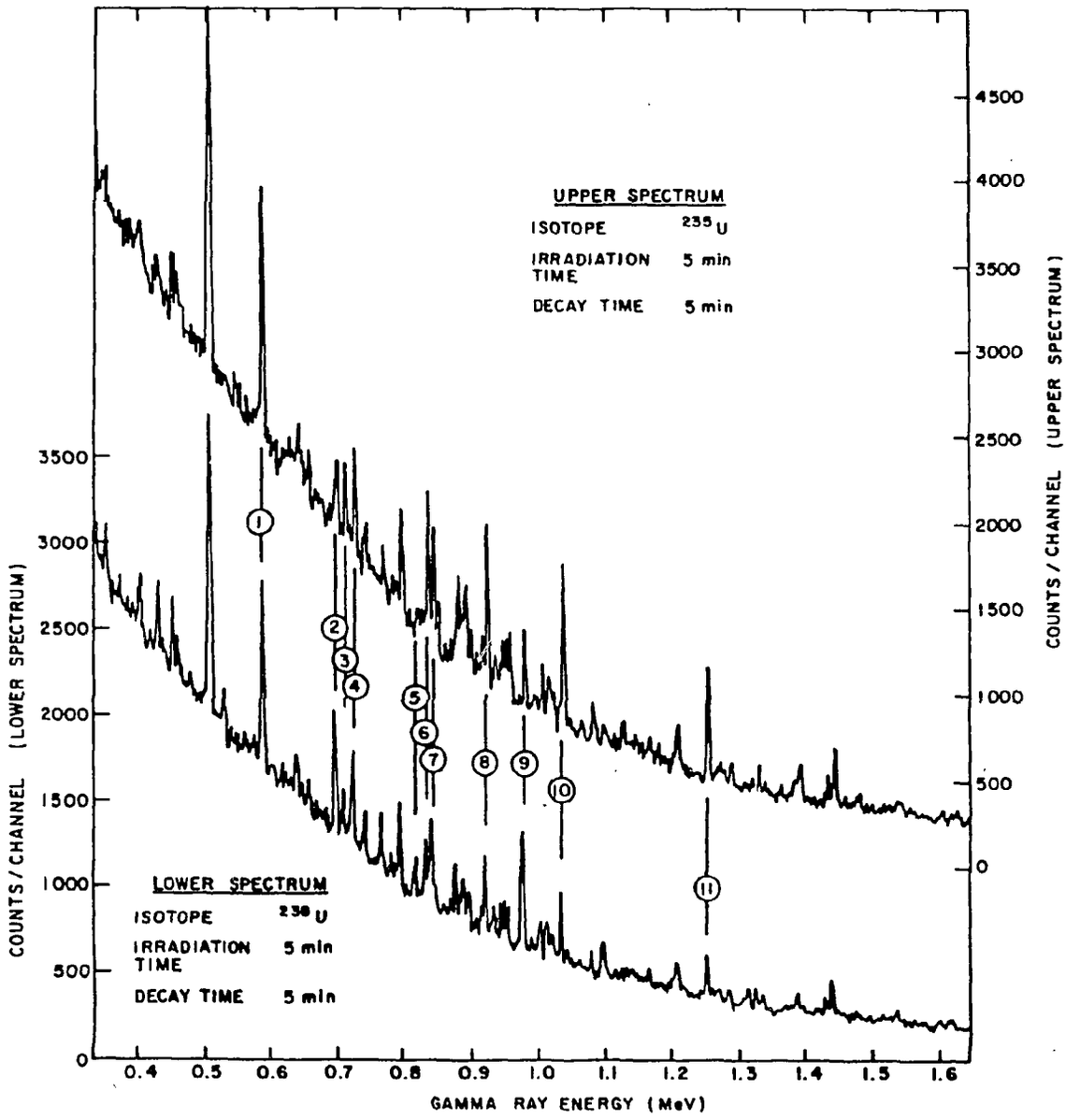

Figure 2. Fission product gamma-ray spectra five minutes after irradiation.

have plotted two spectra from identical samples of ^{238}U and ^{235}U, 17.24 g. The numbered lines are those that are substantially different from ^{235}U to ^{238}U. Assay would be based on the areas under the selected peaks in the two spectra--there are a number of them.

Figure 3 shows the analysis that is applied to both gamma-ray yield and neutron yield data. The analysis is straightforward but there can be a large number of measurements for any one irradiation, hence it is convenient to use a computer to find the values of X_j which minimize χ^2. Both matrix and grid search procedures have been used. The matrix approach is easier to program whereas in a grid search it is easier to put in limits. χ^2 is a sum of M terms whose average is one and is, therefore, on the average equal to M-N. If we remove an adjustable parameter, χ^2 is expected to increase by one. This is the way the error estimate on the data is used to estimate the error in the assay.

An illustration of the procedure is shown in Fig. 4. This is a calculation of χ^2 for four separate "gedanken" experiments. For ease in presentation, the problem is made one-dimensional by assuming that the total weight of ^{235}U plus ^{238}U in the sample is fixed at one gram, but the isotopic composition is varied. Two experiments consisted of 20-second measurements of the prompt neutron yield with 6.5 MeV bremsstrahlung. The discrimination ratio is 2.0 and the precision is ±0.01 gram in the amount of ^{235}U. The other two experiments consisted of 250-second measurements of time delayed neutrons produced by 7.5 MeV bremsstrahlung. The discrimination ratio is 1.83 and the precision is ± 0.05 gram. The better precision using prompt neutrons, even though the irradiation was increased by a factor of 100 for the delayed neutron measurement, can be understood from the next figure.

Figure 5 shows the delayed neutron yield as a function of time after a 300-second irradiation for ^{238}U and ^{235}U. Discrimination is obtained by comparison of yields at two different times. The statistics are good at early times but bad at late times.

Some results for other more realistic gedanken experiments are shown in Table 1. The table shown the precisions that can be attained for 340-gram samples at

YIELD = SUM OF (UNIT YIELDS X UNKNOWN WEIGHTS)

$$Y_i = \sum_{j=1}^{N} A_{ij} \; X_j \qquad i = 1, M \geq N$$

y_i = MEASURED YIELD

δ_i = ERROR IN y_i (FROM STATISICS OR SYSTEMATICS)

δ_{ij} = ERROR IN A_{ij} (FROM STATISTICS OR SYSTEMATICS)

FIND $\{X_j\}$ WHICH MINIMIZE

$$\chi^2 = \sum_{i=1}^{M} \frac{(Y_i - y_i)^2}{\delta_i^2 + \sum_j (X_j \, \delta_{ij})^2}$$

Figure 3. Formulae used in the analysis of the precision of assay data.

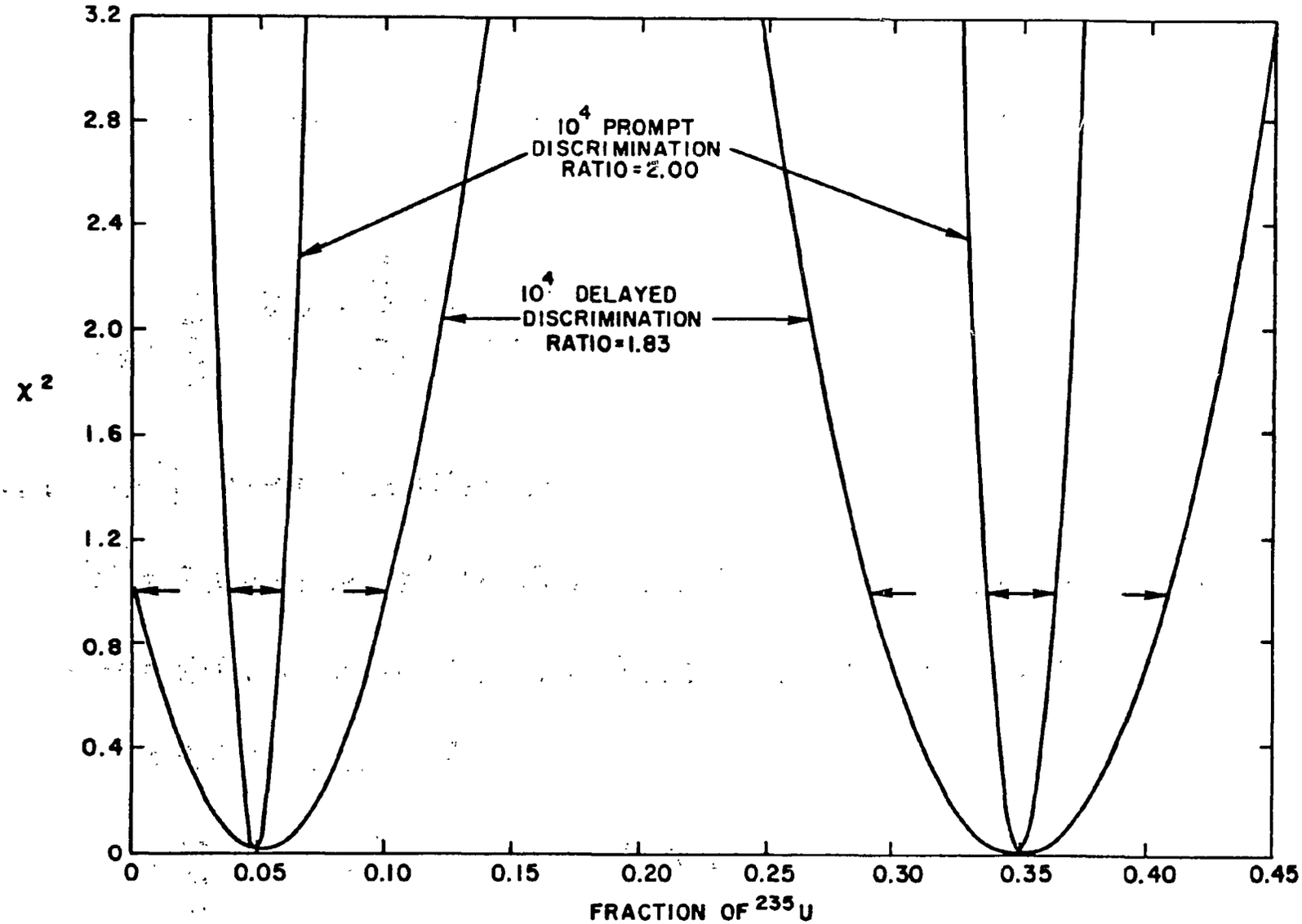

Figure 4. Values of chi square in four hypothetical experiments to determine isotopic composition.

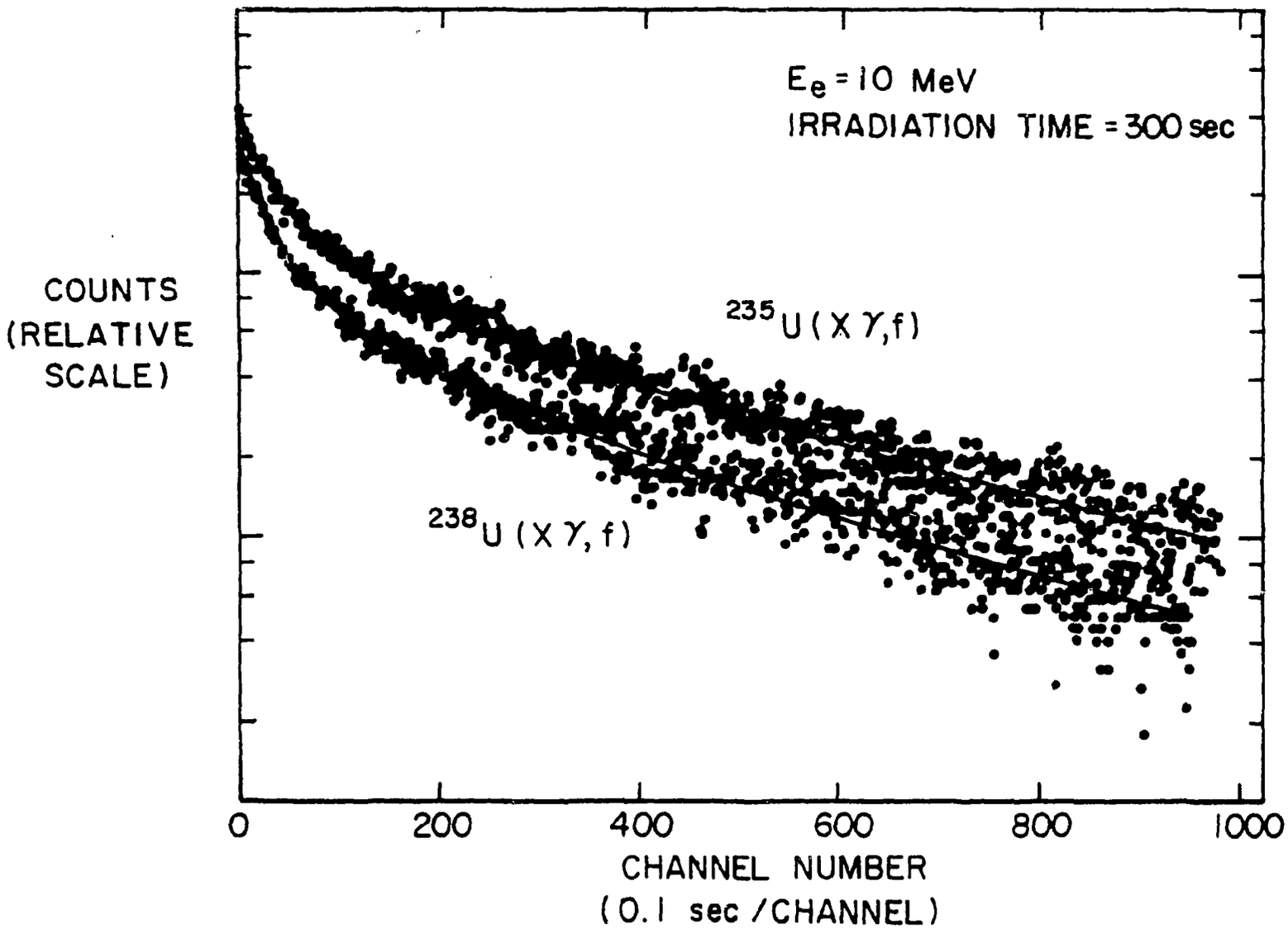

Figure 5. Delayed neutron yields from ^{235}U and ^{238}U photofission the two yields are normalized at t = 0.

TABLE 1A

PRECISIONS CALCULATED FOR ASSAYS ON A 340 g SAMPLE

Isotope Fraction and Precision						Detection Mode*
^{235}U			^{238}U		^{239}Pu	
.50 ±	.0133		.50 ±	.0127	---------	TD
	.0083			.0075		P
	.0038			.0036		P + TD
.33 ±	.078		.34 ±	.025	.33 ± .111	TD
	.0058			.0128	.0055	P
	.0053			.0056	.0026	P + TD

*TD - 10^5 timed delayed neutrons - experiment time 15 min.

P - 10^5 prompt neutrons.

TABLE 1B

PRECISIONS CALCULATED FOR ASSAYS ON A 340 g SAMPLE

Isotope Fraction and Precision				Detection Mode*	Typical Origin of Material
^{235}U	^{238}U	^{239}Pu	^{240}Pu		
---------	.95 ± .0089	.05 ± .0187		TD	Pu Breeder Input
	.95 ± .0100	.05 ± .0046		P	
	.95 ± .0031	.05 ± .0021		P + TD	
.40 ± .097	.40 ± .058	.15 ± .83	.05 ± .69	TD	Breeder Output

*TD - 10^5 timed delayed neutrons - experiment time 15 min.

P - 10^5 prompt neutrons.

TABLE 1C

PRECISIONS CALCULATED FOR ASSAYS ON A 340 g SAMPLE

Isotope Fraction and Precision		Detection Mode*	Typical Origin of Material
.012 ± .0137	.988 ± .0134	TD	
.012 ± .0037	.988 ± .0052	P	PWR input
.012 ± .0033	.988 ± .0040	P + TD	
.930 ± .0130	.070 ± .0123	TD	
.930 ± .0086	.070 ± .0045	P	MTR fuel
.930 ± .0034	.070 ± .0023	P + TD	

*TD - 10^5 timed delayed neutrons - experiment time 15 min.

P - 10^5 prompt neutrons.

six inches from a source of photons produced by a 10-μA beam at 6.5 MeV or by a 10^{11} neutrons/sec 14 MeV neutron source. In the table, TD means timed delayed neutrons and P means prompt neutron measurements at 5.75, 6.25 and 9 MeV bremsstrahlung energies. When both ^{235}U and ^{239}Pu are present, the TD results are poor. Large errors arise when only one technique is used for a sample containing many isotopes. The precision for the MTR fuel assay using delayed neutrons is about the same as those reported by Los Alamos. Note that, for a fixed total weight of the sample, the absolute errors in the amount of isotope are constant.

Table 2 shows the result of an actual assay performed recently at Gulf General Atomic. Prompt neutrons and the time integrated delayed neutrons were used. Two different detectors were used. The sample was 50-50 $^{235}U/^{238}U$. The precision of isotopic assay was 3.5%. The precision on total weight is limited by the systematic error which at present is probably about 2%. The fact that all the answers are much closer to the true weights may be due to overconservative estimates of our systematic errors.

Using the analysis described here, we are able to make informed evaluations of precisions in specified geometries, and we are able to reduce the data of assay experiments in an optimized way.

TABLE 2

ACTUAL ASSAY USING PRESENT FACILITY

Detector	^{235}U (g)	^{238}U (g)	Sum (g)
BF_3	17.40 ± 1.08	16.98 ± 1.37	34.38
^{3}He	17.42 ± 1.00	17.09 ± 1.08	34.51
Both	17.40 ± .62	17.06 ± .70	34.46
Nominal wt.	17.24	17.24	34.48

CALCULATION OF DELAYED NEUTRON MULTIPLICATION IN THIN SUBCRITICAL SYSTEMS*

Clarence E. Lee and Christopher F. Masters

University of California
Los Alamos Scientific Laboratory
Los Alamos, New Mexico

The delayed neutron response in a subcritical fissionable system excited by neutron or photofission sources can be examined in the Zero Prompt Lifetime Approximation.[1] A practical application of this approximation is the calculation of delayed neutron multiplication. The problem has its origin in the LASL nuclear safeguards program,[2] in which delayed neutrons are of primary interest.

Two physical processes dominate delayed neutron multiplication in thin samples. First, neutrons from fission induced by the external source cause further fissions and the accompanying birth of delayed neutron precursors. Second, neutrons from delayed precursors induce fissions, thus further enhancing the neutron signal. These latter neutrons may be considered delayed because their ancestry includes a delayed neutron.

Experiments have been performed to measure this delayed neutron multiplication.[3] A fissionable sample, in the shape of a thin disk, is irradiated to saturation by a neutron source modulated on and off at a frequency much greater than any delayed neutron precursor decay constant. The quantity measured at saturation is the delayed neutron response per unit sample thickness, between source bursts, as a function of sample thickness.

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*Work performed under the auspices of the U.S. Atomic Energy Commission.

The quantity of physical interest is the initial rate of change of the response with respect to sample thickness.

Under the pulsing conditions of the experiment, the Volterra equation resulting from the Zero Prompt Lifetime Approximation to the Boltzmann equation takes on a simplified form. The first term in the Neumann series expansion of the flux becomes

$$\Psi_0 = B^{-1} S \tag{1}$$

where B is the prompt Boltzmann operator in the notation of Ref. 1. The zeroth order flux, Ψ_0, gives rise to delayed neutrons and the resulting detector response is determined from

$$R_1 = < \Psi^+, \sum_i \chi_i \beta_i K \Psi_0 > \tag{2}$$

where χ_i and β_i are the delayed neutron spectra and fraction, respectively, and K is the fission neutron production operator. In this expression Ψ^+ is the solution to the inhomogeneous prompt adjoint Boltzmann equation with the detector as the inhomogeneous source.

The second term in the Neumann series expansion is given by

$$\Psi_1 = B^{-1} \sum_i \chi_i \beta_i K \Psi_0 . \tag{3}$$

The first order flux, Ψ_1, gives rise to delayed neutrons, and the detector response from them is determined from

$$R_2 = <\Psi^+, \sum_i \chi_i \beta_i K \Psi_1 >. \tag{4}$$

The total detector response at this point is $R_1 + R_2$. In this manner the detector response can be constructed to any order in the Neumann series. This series for far subcritical systems appears to converge rapidly.

The adjoint formalism permits the calculation of the n^{th} order detector response from the $n\text{-}1^{th}$ order flux, resulting in greatly reduced computational effort.

The experiments were done in thin disk geometry (2-in. diameter, 10 to 100 mil thicknesses), with 14 MeV source neutrons incident on one of the flat sides and the detector located on the other. Since the h/D ratios were small, the calculations were performed in one-dimensional plane geometry. The Los Alamos DTF-IV transport theory code was modified to incorporate the Zero Prompt Lifetime Approximation. Computational results and experimental data[3] are compared in Table I.

These calculations provide an initial investigation, with experimental comparison, of transport theory in the Zero Prompt Lifetime Approximation for very thin, leakage dominant, systems subjected to 14 MeV neutron sources.

1) W. L. HENDRY, G. I. BELL, "Analysis of an Initial Value Problem for the Neutron Transport Equation with Delayed Neutrons", Trans. Am. Nucl. Soc., 11, 232 (1968).

2) G. R. KEEPIN, "Nondestructive Detection, Identification and Analysis of Fissionable Materials", Proc. Symp. Safeguards Research and Development, AEC Research and Development Report, WASH-1076, p. 150 (1967).

3) R. H. AUGUSTSON et al., private communication.

TABLE I

INITIAL SLOPE, DELAYED NEUTRON MULTIPLICATION

Isotope	Experiment	Calculation
^{238}U	.087 (1 ± .05)%/mil	.060%/mil
93%^{235}U	.248 (1 ± .02)%/mil	.27%/mil

NEW APPROACHES IN NONDESTRUCTIVE FUEL ASSAY USING INDUCED EMISSION OF GAMMA RAYS

W. Michaelis, F. Horsch, H. Leuschner and C. Weitkamp
Institut für Angewandte Kernphysik
Kernforschungszentrum Karlsruhe
Karlsruhe, Germany

A modern nuclear control system based on automatic instrumentation requires nondestructive techniques which are accurate, reliable, tamperproof, and isotope specific. During the last year a broad program has been started at Karlsruhe for the study of new approaches to this problem. Herein use is made of a considerable scientific and technical background from nuclear physics experiments at the Karlsruhe reactor FR-2. It is the purpose of this contribution to summarize very briefly these activities restricting the discussion to unirradiated fuel since nondestructive assay of this material turns out to be the most urgent problem.

In view of the requirements of the methods needed extensive use is made of high-resolution germanium diodes. Detecting characteristic gamma rays of sufficiently high energy may fulfill all requirements if the emission is induced by neutrons or photons of appropriate energy.

One possible reaction for producing characteristic gamma rays is the radiative neutron capture process. Deexcitation from the capturing state occurs preferably by dipole transitions to levels of low excitation energy. Here the level density is small and thus the upper part of the high-energy spectrum shows a simple structure. A typical example is presented in Figure 1 where the high-energy spectrum from a nickel sample is given. In this case the isotopic composition was considerably different from that of the natural element (^{58}Ni 1.62%, ^{60}Ni 5.18%, ^{61}Ni 92.11%, ^{62}Ni 1.08%). The individual isotopes are identified by the energy of the transitions observed and

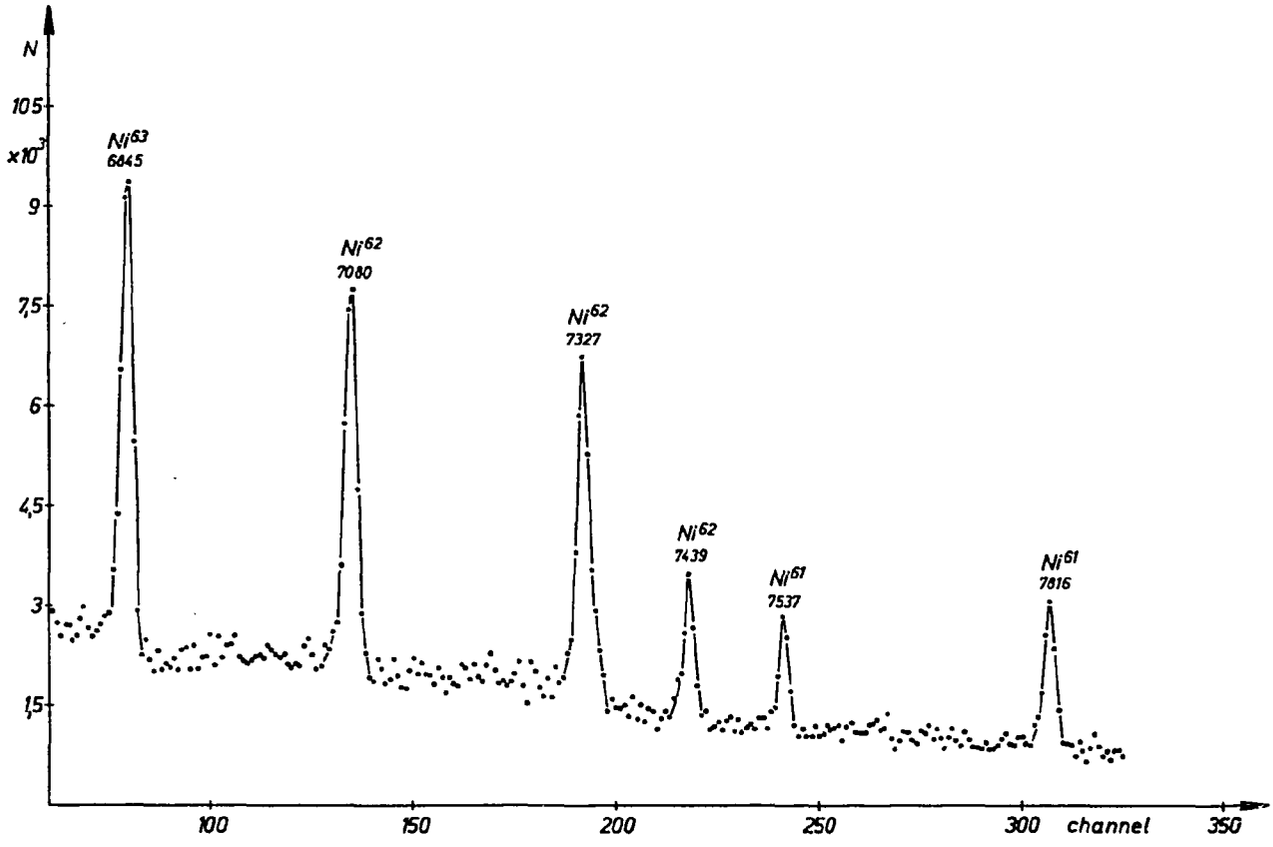

Fig. 1

High-energy capture gamma-rays from a nickel sample.
Energies in keV.

they are analyzed quantitatively via the peak areas of these gamma rays. This method yields the best possible signature of the various isotopes in the sample.

In fissile material this clear capture spectrum is masked by the prompt gamma-ray spectrum from fission and--to a lesser extent--by the delayed gamma rays arising from beta decay of the fission products. As has been proposed by Michaelis[1], the first interfering component may be suppressed by detecting the fast fission neutrons in anticoincidence and in 4π geometry. Since the average number of neutrons emitted per fission is 2.5 or even 3.0, a detection efficiency of about 70% for a single neutron is sufficient to obtain 95% total efficiency. The principle can be sketched as follows: The sample or the irradiated area of the fuel element is surrounded by a plastic or a liquid scintillator of appropriate thickness. Slow neutrons scattered in the fuel are kept off the detectors by a shield of ^{6}Li. The gamma rays are filtered by a few mm of lead and are detected in a germanium counter used as a double-escape spectrometer. A lead shield of about 5 cm thickness between sample material and neutron detector prevents suppression of capture gamma rays coincident with low-energy photons. A detailed analysis of the system is given in reference 1.

The second interfering component arising from beta decay of the fission products is no severe obstacle in the case of short measuring intervals. The most important data on radiative neutron capture in uranium and plutonium are summarized in Table 1. The information is not yet complete, since no systematic studies on radiative capture in fissionable isotopes have been performed till now. From the binding energies and the spin values of target nucleus, capture state and ground state of the product nucleus, the energy intervals can be determined where the most energetic primary gamma rays are to be expected. Assuming that 0.05 photons per MeV and capture process are emitted in the energy range under study--this is a very conservative estimate--we can calculate the intensity in photons per MeV and fission relative to the interfering components. The result is shown in Figure 2.

In order to demonstrate the applicability of the method, an experimental setup has been installed at the reactor FR-2 consisting of a Ge(Li) counter and a 44 cm dia. x 60 cm

TABLE 1

Neutron Capture in Uranium and Plutonium Isotopes[a]

Isotope	Thermal Neutrons /barn/	/barn/	$\sigma_{n\gamma}/\sigma_{nf}$	$\bar{\nu}$	E_B /keV/[b]	J_T	J_c	J_p	E_a /keV/	J_a	E_n + 0.1 eV $\sigma_{n\gamma}$ /barn/c	σ_{nf} /barn/c	$\sigma_{n\gamma}/\sigma_{nf}$
U^{233}	49.4	525.9	0.094	2.51	6783	$5/2^+$		0^+					
U^{234}	95	-	∞	-	5267	0^+	$1/2^+$	$7/2^-$					
U^{235}	100.9	578.3	0.175	2.44	6467	$7/2^-$	(4^-)	0^+	150	4^+	50	250	0.17
U^{236}	6	-	∞	-	5304	0^+	$1/2^+$	$(1/2^+)$					
U^{238}	2.73	-	∞	-	4784	0^+	$1/2^+$				~ 2	-	
Pu^{239}	265.8	743.1	0.358	2.88	6455	$1/2^+$	(1^+)	0^+	597	(1^-)	250	450	0.6
Pu^{240}	250	< 0.1	>2500		5412	0^+	$1/2^+$	$5/2^+$			150		
Pu^{241}	425	950	(0.39)	3.00	6219	$5/2^+$		0^+			300	600	0.5
Pu^{242}	19	< 0.2	> 95		5047	0^+	$1/2^+$						

[a] Here: $\sigma_{n\gamma}$ = (n,γ) cross section

σ_{nf} = fission cross section

$\bar{\nu}$ = fission neutrons per fission

E_B = neutron binding energy

J_T, J_c, J_p = spin of target nucleus, compound nucleus and product nucleus ground state

E_a, J_a = excitation energy and spin of the first level which can be populated by E1 radiation

b From systematics of heavy elements (V. E. Viola and G. T. Seaborg, J. Inorg, Nucl. Chem. 28, 697 (1966)

c Estimated values

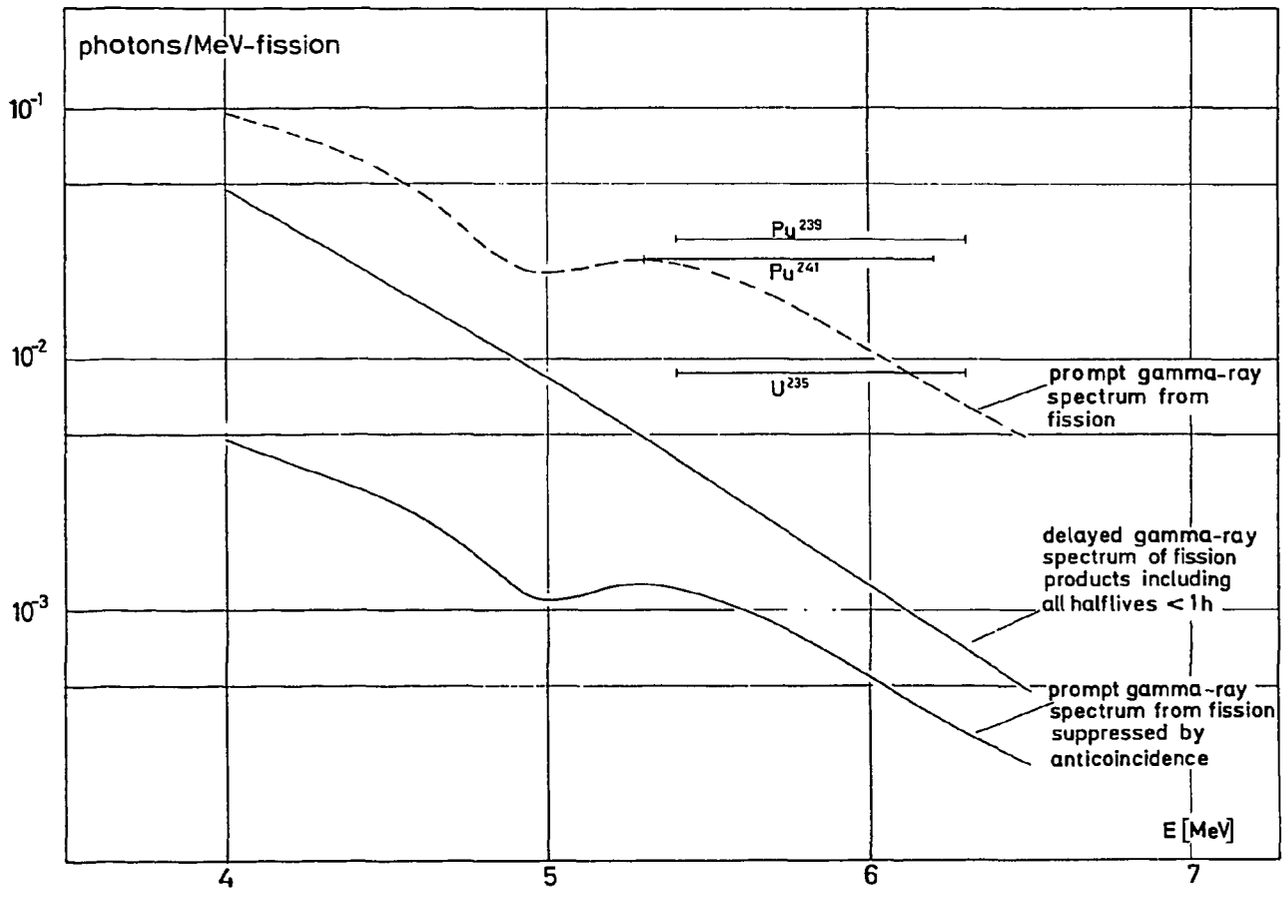

Fig. 2

Estimated intensities of capture gamma-rays together with interfering components.

plastic scintillator which is provided with appropriate wells for the neutron beam and the gamma detector. A schematic drawing of this apparatus is given in Figure 3. For obtaining a maximum signal-to-background ratio the scintillator was bevelled on both ends. Figure 4 shows a photograph which was taken before assembling the system.

The method can be extended without difficulty to neutrons from an electron accelerator. A neutron beam filtered with ^{235}U and/or ^{239}Pu may be most expedient for improved penetrability. An upper limit of about 5 keV is set on the neutron energy by the required gamma-ray resolution, that means the kinetic energy of the inducing particle should have no remarkable influence on the line width. Neutrons of higher energy may be eliminated using the time-of-flight method. The extension of nonthermal neutrons also improves the capture-to-fission ratio.

We have calculated the penetrability as a function of neutron energy for a typical fast breeder fuel element. The results show that for a properly filtered neutron beam there is absolute certainty that no material can be diverted. The radiative capture method offers several advantages the most important of which are as follows:

1. The method provides the optimum distinctive signatures for the various fissile materials.

2. All relevant isotopes including nonfissionable species added to the fuel are detectable.

When operating the device in the coincidence mode high resolution studies of the prompt fission gamma-ray spectrum can be performed without interference from the radiative capture process. Thus the fission spectrum can be scanned for possible differences in its structure for the various fissile isotopes. It is not unlikely that the high-energy part of the spectrum when measured with germanium counters exhibits clearly distinguishing characteristics.

Another group of methods under investigation employs the delayed gamma radiation from fission. According to its origin delayed radiation can be divided into two main categories: radiation arising from isomeric states in the fission products and radiation following beta decay.

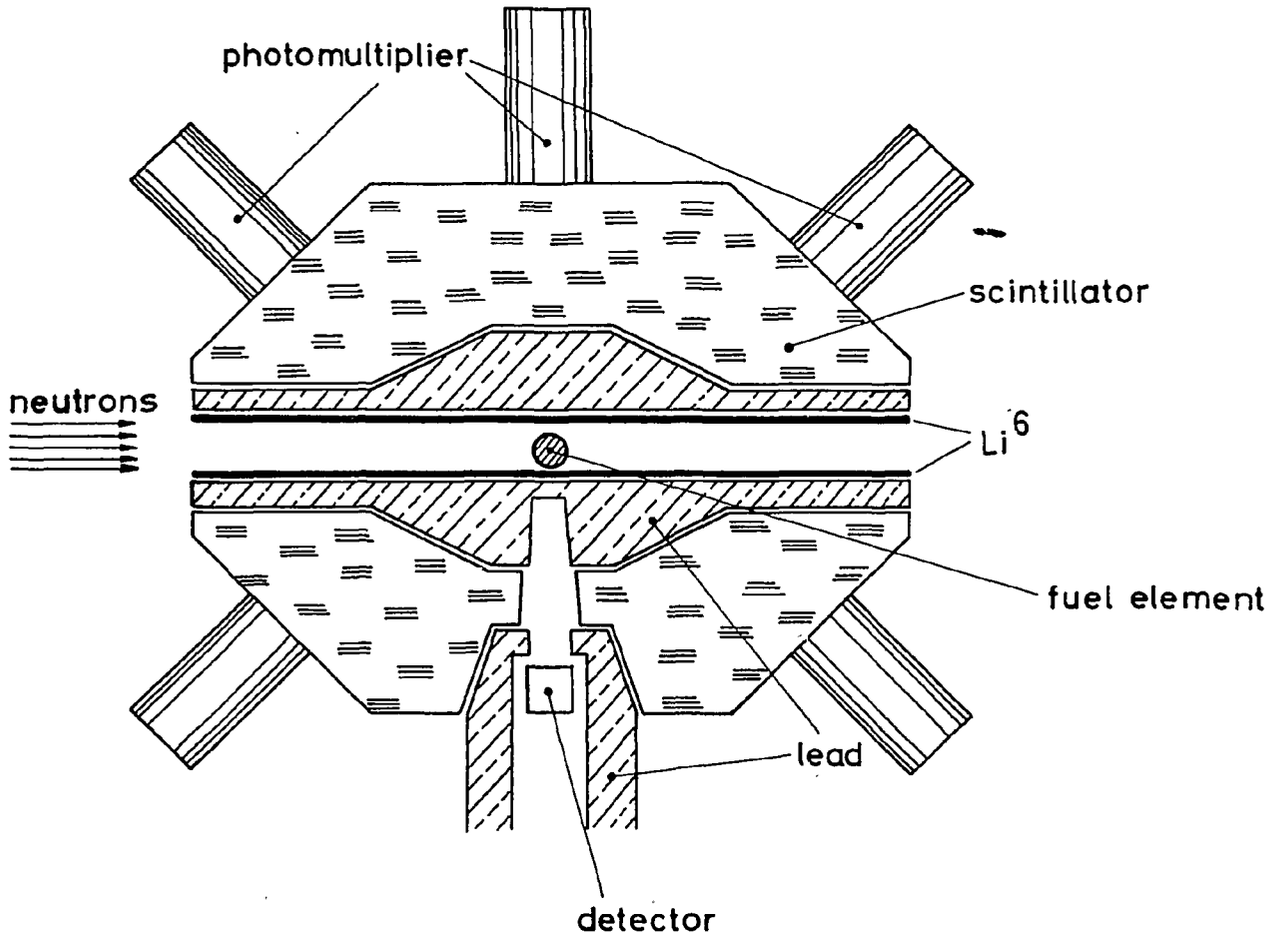

Fig. 3

Schematic drawing of the (n, γ) apparatus.

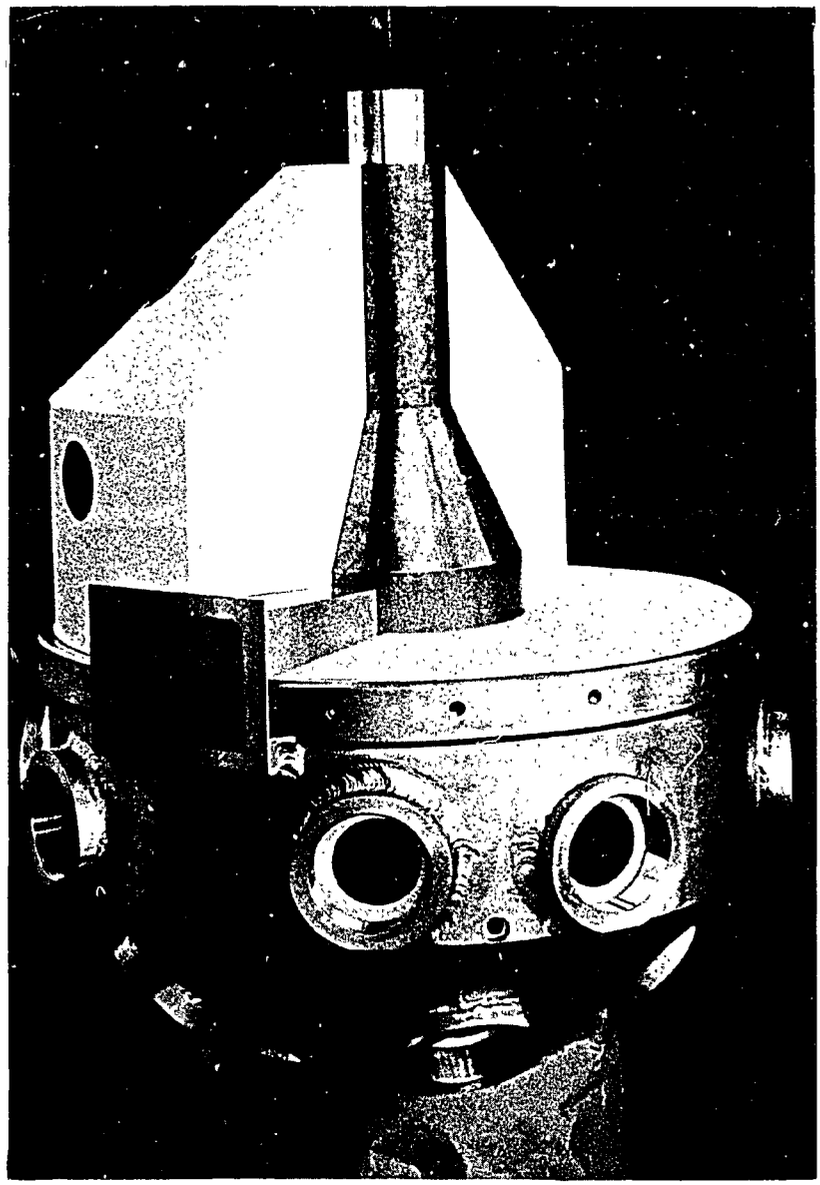

Fig. 4 Photograph of the neutron detector taken before assembling.

Till now there have been no systematic high-resolution studies of both components and the information on the basic data is thus still very poor. The following facts are known:

1. A group of the Ioffe Institute at Leningrad[2] found a comparatively simple structure of the delayed spectrum in the nanosecond region (less than 20 gamma lines below 2 MeV) from fission of ^{235}U. Data on ^{239}Pu do not exist yet.

2. The General Atomic group observed marked differences in the medium-energy spectra from fission-fragment isomers for ^{235}U and ^{239}Pu in the 100 μsec region[3]. The spectra were taken with NaI(Tl) scintillation detectors.

3. The beta component shows a different time behaviour when being observed in several broad energy intervals. Such measurements have been performed at Los Alamos[4].

4. High-resolution spectrometry of the beta component has been concentrated on long decay times (>1 min) and on spent reactor fuel (see e.g., References 5, 6).

Although these data are not exhaustive, they are encouraging enough to make systematic studies over the whole time scale. We have therefore constructed a combined fast-slow chopper system which together with the instrument described in the first part of this paper allows studies over the whole range from a few nanoseconds up to several seconds. While the first instrument covers the nanosecond region by delayed coincidence techniques, the second instrument uses a properly chopped beam for the longer delay times. A schematic view of the apparatus is shown in Figure 5. The main data of the system are:

1. Fast Chopper

Rotation axis	vertical, perpendicular to beam
Number of rotors	1
Maximum speed	12,000 RPM
Diameter of rotor	17 cm
Number of slits	1
Slit area	0.46 cm x 3.00 cm
Material	53.5% B_2O + 46.5% Plexigum 339
Transmission for thermal neutrons	3.4×10^{-5} per cm
Maximum pulse frequency	400 per sec
Minimum pulse width (PWHM)	43 μsec

2. Slow Chopper

Rotation axis	horizontal, parallel to beam
Number of disks	2
Maximum speed	1300/1230 RPM
Diameter of disks	34.5 cm
Number of holes	1 per disk
Diameter of holes	4.0 cm
Material	66.0% $^6Li_2CO_3$ + 34.0% Plexigum 339
Thickness	1.0 cm
Transmission	1.0×10^{-7} cm
Maximum pulse frequency	22 per sec (single disk) 1.0 per sec (dual disk)
Minimum pulse width	21 msec
Fast-to-slow chopper frequency ratio	332 : 1

Since high-energy neutrons or photons can be utilized in the delayed gamma-ray methods sufficient penetrability is obtainable even for highly enriched materials.

The various parts of both the prompt neutron detection device and the combined chopper system are now in a testing procedure. The first final spectra are expected in the very near future.

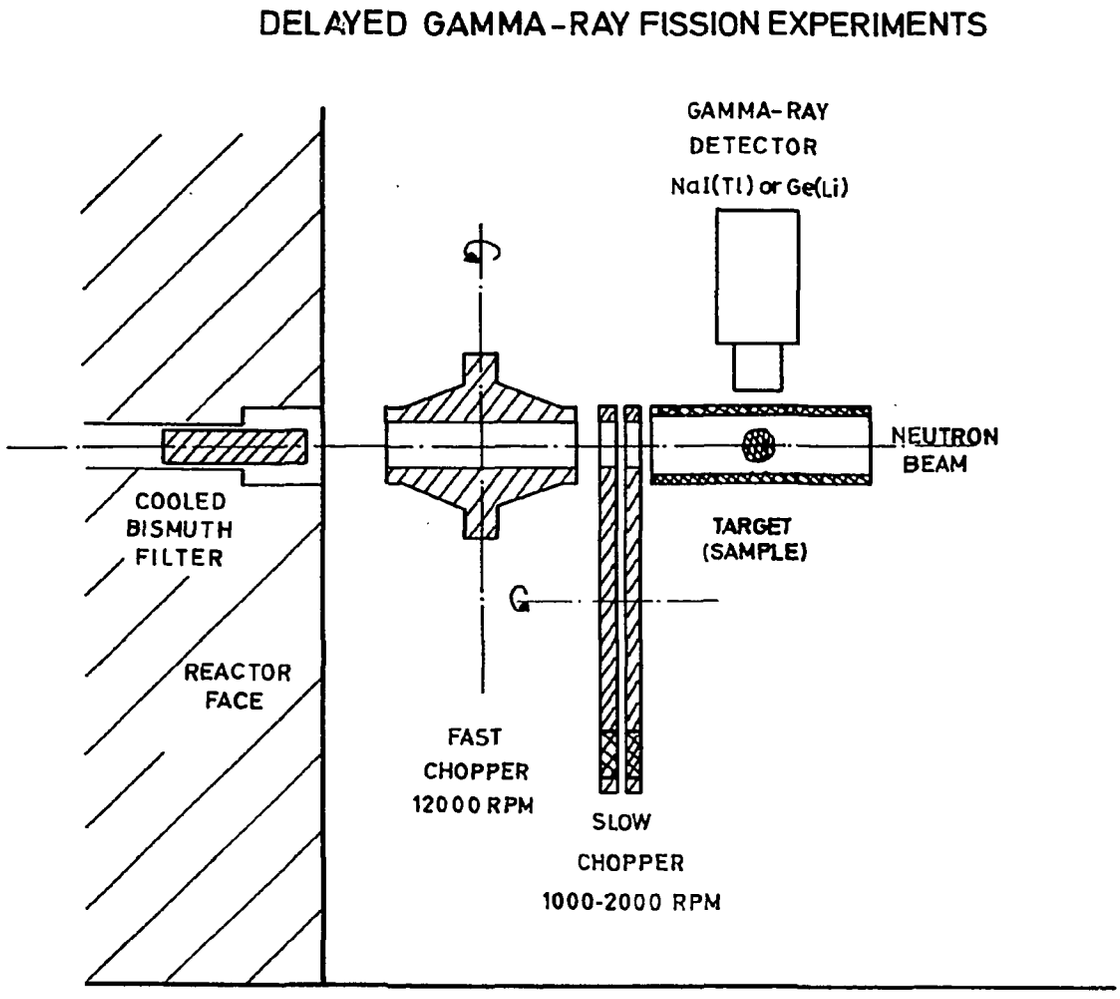

Fig. 5

Schematic view of the apparatus for delayed gamma-ray experiments.

In summary, we may conclude that a lot of new data can be accumulated with the instruments described. It is reasonable to assume that the results will exhibit the feasibility of new techniques in nondestructive fuel assay thus providing useful contributions to the field of safeguards technology.

References:

1) W. MICHAELIS, Atomkernenergie (in press).

2) Private Communication.

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4) P. C. FISHER and L. B. ENGLE, Phys. Rev. 134, 796 (1964).

5) R. L. HEATH, WASH 1076, 115.

6) N. C. RASMUSSEN, WASH 1076, 130.